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3.4 General Standards

3.4.1 Chemical and Galvanic Reactions

The materials used in the fabrication and operation of the Universal Storage System are evaluated to determine whether chemical, galvanic or other reactions among the materials, contents, and environments can occur. All phases of operation — loading, unloading, handling, and storage — are considered for the environments that may be encountered under normal, off-normal, or accident conditions. Based on the evaluation, no potential reactions that could adversely affect the overall integrity of the vertical concrete cask, the fuel basket, the transportable storage canister or the structural integrity and retrievability of the fuel from the canister have been identified. The evaluation conforms to the guidelines of NRC Bulletin 96-04 [18].

3.4.1.1 Component Operating Environment

Most of the component materials of the Universal Storage System are exposed to two typical operating environments: 1) an open canister containing fuel pool water or borated water with a pH of 4.5 and spent fuel or other radioactive material; or 2) a sealed canister containing helium, but with external environments that include air, rain water/snow/ice, and marine (salty) water/air. Each category of canister component materials is evaluated for potential reactions in each of the operating environments to which those materials are exposed. These environments may occur during fuel loading or unloading, handling or storage, and include normal, off-normal, and accident conditions.

The long-term environment to which the canister's internal components are exposed is dry helium. Both moisture and oxygen are removed prior to sealing the canister. The helium displaces the oxygen in the canister, effectively precluding chemical corrosion. Galvanic corrosion between dissimilar metals in electrical contact is also inhibited by the dry environment inside the sealed canister. NAC's operating procedures provide two helium backfill cycles in series separated by a vacuum-drying cycle during the preparation of the canister for storage. Therefore, the sealed canister cavity is effectively dry and galvanic corrosion is precluded.

The control element assembly, thimble plugs and non-fuel components—including start-up sources, instrument segments, and the boronometer source—are non-reactive with the fuel assembly. By design, the control components and non-fuel components, other than the boronometer source, are inserted in the guide tubes of a fuel assembly. During reactor operation, the control and non-fuel components are immersed in acidic water having a high flow rate and are exposed to significantly higher neutron flux, radiation and pressure than will exist in dry storage. The boronometer was housed in a vessel external to the reactor vessel. Reactor coolant (acidic water) was let down from the reactor coolant system, cooled and depressurized prior to being passed through the boronometer vessel. The control and non-fuel components are physically placed in storage in a dry, inert atmosphere in the same configuration as when used in the reactor, with the exception of the boronometer source. However, the boronometer source is the same material as the start-up source, so its storage in a guide tube is acceptable. There are no adverse reactions, such as gas generation, galvanic or chemical reactions or corrosion, that occur in the reactor coolant water, since these components are designed for use and operation in this environment and are non-reactive with the Zircaloy guide tubes and fuel rods. There are no aluminum or carbon steel parts, and no gas generation or corrosion occurs during prolonged water immersion (20 – 40 years). Thus, no adverse reactions occur with the control and non-fuel components over prolonged periods of dry storage.

3.4.1.2 Component Material Categories

The component materials are categorized in this section for their chemical and galvanic corrosion potential on the basis of similarity of physical and chemical properties and component functions. The categories are stainless steels, nonferrous metals, carbon steel, coatings, concrete, and criticality control materials. The evaluation is based on the environment to which these categories could be exposed during operation or use of the canister.

The canister component materials are not reactive among themselves, with the canister's contents, nor with the canister's operating environments during any phase of normal, off-normal, or accident condition, loading, unloading, handling, or storage operations. Since no reactions will occur, no gases or other corrosion by-products will be generated.

The control component and non-fuel component materials are those that are typically used in the fabrication of fuel assemblies, i.e., stainless steels, Inconel 625, and Zircaloy, so no adverse reactions occur in the inert atmosphere that exists in storage. The control element assembly, thimble plugs and non-fuel components—including start-up sources or instrument segments to be inserted into a fuel assembly—are non-reactive among themselves, with the fuel assembly, nor with the canister's operating environment for any storage condition.

3.4.1.2.1 Stainless Steels

No reaction of the canister component stainless steels is expected in any environment except for the marine environment, where chloride-containing salt spray could potentially initiate pitting of the steels if the chlorides are allowed to concentrate and stay wet for extended periods of time (weeks). Only the external canister surface could be so exposed. The corrosion rate will, however, be so low that no detectable corrosion products or gases will be generated. The Universal Storage System has smooth external surfaces to minimize the collection of such materials as salts.

Galvanic corrosion between the various types of stainless steels does not occur because there is no effective electrochemical potential difference between these metals. No coatings are applied to the stainless steels. An electrochemical potential difference does exist between austenitic (300 series) stainless steel and aluminum. However, the stainless steel becomes relatively cathodic and is protected by the aluminum.

The canister confinement boundary uses Type 304L stainless steel for all components, except the shield lid, which is made of Type 304 stainless steel. Type 304L resists chromium-carbide precipitation at the grain boundaries during welding and assures that degradation from intergranular stress corrosion will not be a concern over the life of the canister. Fabrication specifications control the maximum interpass temperature for austenitic steel welds to less than 350°F. The material will not be heated to a temperature above 800°F, other than by welding thermal cutting. Minor sensitization of Type 304 stainless steel that may occur during welding will not affect the material performance over the design life because the storage environment is relatively mild.

Based on the foregoing discussion, no potential reactions associated with the stainless steel canister or basket components are expected to occur.

3.4.1.2.2 Nonferrous Metals

Aluminum is used as a heat transfer component in the Universal Storage System spent fuel basket, and aluminum components in electrical contact with austenitic stainless steel could experience corrosion driven by electrochemical EMF when immersed in water. The conductivity of the water is the dominant factor. BWR fuel pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. PWR pool water, however, does provide a conductive medium. The only aluminum components that will be in contact with stainless steel and exposed to the pool water are the alloy 6061-T651 heat transfer disks in the fuel basket.

Aluminum produces a thin surface film of oxidation that effectively inhibits further oxidation of the aluminum surface. This oxide layer adheres tightly to the base metal and does not react readily with the materials or environments to which the fuel basket will be exposed. The volume of the aluminum oxide does not increase significantly over time. Thus, binding due to corrosion product build-up during future removal of spent fuel assemblies is not a concern. The borated water in a PWR fuel pool is an oxidizing-type acid with a pH on the order of 4.5. However, aluminum is generally passive in pH ranges down to about 4 [19]. Data provided by the Aluminum Association [20] shows that aluminum alloys are resistant to aqueous solutions (1-15 %) of boric acid (at 140°F). Based on these considerations and the very short exposure of the aluminum in the fuel basket to the borated water, oxidation of the aluminum is not likely to occur beyond the formation of a thin surface film. No observable degradation of aluminum components is expected as a result of exposure to BWR or PWR pool water at temperatures up to 200°F, which is higher than the permissible fuel pool water temperature.

Aluminum is high on the electromotive potential table, and it becomes anodic when in electrical contact with stainless or carbon steel in the presence of water. BWR pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. PWR pool water is sufficiently conductive to allow galvanic activity to begin. However, exposure time of the aluminum components to the PWR pool environment is short. The long-term storage environment is sufficiently dry to inhibit galvanic corrosion.

From the foregoing discussion, it is concluded that the initial surface oxidation of the aluminum component surfaces effectively inhibits any potential galvanic reactions.

Heat transfer disks fabricated from 6061-T651 aluminum alloy are used in the NAC-UMS® Universal Storage System PWR and BWR fuel baskets to augment heat transfer from the spent fuel through the basket structure to the canister exterior. Vendor and Nuclear Regulatory Commission safety evaluations of the NUHOMS Dry Spent Fuel Storage System (Docket No. 72-1004) have concluded that combustible gases, primarily hydrogen, may be produced by a chemical reaction and/or radiolysis when aluminum or aluminum flame-sprayed components are immersed in spent fuel pool water. The evaluations further concluded that it is possible, at higher temperatures (above 150 - 160°F), for the aluminum/water reaction to produce a hydrogen concentration in the canister that approaches or exceeds the Lower Flammability Limit (LFL) for hydrogen of 4 percent. The NRC Inspection Reports No. 50-266/96005 and 50-301/96005 dated July 01, 1996, for the Point Beach Nuclear Plant concluded that hydrogen generation by radiolysis was insignificant relative to other sources.

Thus, it is reasonable to conclude that small amounts of combustible gases, primarily hydrogen, may be produced during UMS Storage System canister loading or unloading operations as a result of a chemical reaction between the 6061-T6 aluminum heat transfer disks in the fuel basket and the spent fuel pool water. The generation of combustible gases stops when the water is removed from the cask or canister and the aluminum surfaces are dry.

A galvanic reaction may occur at the contact surfaces between the aluminum disks and the stainless steel tie rods and spacers in the presence of an electrolyte, like the pool water. The galvanic reaction ceases when the electrolyte is removed. Each metal has some tendency to ionize, or release electrons. An Electromotive Force (EMF) associated with this release of electrons is generated between two dissimilar metals in an electrolytic solution. The EMF between aluminum and stainless steel is small and the amount of corrosion is directly proportional to the EMF. Loading operations generally take less than 24 hours, a large portion of which has the canister immersed in and open to the pool water after which the electrolyte (water) is drained and the cask or canister is dried and back-filled with helium, effectively halting any galvanic reaction.

The potential chemical or galvanic reactions do not have a significant detrimental effect on the ability of the aluminum heat transfer disks to perform their function for all normal and accident conditions associated with dry storage.

Loading Operations

After the canister is removed from the pool and during canister closure operations, an air space is created inside the canister beneath the shield lid by the drain-down of 50 gallons of water so that the shield-lid-to-canister-shell weld can be performed. The resulting air space is approximately

66 inches in diameter and 3 inches deep. As there is some clearance between the inside diameter of the canister shell and the outside diameter of the shield lid, it is possible that gases released from a chemical reaction inside the canister could accumulate beneath the shield lid. A bare aluminum surface oxidizes when exposed to air, reacts chemically in an aqueous solution, and may react galvanically when in contact with stainless steel in the presence of an aqueous solution.

The reaction of aluminum in water, which results in hydrogen generation, proceeds as:

$$2\,Al + 3\,H_20 \Rightarrow Al_2O_3 + 3\,H_2$$

The aluminum oxide (Al_2O_3) produces the dull, light gray film that is present on the surface of bare aluminum when it reacts with the oxygen in air or water. The formation of the thin oxide film is a self limiting reaction as the film isolates the aluminum metal from the oxygen source acting as a barrier to further oxidation. The oxide film is stable in pH neutral (passive) solutions, but is soluble in borated PWR spent fuel pool water. The oxide film dissolves at a rate dependent upon the pH of the water, the exposure time of the aluminum in the water, and the temperatures of the aluminum and water.

PWR spent fuel pool water is a boric acid and demineralized water solution. BWR spent fuel pool water does not contain boron and typically has a neutral pH (approximately 7.0). The pH, water chemistry, and water temperature vary from pool to pool. Since the reaction rate is largely dependent upon these variables, it may vary considerably from pool to pool. Thus, the generation rate of combustible gas (hydrogen) that could be considered representative of spent fuel pools in general is very difficult to accurately calculate, but the reaction rate would be less in the neutral pH BWR pool.

The BWR basket configuration incorporates carbon steel support plates that are coated with electroless nickel. The coating protects the carbon steel during the comparatively short time that the canister is immersed in, or contains, water. The coating is described in Section 3.8.3. The coating is non-reactive with the BWR pool water and does not off-gas or generate gases as a result of contact with the pool water. Consequently, there are no flammable gases that are generated by the coating. A coating is not used in PWR basket configurations.

To ensure safe loading and/or unloading of the UMS transportable storage canister, the loading and unloading procedures defined in Chapter 8 are revised to provide for the monitoring of hydrogen gas before and during the welding operations joining the shield lid to the canister shell, and joining the vent and drain port covers to the shield lid. The monitoring system shall be capable of detecting hydrogen at 60% of the lower flammability limit for hydrogen (i.e. 0.6 x 4.0 = 2.4%). The hydrogen detector shall be mounted so as to detect hydrogen prior to initiation of the weld, and continuously during the welding operation. Detection of hydrogen in a concentration exceeding 2.4% shall be cause for the welding operation to stop. If hydrogen gas is detected at concentrations above 2.4% at any time, the hydrogen gas shall be removed by flushing ambient air into the region below the shield lid or port cover. To remove hydrogen from below the shield lid, the vacuum pump is attached to the vent port and operated for a sufficient period of time to remove at least five times the air volume of the space below the lid by drawing ambient air through the gap between the shield lid and the canister shell, thus removing or diluting any combustible gas concentrations.

The vacuum pump shall exhaust to a system or area where hydrogen flammability is not an issue. If hydrogen gas is detected at the port covers, the cover is removed and service air is used to flush combustible gases from the port. Once the root pass weld is completed there is no further likelihood of a combustible gas burn because the ignition source is isolated from the combustible gas. Once welding of the shield lid has been completed, the canister is drained, vacuum dried and back-filled with helium.

No hydrogen is expected to be detected prior to, or during, the welding operations. The vent port in the shield lid remains open from the time that the loaded canister is removed from the spent fuel pool until the time that the vent port cover is ready to be welded to the shield lid. Since the postulated combustible gases are very light, the open vent port provides an escape path for any gases that are generated prior to the time that the canister is vacuum dried. Once the canister is dry, no combustible gases form within the canister. The mating surfaces of the support ring and inner lid are machined to provide a good level fitup, but are not machined to provide a metal to metal seal. Consequently, additional exit paths for the combustible gases exist at the circumference of the shield lid.

Unloading Operations

It is not expected that the canister will contain a measurable quantity of combustible gases during the time period of storage. The canister is vacuum dried and backfilled with helium immediately prior to being welded closed. There are only minor mechanisms by which hydrogen is generated after the canister is dried and sealed.

As shown in Section 8.3, the principal steps in opening the canister are the removal of the structural lid, the removal of the vent and drain port covers, and the removal of the shield lid. These steps are expected to be performed by cutting or grinding. The design of the canister precludes monitoring for the presence of combustible gases prior to the removal of the structural lid and the vent or drain port covers. Following removal of the vent port cover, a vent line is connected to the vent port quick disconnect. The vent line incorporates a hydrogen gas detector which is capable of detecting hydrogen at a concentration of 2.4% (60% of its lower flammability limit of 4%). The pressurized gases (expected to be greater than 96% helium) in the canister are expected to carry combustible gases out of the vent port. If the exiting gases in the vent line contain no hydrogen at concentrations above 2.4%, the drain port cover weld is cut and the cover removed. If levels of hydrogen gas above 2.4% concentration are detected in the vent line, then the vacuum system is used to remove all residual gas prior to removal of the drain port cover. During the removal of the drain port cover, the hydrogen gas detector is attached to the vent port to ensure that the hydrogen gas concentration remains below 2.4%. Following removal of the drain port cover, the canister is filled with water using the vent and drain ports. Prior to cutting the shield lid weld, 50 gallons of water are removed from the canister to permit the removal of the shield lid. Monitoring for hydrogen would then proceed as described for the loading operations.

3.4.1.2.3 Carbon Steel

Carbon steel support disks are used in the BWR basket configuration. There is a small electrochemical potential difference between carbon steel (SA-533) and aluminum and stainless steel. When in contact in water, these materials exhibit limited electrochemically-driven corrosion. BWR pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. In addition, the carbon steel support disks are coated with electroless nickel to protect the carbon steel surface during exposure to air or to spent fuel pool water, further reducing the possibility of corrosion. Once the canister is loaded, the water is drained from the cavity, the air is evacuated, and the canister is backfilled with helium and sealed. Removal of the water and the moisture eliminates the catalyst for galvanic corrosion. The canister operating procedures (see Chapter 8) provide two backfill cycles in series separated by a vacuum drying cycle during closing of the canister. The displacement of oxygen by helium effectively inhibits corrosion.

The transfer cask structural components are fabricated primarily from ASTM A588 and A36 carbon steel. The exposed carbon steel components are coated with either Keeler & Long E-Series Epoxy Enamel or Carboline 890 to protect the components during in-pool use and to provide a smooth surface to facilitate decontamination.

The concrete shell of the vertical concrete cask contains an ASTM A36 carbon steel liner, as well as other carbon steel components. The exposed surfaces of the base of the concrete cask and the liner are coated with either Keeler & Long E-Series Epoxy Enamel, or Carboline 890, to provide protection from weather related moisture.

No potential reactions associated with the BWR basket carbon steel disks, the transfer cask components or vertical concrete cask components are expected to occur.

3.4.1.2.4 Coatings

The exposed carbon steel surfaces of the transfer cask, the transfer cask adaptor plate and the vertical concrete cask are coated with either Keeler & Long E-Series Epoxy Enamel or Carboline 890. These coatings are approved for Nuclear Service Level 2 use. Load bearing surfaces (i.e., the bottom surface of the trunnions and the contact surfaces of the transfer cask doors and rails) are not painted, but are coated with an appropriate nuclear grade lubricant, such as Neolube®. The technical specifications for these coatings are provided in Sections 3.8.1 and 3.8.2, respectively.

Carbon steel support disks used in the BWR canister basket are coated with electroless nickel. The coating is applied in accordance with ASTM B733-SC3, Type V, Class 1[37]. As described in Section 3.8.3, the electroless nickel coating process uses a chemical reducing agent in a hot aqueous solution to deposit nickel on a catalytic surface. The deposited nickel coating is a hard alloy of uniform thickness of 25 μm (0.001 inch), containing from 4% to 12% phosphorus. Following its application, the nickel coating combines with oxygen in the air to form a passive oxide layer that effectively eliminates free electrons on the surface that would be available to cathodically react with water to produce hydrogen gas. Consequently, the production of hydrogen gas in sufficient quantities to facilitate combustion is highly unlikely.

3.4.1.2.5 Concrete

The vertical concrete storage cask is fabricated of 4000 psi, Type 2 Portland cement that is reinforced with vertical and circumferential carbon steel rebar. Quality control of the proportioning, mixing, and placing of the concrete, in accordance with the NAC fabrication specification, will make the concrete highly resistant to water. The concrete shell is not expected to experience corrosion, or significant degradation from the storage environment through the life of the cask.

3.4.1.2.6 Criticality Control Material

The criticality control material is boron carbide mixed in an aluminum alloy matrix. Sheets of this material are affixed to one or more sides of the designated fuel tubes and completely enclosed by a welded stainless steel cover. The material resists corrosion similar to aluminum, and is protected by an oxide layer that forms shortly after fabrication and inhibits further interaction with the stainless steel. Consequently, no potential reactions associated with the aluminum-based criticality control material are expected.

3.4.1.2.7 Neutron Shielding Material

The neutron shielding material is a hydrogenated polymer, NS-4-FR, consisting primarily of aluminum, carbon, oxygen and hydrogen, to which boron carbide (B_4C) is added to improve shielding effectiveness. It is used in the transfer cask and in the shield plug of the vertical concrete storage cask to provide radiation shielding. The acceptable performance of the material has been demonstrated by use and testing. The material has been used in two licensed storage casks in the United States for up to 10 years and in more than 50 licensed casks in Japan, Spain and the United Kingdom. There are no reports that the shielding effectiveness of NS-4-FR material has degraded in these applications, demonstrating the long-term reliability for the purpose of shielding neutrons from personnel and the environment. There are no potential reactions associated with the polymer structure of the material and the stainless steel or carbon steel in which it is encapsulated during use.

The chemistry of the material (e.g., the way the elements are bonded to one another) contributes significantly to the fire retardant capability of the NS-4-FR. Even though the material contains hydrogen, the ingredients were selected so that the NS-4-FR resists fire. Approximately 90% of the off-gassing that does occur consists of water vapor.

The thermal performance of the NS-4-FR has been demonstrated by long-term functional stability tests of the material at temperatures from -40°F to 338°F. These tests included specimens open to the atmosphere and enclosed in a cavity at both constant and cyclic thermal loads. The tests evaluated material loss though off-gassing and material degradation. The results of the tests demonstrate that, in the temperature range of interest, the NS-4-FR does not exhibit loss of material by off-gassing, does not generate any significant gases, and does not suffer degradation or embrittlement. Further, the tests demonstrated that encased material, as it is used in the NAC-UMS®, performed significantly better than exposed material. Consequently, the formation of flammable gases is not a concern.

Radiation exposure testing of NS-4-FR in reactor pool water demonstrated no physical deterioration of the material and no significant loss of hydrogen (less than 1%). The tests also demonstrated that the NS-4-FR retains its neutron shield capability over the cask's 50-year design life with substantial margin. The radiation testing has shown that detrimental embrittlement and loss of hydrogen from the material do not occur at dose rates (9×10^{14} n/cm^2) that exceed those that would occur assuming the continuous storage of design basis fuel for a 50-year life (estimated to be 1.7×10^{12} cm^2/yr). Consequently, detrimental deterioration or embrittlement due to radiation flux does not occur.

Since the NS-4-FR in the NAC-UMS® transfer cask is sandwiched between the shell and the lead shield and enclosed within a welded steel shell where the shell seams are welded to top and bottom plates with full penetration or fillet welds, it will maintain its form over the expected lifetime of the transfer cask's radiation exposure. The material's placement between the lead shield and the outer shell does not allow the material to redistribute within the annulus.

The NS-4-FR shield material is similarly enclosed in the storage cask shield plug, since a disk of NS-4-FR is captured in an annulus formed by a carbon steel ring and two carbon steel plates. This material cannot redistribute within this volume.

3.4.1.3 General Effects of Identified Reactions

No potential chemical, galvanic, or other reactions have been identified for the Universal Storage System. Therefore, no adverse conditions, such as the generation of flammable or explosive quantities of combustible gases or an increase in neutron multiplication in the fuel (criticality) because of boron precipitation, can result during any phase of canister operations for normal, off-normal, or accident conditions.

3.4.1.4 Adequacy of the Canister Operating Procedures

Based on this evaluation, which results in no identified reactions, it is concluded that the Universal Storage System operating controls and procedures presented in Chapter 8.0 are adequate to minimize the occurrence of hazardous conditions.

3.4.1.5 Effects of Reaction Products

No potential chemical, galvanic, or other reactions have been identified for the Universal Storage System. Therefore, the overall integrity of the canister and the structural integrity and retrievability of the spent fuel are not adversely affected for any operations throughout the design basis life of the canister. Based on the evaluation, no change in the canister or fuel cladding thermal properties is expected, and no corrosion of mechanical surfaces is anticipated. No change in basket clearances or degradation of any safety components, either directly or indirectly, is likely to occur since no potential reactions have been identified.

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. Damaged fuel is separately containerized in Maine Yankee fuel cans.

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.

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:

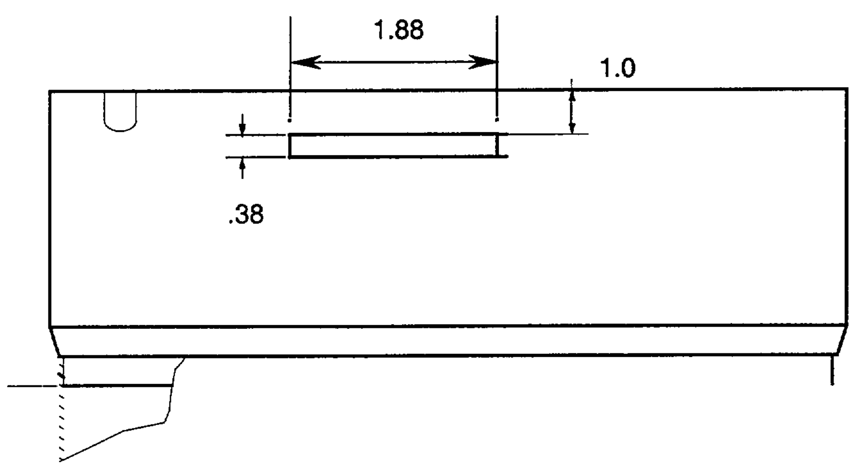
$$\text{M.S.} = 20{,}000/(143/1.714) - 1$$
$$\text{M.S.} = +\text{LARGE}$$

3.6.1.2.2 Lifting Evaluation

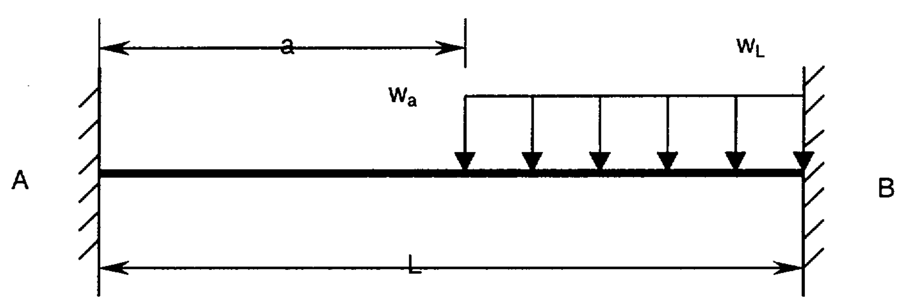
Based on the loaded weight of the fuel can, the lift evaluation does not require the use of the design criteria of ANSI N14.6 or NUREG-0612. However, for purposes of conservatism and good engineering practice, a factor of safety of three on material yield strength is used for the stress evaluations for 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.
$w_a = w_L = (2{,}230 \text{ lbs}/2)(1.10)/1.0$ 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:

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

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:

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

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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8) account for the preferential loading configuration for Maine Yankee site specific high burnup fuel (Section 4.5.1.2.2), with case 8 being the bounding case for the Maine Yankee high burnup fuel. Based on the analysis results of the 17.6 kW heat load cases, only two loading cases are required to establish the bounding condition for the 20, 14, 11 and 8 kW heat loads.

Canister Heat Load (kW)	Heat Load Case	Heat Load (kW) Evaluated in Each Basket Location (See Figure 4.4.3-7)					
		1	2	3	4	5	6
20	1	0.958	0.958	0.709	0.958	0.709	0.709
20*	2	0.958	0.958	0.958	0.958	0.958	0.210
17.6	3	0.958	0.958	0.509	0.958	0.509	0.509
17.6*	4	0.958	0.958	0.568	0.958	0.958	0.000
17.6	5	0.958	0.958	0.958	0.958	0.568	0.000
17.6	6	0.958	0.958	0.284	0.958	0.958	0.284
17.6	7	0.958	0.146	1.050	0.146	1.050	1.050
17.6	8	0.958	0.958	1.050	0.384	1.050	0.000
14	9	0.958	0.958	0.209	0.958	0.209	0.209
14*	10	0.958	0.958	0.000	0.958	0.626	0.000
11	11	0.958	0.896	0.000	0.896	0.000	0.000
11*	12	0.958	0.958	0.000	0.834	0.000	0.000
8	13	0.958	0.521	0.000	0.521	0.000	0.000
8*	14	0.958	0.958	0.000	0.084	0.000	0.000

The heat load (23 kW/24 = 0.958 kW) at the four (4) central basket locations corresponds to the maximum allowable canister heat load for 5-year cooled fuel (Table 4.4.7-8). The non-uniform heat loads evaluated in this section bound the equivalent uniform heat loads, since they result in higher maximum temperatures of the fuel cladding and heat transfer disk.

Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region in each fuel assembly location of the model using the axial power distribution for PWR fuel (Figure 4.4.1.1-3) in the axial direction.

The thermal analysis results for the closure and transfer of a loaded PWR fuel canister in the transfer cask for the reduced heat load cases are shown in Table 4.4.3-5. The temperatures shown are the maximum temperatures for the limiting components (fuel cladding and heat transfer disk). The maximum temperatures of the fuel cladding and the heat transfer disk are less than the allowable temperatures (Table 4.1-3) of these components for the short-term conditions of vacuum drying and helium backfill. As shown in Table 4.4.3-5, there is no time limit for movement of the canister out of the transfer cask for the cases with a heat load less than 14 kW, after the canister is filled with helium. For heat loads equal to or less than 14 kW, the maximum fuel cladding/heat transfer disk temperatures for the steady state condition are well below the short term allowable temperatures of the fuel cladding and the heat transfer disk. Note that the maximum water temperature at the end of the "water period" is considered to be the volumetric average temperature of the calculated cladding temperatures in the active fuel region of the hottest fuel assembly. The results indicate that the volumetric average water temperature is below 212°F for all cases evaluated. This is consistent with the thermal model that only considers conduction in the fuel assembly region and between the disks. This approach does not include consideration of convection of the water or the energy absorbed by latent heat of vaporization.

The Technical Specifications specify the remedial actions, either in-pool or forced air cooling, required to ensure that the fuel cladding and basket component temperatures do not exceed their short-term allowable temperatures, if the time limits are not met. LCOs 3.1.1 and 3.1.4 incorporate the operating times for heat loads that are less than the design basis heat loads as evaluated in this section.

Using the same three-dimensional transfer cask/canister model, analysis is performed for the conditions of in-pool cooling followed by the vacuum drying and helium backfill operation (LCO 3.1.1). The condition at the end of the vacuum drying as shown in Table 4.4.3-5 is used as the initial condition of the analysis. The LCO 3.1.1 "Action" analysis results are shown in Table 4.4.3-6. The maximum temperatures for the fuel cladding and the heat transfer disk are below the short-term allowable temperatures.

The in-pool cooling followed by the helium backfill operation in LCO 3.1.4 is also evaluated. The condition at the end of the helium condition as shown in Table 4.4.3-5 is used as the initial condition. Based on the in-pool cooling analysis for LCO 3.1.1, the minimum temperature reduction due to in-pool cooling is 216°F (706-490) for the 20 kW heat load case. The evaluation for LCO 3.1.4 in-pool cooling conservatively considers a temperature reduction of 150°F for in-pool cooling and a heat up rate of 6°F/hour (helium condition) for an additional 16 hours and 20 hours for 20 kW and 17.6 kW heat load cases, respectively. The maximum fuel temperature and heat transfer disk temperatures at the end of the helium condition for the governing case of 17.6 kW are determined to be 668°F ((698-150)+(20×6)) and 612°F ((642-150)+(20×6)), respectively, which are well below the short-term allowable temperatures.

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. Standard fuel assemblies with inserted start-up 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.

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, 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

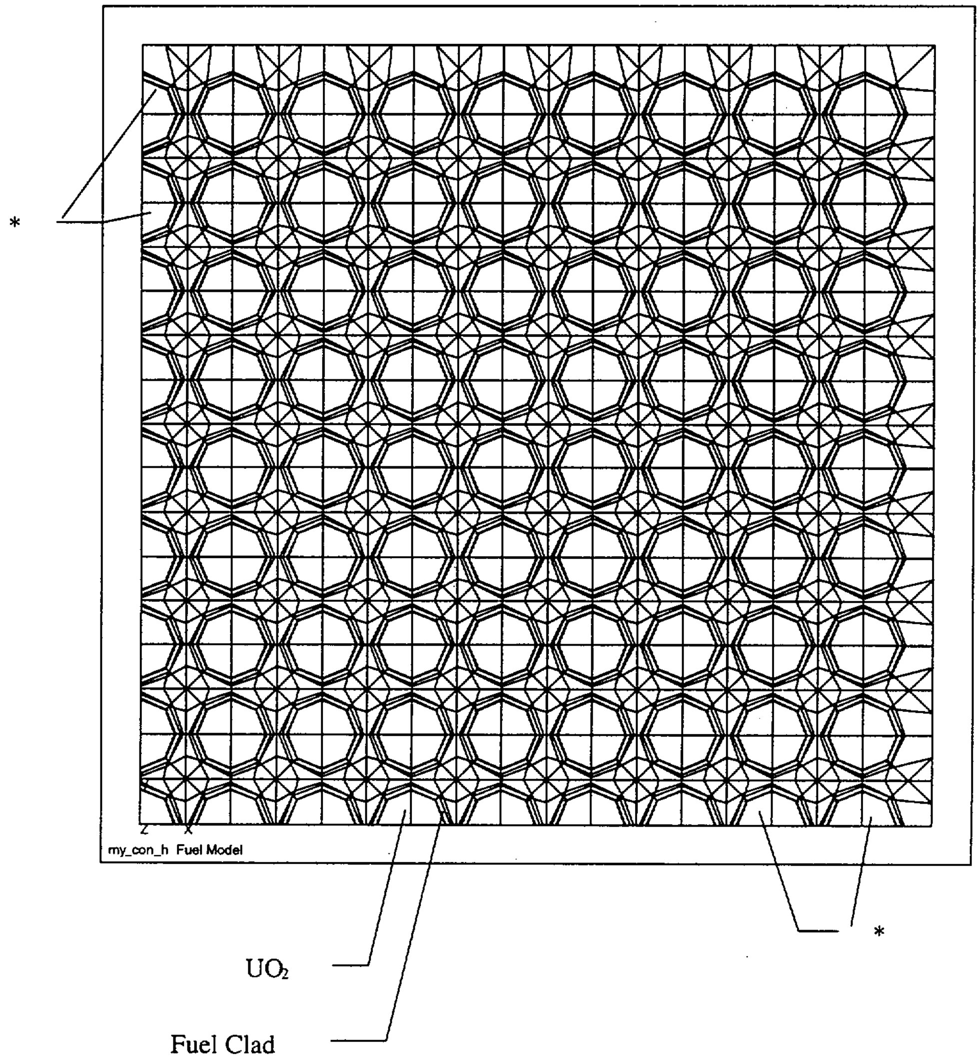
Similar to fuel assemblies that have damaged fuel rods, fuel assemblies that have up to two damaged fuel rods or poison rods stored in each guide tube are placed in Maine Yankee fuel cans and their loading positions are restricted to the four corner fuel tubes in the basket. The rods inserted in the 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 Standard Fuel Assemblies with Inserted Start-up Sources and Other Non-Fuel Items

Five Control Element Assembly (CEA) fingertips, a 24-inch ICI segment, and a neutron source associated with the boronometer may be placed into the guide tubes of a fuel assembly. In addition, four irradiated start-up neutron sources and one unirradiated source, having a combined total heat load of 15.4 watts, 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. In addition, the fuel assemblies that hold these non-fuel items are restricted to basket corner loading locations, which have an insignificant effect on the maximum fuel cladding and basket component temperatures at the center of the basket.

Note that the total heat load of the fuel assembly, including the small amount of extra heat generated by the CEA fingertips, ICI 24-inch segment, and the neutron sources, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with these non-fuel items 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

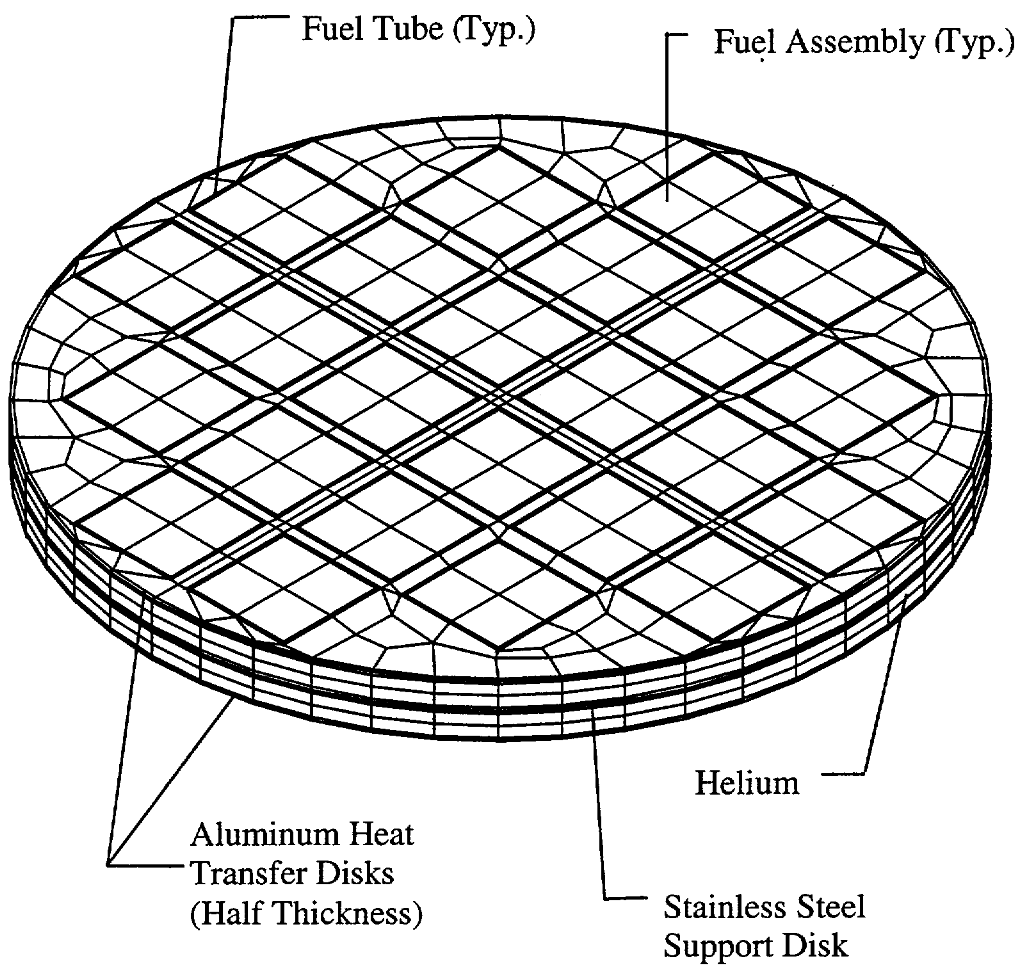


Figure 4.5.1.1-3 Evaluated Locations for the Maine Yankee Consolidated Fuel Lattice in the PWR Fuel Basket

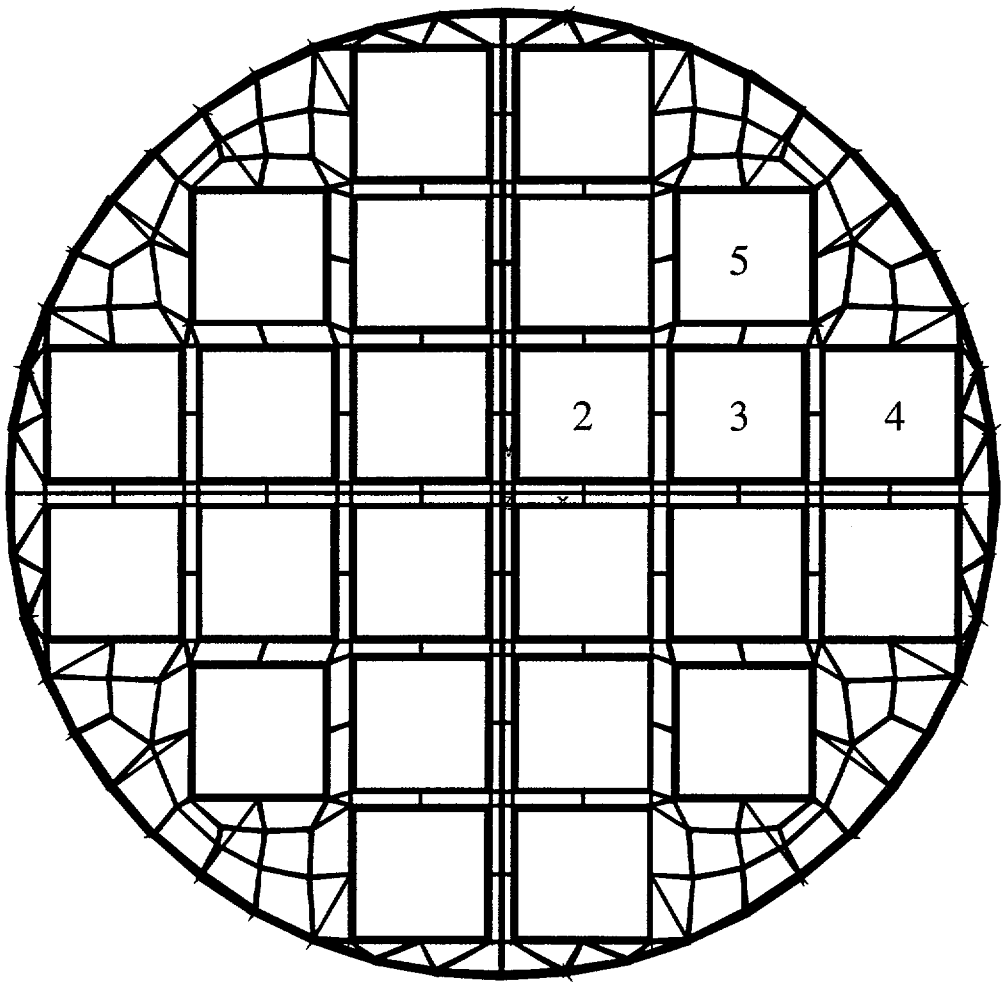
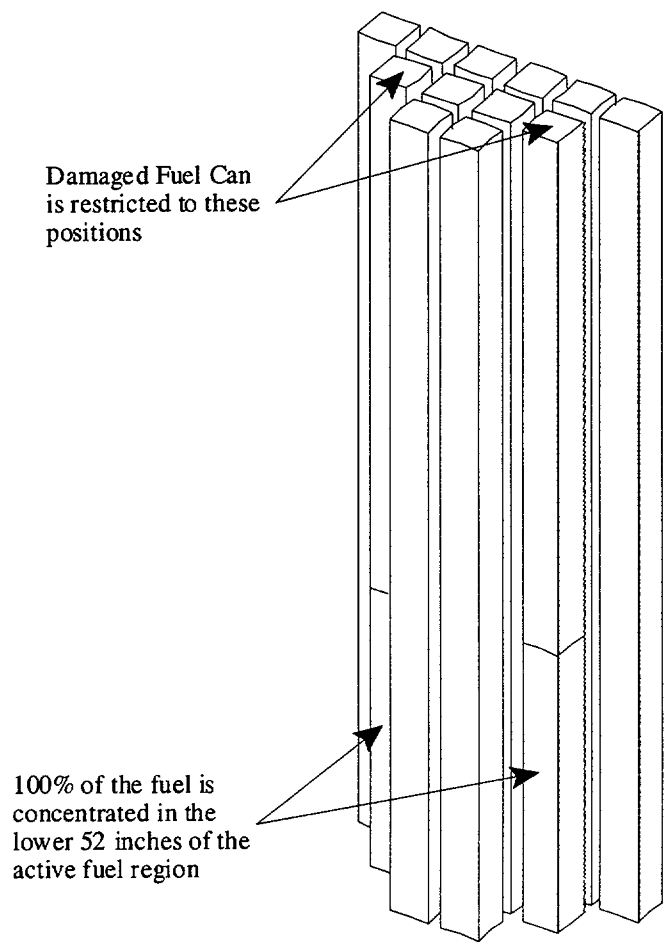
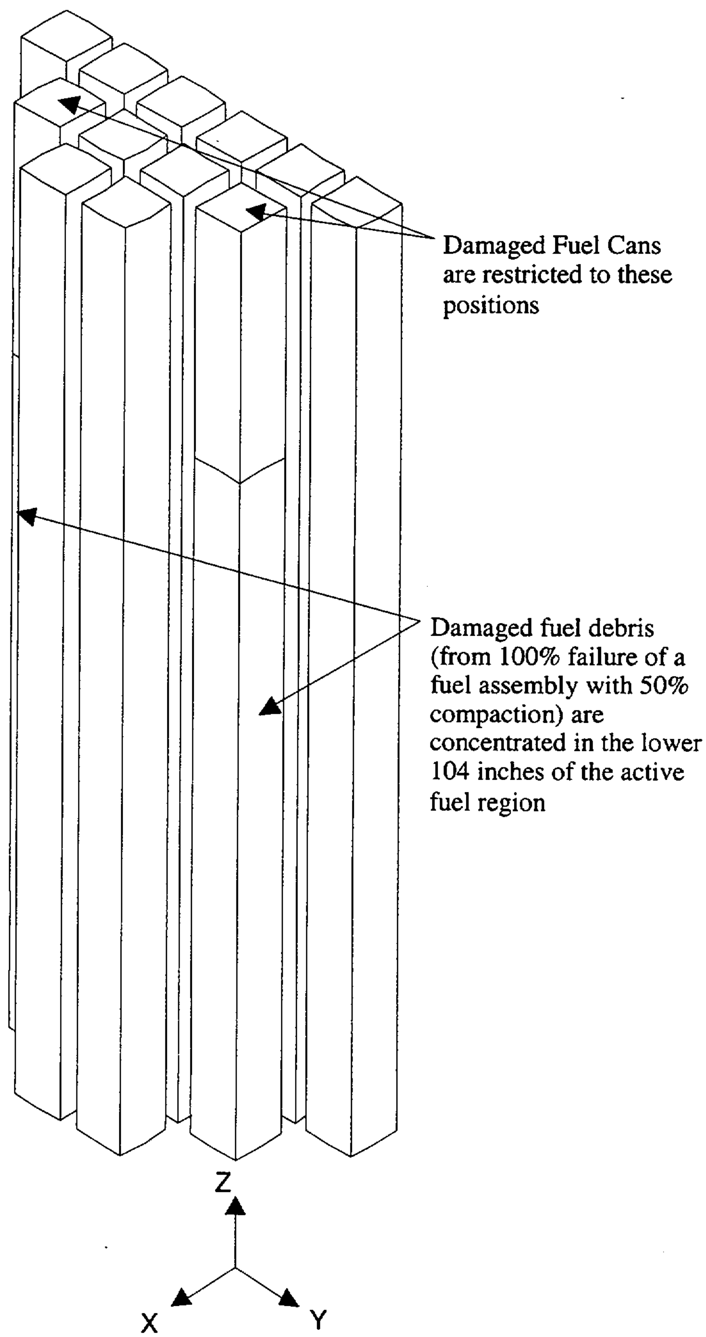


Figure 4.5.1.1-4 Active Fuel Region in the Three-Dimensional Canister Model



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.

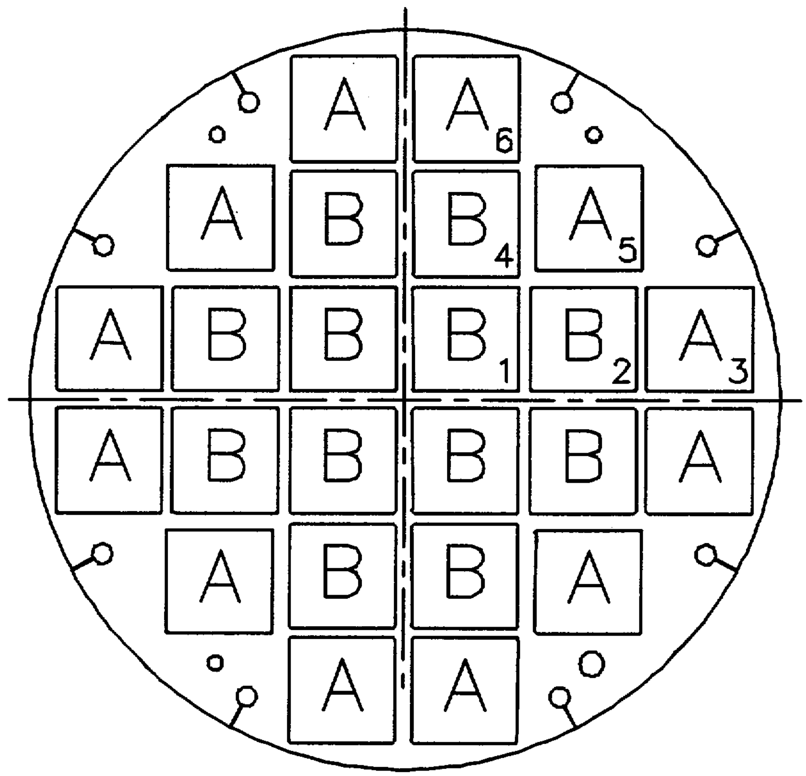
The maximum basket heat load for this configuration is restricted to 23 kW. Given the higher than design basis heat load in peripheral basket locations, an evaluation is performed to assure that maximum cladding allowable temperatures are not exceeded.

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)

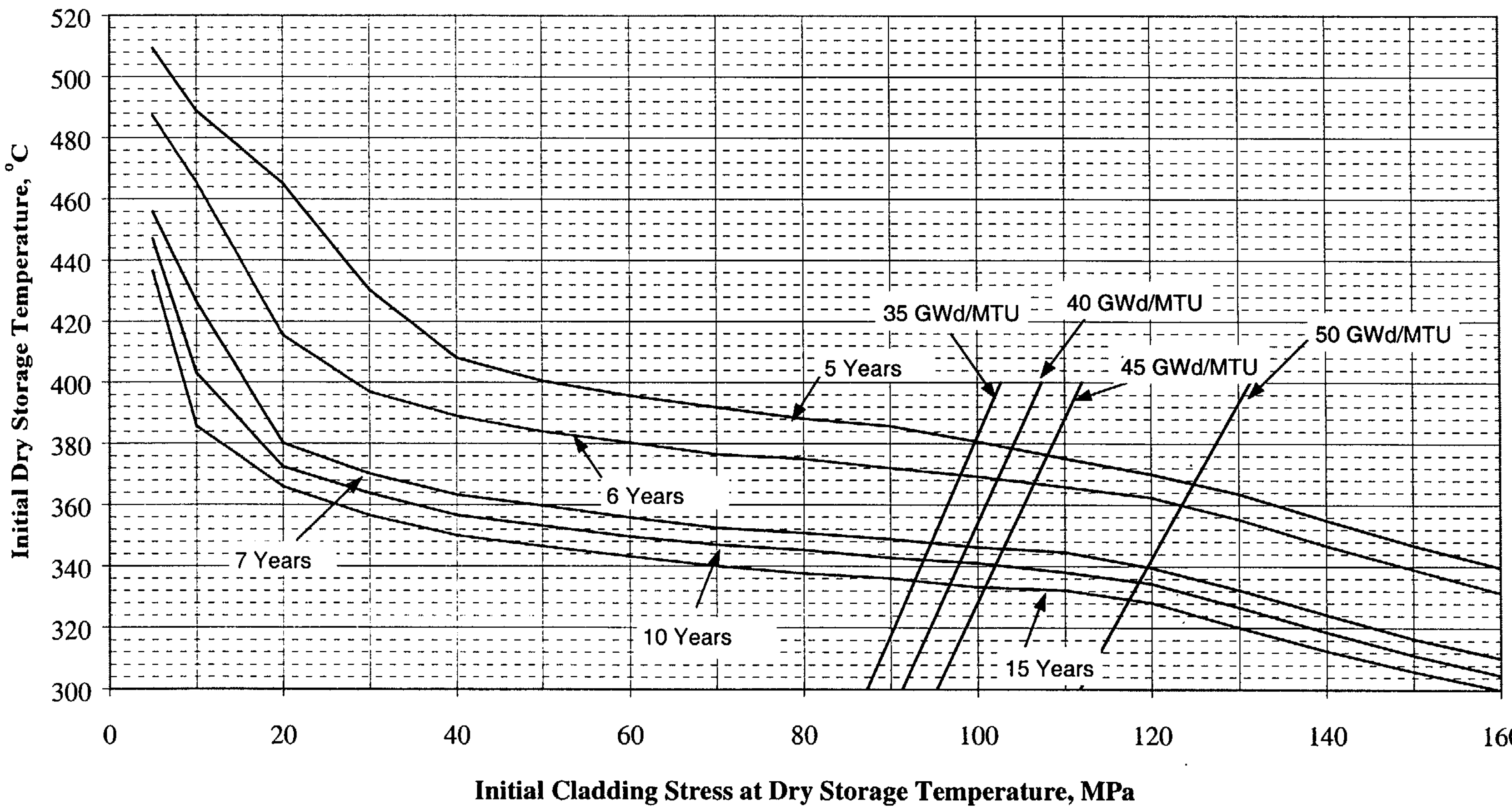


Table 4.5.1.2-1 Cladding Stress for 50,000 MWD/MTU Burnup Fuel

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 Cool Time (years)	Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000
	---	---	---	---
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 Cool Time (years)	Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000
	---	---	---	---
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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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) segments, 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 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 includes fuel assemblies containing damaged fuel rods, damaged fuel rods and fuel debris. Fuel assemblies containing damaged fuel rods, damaged fuel rods and fuel debris will be placed into a screened Maine Yankee fuel can prior to loading in the UMS® basket. 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.

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.

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	4.4833E+13

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

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Corresponding value for the cask containing BWR fuel assemblies is 0.38168 under normal storage conditions, 0.38586 under off-normal conditions and 0.92332 under accident conditions involving full moderator intrusion. These values reflect the following conditions:

- A method bias and uncertainty associated with KENO-Va and the 27 group ENDF/B-IV library
- An infinite cask array
- Normal conditions is defined to be a dry basket, dry heat transfer annulus and dry exterior
- Accident conditions is defined to be full interior, exterior and fuel clad gap moderator (water) intrusion
- Westinghouse 17x17 OFA fuel assemblies at 4.2 wt % ^{235}U (most reactive PWR fuel assembly type) or 56 Ex/ANF 9x9-79 rod fuel assemblies at 4.0 wt % ^{235}U (most reactive BWR fuel assembly type)
- No fuel burnup
- 75% of nominal ^{10}B loading in BORAL
- Most reactive mechanical configuration for PWR (Assemblies and fuel tubes moved toward the center of the basket; maximum fuel tube openings; minimum BORAL sheet widths and closely packed disk openings)
- Most reactive mechanical configuration for BWR (Assemblies and fuel tubes moved toward the center of the basket)

Analysis of simultaneous moderator density variation inside and outside the concrete cask shows a monotonic decrease in reactivity with decreasing moderator density. Thus, the full moderator density situation bounds any off normal or accident condition. Analysis of moderator intrusion into the cask heat transfer annulus with a dry canister shows a slight decrease in reactivity from the completely dry situation. This is due to better neutron reflection from the concrete cask steel shell and concrete shielding with no moderator present.

Analysis of the BWR cask reactivity of the fuel assemblies in the axial region above the top of partial length rods shows this region to be less reactive than the region with all of the fuel rods present. Therefore, it is appropriate to represent partial length rods as full length rods in the BWR fuel models.

6.4.3.2 Criticality Results for PWR

Transfer Cask

Results of the calculations for the transfer cask containing PWR fuel are provided in Tables 6.4-11 through 6.4-13. The tables list k_s without the Δk penalty associated with BORAL plates. A Δk of 0.00246 is added in the k_s listed below. CSAS input for the normal conditions analysis for the transfer cask is provided in Figure 6.8-1. Figure 6.8-2 provides CSAS input for the transfer cask analysis under hypothetical accident conditions.

Under normal conditions involving loading, draining and drying, the maximum k_{eff} including bias and uncertainties (k_s) is 0.93921 for the transfer cask. In the accident situation involving fuel failure and moderator intrusion, the maximum k_{eff} including biases and uncertainties (k_s) is 0.94749. Thus, the multiplication factor for the transfer cask containing 24 design basis PWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, and accident conditions.

Vertical Concrete Cask

Results of the calculations for the Vertical Concrete Cask containing PWR fuel are provided in Tables 6.4-14 through 6.4-16. Figure 6.8-3 provides CSAS input for the analysis of the concrete cask under normal conditions. Figure 6.8-4 provides CSAS input for the concrete cask analysis for hypothetical accident conditions.

Under normal dry conditions, maximum k_{eff} including biases and uncertainty (k_s) is 0.38329 for the concrete cask. Under off-normal conditions involving flooding of the heat transfer annulus, the k_s of the cask is even less (0.37420). Under accident conditions involving full moderator intrusion into the canister and fuel clad gap, the maximum k_s of the concrete cask is 0.94704. Thus, the multiplication factor for the concrete cask containing 24 design basis PWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, off-normal, and accident conditions.

6.4.3.3 Criticality Results for BWR

Transfer Cask

Results of the criticality calculations for the transfer cask containing BWR fuel are provided in Tables 6.4-17 through 6.4-19. CSAS input for the normal conditions analysis are provided in Figure 6.8-5. Figure 6.8-6 provides CSAS input for the analysis for hypothetical accident conditions.

As the tables show, under normal conditions involving loading, draining and drying, the maximum k_{eff} including bias and uncertainties is 0.91919 for the transfer cask. In the accident condition involving fuel failure and moderator intrusion, the maximum k_{eff} including biases and uncertainties is 0.92235. Thus, the multiplication factor for the transfer cask containing 56 design basis BWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, and accident conditions.

Vertical Concrete Cask

Tables 6.4-20 through 6.4-22 provide results of the criticality calculations for the Vertical Concrete Cask containing BWR fuel assemblies. CSAS input for the normal condition analysis for the concrete cask are provided in Figure 6.8-7. Figure 6.8-8 provides CSAS input under hypothetical accident conditions.

For the concrete cask containing BWR fuel, under normal dry conditions, maximum k_{eff} including biases and uncertainty is calculated to be 0.38168. Under off-normal conditions involving flooding of the heat transfer annulus, the k_{eff} of the cask is 0.38586. Under accident conditions involving full moderator intrusion into the canister and fuel clad gap, the maximum k_{eff} of the concrete cask is 0.92332. Thus, the multiplication factor for the concrete cask containing 56 design basis BWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, off-normal, and accident conditions.

6.4.4 Fuel Assembly Lattice Dimension Variations

The nominal lattice dimensions for the most reactive PWR and BWR fuel under the most reactive accident conditions are varied to determine if dimensional perturbations significantly affect the reactivity of the system. Accident conditions are defined to be full interior, exterior and fuel-clad gap moderator (water) intrusion at a density of 1 g/cc and a temperature of 70 °F. Flooding the fuel-clad gap magnifies the effect on reactivity from lattice dimensional variations by adding or removing moderator from the undermoderated fuel lattice. The conclusions drawn are then used to establish fuel dimension limits for the PWR and BWR fuel assemblies previously evaluated as UMS® contents nominal fuel assembly dimensions.

The PWR analysis is performed modeling a Westinghouse 17x17 OFA fuel assembly in an infinite array of infinitely tall fuel tube cells. This prevents any leakage of neutrons from the system. The BWR analysis is performed modeling an infinite array of infinitely tall Vertical Concrete Casks filled with Exxon\ANF 9x9 fuel assemblies. The following fuel assembly nominal lattice dimensions are modified to determine if these perturbations significantly affect the reactivity of the system:

a) Pellet Radius
b) Clad Inner Radius
c) Clad Outer Radius
d) Water Rod Inner Radius
e) Water Rod Outer Radius

As shown in Table 6.4-22 and 6.4-23 the following dimensional perturbations were determined to significantly decrease the reactivity of both the PWR and the BWR systems: decreasing the clad inner radius and increasing the clad outer radius. Decreasing the pellet radius of the BWR fuel assembly was also determined to significantly decrease the reactivity. The results are as expected as these perturbations decrease the H/U ratio in the undermoderated fuel lattice. Additionally, varying the BWR water rod dimensions was determined to have an insignificant effect on the reactivity of the system. Therefore, these nominal dimension variations are of no concern with regards to the criticality safety of the system.

6.5 Critical Benchmark Experiments

This section provides the validation of the CSAS25 criticality analysis sequence contained in Version 4.3 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.

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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.

6.6.1.4 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 start-up and boronometer sources, Control Element Assembly (CEA) fingertips, and a 24-inch ICI segment. Start-up 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 Start-up 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 start-up 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 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 6.6.1-12, 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 6.6.1-13, 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 6.6.1-3 and 6.6.1-4 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 6.6.1-14, 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 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 start-up sources, a boronometer source, CEA fingertips, or an ICI 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) Start-up 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 start-up 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 start-up 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 four intact fuel assemblies with inserted start-up 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

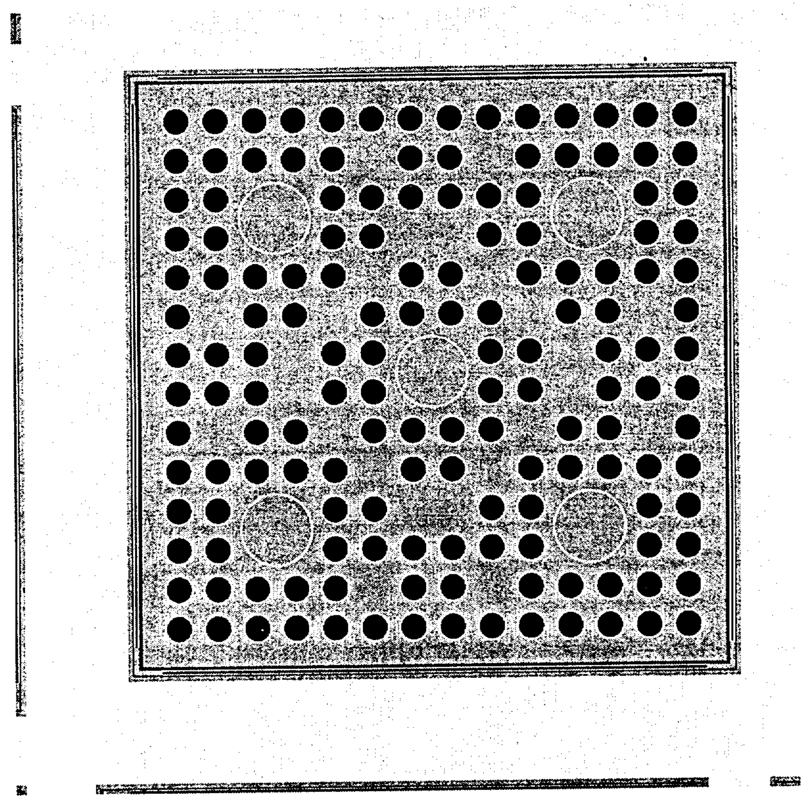


Figure 6.6.1-2 Consolidated Fuel Geometry, 113 Empty Fuel Rod Positions, Maine Yankee Site Specific Fuel

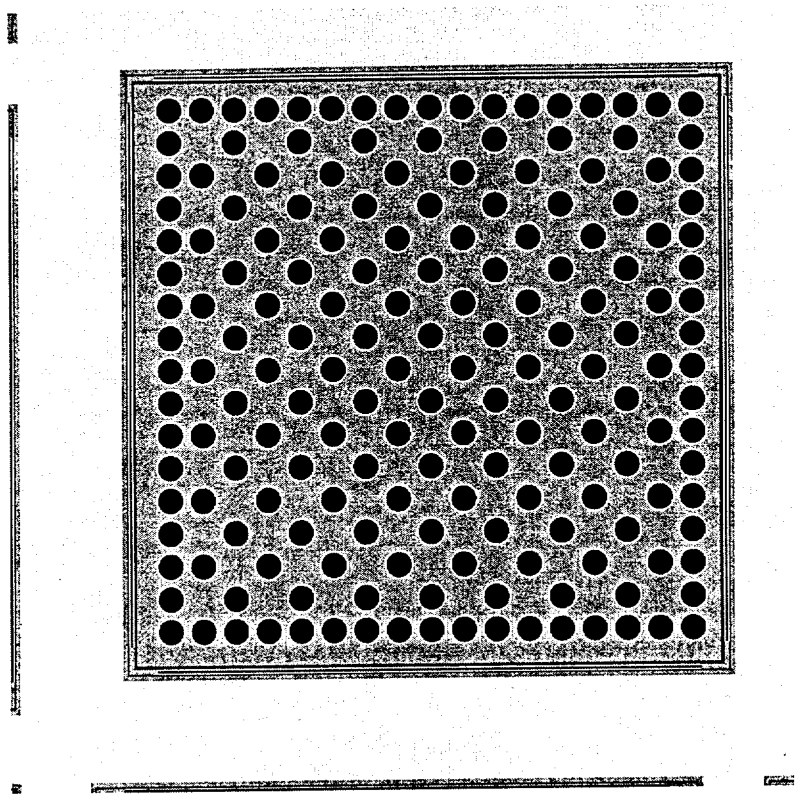


Figure 6.6.1-3 Top View of Boronometer Source Heterogeneous Model Geometry

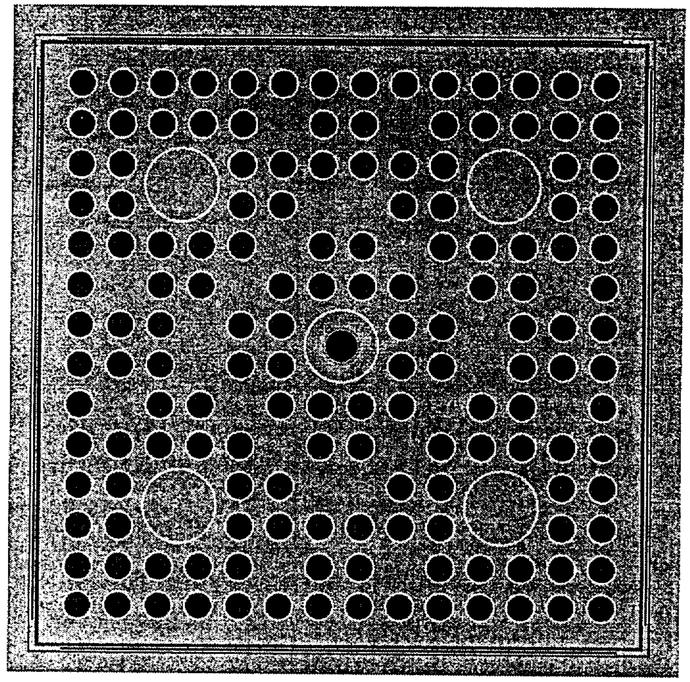


Figure 6.6.1-4 Side View of Boronometer Source Heterogeneous Model Geometry

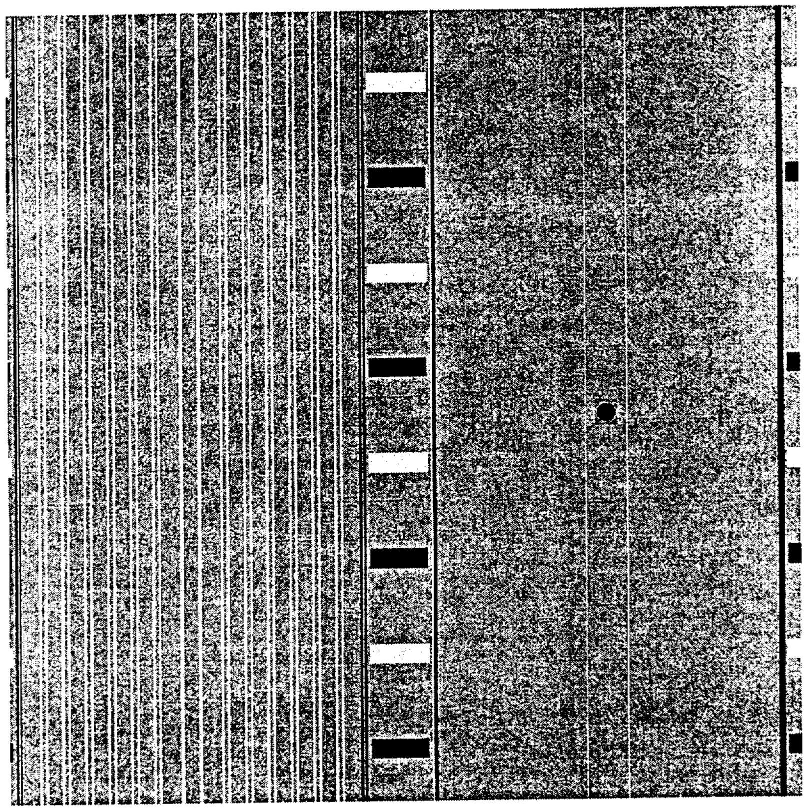


Table 6.6.1-1 Maine Yankee Standard Fuel Characteristics

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 Infinite Height Analysis of Maine Yankee Start-up 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 6.6.1-13 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 6.6.1-14 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.

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 26.0×10^6 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 13.0×10^6 psi. The value of EX (10.47×10^6 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 219.0 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 0.240 ($\beta = 8.44$).

<u>120 Micron Oxide Layer Thickness Evaluation</u>

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.

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:

$$EI_{clad} = 10.47E6\frac{\pi(0.217^4 - 0.194^4)}{4} = 6{,}586 \text{ lb-in}^2$$

$$EI_{fuel} = 13.0E6\frac{\pi(0.194^4)}{4} = 14{,}462 \text{ lb-in}^2$$

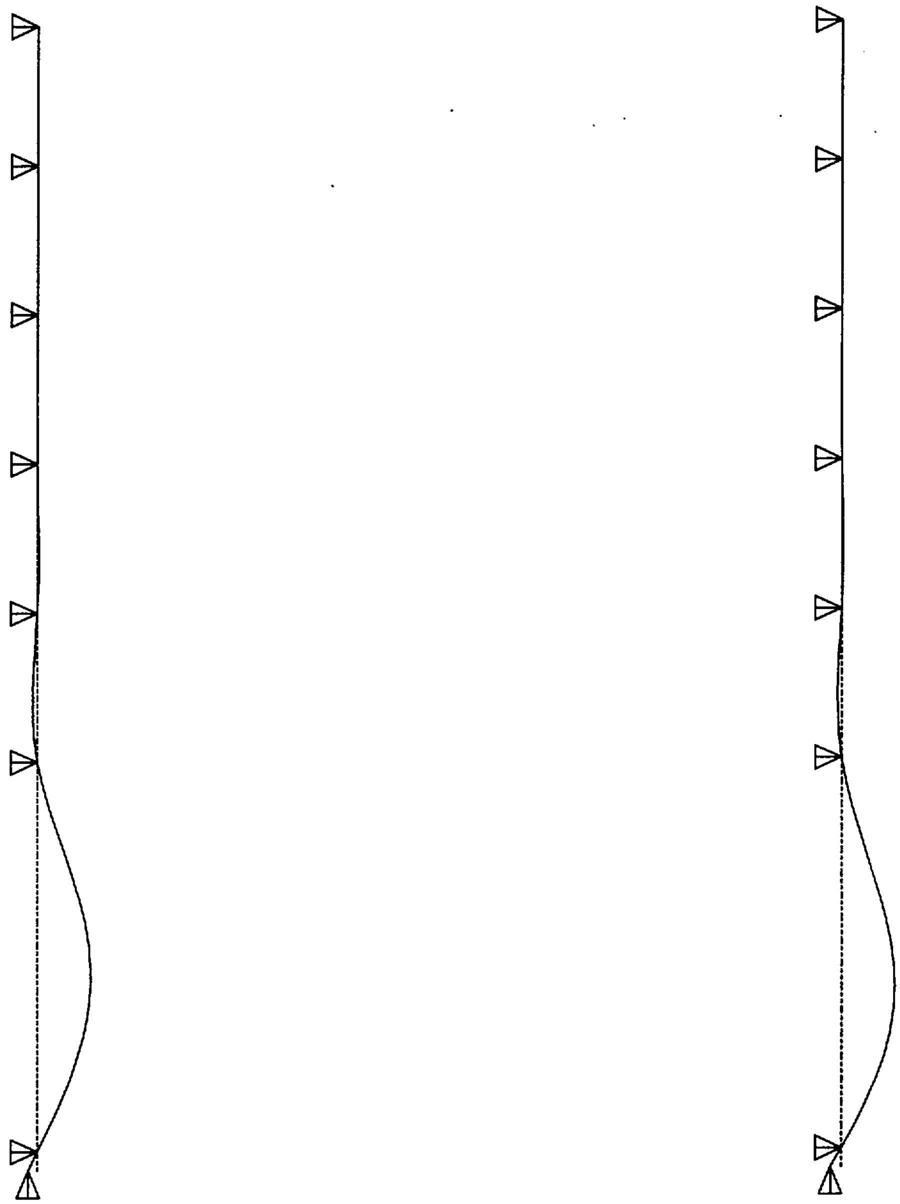
$$EI = 6{,}586 + 14{,}462 = 21{,}048 \text{ lb-in}^2$$

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:

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 [deleted]

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.

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 or HIGH BURNUP FUEL, is administratively controlled in accordance with Section B 2.1.

Figure 12B2-1 PWR Basket Fuel Loading Positions

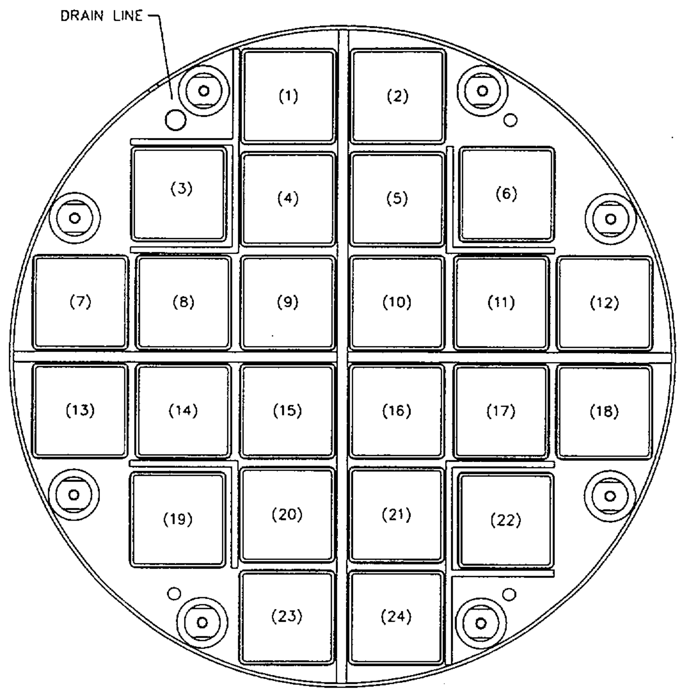


Figure 12B2-2 BWR Basket Fuel Loading Positions

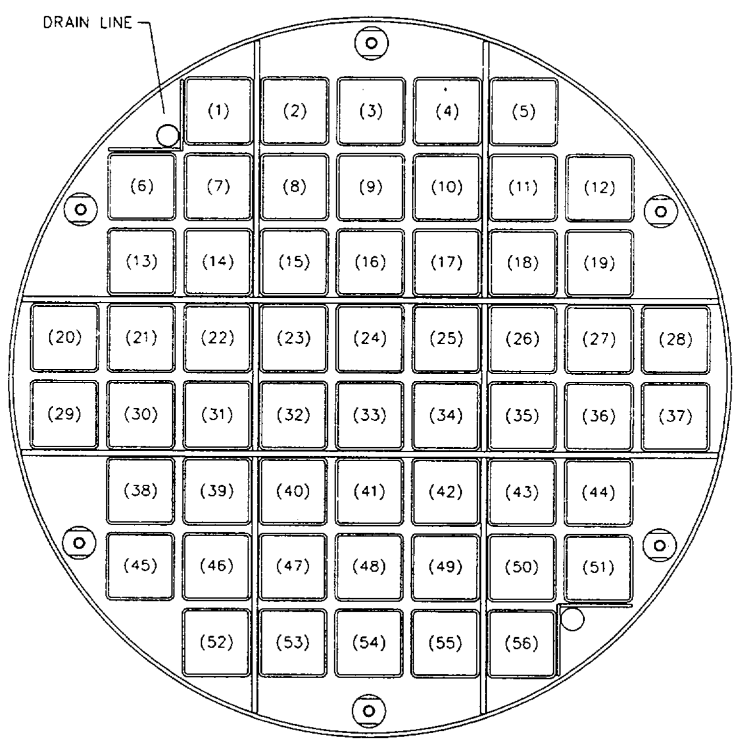


Table 12B2-5 Minimum Cooling Time Versus Burnup/Initial Enrichment for BWR Fuel

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

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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]
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. INTACT FUEL ASSEMBLIES with a start-up 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.
5. PWR INTACT FUEL ASSEMBLIES with CEA ends (fingertips) 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.
6. INTACT FUEL ASSEMBLIES may be loaded in a MAINE YANKEE FUEL CAN.
7. 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.