Chapter 5

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5.0 SHIELDING EVALUATION

Specific dose rate limits for individual casks in a storage array are not established by 10 CFR 72 [1]. Annual dose limit criteria for the independent spent fuel storage installation (ISFSI) controlled area boundary are established by 10 CFR 72.104 and 10 CFR 72.106 for normal conditions and for design basis accidents. These regulations require that, for an array of casks in an ISFSI, the annual dose to an individual outside the controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ during normal operations. For a design basis accident, the dose to an individual outside the area boundary must not exceed 5 rem to the whole body. The ISFSI must be at least 100 meters from the owner controlled area boundary. In addition, the occupational dose limits and radiation dose limits established in 10 CFR Part 20 (Subparts C and D) [2] for individual members of the public must be met.

This chapter describes the shielding design and the analysis used to establish bounding radiological dose rates for the storage of various types of PWR and BWR fuel assemblies. The analysis shows that the Universal Storage System meets the requirements of 10 CFR 72.104 and 10 CFR 72.106 when the system is configured and used in accordance with the design basis established by this Safety Analysis Report.

The Universal Storage System compliance with the requirements of 10 CFR 72 with regard to annual and occupational doses at the owner controlled area boundary is demonstrated in Section 10.3 and 10.4.

5.1 Discussion and Results

The Universal Storage System is comprised of a transportable storage canister, a transfer cask, and a vertical concrete cask. A multi-walled shielding arrangement is employed in both the radial and axial shields of the concrete cask and the transfer cask. The transfer cask has a radial shield comprised of 0.75 inch of low alloy steel, 3.75 inches of lead, 3 inches of solid borated polymer (NS-4-FR), and 1.25 inches of low alloy steel. An additional 0.625 inch of stainless steel shielding is provided, radially, by the canister shell. Gamma shielding is provided primarily by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. The transfer cask bottom shield design is a solid section of 7.5 inches of low alloy steel and 1.5 inches of NS-4-FR. The top shielding of the transfer cask is provided by the stainless steel canister shield and structural lids, which are 7 inches and 3 inches thick, respectively. In addition, 5 inches of steel is used as temporary shielding during welding, draining, drying, helium backfill, and other operations related to closing the canister. This temporary shielding is removed prior to storage.

The vertical concrete cask radial shield design is comprised of a 2.5-inch thick carbon steel inner liner surrounded by 28.25 inches of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. As in the transfer cask, an additional 0.625 inch thickness of stainless steel radial gamma shielding is provided by the canister shell. The concrete cask top shielding design is comprised of 10 inches of stainless steel from the canister lids, a shield plug containing a 1 inch thickness of NS-4-FR and 4.1 inches of carbon steel, and a 1.5 inch thick carbon steel lid. Since the bottom of the concrete cask rests on a concrete pad, the cask bottom shielding is comprised of 1.75 inch of stainless steel from the canister bottom plate, 2 inches of carbon steel (pedestal plate) and 1 inch of carbon steel cask base plate. The base plate and pedestal base are structural components that position the canister above the air inlets. The cask base supports the concrete cask during lifting, and forms the cooling air inlet channels at the cask bottom. An optional carbon steel supplemental shielding fixture, shown in Drawing 790-613, may be installed to reduce the radiation dose rates at the air inlets.

The spent fuel that may be stored in the Universal Storage System is divided into 5 classes, three PWR and two BWR, depending on the length of the fuel assembly. The transportable storage canister, transfer cask, and vertical concrete cask are provided in 5 lengths, corresponding to the lengths of the fuel assemblies.

The designs for PWR and BWR fuel are similar, but differ slightly in the design of the basket structure. The shielding analysis is based on the use of bounding dose rates for the design basis PWR and BWR fuel assembly, and its associated canister, transfer cask, and concrete cask.

The design basis PWR fuel for the shielding evaluation is the Westinghouse 17×17 standard assembly with an average burnup of 40,000 MWD/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time. The shielding design basis BWR fuel is a GE9×9 assembly with a burnup of 40,000 MWD/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time. The source term specification is provided in Section 5.2. The shielding evaluation for fuel having a higher burnup is provided in Section 2.5.

Table 5.3-5 Transfer Cask Material Densities

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Carbon and Low-Alloy Steel	11	CARBONSTEEL	7.8212	CARBON-12 IRON	3.9250E-03 8.3498E-02
Stainless Steel	12	SS304	7.9200	CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	1.7429E-02 1.7363E-03 5.9358E-02 7.7207E-03
Lead	13	PB	11.3440	LEAD	3.2969E-02
NS-4-FR	14	H B-10 B-11 C N O AL	1.63	HYDROGEN BORON-10 BORON-11 CARBON-12 NITROGEN-14 OXYGEN-16 ALUMINUM	5.8540E-02 8.5530E-05 3.4220E-04 2.2640E-02 1.3940E-03 2.6090E-02 7.7630E-03
Aluminum	17	AL	2.7020	ALUMINUM	6.0307E-02
Concrete	18	REG-CONCRETE	2.2426	HYDROGEN OXYGEN-16 SODIUM-23 ALUMINUM SILICON CALCIUM IRON	1.3401E-02 4.4931E-02 1.7036E-03 1.7018E-03 1.6205E-02 1.4826E-03 3.3857E-04
Canister Void (Dry Conditions)	19	N	VF=1.0E-6	NITROGEN-14	4.3006E-08
Canister Water (Wet Conditions)	19	H2O	0.9982	HYDROGEN OXYGEN-16	6.6769E-02 3.3385E-02

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5.6.1 Shielding Evaluation for Maine Yankee Site Specific Spent Fuel

This analysis considers both assembly fuel 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

5.6.1.4.1.1 Establishment of Limiting Values

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

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

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

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

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

5.6.1.4.1.2 Three-Dimensional Model Results

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

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

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

5.6.1.4.1.3 Decay Heat Limits

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

5.6.1.4.1.4 Loading Table Analysis

With the adjusted one-dimensional dose and heat generation rate limits established above, the loading table analysis proceeds following the methodology developed in Section 5.5. Each combination of initial enrichment and burnup is analyzed to determine the minimum required cool time in order for an assembly to either 1) contain a design basis CEA cooled 5, 10, 15, or 20 loading pattern, permitting 1.05 kW per peripheral assembly, reduces the minimum cool time based on thermal constraints to 6 years. The storage cask dose rate constraint is satisfied for the preferentially loaded assembles after 5 years cooling. Recognizing that only two of the assemblies in the Maine Yankee spent fuel inventory, R439 and R444, require peripheral loading, the transfer cask dose rate limit is not applied for these two assemblies. Since the dose rate comparisons are made on the basis of an assumed fuel cask of assemblies, the transfer cask dose rate limit is unnecessarily restrictive.

5.6.1.4.4 Consolidated Fuel

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

A limiting cool time analysis is conducted by identifying a fuel assembly description analyzed in the loading table analysis that bounds the parameters of the fuel rods in the consolidated fuel lattices. The parameters of those fuel rods are shown in Table 5.6.1-15. The CE 14 x 14 fuel at 30,000 MWD/MTU and 1.9 wt % enrichment represents a bounding assembly type since it has a significantly higher burnup and a lower enrichment than the original assemblies. This fuel requires 6-year 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-year 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 and fuel debris. Damaged fuel rods and fuel debris will be placed into a screened Maine Yankee fuel can prior to loading in the UMS® basket. 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.

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 and ICI string segment 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.

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 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 fingertips and the ICI string segment will be loaded into one assembly. Therefore, the gamma spectra of these items are summed and only one gamma spectrum is 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 are presented in Table 5.6.1-21.

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

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

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

Table 5.6.1-1 Maine Yankee CEA Exposure History by Group

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

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

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

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

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

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

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

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

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

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

1. MWD/MTU.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Table 5.6.1-15 Maine Yankee Consolidated Fuel Model Parameters

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

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

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

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

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

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

Non Fuel Material	Pu-238 grams	Pu-238 Curies	Pu-239 grams	Pu-239 Curies
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]	Total Gamma [g/sec]
1	0.0000E+00	0.0000E+00	0.00E+00
2	0.0000E+00	0.0000E+00	0.00E+00
3	0.0000E+00	0.0000E+00	0.00E+00
4	0.0000E+00	0.0000E+00	0.00E+00
5	0.0000E+00	0.0000E+00	0.00E+00
6	5.6364E+04	1.4995E+04	7.14E+04
7	3.6350E+07	9.6704E+06	4.60E+07
8	0.0000E+00	0.0000E+00	0.00E+00
9	1.5317E+12	4.0749E+11	1.94E+12
10	5.4239E+12	1.4430E+12	6.87E+12
11	2.4164E+08	6.4285E+07	3.06E+08
12	6.4084E+06	1.7049E+06	8.11E+06
13	1.8453E+07	4.9092E+06	2.34E+07
14	2.9197E+08	7.7675E+07	3.70E+08
15	2.2253E+08	5.9201E+07	2.82E+08
16	4.4816E+09	1.1923E+09	5.67E+09
17	1.8576E+10	4.9418E+09	2.35E+10
18	9.3171E+10	2.4787E+10	1.18E+11
Total	7.0726E+12	1.8816E+12	8.95E+12

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

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

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

Group	Storage - Surface Gamma Dose [mrem/hr]	Transfer - Surface Gamma Dose [mrem/hr]
1	3.66E-10	1.51E-10
2	1.41E-09	8.97E-10
3	4.92E-09	5.00E-09
4	7.10E-09	1.20E-08
5	1.08E-08	2.99E-08
6	4.21E-08	1.91E-07
7	9.96E-06	6.12E-05
8	2.24E-09	1.72E-08
9	4.59E-02	3.77E-01
10	3.49E-02	2.24E-01
11	2.31E-07	6.42E-07
12	1.82E-09	1.02E-09
13	2.68E-10	9.13E-13
14	9.84E-11	3.12E-19
15	2.65E-12	1.49E-40
16	1.11E-14	0.00E+00
17	1.91E-41	0.00E+00
18	0.00E+00	0.00E+00
Total	8.09E-02	6.01E-01

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

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

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

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

Chapter 6

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6.2 Spent Fuel Loading

The Universal Storage System is designed to store Transportable Storage Canisters containing spent nuclear fuel. Canisters of five different lengths are designed, each to accommodate one of three classes of PWR fuel assemblies or one of two classes of BWR fuel assemblies. The classification of the fuel assemblies is based primarily on fuel assembly length and cross section. The classes of major fuel assemblies to be stored in the Universal Storage System and their characteristics are shown in Tables 6.2-1 (PWR) and 6.2-2 (BWR).

Class 1 Westinghouse fuel assemblies and Class 2 B&W fuel assemblies include inserts. Fuel assembly inserts are non-fuel bearing components, such as thimble plugs and burnable poison rods, inserted in the fuel assembly guide tubes. The criticality analysis conservatively takes no credit for displacement of moderator by the inserts. The presence of a poison insert reduces reactivity by further decreasing the unborated water moderator to fuel ratio in the fuel assembly lattice.

To preclude a potential increase in reactivity as a result of empty fuel rod positions in the assembly, any empty fuel rod positions are to be filled with solid filler rods fabricated from either solid Zircaloy or solid Type 304 stainless steel.

Table 6.2-1 PWR Fuel Assembly Characteristics (Zirc-4 Clad)

Fuel Class	Vendor	Array	Version	Max MTU	No of Fuel Rods	Pitch (in)	Rod Dia. (in)	Clad Thick (in)	Pellet Dia (in)	Active Length (in)
1	CE	14 × 14	Std.	0.4037	176	0.5800	0.440	0.0280	0.3765	137.0
1	CE	14 × 14	Ft Cal.	0.3772	176	0.5800	0.440	0.0280	0.3765	128.0
1	CE	15 × 15	Palis.	0.4317	216	0.5500	0.418	0.0260	0.3580	132.0
1	CE	16 × 16	Lucie 2	0.4025	236	0.5060	0.382	0.0230	0.3255	136.7
1	Ex/ANF	14 × 14	WE	0.3689	179	0.5560	0.424	0.0300	0.3505	142.0
1	Ex/ANF	14 × 14	CE	0.3814	176	0.5800	0.440	0.0310	0.3700	134.0
1	Ex/ANF	14 × 14	Praire Isl.	0.3741	179	0.5560	0.417	0.0300	0.3505	144.0
1	Ex/ANF	15 × 15	WE	0.4410	204	0.5630	0.424	0.0300	0.3565	144.0
1	Ex/ANF	15 × 15	Palis	0.4310	216	0.5500	0.417	0.0300	0.3580	131.8
1	Ex/ANF	17 × 17	WE	0.4123	264	0.4960	0.360	0.0250	0.3030	144.0
1	WE	14 × 14	Std/ZCA	0.4144	179	0.5560	0.422	0.0225	0.3674	145.2
1	WE	14 × 14	OFA	0.3612	179	0.5560	0.400	0.0243	0.3444	144.0
1	WE	14 × 14	Std/ZCB	0.4144	179	0.5560	0.422	0.0225	0.3674	145.2
1	WE	14 × 14	CE Model	0.4115	176	0.5800	0.440	0.0260	0.3805	136.7
1	WE	15 × 15	Std	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0
1	WE	15 × 15	Std/ZC	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0
1	WE	15 × 15	OFA	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0
1	WE	17 × 17	Std	0.4671	264	0.4960	0.374	0.0225	0.3225	144.0
1	WE	17 × 17	OFA	0.4282	264	0.4960	0.360	0.0225	0.3088	144.0
1	WE	17 × 17	Vant 5	0.4282	264	0.4960	0.360	0.0225	0.3088	144.0
2	B&W	15 × 15	Mark B	0.4807	208	0.5680	0.430	0.0265	0.3686	144.0
2	B&W	15 × 15	Mark BZ	0.4807	208	0.5680	0.430	0.0265	0.3686	144.0
2	B&W	17 × 17	Mark C	0.4658	264	0.5020	0.379	0.0240	0.3232	143.0
3	CE	16 × 16	Sono 2&3	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0
3	CE	16 × 16	ANO2	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0
3	CE	16 × 16	SYS80	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0

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 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.

The following perturbations were determined to significantly increase the reactivity of both the PWR and BWR systems: increasing the clad inner radius and decreasing the clad outer radius, increasing the guide tube inner radius, decreasing the guide tube outer radius. The increase in reactivity is due to the fact that these perturbations increase the H/U ratio in the undermoderated fuel lattice.

An increase in reactivity was also seen in the PWR system when decreasing the pellet diameter. This slight increase in reactivity, 0.004 Δk, is due to flooding of the pellet-to-clad gap in the accident model, which provides additional moderator to the lattice. Since 100% of clad failure is not expected during normal or accident operating conditions, no lower bound limit is placed on the fuel pellet diameter.

The effect on reactivity from perturbations in the nominal fuel dimensions requires the following limits on the fuel assembly lattice parameters in order to retain the maximum reactivity of the UMS system below existing design basis results:

PWR

a) Fuel Rod Diameter ≥ Nominal Dimension

b) Clad Thickness ≥ Nominal Dimension

c) Fuel Rod Pitch ≤ Nominal Dimension

d) Guide Tube (Instrument Tube) Thickness ≥ Nominal Dimension

e) Pellet Diameter ≤ Nominal Dimension

BWR

a) Fuel Rod Diameter ≥ Nominal Dimension

b) Clad Thickness ≥ Nominal Dimension

c) Fuel Rod Pitch ≤ Nominal Dimension

d) Pellet Diameter ≤ Nominal Dimension

Table 6.4-1 k_{eff} for Most Reactive PWR Fuel Assembly Determination

Assembly Type	Dry Pellet Clad Gap		Wet Pellet Clad Gap		Δk_{eff}[1]
	k_{eff}	σ	k_{eff}	σ	Wet - Dry
B&W 15×15 Mark B	0.9613	0.0011	0.9692	0.0012	0.0079
B&W 17×17 Mark C	0.9621	0.0012	0.9705	0.0011	0.0084
CE 14×14	0.9295	0.0013	0.9381	0.0011	0.0085
CE 16×16 SYS 80	0.9348	0.0012	0.9442	0.0012	0.0095
West 14×14	0.9177	0.0013	0.9264	0.0012	0.0086
West 14×14 OFA	0.9238	0.0012	0.9326	0.0012	0.0088
West 15×15	0.9662	0.0011	0.9712	0.0012	0.0050
West 17×17	0.9596	0.0012	0.9673	0.0012	0.0077
West 17×17 OFA	0.9656	0.0013	0.9727	0.0012	0.0070
Ex/ANF 14×14 CE	0.9309	0.0012	0.9362	0.0011	0.0053
Ex/ANF 14×14 WE	0.9065	0.0012	0.9176	0.0011	0.0111
Ex/ANF 15×15 WE	0.9559	0.0012	0.9634	0.0013	0.0074
Ex/ANF 17×17 WE	0.9631	0.0012	0.9704	0.0012	0.0073

1. Infinite Array of Basket Cells with a 1.0-inch Web.

Table 6.4-2 k_{eff} for Highest Reactivity PWR Fuel Assemblies

Assembly Type	k_{eff}[1]	σ
B&W 15×15 Mark B4	0.9119	0.0011
B&W 17×17 Mark C	0.9141	0.0011
West 15×15	0.9147	0.0013
West 17×17	0.9116	0.0012
West 17×17 OFA	0.9196	0.0012
Ex/ANF 17×17 WE	0.9172	0.0011

1. Infinite Array of Basket Cells with a 1.5-inch Web.

6.5.1 SCALE 4.3 Benchmark Experiments and Applicability

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

6.5.1.1 Description of Experiments

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

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

6.5.1.2 Applicability of Experiments

All of the experiments chosen in this validation are applicable to either PWR or BWR fuel. Fuel enrichments have covered a range from 2.35 up to 4.74 wt % ^{235}U, typical of light water reactor fuel presently used. The experiment fuel rod and pitch characteristics are within the range of standard PWR or BWR fuel rods (i.e., pellet OD from 0.78 to 1.2 cm, rod OD from 0.95 to 1.88 cm, and pitch from 1.26 to 1.87 cm). This is particularly true of the VCML (PWR rod type) and B&W experiments (BWR rod type). The H/U volume ratios of the experimental fuel arrays are within the range of PWR fuel assemblies (1.6 to 2.32) and BWR fuel assemblies (1.6 to 1.9). Experiments covered the geometry and neutron absorber sheet arrangements typical of NAC basket designs. Flux trap gap spacings of 3.81 cm such as those in the NAC-STC and UMS® PWR baskets and gap spacings as low 1.91 cm as in the NAC-MPC were included. ^{10}B neutron absorber loadings, also typical of NAC basket designs (0.005 to 0.025 $^{10}B/cm^2$), were included as well. The experiments addressed the influence of water and metal reflector regions, including steel and lead, that would be present in storage and transport cask shielding.

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

6.5.1.3 Results of Benchmark Calculations

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

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

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

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

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

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

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

6.5.1.4 Trends

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

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

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

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

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

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

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

where:

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

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

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 the computational method.

Δk_m= arbitrary administrative margin to ensure subcriticality of k_s

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

Equation 1 can be rewritten as:

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

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

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

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

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

where:

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

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

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

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

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

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

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

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

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

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

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

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

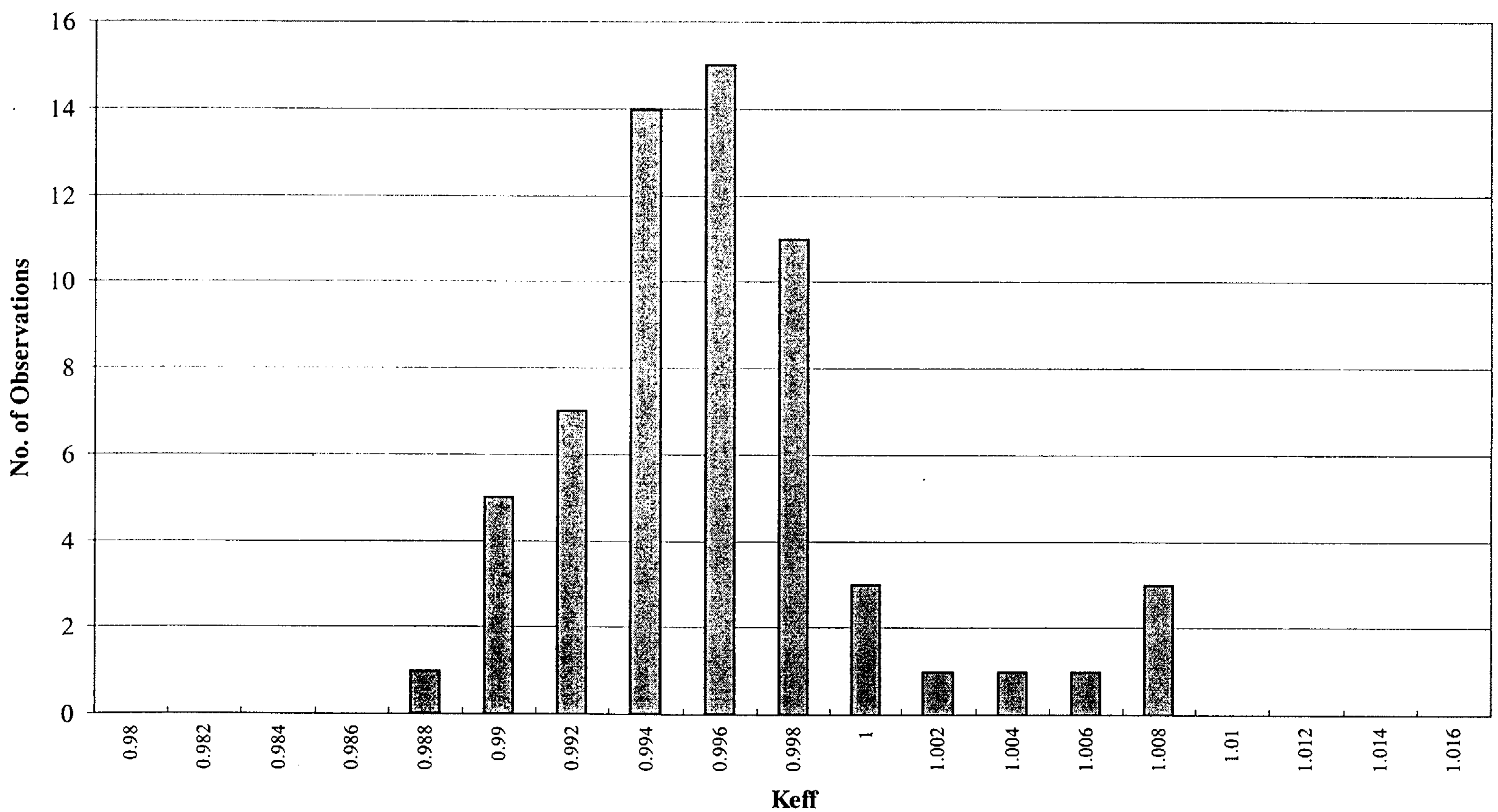

Figure 6.5.1-2 KENO-Va Validation—27-Group Library Results: k_{eff} versus Enrichment

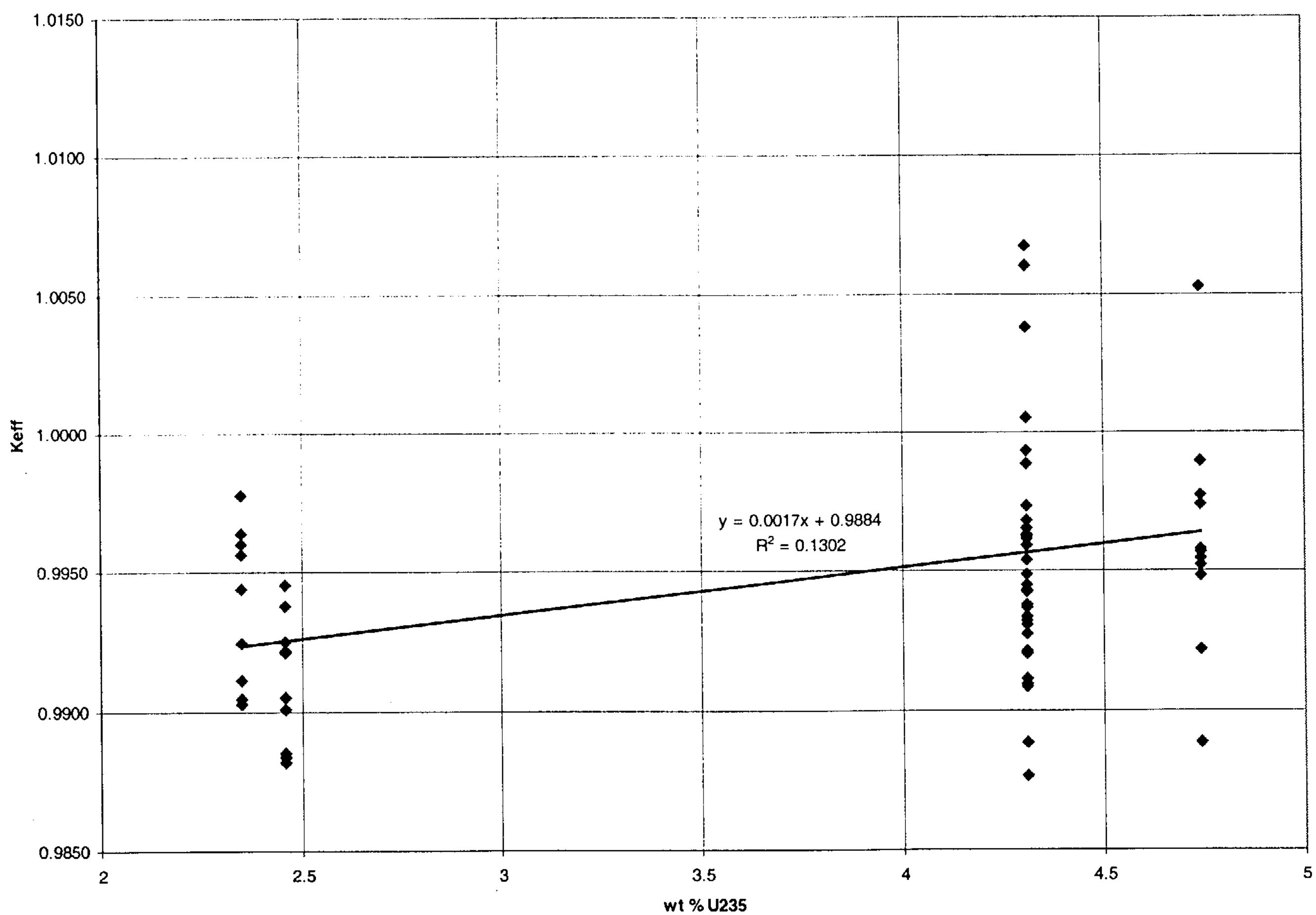

Figure 6.5.1-3 KENO-Va Validation—27-Group Library Results: k_{eff} versus Rod Pitch

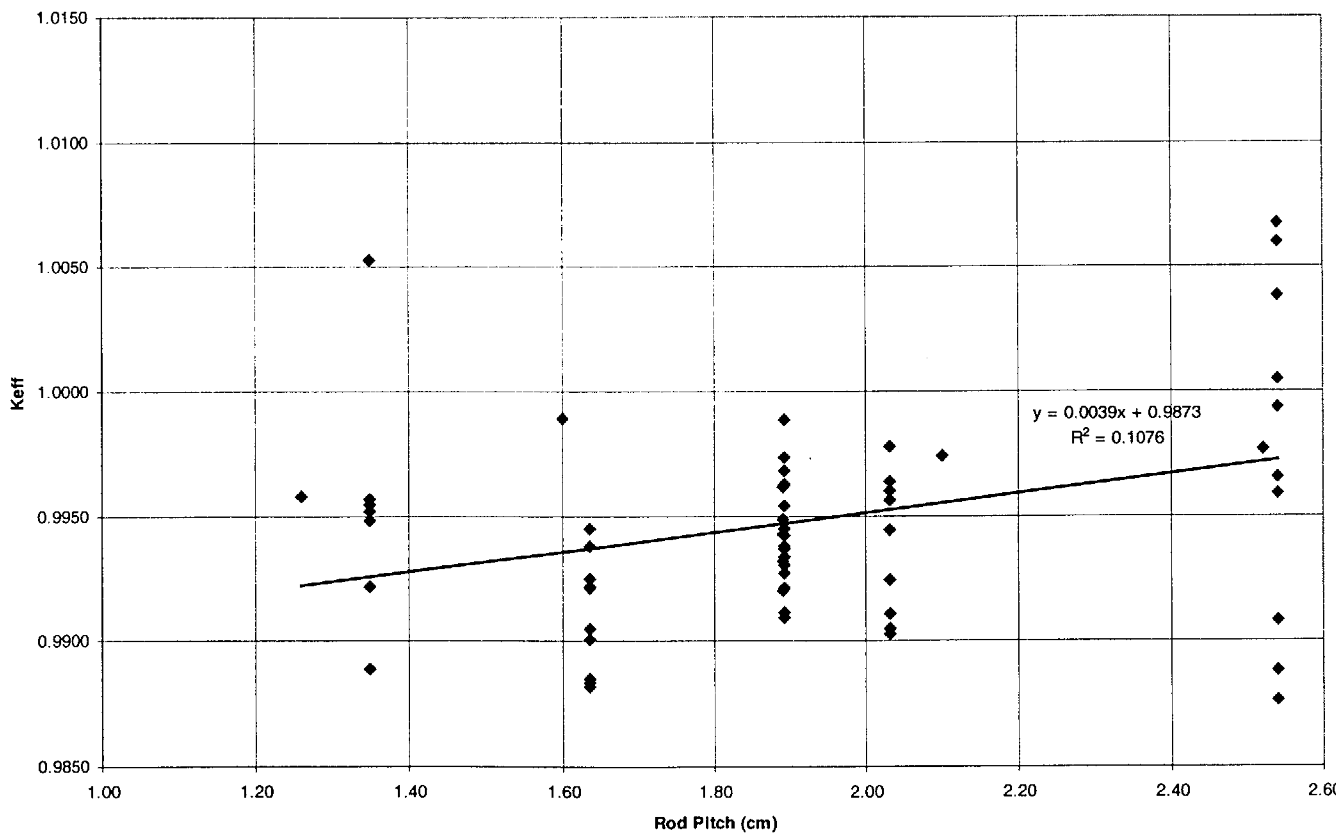

Figure 6.5.1-4 KENO-Va Validation—27-Group Library Results: k_{eff} versus H/U Volume Ratio

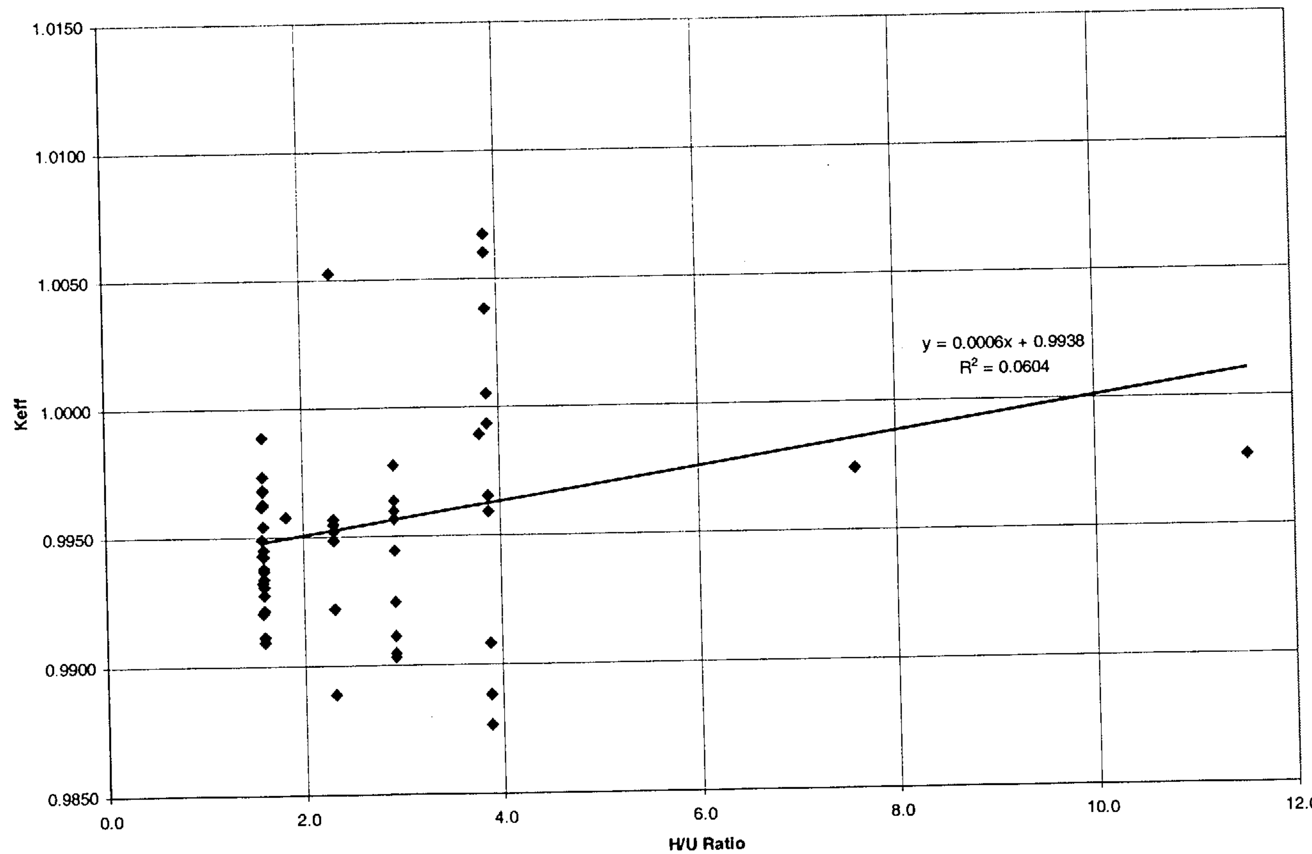

Figure 6.5.1-5 KENO-Va Validation—27-Group Library Results: k_{eff} versus Average Group of Fission

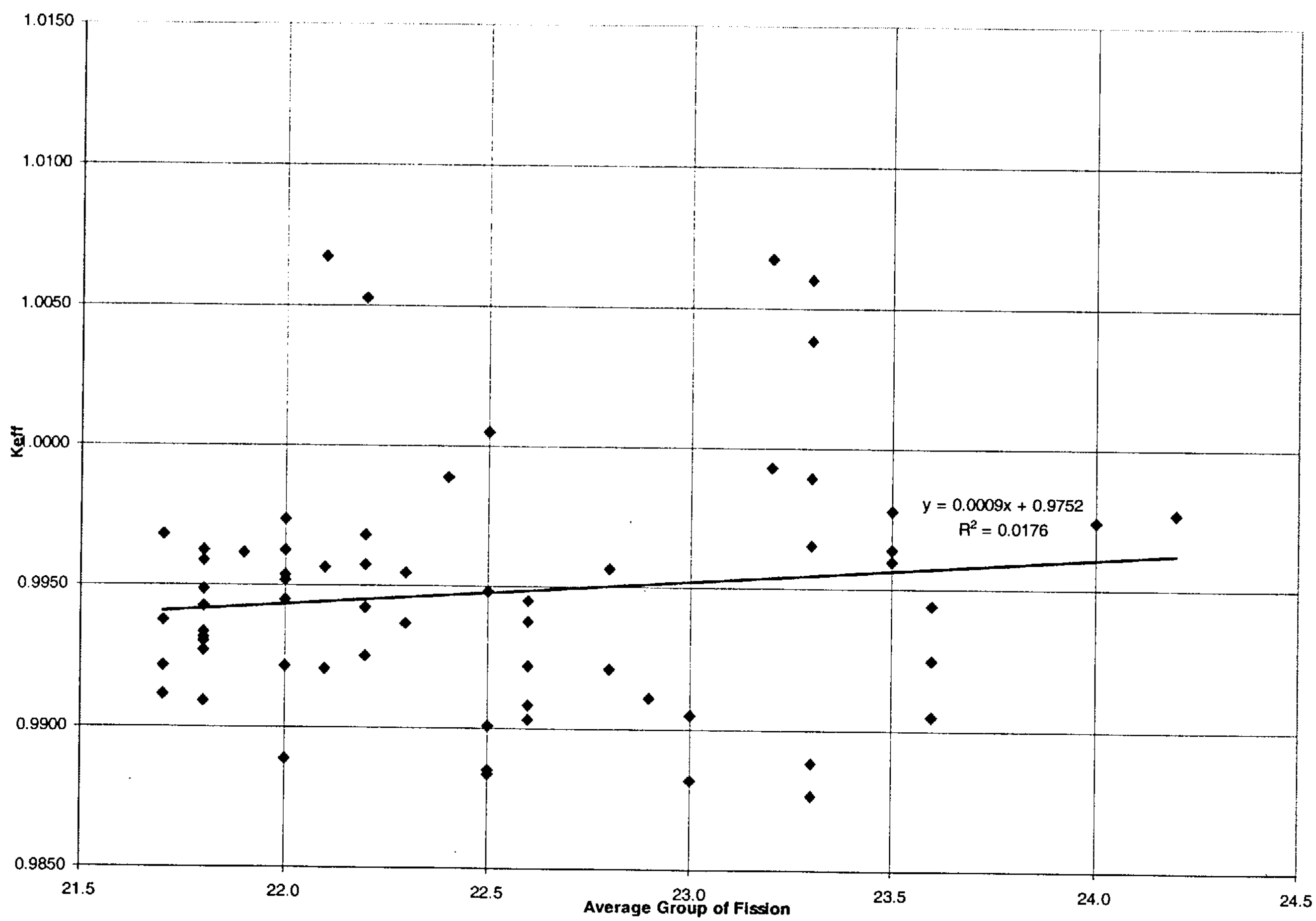

Figure 6.5.1-6 KENO-Va Validation—27-Group Library Results: k_{eff} versus ^{10}B Loading for Flux Trap Criticals

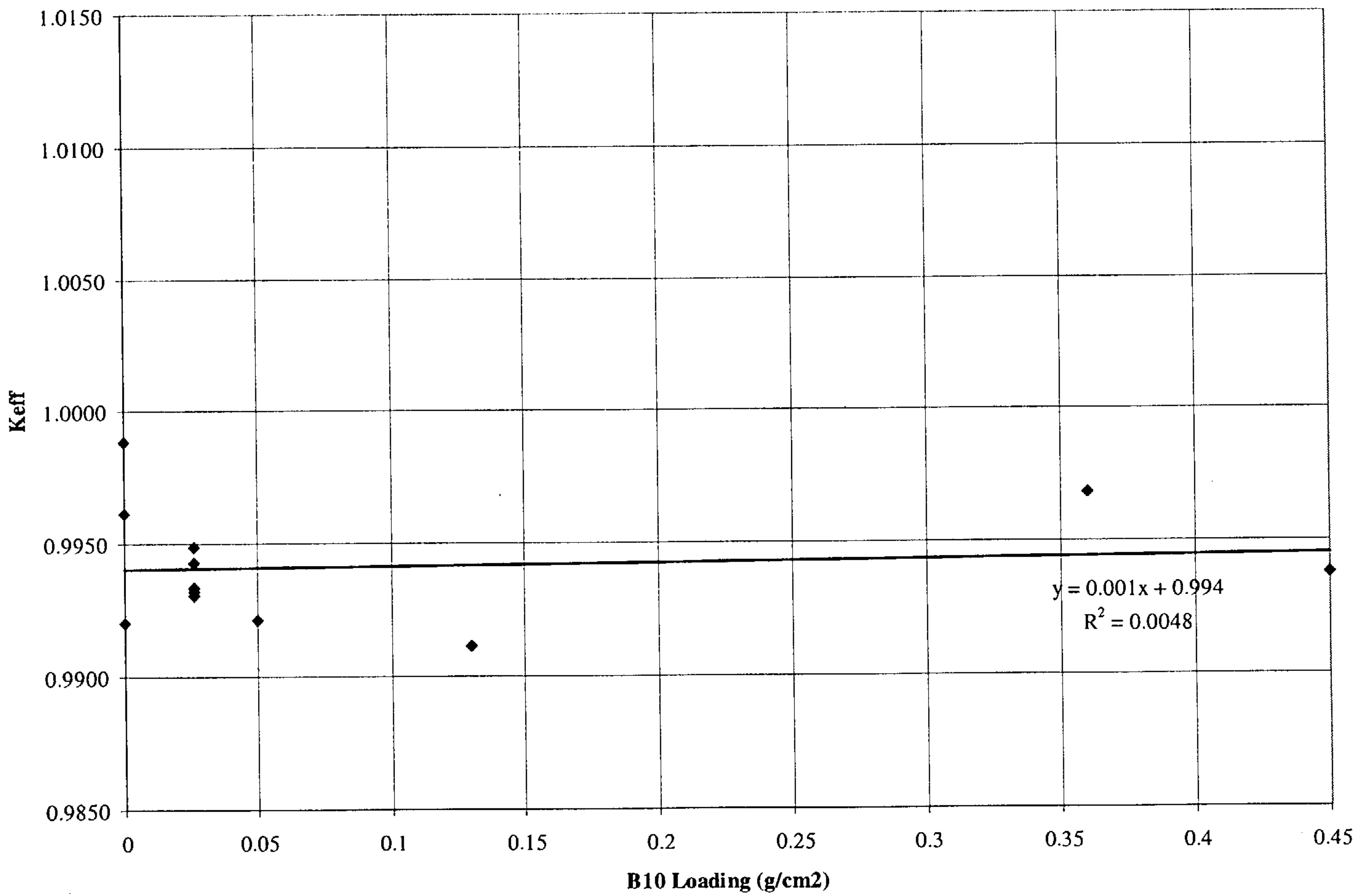

Figure 6.5.1-7 KENO-Va Validation—27-Group Library Results: k_{eff} versus Flux Trap Critical Gap Thickness

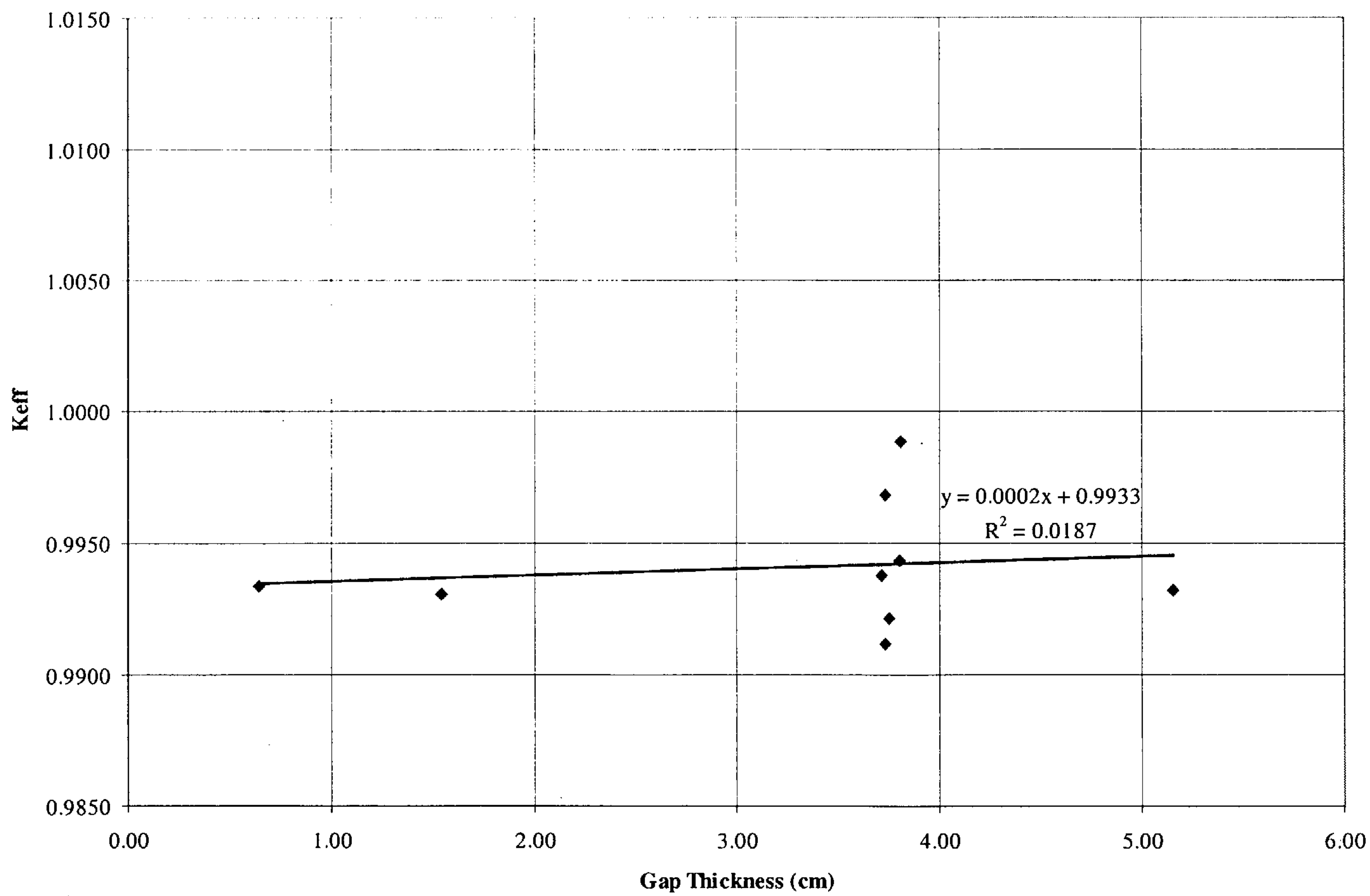

Figure 6.5.1-8 USLSTATS Output for Fuel Enrichment Study

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

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

Input to statistical treatment from file:EN_KEFF.TXT

Title: 63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

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

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

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

Output from statistical treatment

63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

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

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

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

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

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

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


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

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

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

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

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

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

Criticals Set 2	Configuration	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
PNL-043	17×13 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9954	0.0014
PNL-044	16×14 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9945	0.0013
PNL-045	14×16 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9974	0.0013
PNL-046	12×19 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9963	0.0013
PNL-087	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.066	2.83	0.9982	21.8	0.9927	0.0012
PNL-079	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.030	2.83	0.9982	21.8	0.9909	0.0012
PNL-093	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.026	2.83	0.9982	21.8	0.9962	0.0012
PNL-115	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Aluminum	0	2.83	0.9982	22.3	0.9937	0.0013
PNL-064	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.302)	0	2.83	0.9982	22.2	0.9942	0.0012
PNL-071	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.485)	0	2.83	0.9982	22.2	0.9968	0.0012
												Average	0.9948	0.0020

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

Criticals Set 3	Configuration	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap Cluster (cm)	Gap Wall/ Cluster (cm)	Ave. Group Fission	k_{eff}	σ
PNL-STA	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.65	0.00	23.5	0.9964	0.0010
PNL-STB	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.20	1.32	23.6	0.9944	0.0010
PNL-STC	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.36	2.62	23.6	0.9905	0.0010
PNL-PBA	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.84	0.00	23.5	0.9960	0.0011
PNL-PBB	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.72	0.66	23.5	0.9978	0.0010
PNL_PBC	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.25	2.62	23.6	0.9925	0.0010
PNL-DUA	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.83	0.00	22.6	0.9903	0.0009
PNL-DUB	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	14.11	1.96	22.8	0.9957	0.0010
PNL-DUC	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.70	2.62	22.9	0.9911	0.0010

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

Criticals	Configura-tion	wt % ^{235}U	itch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap (cm)	Ave. Group Fission	k_{eff}	σ
Set 3 (Contd.)										Cluster	Wall/ Cluster			
PNL-H20	3×1 H2O Refl	4.31	2.54	1.265	1.415	3.9	0	na	na	8.24	inf	23.3	0.9877	0.0023
PNL-ST0	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.89	0	23.2	0.9993	0.0012
PNL-ST1	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	14.12	1.32	23.3	1.0060	0.0022
PNL-ST26	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.44	2.62	23.3	0.9965	0.0011
PNL-PB0	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	20.62	0	23.2	1.0068	0.0021
PNL-PB13	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	23.3	1.0038	0.0012
PNL-PB5	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	10.3	5.41	23.3	0.9889	0.0011
PNL-DU0	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	15.38	0	21.8	0.9959	0.0011
PNL-DU13	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	22.1	1.0067	0.0010
PNL-DU39	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	18.05	3.91	22.5	1.0005	0.0011
PNL-DU54	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	13.49	5.41	22.6	0.9908	0.0011
												Average	0.9964	0.0060

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

Criticals	Configuration	wt % U^{235}	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 4														
PNL-229	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Aluminum	0	3.81	0.9982	22.4	0.9989	0.0012
PNL-230	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.05	3.75	0.9982	21.7	0.9921	0.0012
PNL-228	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.13	3.73	0.9982	21.7	0.9911	0.0012
PNL-214	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.36	3.73	0.9982	21.7	0.9968	0.0013
PNL-231	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.45	3.71	0.9982	21.7	0.9938	0.0012
PNL-127	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	0.64	0.9982	21.8	0.9934	0.0010
PNL-126	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	1.54	0.9982	21.8	0.9931	0.0010
PNL-123	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	3.80	0.9982	21.8	0.9943	0.0010
PNL-125	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	5.16	0.9982	21.8	0.9932	0.0010
PNL-124	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	INF	0.9982	21.8	0.9949	0.0010
PNL-123-S	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	3.80	0.9982	22.1	0.9920	0.0010
PNL-124-S	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	INF	0.9982	21.9	0.9962	0.0010
												Average	0.9941	0.0022

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

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

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

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

Table 6.5.1-3 Most Reactive Configuration System Parameters

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

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6.6 Criticality Evaluation for Site Specific Spent Fuel

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

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

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6.6.1 Criticality Evaluation for Maine Yankee Site Specific Spent Fuel

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

6.6.1.1 Maine Yankee Fuel Criticality Model

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

6.6.1.2 Maine Yankee Intact Spent Fuel

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

6.6.1.2.1 Fuel Assembly Lattice Dimensional Variations

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

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

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

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

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

6.6.1.2.2 Variably Enriched Fuel Assemblies

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

6.6.1.2.3 Assemblies with Annular Axial End Blankets

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

This geometry is discretely modeled as approximately 5% annular fuel, 90% solid fuel and then 5% annular fuel, with all fuel materials enriched to 4.2 wt % ^{235}U. The diameter of all pellets is initially modeled as the most reactive pellet diameter. The accident case model, which includes flooding of the fuel cladding annulus, is used in this evaluation. Axial periodic boundary conditions are placed on the model, retaining the conservatism of the infinite fuel length. Use of a smaller pellet diameter is not considered to be conservative when evaluating the annular fuel pellets. The smaller pellet diameter is the most reactive diameter under the assumption that it is solid and not an annulus. Flooding the axial end blanket annulus provides additional moderator to the fuel lattice. Therefore, the diameter of the annular pellets is also modeled as the maximum pellet diameter of 0.380 inch. The 0.380-inch diameter is applied to the annular pellets, while the smaller diameter is applied to the solid pellets. The results of both evaluations are reported in Table 6.6.1-4.

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

6.6.1.2.4 Assemblies with Removed Fuel Rods

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

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

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

6.6.1.2.5 Assemblies with Fuel Rods in the Guide Tubes

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

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

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

6.6.1.2.6 Consolidated Fuel

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

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

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

6.6.1.2.7 Conclusions

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

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

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

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

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

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

6.6.1.3 Maine Yankee Damaged Spent Fuel and Fuel Debris

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

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

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

6.6.1.3.1 Damaged Fuel Rods

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

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

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

The transfer and the storage casks loaded with the Westinghouse 17 × 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 sources, Control Element Assembly (CEA) fingertips, and a 24-inch ICI segment. Start-up sources must be inserted in the center guide tube. The 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 × 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 increase the reactivity of the system beyond that reported for the accident condition. The antimony (Sb) content is ignored. The start-up 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 geometry with fuel assemblies that have 24 empty rod positions in the most reactive geometry, in the four corner locations of the basket, i.e., Case "24 (Four Corners)" reported in Table 6.6.1-6. The center guide tube of this model is filled with 50% water and 50% Be. The analysis assumes that assemblies with start-up sources are loaded in all four of the basket corner fuel positions. This configuration, resulting in a system reactivity of $k_{eff} \pm \sigma$, or 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 start-up 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 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 × 14 assembly. 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.

6.6.1.4.3 Maine Yankee Miscellaneous Component Loading Restrictions

Based on the evaluation of Maine Yankee fuel assemblies with start-up sources, 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 CEA finger tips and ICI segment 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.
5) Fuel assemblies having a CEA flow plug installed must be loaded in a Class 2 canister.
6) 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

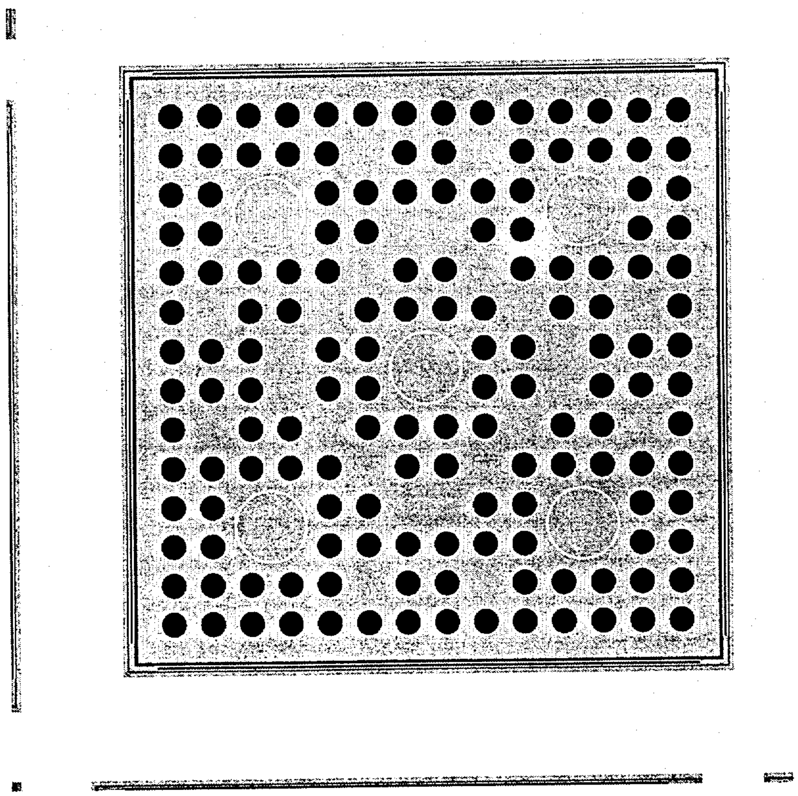

Figure 6.6.1-2 Consolidated Fuel Geometry, 113 Empty Fuel Rod Positions, Maine Yankee Site Specific Fuel

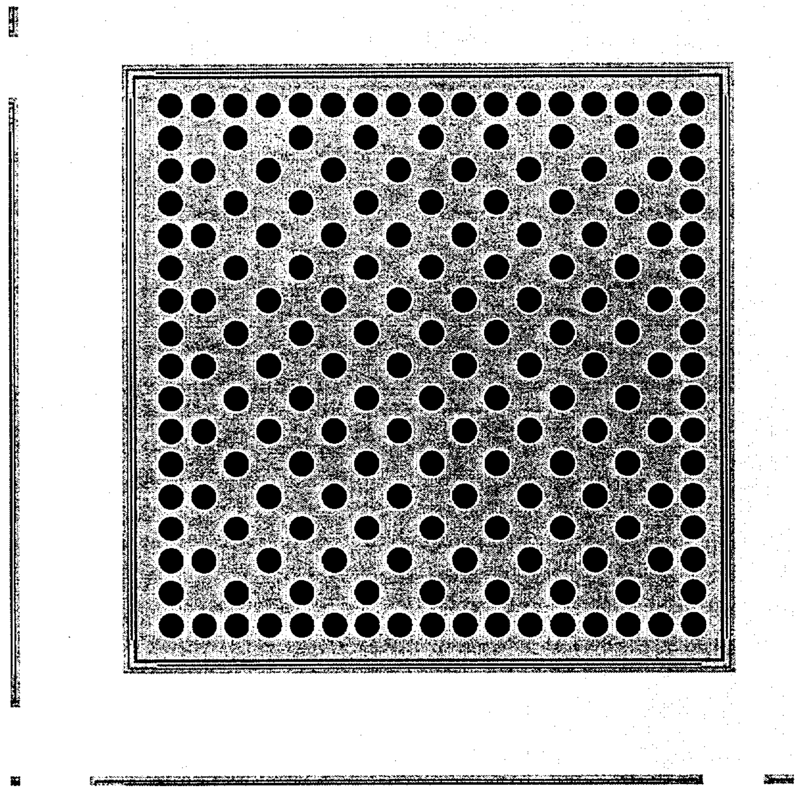

Table 6.6.1-1 Maine Yankee Standard Fuel Characteristics

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	14×14	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	14×14	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	14×14	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 × 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 × 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 × 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 × 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

1. See Table 6.6.1-6.

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[1]	Δk_{eff} to 24 (Four Corners) Infinite Height Model[2]
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

1. The k_{eff} comparison basis for this column is the finite height model with the four corner locations of the basket loaded with Maine Yankee assemblies in the most reactive missing rod geometry. This case is the first case presented in Table 6.6.1-10 with 0% UO_2 in the water above and below the active fuel of the missing rod array.
2. The k_{eff} comparison basis for this column is the infinite height model with the four corner locations of the basket loaded with Maine Yankee assemblies in the most reactive missing rod geometry, the case presented in Table 6.6.1-6 labeled "24 (Four Corners)", k_{eff} = 0.91153.

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.

Chapter 7

7.5 References

1. ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," American National Standards Institute, 1997.

2. Title 10 of the Code of Federal Regulations, Part 72 (10 CFR 72), "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," April 1996 Edition.

3. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Rules for Construction of Nuclear Power Plant Components," 1995 Edition with 1997 Addenda.

4. ASME Boiler and Pressure Vessel Code, Section IX, "Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators," 1995 Edition with 1997 Addenda.

5. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1997 Addenda.

6. Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," The American Society for Nondestructive Testing, Inc., edition as invoked by the applicable ASME Code.

7. PNL-6365, "Evaluation of Cover Gas Impuritities and Their Effects on the Dry Storage of LWR Spent Fuel," Pacific Northwest Laboratory, Richland, Washington, November, 1987.

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Chapter 8

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8.1 Procedures For Loading the Universal Storage System

The Universal Storage System consists of three principal components: the transportable storage canister (canister), the transfer cask, and the vertical concrete cask. The transfer cask is used to hold the canister during loading and while the canister is being closed and sealed. The transfer cask is also used to transfer the canister to the concrete cask and to load the canister into the transport cask. The principal handling operations involve closing and sealing the canister by welding, and placing the loaded canister in the vertical concrete cask. The vent and drain port locations are shown in Figure 8.1.1-1.

This procedure assumes that the canister with an empty basket is installed in the transfer cask, that the transfer cask is positioned in the decontamination area or other suitable work station, and that the vertical concrete cask is positioned in the plant cask receiving area or other suitable staging area. The transfer cask extension must be installed on the transfer cask if its use is required. To facilitate movement of the transfer cask to the concrete cask, the staging area should be within the operational "footprint" of the cask handling crane. The concrete cask may be positioned on a heavy-haul transporter, or on the floor of the work area.

The User must ensure that the fuel assemblies selected for loading conform to the Approved Contents provisions of Section B2.0 of Appendix 12B and to the Certificate of Compliance. Fuel assembly loading may also be administratively controlled to ensure that fuel assemblies with specific characteristics are preferentially loaded in specified positions in the canister. Preferential loading requirements are described in Appendix 12B, Section B2.1.2 and B2.1.3.

Note: Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications or the NAC-UMS® Storage FSAR. These steps include the following:

- Placement and installation of air pads
- Sequence and use of an annulus fill system including optional seals and/or foreign material exclusion devices.

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8.1.1 Loading and Closing the Transportable Storage Canister

1. Visually inspect the basket fuel tubes to ensure that they are unobstructed and free of debris. Ensure that the welding zones on the canister, shield, and structural lids, and the port covers are prepared for welding. Ensure transfer cask door lock bolts/lock pins are installed and secure.
2. Fill the canister with clean or filtered pool water until the water is about 4 inches from the top of the canister.
   Note: Do not fill the canister completely in order to avoid spilling water during the transfer to the spent fuel pool.
3. Install the annulus fill system to transfer cask including the clean or filtered pool water lines.
4. If it is not already attached, attach the transfer cask lifting yoke to the cask handling crane, and engage the transfer cask lifting trunnions.
   Note: The minimum temperature of the transfer cask (i.e., surrounding air temperature) must be verified to be higher than 0°F prior to lifting, in accordance with Appendix 12B, Section B3.4 (8).
5. Raise the transfer cask and move it over the pool, following the prescribed travel path.
6. Lower the transfer cask to the pool surface and turn on the clean or filtered pool water line to fill the canister and the annulus between the transfer cask and canister.
7. Lower the transfer cask as the annulus fills with clean or filtered pool water until the trunnions are at the surface, and hold that position until the clean or filtered pool water overflows through the upper fill lines or annulus of the transfer cask. Then lower the transfer cask to the bottom of the pool cask loading area.
   Note: If an intermediate shelf is used to avoid wetting the cask handling crane hook, follow the plant procedure for use of the crane lift extension piece.
8. Disengage the transfer cask lifting yoke to provide clear access to the canister.
9. Load the previously designated fuel assemblies into the canister.
   Note: Contents must be in accordance with the Approved Contents provisions of Appendix 12B, Section B2.0.
   Note: Contents shall be administratively controlled to ensure that fuel assemblies with certain characteristics are preferentially loaded in specified positions in the basket. Preferential loading requirements are presented in Appendix 12B, Section B2.1.2 and B2.1.3.
10. Attach a three-legged sling to the shield lid using the swivel hoist rings. Torque hoist rings in accordance with Table 8.1-2. Attach the suction pump fitting to the vent port.
    Caution: Verify that the hoist rings are fully seated against the shield lid.
    Note: Ensure that the shield lid key slot aligns with the key welded to the canister shell.

11. Using the cask handling crane, or auxiliary hook, lower the shield lid until it rests in the top of the canister.
12. Raise the transfer cask until its top just clears the pool surface. Hold at that position, and using a suction pump, drain the pool water from above the shield lid. After the water is removed, continue to raise the cask. Note the time that the transfer cask is removed from the pool. Operations through Step 28 must be completed in accordance with the time limits presented in Table 8.1.1-3.

    Note: Alternately, the temperature of the water in the canister may be used to establish the time for completion through Step 28. Those operations must be completed within 2 hours of the time that the canister water temperature is 200°F. For this alternative, the water temperature must be determined every 2 hours beginning 17 hours after the time the transfer cask is removed from the pool.
13. As the cask is raised, spray the transfer cask outer surface with clean or filtered pool water to wash off any gross contamination.
14. When the transfer cask is clear of the pool surface, but still over the pool, turn off the clean or filtered pool water flow to the annulus, remove hoses and allow the annulus water to drain to the pool. Move the transfer cask to the decontamination area or other suitable work station.

    Note: Access to the top of the transfer cask is required. A suitable work platform may need to be erected.
15. Verify that the shield lid is level and centered.
16. Attach the suction pump to the suction pump fitting on the vent port. Operate the suction pump to remove free water from the shield lid surface. Disconnect the suction pump and suction pump fitting. Remove any free standing water from the shield lid surface and from the vent and drain ports.
17. Decontaminate the top of the transfer cask and shield lid as required to allow welding and inspection activities.

    Note: Supplemental shielding may be used for activities around the shield lid.
18. Insert the drain tube assembly with a female quick-disconnect attached through the drain port of the shield lid into the basket drain tube sleeve. Remove the female quick-disconnect. Torque the drain tube assembly to 125 ± 5 ft-lbs. Install a quick-disconnect to open the valve in the vent port.
19. Connect the suction pump to the drain port. Verify that the vent port is open. Remove approximately 50 gallons of water from the canister. Disconnect and remove the pump.

    Caution: Radiation level may increase as water is removed from the canister.
20. Install the automatic welding equipment, including the supplemental shield plate.

21. Attach the hydrogen gas detector to the vent port. Verify that the concentration of any detectable hydrogen gas is below 2.4%.

    Note: If the concentration exceeds 2.4%, connect and operate the vacuum system to remove gases from the underside of the shield lid and re-verify the hydrogen gas concentration. Disconnect and remove vacuum system.

22. Operate the welding equipment to complete the root weld joining the shield lid to the canister shell following approved procedures. Remove the hydrogen detector from the vent tube. Leave the connector and vent tube installed to vent the canister.
23. Examine the root weld using liquid penetrant and record the results.
24. Complete welding of the shield lid to the canister shell.
25. Liquid penetrant examine the final weld surface and record the results.
26. Attach a regulated air or nitrogen supply line to the vent port. Install a valved fitting on the drain port and ensure the valve is closed. Pressurize the canister to 35 psia and hold the pressure. There must be no loss of pressure for a minimum of 60 minutes.
27. Release the pressure.

    Note: As an option, an informational helium leak test may be conducted at this point using the following steps (the record leak test is performed at Step 49).

    27a. Evacuate and backfill the canister with helium having a minimum purity of 99.9% to a pressure of 18.0 psia.

    27b. Using a helium leak detector ("sniffer" detector) with a test sensitivity of 5×10^{-5} cm^3/sec (helium), survey the weld joining the shield lid and canister shell.

    27c. At the completion of the survey, vent the canister helium pressure to one atmosphere (0 psig).

28. Drain the canister.

    28a. Attach the suction pump to the drain line. Ensure that the vent line is open. Using the pump, remove the remaining free water from the canister cavity.

    28b. Remove the suction pump from the drain line and close the drain line.

    28c. Using the vent port, pressurize the canister with nitrogen to 15 (+3, -0) psig.

    28d. Open the drain line to blow any remaining free water from the canister.

    28e. When free water is no longer present at the drain line, stop the flow of nitrogen, vent the remaining pressure and remove the nitrogen supply line. Note the time that the last free water is removed from the canister cavity.

    Caution: Radiation levels at the top and sides of the transfer cask will rise as water is removed.

    Note: The time duration from completion of draining through the completion of helium backfill (Step 34) shall be monitored in accordance with LCO 3.1.1.

29. Attach the vacuum equipment to the vent and drain ports. Dry any free standing water in the vent and drain port recesses.
30. Operate the vacuum equipment until a vacuum of 3 mm of mercury exists in the canister.
    Note: Vacuum drying pressure must conform to the requirements of LCO 3.1.2.
31. Verify that no water remains in the canister by holding the vacuum for 30 minutes. If water is present in the cavity, the pressure will rise as the water vaporizes. Continue the vacuum/hold cycle until the conditions of LCO 3.1.2 are met.
32. Backfill the canister cavity with helium having a minimum purity of 99.9% to a pressure of one atmosphere (0 psig).
33. Restart the vacuum equipment and operate until a vacuum of 3 mm of mercury exists in the canister.
34. Backfill the canister with helium having a minimum purity of 99.9% to a pressure of one atmosphere (0 psig).
    Note: Canister helium backfill pressure must conform to the requirements of LCO 3.1.3.
    Note: Monitor the time from this step (completion of helium backfill) until completion of canister transfer to the concrete cask in accordance with LCO 3.1.4.
35. Disconnect the vacuum and helium supply lines from the vent and drain ports. Dry any residual water that may be present in the vent and drain port cavities.
36. Install the vent and drain port covers.
37. Complete the root pass weld of the drain port cover to the shield lid.
    Note: If the drain port cover weld is completed in a single pass, the weld final surface is liquid penetrant inspected in accordance with Step 40.
38. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.
39. Complete welding of the drain port cover to the shield lid.
40. Prepare the weld and perform a liquid penetrant examination of the drain port cover weld final pass. Record the results.
41. Complete the root pass weld of the vent port cover to the shield lid.
    Note: If the drain port cover weld is completed in a single pass, the weld final surface is liquid penetrant inspected in accordance with Step 44.
42. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.
43. Complete welding of the vent port cover to the shield lid.
44. Prepare the weld and perform a liquid penetrant examination of the weld final surface. Record the results.

45. Remove the welding machine and any supplemental shielding used during shield lid closure activities.
46. Install the helium leak test fixture.
47. Attach the vacuum line and leak detector to the leak test fixture fitting.
48. Operate the vacuum system to establish a vacuum in the leak test fixture.
49. Operate the helium leak detector to verify that there is no indication of a helium leak exceeding 2×10^{-7} cm^3/second, at a minimum test sensitivity of 1×10^{-7} cm^3/second helium, in accordance with the requirements of LCO 3.1.5.
50. Release the vacuum and disconnect the vacuum and leak detector lines from the fixture.
51. Remove the leak test fixture.
52. Attach a three-legged sling to the structural lid using the swivel hoist rings.

    Caution: Ensure that the hoist rings are fully seated against the structural lid. Torque the hoist rings in accordance with Table 8.1.1-2. Verify that the spacer ring is in place on the structural lid.

    Note: Verify that the structural lid is stamped or otherwise marked to provide traceability of the canister contents.
53. Using the cask handling crane or the auxiliary hook, install the structural lid in the top of the canister. Verify that the structural lid is flush with, or protrudes slightly above, the canister shell. Verify that the gap in the spacer ring is not aligned with the shield lid alignment key. Remove the hoist rings.
54. Install the automatic welding equipment on the structural lid including the supplemental shield plate.
55. Operate the welding equipment to complete the root weld joining the structural lid to the canister shell.
56. Prepare the weld and perform a liquid penetrant examination of the weld root pass. Record the results.
57. Continue with the welding procedure, examining the weld at 3/8-inch intervals using liquid penetrant. Record the results of each intermediate examination.

    Note: If ultrasonic testing of the weld is used, testing is performed after the weld is completed.
58. Remove the weld equipment and supplemental shielding.
59. Perform a smear survey of the accessible area at the top of the canister to ensure that the surface contamination is less than the limits established for the site. Smear survey results shall meet the requirements of Technical Specification LCO 3.2.1.
60. Install the transfer cask retaining ring. Torque bolts to 155 ± 10 ft-lbs. (Table 8.1.1-2).
61. Decontaminate the external surface of the transfer cask to the limits established for the site.

Figure 8.1.1-1 Vent and Drain Port Locations

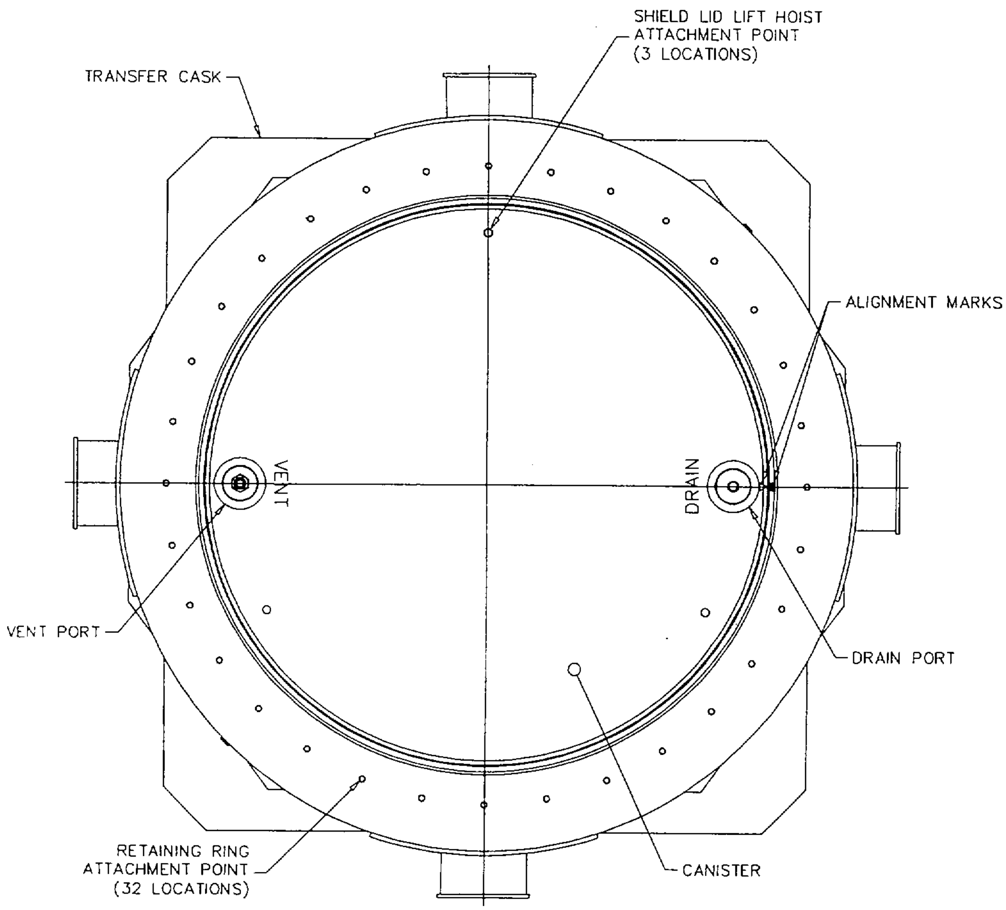

Table 8.1.1-1 List of Ancillary Equipment

Item	**Description**
Transfer Cask Lifting Yoke	Required for lifting and moving the transfer cask.
Heavy-Haul Transporter (Optional)	Heavy-haul (double drop frame) trailer required for moving the loaded and empty vertical concrete cask to and from the ISFSI pad.
Mobile Lifting Frame (Optional)	A self-propelled or towed A-frame lifting device for the concrete cask. Mobile Lifting Frame is used to lift the cask and move it using four lifting lugs in the top of the concrete cask.
Helium Supply System	Supplies helium to the canister for helium backfill and purging operations.
Vacuum Drying System	Used for evacuating the canister. Used to remove residual water, air and initial helium backfill.
Automated Welding System	Used for welding the shield lid and structural lid to the canister shell.
Self-Priming Pump	Used to remove water from the canister.
Shield Lid Sling	A three-legged sling used for lifting the shield lid. It is also used to lift the concrete cask shield plug and lid.
Canister Sling	A set of 2 three-legged slings joined by a master link, used for lifting the structural lid by itself, or for lifting the canister when the structural lid is welded to it. The master link allows the slings to be loaded simultaneously during the lift.
Transfer Adapter	Used to align the transfer cask to the vertical concrete cask or the Universal Transport Cask. Provides the platform for the operation of the transfer cask shield doors.
Transfer Cask Extension	A carbon steel ring used to extend the height of the transfer cask when using the next larger size canister.
Hydraulic Unit	Operates the shield doors of the transfer cask.
Lift Pump Unit	Jacking system for raising and lowering the concrete cask.
Air Pad Rig Set	Air cushion system used for moving the concrete cask.
Supplemental Shielding Fixture	An optional carbon steel fixture inserted in the Vertical Concrete Cask air inlets to reduce radiation dose rates at the inlets.

Table 8.1.1-2 Torque Values

Fastener	Torque Value (ft-lbs)	Torque Pattern
Transfer Adapter Bolts (Optional)	40 ± 5	None
Transfer Cask Retaining Ring	155 ±10	0°, 180°, 270° and 90° in two passes
Transfer Cask Extension	155 ±10	None
Vertical Concrete Cask Lid	40 ± 5	None
Lifting Hoist Rings (Loaded Canister) Canister Structural Lid	800 +80, -0	None
Canister Lid Plug Bolts Shield Lid Plug Bolts	Hand Tight Hand Tight	None None
Transfer Cask Door Lock Bolts	Hand Tight	None
Canister Drain Tube	125 ± 5	None

Table 8.1.1-3 Handling Time Limits Based on Decay Heat Load with Canister Full of Water

Total Heat Load (L) (kW)	**PWR Time Limit (Hours)**	**BWR Time Limit (Hours)**
$20.0 < L \leq 23.0$	17	17
$17.6 < L \leq 20.0$	18	17
$14.0 < L \leq 17.6$	20	17
$11.0 < L \leq 14.0$	22	17
$8.0 < L \leq 11.0$	24	17
$L \leq 8.0$	26	17

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8.1.2 Loading the Vertical Concrete Cask

This section of the loading procedure assumes that the vertical concrete cask is located on the bed of a heavy-haul transporter, or on the floor of the work area, under a crane suitable for lifting the loaded transfer cask. The vertical concrete cask shield plug and lid are not in place, and the bottom pedestal plate cover is installed.

1. Using a suitable crane, place the transfer adapter on the top of the concrete cask.
2. If using the transfer adapter bolt hole pattern for alignment, align the adapter to the concrete cask. Bolt the adapter to the cask using four (4) socket head cap screws (Note: Use of socket head cap screws is optional).
3. Verify that the shield door connectors on the adapter plate are in the fully extended position.
   Note: Steps 4 through 6 may be performed in any order, as long as all items are completed.
4. If not already done, attach the transfer cask lifting yoke to the cask handling crane. Verify that the transfer cask retaining ring is installed.
5. Install six (6) swivel hoist rings in the structural lid of the canister and torque to the value specified in Table 8.1.1-2. Attach two (2) three-legged slings to the hoist rings.
   Caution: Ensure that the hoist rings are fully seated against the structural lid.
6. Stack the slings on the top of the canister so they are available for use in lowering the canister into the storage cask.
7. Engage the transfer cask trunnions with the transfer cask lifting yoke. Ensure that all lines are disconnected from the transfer cask.
   Note: The minimum temperature of the transfer cask (i.e., temperature of the surrounding air) must be verified to be higher than 0°F prior to lifting, in accordance with Appendix 12B, Section B3.4(8).
   Note: Verify that the transfer cask extension is installed if required.
8. Raise the transfer cask and move it over the concrete cask. Lower the transfer cask, ensuring that the transfer cask shield door rails and connector tees align with the adapter plate rails and door connectors. Prior to final set down, remove transfer cask shield door lock bolts/lock pins (there is a minimum of one per door).
9. Ensure that the shield door connector tees are engaged with the adapter plate door connectors.
10. Disengage the transfer cask yoke from the transfer cask and from the cask handling crane hook.

11. Return the cask handling crane hook to the top of the transfer cask and engage the two (2) three-legged slings attached to the canister.
    Caution: The top connection of the three-legged slings must be at least 75 inches above the top of the canister.
12. Lift the canister slightly (about ½ inch) to take the canister weight off of the transfer cask shield doors.
    Note: A load cell may be used to determine when the canister is supported by the crane.
    Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.
13. Using the hydraulic system, open the shield doors to access the concrete cask cavity.
14. Lower the canister into the concrete cask, using a slow crane speed as the canister nears the pedestal at the base of the concrete cask.
15. When the canister is properly seated, disconnect the slings from the canister at the crane hook, and close the transfer cask shield doors.
16. Retrieve the transfer cask lifting yoke and attach the yoke to the transfer cask.
17. Lift the transfer cask off of the vertical concrete cask and return it to the decontamination area or designated work station.
18. Using the auxiliary crane, remove the adapter plate from the top of the concrete cask.
19. Remove the swivel hoist rings from the structural lid and replace them with threaded plugs.
20. Install three swivel hoist rings in the shield plug and torque in accordance with Table 8.1.1-2.
21. Using the auxiliary crane, retrieve the shield plug and install the shield plug in the top of the concrete cask. Remove swivel hoist rings and insert threaded plugs.
22. Install seal tape around the diameter of the lid bolting pattern on the concrete cask flange.
23. Using the auxiliary crane, retrieve the concrete cask lid and install the lid in the top of the concrete cask. Secure the lid using six stainless steel bolts. Torque bolts in accordance with Table 8.1.1-2.
24. Ensure that there is no foreign material left at the top of the concrete cask. Install the tamper-indicating seal.
25. If used, install a supplemental shielding fixture in each of the four inlets. Note: The supplemental shielding fixtures may also be shop installed.

8.1.3 Transport and Placement of the Vertical Concrete Cask

This procedure assumes that the loaded vertical concrete cask is positioned on a heavy-haul transporter and is to be positioned on the ISFSI pad using the air pad set. Alternately, the concrete cask may be lifted and moved using a mobile lifting frame. The mobile lifting frame lifts the cask using four lifting lugs at the top of the concrete cask. The lifting frame may be self-propelled or towed, and does not use the air pad set.

The vertical concrete cask lift height limit is 24-inches when the cask is moved using the air pad set or the mobile lifting frame in accordance with the requirements of Appendix 12A, Section A5.6. Because of lift fixture configuration, the maximum lift height of the concrete cask using the jacking arrangement is approximately 4 inches.

The concrete cask surface dose rates must be verified in accordance with the requirements of LCO 3.2.2. These measurements may be made prior to movement of the cask, at a location along the transport path, or at the ISFSI. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the concrete cask air inlets to reduce the radiation dose rate at the inlets.

1. Using a suitable towing vehicle, tow the heavy-haul transporter to the dry storage pad (ISFSI). Verify that the bed of the transporter is approximately at the same height as the pad surface. Install four (4) hydraulic jacks at the four (4) designated jacking points at the air inlets in the bottom of the vertical concrete cask.
2. Raise the concrete cask approximately 4 inches using the hydraulic jacks.

   Caution: Do not exceed a maximum lift height of 24 inches, in accordance with the requirements of Administrative Control A5.6.
3. Move the air-bearing rig set under the cask.

   Note: A hydraulic skid may also be used to move the concrete cask. The height the concrete cask is raised depends upon the height of the skid or air pad set used, but may not exceed 20 inches.
4. Inflate the air-bearing rig set. Remove the four (4) hydraulic jacks.
5. Using a suitable towing vehicle, move the concrete cask from the bed of the transporter to the designated location on the storage pad.

   Note: Spacing between concrete casks must not be less than 15 feet (center-to-center).
6. Turn off the air-bearing rig set, allowing it to deflate.

7. Reinstall the four (4) hydraulic jacks and raise the concrete cask approximately 4 inches.

   Caution: Do not exceed a maximum lift height of 24 inches, in accordance with the requirements of Administrative Control A5.6.

8. Remove the air-bearing rig set pads. Ensure that the surface of the dry storage pad under the concrete cask is free of foreign objects.
9. Lower the concrete cask to the surface and remove the four (4) hydraulic jacks.
10. Install the screens in the inlets and outlets.
11. Install/connect temperature monitoring equipment and verify operation in accordance with LCO 3.1.6.
12. Scribe/stamp concrete cask name plate to indicate loading information.

8.2 Removal of the Loaded Transportable Storage Canister from the Vertical Concrete Cask

Removal of the loaded canister from the vertical concrete cask is expected to occur at the time of shipment of the canistered fuel off site. Alternately, removal could be required in the unlikely event of an accident condition that rendered the concrete cask or canister unsuitable for continued long-term storage or for transport. This procedure assumes that the concrete cask is being returned to the reactor cask receiving area. However, the cask may be moved to another facility or area using the same operations. It identifies the general steps to return the loaded canister to the transfer cask and return the transfer cask to the decontamination station, or other designated work. area or facility. Since these steps are the reverse of those undertaken to place the canister in the concrete cask, as described in Section 8.1.2, they are only summarized here.

The concrete cask may be moved using the air pad set or a mobile lifting frame. This procedure assumes the use of the air pad set. If a lifting frame is used, the concrete cask is lifted using four lifting lugs in the top of the cask, and the air pad set and heavy haul transporter are not required. The mobile lifting frame may be self-powered or towed.

At the option of the user, the canister may be removed from the concrete cask and transferred to another concrete cask or to the Universal Transport Cask at the ISFSI site. This transfer is done using the transfer cask, which provides shielding for the canister contents during the transfer.

Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications or the NAC-UMS® FSAR. This includes the placement and installation of the air pads.

1. Remove the screens and instrumentation.
2. Using the hydraulic jacking system and the air pad set, move the concrete cask from the ISFSI pad to the heavy-haul transporter. The bed of the transporter must be approximately level with the surface of the pad and sheet metal plates are placed across the gap between the pad and the transporter bed.

   Caution: Do not exceed a maximum lift height of 24 inches when raising the concrete cask.
3. Tow the transporter to the cask receiving area or other designated work area or facility.

4. Remove the concrete cask shield plug and lid. Install the hoist rings in the canister structural lid and torque to the value specified in Table 8.1.1-2. Verify that the hoist rings are fully seated against the structural lid and attach the lift slings. Install the transfer adapter on the top of the concrete cask.
5. Retrieve the transfer cask with the retaining ring installed, and position it on the transfer adapter. Attach the shield door hydraulic cylinders.

   Note: The surrounding air temperature for cask unloading operations shall be ≥ 0°F.
6. Open the shield doors. Attach the canister lift slings to the cask handling crane hook.

   Caution: The master link must be at least 75 inches above the top of the canister.
7. Raise the canister into the transfer cask.

   Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.
8. Close the shield doors. Lower the canister to rest on the shield doors. Disconnect the canister slings from the crane hook. Install and secure door lock bolts/lock pins.
9. Retrieve the transfer cask lifting yoke. Engage the transfer cask trunnions and move the transfer cask to the decontamination area or designated work station.

After the transfer cask containing the canister is in the decontamination area or other suitable work station, additional operations may be performed on the canister. It may be opened, transferred to another storage cask, or placed in the Universal Transport Cask.

8.3 Unloading the Transportable Storage Canister

This section describes the basic operations required to open the sealed canister if circumstances arise that dictate the opening of a previously loaded canister and the removal of the stored spent fuel. It is assumed that the canister is positioned in the transfer cask and that the transfer cask is in the decontamination station or other suitable work station in the facility. The principal mechanical operations are the cutting of the closure welds, filling the canister with water, cooling the fuel contents, and removing the spent fuel. Supplemental shielding is used as required. The time duration for holding the canister in the transfer cask shall not exceed 4 hours without forced air cooling. Once forced air cooling is initiated, the amount of time that the canister may be in the transfer cask is not limited. The canister cooling water temperature, flow rate and pressure must be limited in accordance with this procedure.

Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications of the NAC-UMS® Storage FSAR. This includes the sequence and use of an annulus fill system including optional seals and/or foreign material exclusion devices.

1. Remove the transfer cask retaining ring.
2. Survey the top of the canister to establish the radiation level and contamination level at the structural lid.
3. Set up the weld cutting equipment to cut the structural lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment).
4. Enclose the top of the transfer cask in a radioactive material retention tent, as required.
   Caution: Monitor for any out-gassing. Wear respiratory protection as required.
5. Operate the cutting equipment to cut the structural lid weld.
6. After proper monitoring, remove the retention tent. Remove the cutting equipment and attach a three-legged sling to the structural lid.
7. Using the auxiliary crane, lift the structural lid from the canister and out of the transfer cask.
8. Survey the top of the shield lid to determine radiation and contamination levels. Use supplemental shielding as necessary. Decontaminate the top of the shield lid, if necessary.
9. Reinstall the retention tent. Using an abrasive grinder or hydrolaser, and wearing suitable respiratory protection, cut the welds joining the vent and drain port covers to the shield lid.
   Caution: The canister could be pressurized.

10. Remove the port covers. Monitor for any out-gassing and survey the radiation level at the quick-disconnect fittings.
11. Attach a nitrogen gas line to the drain port quick-disconnect and a discharge line from the vent port quick-disconnect to an off-gas handling system in accordance with the schematic shown in Figure 8.3-1. Set up the vent line with appropriate instruments so that the pressure in the discharge line and the temperature of the discharge gas are indicated. Continuously monitor the radiation level of the discharge line.

    Caution: The discharge gas temperature could initially be above 400°F. The discharge line and fittings may be very hot.

    Note: Any significant radiation level in the discharge gas indicates the presence of fission gas products. The temperature of the gas indicates the thermal conditions in the canister.
12. Start the flow of nitrogen through the line until there is no evidence of fission gas activity in the discharge line. Continue to monitor the gas discharge temperature. When there is no additional evidence of fission gas, stop the nitrogen flow and disconnect the drain and vent port line connections. The nitrogen gas flush must be maintained for at least 10 minutes.

    Note: See Figure 8.3-1 for Canister Reflood Piping and Control Schematic.
13. Perform canister refill and fuel cooldown operations. Attach a source of clean or filtered pool water with a minimum temperature of 70°F and a maximum supply pressure of 25 (+10, -0) psig to the drain port quick-disconnect. Attach a steam rated discharge line to the vent port quick-disconnect and route it to a fuel pool cooler or an in-pool steam condensing unit. Slowly start the flow of clean or filtered pool water to establish a flow rate at 5 (+3, -0) gpm. Monitor the discharge line pressure gage during canister flooding. Stop filling the canister if the canister vent line pressure exceeds 45 psig. Re-establish water flow when the canister pressure is below 35 psig. The discharge line will initially discharge hot gas, but after the canister fills, it will discharge hot water.

    Caution: Relatively cool water may flash to steam as it encounters hot surfaces within the canister.

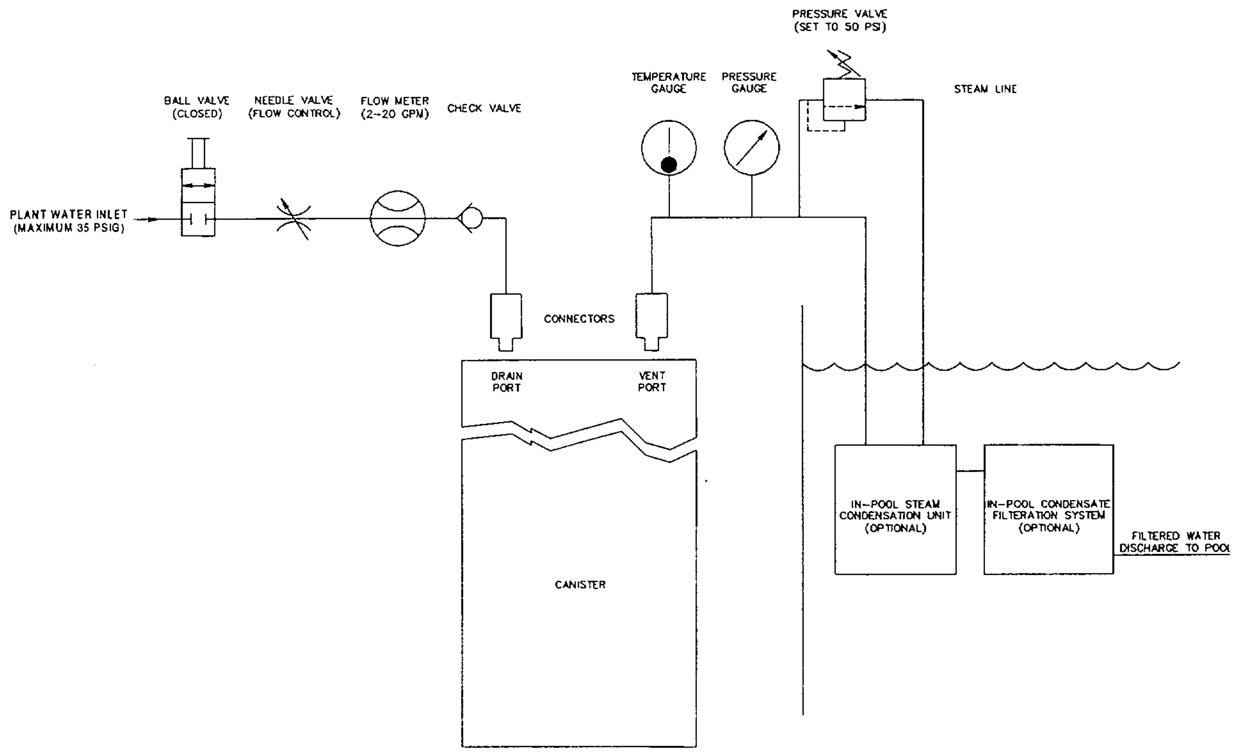
    Caution: If there are grossly failed or ruptured fuel rods within the canister, very high levels of radiation could rapidly appear at the discharge line. The radiation level of the discharge gas or water should be continuously monitored.
14. Monitor water flow through the canister until the water discharge temperature is below 200°F. Stop the flow of water and remove the connection to the drain line.

    Note: Monitor canister water temperature and reinitiate cooldown operations if temperature exceeds 200°F.
15. Connect a suction pump to the drain port and a vent line to the vent port. Operate the pump and remove approximately 50 gallons of water. Disconnect and remove the pump.

16. Set up the weld cutting equipment to cut the shield lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment.). Route the vent line to avoid interference with the weld cutting operation.
17. Tent the top of the transfer cask and wear respiratory protection equipment as required. Attach a hydrogen gas detector to the vent port line. Verify that the concentration of hydrogen gas is less than 2.4%.
18. Operate the cutting equipment to cut the shield lid weld.
    Note: Stop the cutting operation if the hydrogen gas detector indicates a concentration of hydrogen gas above 2.4%. Connect the vacuum drying system and evacuate gas before proceeding with the cutting operation.
19. Remove the cutting equipment. Remove supplemental shielding if used. Install the shield lid lifting hoist rings, verifying that the hoist rings are fully seated against the shield lid, and attach a three-legged sling. Attach a tag line to the sling set to aid in attaching the sling to the crane hook (at Step 24).
20. Install the annulus fill system to the transfer cask, including the clean or demineralized water lines.
21. Retrieve the transfer cask lifting yoke and engage the transfer cask lifting trunnions.
22. Move the transfer cask over the pool and lower the bottom of the transfer cask to the surface. Start the flow of clean or filtered pool water to the transfer cask annulus. Continue to lower the transfer cask, as the annulus fills with water, until the top of the transfer cask is about 4 inches above the pool surface. Hold this position until clean or filtered pool water fills to the top of the transfer cask.
23. Lower the transfer cask to the bottom of the cask loading area and remove the lifting yoke.
24. Attach the shield lid lifting sling to the crane hook.
    Caution: The drain line tube is suspended from the under side of the shield lid. The lid should be raised as straight as possible until the drain tube clears the canister basket. The under side of the shield lid could be highly contaminated.
25. Slowly lift the shield lid. Move the shield lid to one side after it is raised clear of the transfer cask.
26. Visually inspect the fuel for damage.

At this point, the spent fuel could be transferred from the canister to the fuel racks. If the fuel is damaged, special handling equipment may be required to remove the fuel. In addition, the bottom of the canister could be highly contaminated. Care must be exercised in the handling of the transfer cask when it is removed from the pool.

Figure 8.3-1 Canister Reflood Piping and Controls Schematic



Chapter 9

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9.1.2 Structural and Pressure Test

The transportable storage canister is pressure tested at the time of use. After loading of the canister basket with spent fuel, the shield lid is welded in place after approximately 50 gallons of water are removed from the canister. Removal of the water ensures that the water level in the canister is below the bottom of the shield lid during welding of the shield lid to the canister shell. Prior to removing the remaining spent fuel pool water from the canister, the canister is pressure tested at 35 psia. This pressure is held for a minimum 10 minutes. Any loss of pressure during the test period is unacceptable. The leak must be located and repaired. The pressure test procedure is described in Section 8.1.1.

Transfer Cask

The transfer cask lifting trunnions and the bottom shield doors shall be tested in accordance with the requirements of ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kg) or More for Nuclear Materials" [8].

The lifting trunnion load test shall consist of applying a vertical load of 660,000 pounds, which is greater than 300 percent of the maximum service load (207,616 pounds) for the loaded canister with the shield lid and full of water. The bottom shield door load test shall consist of applying a vertical load of 266,000 pounds, which is over 300 percent of the maximum service load (88,376 pounds) for the loaded canister with the shield lid and full of water. These maximum service loads are based on the Class 5 BWR configuration, which is the heaviest configuration and, thus, bounds all of the other configurations.

The load tests shall be held for a minimum of 10 minutes and shall be performed in accordance with approved, written procedures.

Following completion of the lifting trunnion load tests, all trunnion welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Magnetic particle or liquid penetrant examinations shall be performed in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5340 or NF-5350, as applicable. Similarly, following completion of the bottom shield door load tests, all door rail welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking.

Any evidence of permanent deformation, cracking or galling of the load bearing surfaces or unacceptable liquid penetrant examination results, shall be cause for evaluation, rejection, or rework of the affected component. Liquid penetrant or magnetic particle examinations of all load bearing welds shall be performed in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5350 or NF-5340, as applicable.

Concrete Cask

The concrete cask, at the option of the user/licensee, may be provided with lifting lugs to allow for the vertical handling and movement of the concrete cask. The lifting lugs are provided as two sets of two lugs each, through which a lifting pin is inserted and connected to a specially designed mobile lifting frame. The concrete cask lifting lug system and mobile lifting frame and pins are designed, analyzed, and load tested in accordance with ANSI N14.6. The concrete cask lifting lug load test shall consist of applying a vertical load of 515,200 pounds, which is greater than 150 percent of the maximum concrete cask weight of 312,210 pounds plus a 10 percent dynamic load factor.

The test load shall be applied for a minimum of 10 minutes in accordance with approved, written procedures. Following completion of the load test, all load bearing surfaces of the lifting lugs shall be visually inspected for permanent deformation, galling, or cracking. Liquid penetrant or magnetic particle examinations of load bearing surfaces shall be performed in accordance with ASME Code, Section V, Articles 1, 6 and/or 7, with acceptance criteria in accordance with ASME Code, Section III, Subsection NF, NF-5350 or NF-5340, as applicable.

Any evidence of permanent deformation, cracking, or galling, or unacceptable liquid penetrant or magnetic particle examination results for the load bearing surfaces of the lifting anchors shall be cause for evaluation, rejection, or rework and retesting.

9.1.3 Leak Tests

The canister is leak tested at the time of use. After the pressure test described in Section 9.1.2, the canister is drained of residual water, vacuum dried and backfilled with helium. The canister is pressurized with helium to 0 psig. The shield lid to canister shell weld and the weld joining the port covers to the shield lid, are helium leak tested using a leak test fixture installed above the shield lid. The leaktight criteria of 2.0×10^{-7} cm^3/sec (helium) of ANSI N14.5[1] is applied. The leak test is performed at a sensitivity of 1.0×10^{-7} cm^3/sec (helium). Any indication of a leak of 2.0 $\times 10^{-7}$ cm^3/sec (helium), or greater, is unacceptable and repair is required as appropriate.

9.1.4 Component Tests

The components of the Universal Storage System do not require any special tests in addition to the material receipt, dimensional, and form and fit tests described in this chapter.

9.1.4.1 Valves, Rupture Disks and Fluid Transport Devices

The transportable storage canister and the vertical concrete cask do not contain rupture disks or fluid transport devices. There are no valves that are part of the confinement boundary for transport or storage. Quick-disconnect valves are installed in the vent and drain ports of the shield lid. These valves are convenience items for the operator, as they provide a means of quickly connecting ancillary drain and vent lines to the canister. During storage and transport, these fittings are not accessible, as they are covered by port covers that are welded in place when the canister is closed. As presented for storage and transport, the canister has no accessible valves or fittings.

9.1.4.2 Gaskets

The transportable storage canister and the vertical concrete cask have no mechanical seals or gaskets that form an integral part of the system, and there are no mechanical seals or gaskets in the confinement boundary.

9.1.5 Shielding Tests

Based on the conservative design of the Universal Storage System for shielding criteria and the detailed construction requirements, no shielding tests of the vertical concrete cask are required.

9.1.6 Neutron-Absorber Tests

Neutron-absorbing material, BORAL®, is used as a poison in the BWR and PWR fuel tubes. BORAL is manufactured by AAR Advanced Structures (AAR) of Livonia, Michigan, under a Quality Assurance/Quality Control program in conformance with the requirements of 10 CFR 50, Appendix B. The computer-aided manufacturing process consists of several steps - the first being the mixing of the aluminum and boron-carbide powders that form the core of the finished material, with the amount of each powder a function of the desired ^{10}B areal density. The methods used to control the weight and blend the powders are patented and proprietary processes of AAR.

After manufacturing, test samples from each batch of BORAL® neutron absorber (poison) sheets shall be tested using neutron absorption and/or wet chemistry techniques to verify the presence, proper distribution, and minimum weight percent of ^{10}B. The tests shall be performed in accordance with approved written procedures.

Preparation of Samples

Detailed written procedures to perform wet chemistry and/or neutron absorption tests of each batch of BORAL® sheets shall be established by the manufacturer and approved by NAC. For each batch of BORAL® sheets, a sample shall be taken from each end of randomly selected sheets. The samples shall be indelibly marked and recorded for identification. At least 2 percent of the sheets in a batch shall be fully tested as described, with the remaining sheets to be tested at one location to ensure the presence of boron in those sheets.

Neutron-Absorber Test Performance

Neutron attenuation testing of this material is performed on test samples from each BORAL sheet pour to verify the presence, proper distribution, and areal density of neutron-absorbing material. This real-time radiographic test of the samples is performed in accordance with approved, written procedures by an approved facility with a neutron beam capability. For each batch of neutron absorber, a 2-inch-wide sample is taken from each end of a sheet. The samples are indelibly marked and recorded for identification. A reference BORAL standard for the appropriate ^{10}B areal density is used as the test acceptance standard. For system calibration, a camera is placed in the neutron beam path and the reference BORAL standard plate is placed in a stationary, fixed location between the camera and beamport. A luminance level is then determined at a location near the center of the specimen. The BORAL standard is then replaced by the BORAL test specimen and a luminance level is determined at a location near the center of the specimen.

The test results are considered acceptable if the luminance level determined for each test specimen is equal to or less than that of the BORAL standard. The minimum acceptable areal density for BORAL ^{10}B loading is of 0.011 g/cm^2 for the BORAL sheets used in the BWR fuel tubes and 0.025 g/cm^2 for the PWR BORAL sheets. Any specimen not meeting the acceptance criterion is rejected, and the sheets from that pour are similarly rejected or individually tested for acceptance.

Wet Chemistry Test Performance

An approved facility with chemical analysis capability shall be selected to perform the wet chemistry tests. The tests will ensure the presence of boron and enable the calculation of the ^{10}B areal density.

The most common method of verifying the acceptability of neutron absorber material is the wet chemistry method—a chemical analysis where the aluminum is separated from a sample with known thickness and volume. The remaining boron-carbide material is weighed and the areal density of ^{10}B is computed. A statistical conclusion about the BORAL® sheet from which the sample was taken and that batch of BORAL® sheets may then be drawn based on the test results and the established manufacturing processes previously noted.

Acceptance Criteria

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube drawings.

The neutron absorption test shall be considered acceptable if the neutron count determined for each test specimen is less than or equal to the highest permissible neutron count rate determined from the reference sheet(s), which are based on the ^{10}B areal density specified on the fuel tube drawings.

Any specimen not meeting the acceptance criteria shall be rejected and all of the sheets from that batch shall be similarly rejected.

9.1.7 Thermal Tests

No thermal acceptance testing of the Universal Storage System is required during construction. Thermal performance of the system is confirmed in accordance with the procedure specified in Section 9.2.3. In addition, temperature measurements are taken at the air outlets of the concrete cask(s) placed in service, in accordance with Chapter 12.0, as verification of the thermal performance of the storage system.

9.1.8 Cask Identification

A stamped, stainless steel nameplate is permanently attached on the outer surface of the concrete cask as shown on Drawing No. 790-562.

The nameplate is installed at approximately eye level and includes the following information:

Vertical Concrete Cask	
Owner:	(Utility Name)
Designer:	NAC International Inc.
Fabricator:	(Vendor Name)
Date of Manufacture:	(mm/dd/yy)
Model Number:	(UMS-XXX)
Cask No.:	(XXX)
Date of Loading:	(mm/dd/yy)
Empty Weight:	(Pounds [kilograms])

9.2 Maintenance Program

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

9.2.1 UMS® Storage System Maintenance

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

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

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

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

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

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

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

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

9.2.2 Transfer Cask Maintenance

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

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

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

9.2.3 Required Surveillance of First Storage System Placed in Service

For the first Universal Storage System placed in service with a heat load equal to or greater than 10 kW, the canister is loaded with spent fuel assemblies and the decay heat load calculated for that canister. The canister is then loaded into the vertical concrete cask, and the cask's thermal performance is evaluated by measuring the ambient and air outlet temperatures for normal air flow. The purpose of the surveillance is to measure the heat removal performance of the Universal Storage System and to establish baseline data. In accordance with 10 CFR 72.4, a letter report summarizing the results of the surveillance and evaluation will be submitted to the NRC within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® system heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® systems that are subsequently loaded, provided that the performance of the first system placed in service with a heat load $\geq$ 10 kW, is demonstrated by the comparison of the calculated and measured temperatures.

Chapter 10

10.2 Radiation Protection Design Features

The description of the radiation shielding design is provided in Section 5.3.1. The design criteria radiation exposure rates are summarized in Table 2-1. The principal radiation protection design features are the shielding necessary to meet the design objectives, the placement of penetrations near the edge of the canister shield lid to reduce operator exposure and handling time, and the use of shaped supplemental shielding for work on and around the shield lid, as necessary. This supplemental shielding reduces operator dose rates during the welding, inspection, draining, drying and backfilling operations that seal the canister. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the air inlets to reduce the radiation dose rate at the base of the vertical concrete cask.

Radiation exposure rates at various work locations are determined for the principal Universal Storage System operational steps using a combination of the SAS4 [3] and SKYSHINE III [4] computer codes. The use of SAS4 is described in Section 5.1.2. The SKYSHINE-III code is discussed in Section 10.4. The calculated dose rates decrease with time.

10.2.1 Design Basis for Normal Storage Conditions

The radiation protection design basis for the Universal Storage System vertical concrete cask is derived from 10 CFR 72 and the applicable ALARA guidelines. The design basis surface dose rates, and the calculated surface and 1 meter dose rates are:

Vertical Concrete Cask	Design Basis Surface Dose Rate (mrem/hr)	Surface Dose Rate (mrem/hr)		1 Meter Maximum Dose Rate (mrem/hr)	
		PWR	BWR	PWR	BWR
Side wall	50.0 (avg.)	37.3	22.7	25.3	15.4
Air inlet	100.0	6.8	8.5	<5.0	5.0
Air outlet	100.0	65.6	50.6	12.5	7.5
Top lid	50.0 (avg.)	26.1	19.7	13.3	8.5

The calculated dose rates at these, and at other dose points, are reported in Sections 5.1.3 and 5.4.3. The dose rates presented are for the design basis 40,000 MWD/MTU, 5-year cooled, fuel. These dose rates bound those of the higher burnup, but longer cooled, fuel described in Sections 2.1 and 2.5.

Activities associated with closing the canister, including welding of the shield and structural lids, draining, drying, backfilling and testing, may employ temporary shielding to minimize personnel dose in the performance of those tasks.

10.2.2 Design Basis for Accident Conditions

Damage to the vertical concrete cask after a design basis accident does not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ. The high energy missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by approximately 6 inches. Localized cask surface dose rates for the removal of 6 inches of concrete are estimated to be less than 250 mrem/hr for the PWR and BWR configurations.

A hypothetical accident event, tip-over of the vertical concrete, is considered in Section 11.2.12. There is no design basis event that would result in the tip-over of the vertical concrete cask.

10.3 Estimated On-Site Collective Dose Assessment

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

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

10.3.1 Estimated Collective Dose for Loading a Single Universal Storage System

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

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

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

10.3.2 Estimated Annual Dose Due to Routine Operations

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

Routine operations are expected to include:

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

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

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

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

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

Chapter 11

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11.2.6 Fire Accident

This section evaluates the effects of a bounding condition hypothetical fire accident, although a fire accident is a very unlikely occurrence in the lifetime of the Universal Storage System. The evaluation demonstrates that for the hypothetical thermal accident (fire) condition the cask meets its storage performance requirements.

11.2.6.1 Cause of Fire

A fire may be caused by flammable material or by a transport vehicle. While it is possible that a transport vehicle could cause a fire while transferring a loaded storage cask at the ISFSI, this fire will be confined to the vehicle and will be rapidly extinguished by the persons performing the transfer operations or by the site fire crew. The maximum permissible quantity of fuel in the combined fuel tanks of the transport vehicle and prime mover is the only means by which fuel (maximum 50 gallons) would be next to a cask, and potentially at, or above, the elevation of the surface on which the cask is supported.

The fuel carried by other on-site vehicles or by other equipment used for ISFSI operations and maintenance, such as air compressors or electrical generators, is considered not to be within the proximity of a loaded cask on the ISFSI pad. Site-specific analysis of fire hazards will evaluate the specific equipment used at the ISFSI and determine any additional controls required.

11.2.6.2 Detection of Fire

A fire in the vicinity of the Universal Storage System will be detected by observation of the fire or smoke.

11.2.6.3 Analysis of Fire

The vertical concrete cask with its internal contents, initially at the steady state normal storage condition, is subject to a hypothetical fire accident. The fire is due to the ignition of a flammable fluid, and operationally, the volume of flammable fluid that is permitted to be on the ISFSI pad (at, or above, the elevation of the surface on which a cask is supported and within approximately two feet of an individual cask) is limited to 50 gallons. The lowest burning rate (change of depth per unit time of flammable fluid for a pool of fluid) reported in the Edition of the Fire Protection Handbook [37] is 5 inches/hour for kerosene. The flammable liquid is assumed to cover a 15 foot

Handbook [37] is 5 inches/hour for kerosene. The flammable liquid is assumed to cover a 15-foot square area, corresponding to the center to center distance of the concrete casks less the footprint of the concrete cask, which is a 128 inch diameter circle. The depth (D) of the 50 gallons of flammable liquid is calculated as:

$$D = \frac{50\,(\text{gallons}) \times 231\,(\text{in}^3/(\text{gallon})}{15 \times 15 \times 144\,(\text{in}^2) - 3.14 \times 128^2/4\,(\text{in}^2)}$$

D = 0.6 inches

With a burning rate of 5 inches/hour, the fire would continue for 7.2 minutes. The fire accident evaluation in this section conservatively considers an 8-minute fire. The temperature of the fire is taken to be 1475°F, which is specified for the fire accident condition in 10 CFR 71.73c(3).

The fire condition is an accident condition and is initiated with the concrete cask in a normal operating steady state condition. To determine the maximum temperatures of the concrete cask components, the two-dimensional axisymmetric finite element model for the BWR configuration described in Section 4.4.1.1 is used to perform a transient analysis. However, the effective properties for the canister content for specific heat, density and thermal conductivity for the PWR are used, to conservatively maximize the thermal diffusivity, which results in higher temperatures for the canister contents during the fire accident condition.

The initial condition of the fire accident transient analysis is based on the steady state analysis results for the normal condition of storage, which corresponds to an ambient temperature of 76°F in conjunction with solar insolation (as specified in Section 4.4.1.1). The fire condition is implemented by constraining the nodes at the inlet to be 1475°F for 8 minutes (see Figure 11.2.6-1). One of the nodes at the edge of the inlet is attached to an element in the concrete region. This temperature boundary condition is applied as a stepped boundary condition. During the 8-minute fire, solar insolation is also applied to the outer surface of the concrete cask. At the end of the 8 minutes, the temperature of the nodes at the inlet is reset to the ambient temperature of 76°F. The cool down phase is continued for an additional 10.7 hours to observe the maximum canister shell temperature and the average temperature of the canister contents.

The maximum temperatures of the fuel cladding and basket are obtained by adding the maximum temperature change due to the fire transient to the maximum component temperature for the normal operational condition. The maximum component temperature are presented in Table 11.2.6-1, which shows that the component temperatures are below the allowable temperatures. The limited duration of the fire and the large thermal capacitance of the concrete cask restricted the temperatures above 244°F to a region less than 3 inches above the top surface of the air inlets. The maximum bulk concrete temperature is 138°F during and after the fire accident. This corresponds to an increase of less than 3°F compared to the bulk concrete temperature for normal condition of storage. These results confirm that the operation of the concrete cask is not adversely affected during and after the fire accident condition.

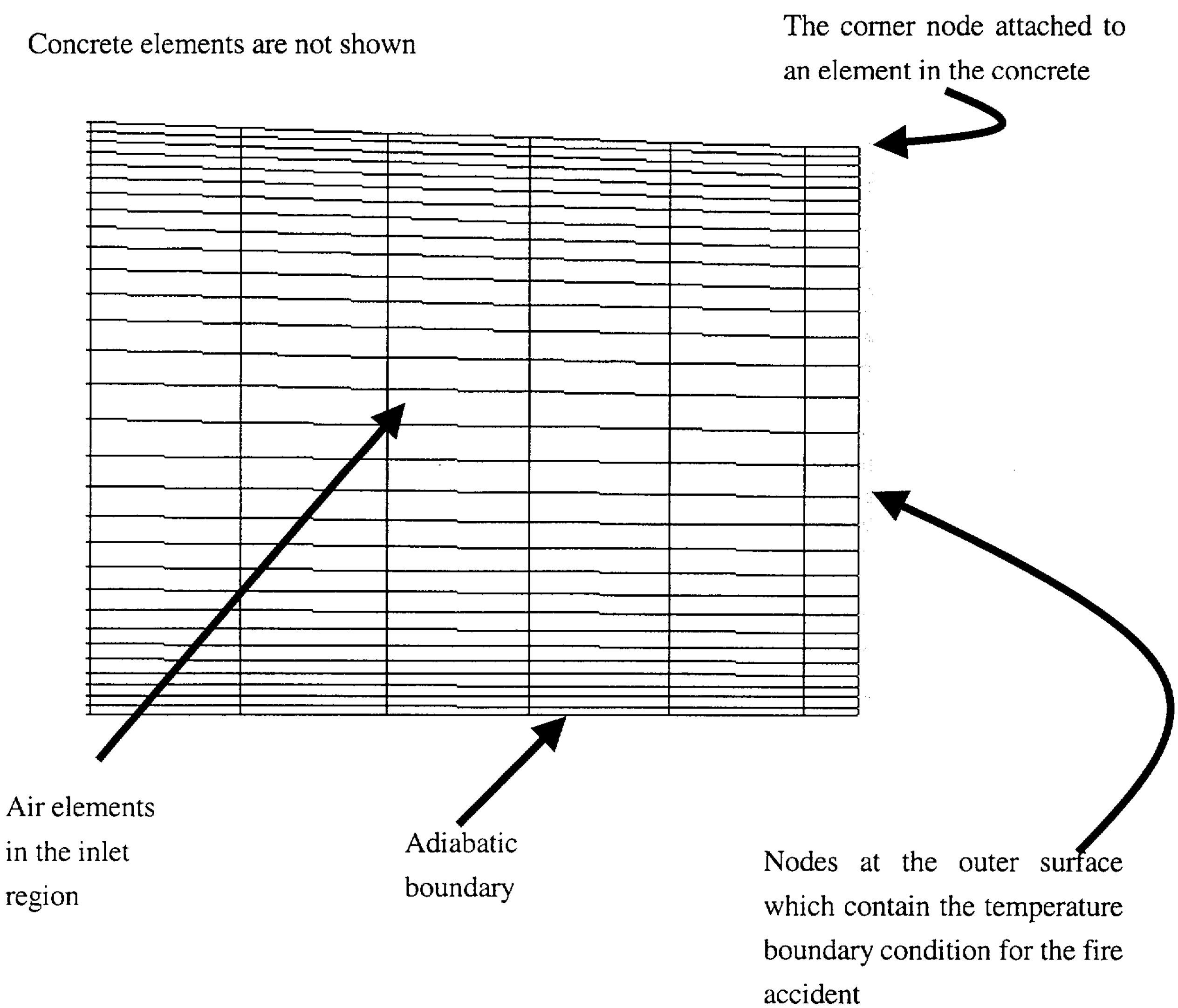
11.2.6.4 Corrective Actions

Immediately upon detection of the fire, appropriate actions should be taken by site personnel to extinguish the fire. The concrete cask should then be inspected for general deterioration of the concrete, loss of shielding (spalling of concrete), exposed reinforcing bar, and surface discoloration that could affect heat rejection. This inspection will be the basis for the determination of any repair activities necessary to return the concrete cask to its design basis configuration.

11.2.6.5 Radiological Impact

There are no significant radiological consequences for this accident. There may be local spalling of concrete during the fire event, which could lead to some minor reduction in shielding effectiveness. The principal effect would be local increases in radiation dose rate on the cask surface.

Figure 11.2.6-1 Temperature Boundary Condition Applied to the Nodes of the Inlet for the Fire Accident Condition



11.2.12 Tip-Over of Vertical Concrete Cask

Tip-over of the Vertical Concrete Cask (cask) is a non-mechanistic, hypothetical accident condition that presents a bounding case for evaluation. There are no design basis accidents that result in the tip-over of the cask.

Functionally, the cask does not suffer significant adverse consequences due to this event. The concrete cask, canister, and basket maintain design basis shielding, geometry control of contents, and contents confinement performance requirements.

Results of the evaluation show that supplemental shielding will be necessary, following the tip-over and until the cask can be righted, because the bottom ends of the concrete cask and the canister have significantly less shielding than the sides and tops of these components.

11.2.12.1 Cause of Cask Tip-Over

A tip-over of the cask is possible in an earthquake that significantly exceeds the design basis described in Section 11.2.8. No other events related to design bases are expected to result in a tip-over of the cask.

11.2.12.2 Detection of Cask Tip-Over

The tipped-over configuration of the concrete cask will be obvious during site inspection following the initiating event.

11.2.12.3 Analysis of Cask Tip-Over

For a tip-over event to occur, the center of gravity of the concrete cask and loaded canister must be displaced beyond its outer radius, i.e., the point of rotation. When the center of gravity passes beyond the point of rotation, the potential energy of the cask and canister is converted to kinetic energy as the cask and canister rotate toward a horizontal orientation on the ISFSI pad. The subsequent motion of the cask is governed by the structural characteristics of the cask, the ISFSI pad and the underlying soil.

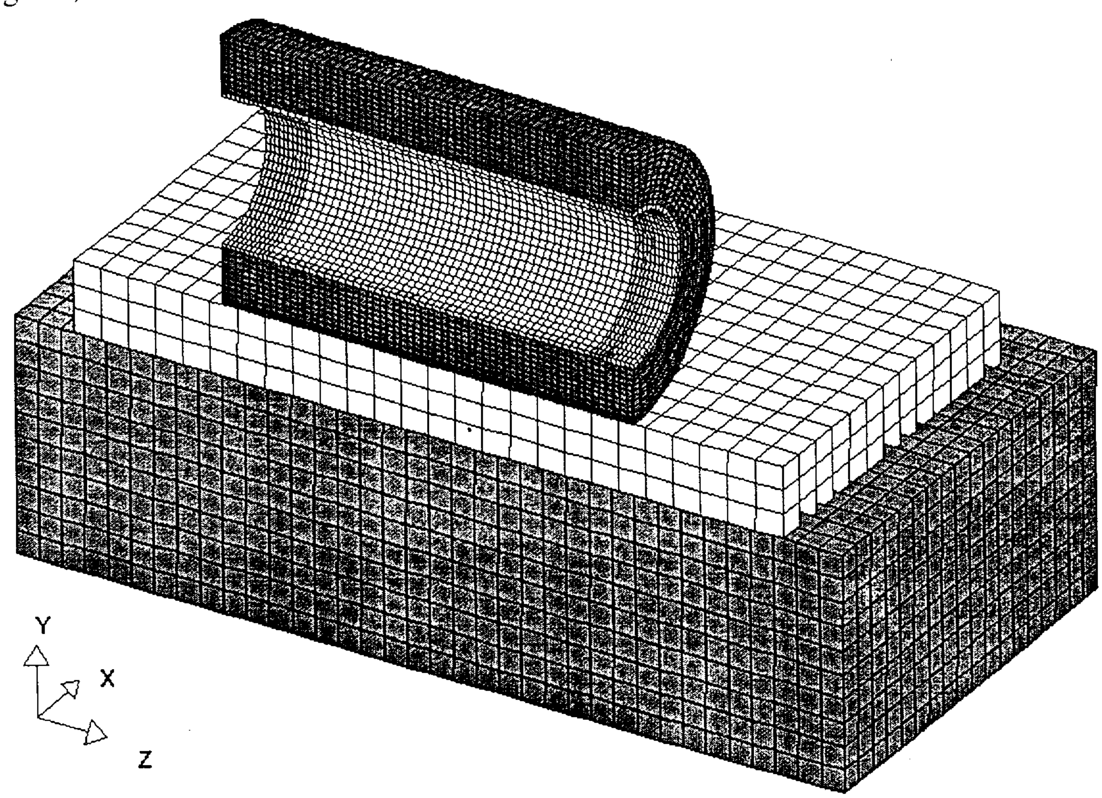
The objective of the evaluation of the response of the concrete cask in the tip-over event is to determine the maximum acceleration to be used in the structural evaluation of the loaded canister and basket (Section 11.2.12.4). The methodology to determine the concrete cask response follows the methodology contained in NUREG/CR-6608, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads" [38]. The LS-DYNA program is used in the evaluation. The validation of the analysis methodology is shown in Section 11.2.12.3.3.

The parameters of the ISFSI pad and foundation are:

Concrete thickness	36 inches maximum
Pad subsoil thickness	10 feet minimum
Specified concrete compressive strength	≤ 4,000 psi at 28 days
Concrete dry density (ρ)	$125 \le \rho \le 145$ lbs/ft^3
Soil in place density (ρ)	$100 \le \rho \le 145$ lbs/ft^3
Soil Modulus of Elasticity	≤ 60,000 psi (PWR) or ≤ 30,000 psi (BWR)

11.2.12.3.1 Analysis of Cask Tip-Over for PWR Configurations

The finite element model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



$$f_c = 805819 \frac{\lambda^2}{L^2}$$
$$= 284 \text{ Hz (PWR Class 1)}$$
$$= 261 \text{ Hz (PWR Class 2)}$$
$$= 244 \text{ Hz (PWR Class 3)}$$

Area of steel liner = $\pi \{(39.75)^2 - (37.25)^2\} = 604.8 \text{ in}^2$

Moment of inertia of steel liner = $\frac{\pi}{4}\{(39.75)^4 - (37.25)^4\} = 448{,}673 \text{ in}^4$

$$f_s = 861068 \frac{\lambda^2}{L^2}$$
$$= 303 \text{ Hz (PWR Class 1)}$$
$$= 279 \text{ Hz (PWR Class 2)}$$
$$= 260 \text{ Hz (PWR Class 3)}$$

Since the concrete cask is short compared to its diameter, the contribution of the flexibility due to shear is also incorporated. This is accomplished by using Dunkerley's formula (Blevins). The system frequency is:

$$\frac{1}{f^2} = \frac{1}{f_c^2} + \frac{1}{f_s^2}$$

Thus, the system frequencies are 207 Hz (PWR Class 1), 191 Hz (PWR Class 2), and 178 Hz (PWR Class 3). A cut-off frequency of 210 Hz (PWR Class 1), 190 Hz (PWR Class 2), and 180 Hz (PWR Class 3) is conservatively applied to filter the analysis results and measure the peak accelerations.

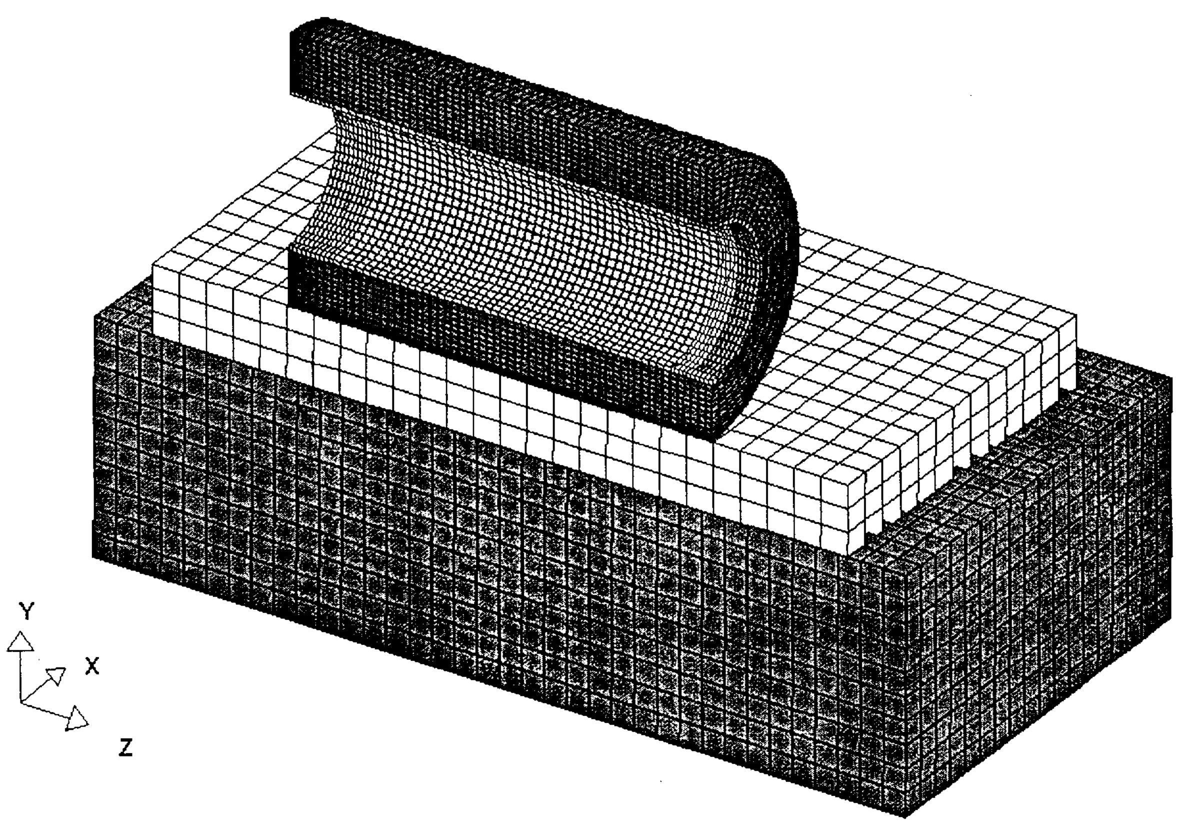
Results of the Transient Analysis

The accelerations at key locations of the concrete cask liner, which are required in the evaluation of the loaded canister/basket model (Section 11.2.12.4) are:

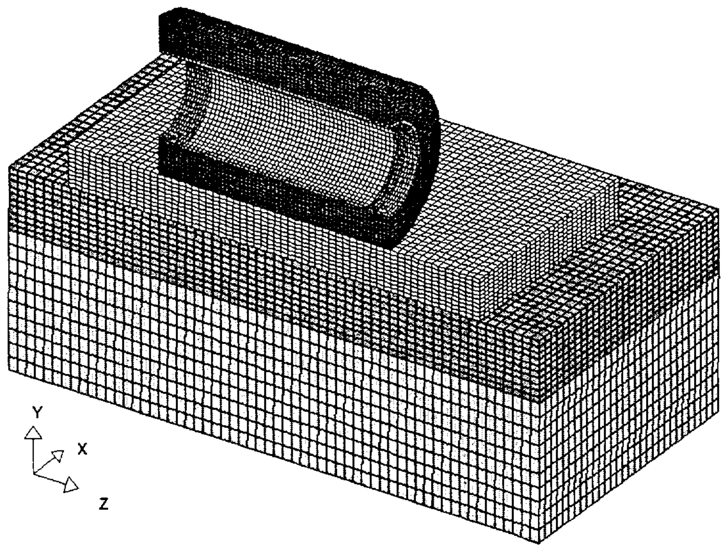
Location on Component	Position Measured from the Bottom of the Concrete Cask (inches)			Acceleration (g)		
	PWR Class 1	PWR Class 2	PWR Class 3	PWR Class 1	PWR Class 2	PWR Class 3
Top support disk	176.7	185.2	196.3	29.1	35.4	34.3
Top of the canister structural lid	197.9	207.0	214.6	31.8	35.1	36.9

11.2.12.3.2 Analysis of Cask Tip-Over for BWR Configurations

The BWR finite element model is similar to that for the PWR configuration. The concrete pad in this model corresponds to a pad 30-feet by 30-feet and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 35-feet by 35-feet in area and 10-feet thick.



The model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



Concrete Pad Properties

Vertical concrete cask tip-over analyses are performed for ISFSI pad concrete compressive strengths of 3,000 and 4,000 psi. The Poisson's Ratio (ν_c) is 0.22. The concrete dry density is considered to be between 135 pcf and 145 pcf. To account for the weight of reinforcing bar in the pad, three values of Density (ρ) are used in the model:

ρ (lbs/ft^3)	E_c (psi)	K_c (psi)
140	2.994×10^6	1.782×10^6
145	3.156×10^6	1.879×10^6
152	3.387×10^6	2.016×10^6

The corresponding values of Modulus of Elasticity (E_c) and Bulk Modulus (K_c) are also provided, where:

Modulus of Elasticity (E_c) $= 33\rho_c^{1.5}\sqrt{f'_c}$ (ACI 318-95)

Bulk Modulus (K_c) $= \dfrac{E_c}{3(1-2\nu_c)}$ (Blevins [19])

Soil Properties

The soil properties used in the model are based on three soil sets. The vertical concrete cask tip-over analyses are performed for three different combinations of soil densities: (1) 4.5-foot thick upper layer density of 135 pcf (Modulus of Elasticity, E = 162,070 psi), with a 10-foot thick lower layer density of 127 pcf (E = 31,900 psi); (2) 4.5-foot thick upper layer density of 130 pcf, with a 10-foot thick lower layer density of 127 pcf; and (3) 15-foot depth with density of 145 pcf (E $\leq$ 60,000 psi). The Poisson's Ratio (ν_s) of the soil is 0.45.

Summary of Design Basis ISFSI Pad Parameters

The ISFSI pads and foundation shall include the following characteristics as applicable to the end drop and tip-over analyses:

Concrete thickness	36 inches maximum
Pad subsoil thickness	15 feet minimum
Specified concrete compressive strength	$\leq$ 4,000 psi at 28 days
Soil in place density (ρ)	$\rho \leq 145$ lbs/ft^3 (upper layer)
Concrete dry density (ρ)	$135 \leq \rho \leq 145$ lbs/ft^3
Soil Modulus of Elasticity	$\leq$ 60,000 psi

The concrete pad maximum thickness excludes the ISFSI pad footer. The compressive strength of the concrete is determined in accordance with Section 5.6 of ACI-318 with concrete acceptance in accordance with the same section. Steel reinforcement is used in the pad and footer. The soil modulus of elasticity is determined according to the test method described in ASTM D4719.

Vertical Concrete Cask Properties

The material properties used in the model for the Vertical Concrete Cask are the same as the properties used in the PWR models in Section 11.2.12.3. The tip-over impact is simulated by applying an initial angular velocity of 1.485 rad/sec (PWR Class 1) and 1.483 rad/sec (PWR Class 2), respectively, to the entire cask. The angular velocity values are determined by the method used in Section 11.2.12 based on the weight of the loaded concrete cask with Maine Yankee fuel (285,513 pounds and 297,509 pounds for PWR Class 1 and PWR Class 2, respectively).

A cut-off frequency of 210 Hz (PWR Class 1) and 190 Hz (PWR Class 2) is applied to filter the analysis results from the LS-DYNA models and determine the peak accelerations. The resulting calculated accelerations on the canister at the location of the top support disk and of the top of the structural lid are tabulated for all of the analysis cases that were run. The maximum accelerations at the two key locations on the canister for the PWR Class 1 and Class 2 configurations are:

Component Location	Position Measured from the Bottom of the Concrete Cask (inches)		Acceleration (g)	
	Class 1	Class 2	Class 1	Class 2
Top Support Disk	176.7	185.2	32.3	34.2
Top of the Canister Structural Lid	197.9	207.0	35.3	37.6

The impact accelerations for the vertical concrete cask tip-over on the Maine Yankee ISFSI pad site are observed to be slightly higher than those reported in Section 11.2.12.3.1 for the design-basis ISFSI pad. Therefore, peak accelerations are calculated for the top support disk and are evaluated with respect to the analysis presented in Section 11.2.12.4.1.

To determine the effect of the rapid application of the inertia loading for the support disk, a dynamic load factor (DLF) is computed using the method presented in Section 11.2.12.4. The DLF is computed to be 1.07 and 1.02 for PWR Class 1 and Class 2, respectively. Applying the DLFs to the 32.3g and 35.4g results in peak accelerations of 34.6g and 36.1g for the top support disk PWR Class 1 and Class 2, respectively. The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid. Additional sensitivity evaluations considering varying values of the ISFSI concrete pad density have been performed. The results of those evaluations demonstrate that the maximum acceleration for the canister and basket are below 40g. Therefore, the maximum acceleration for the canister and basket for the cask tipover accident on the Maine Yankee site ISFSI pad is bounded by the 40g used in Section 11.2.12.4.1 (analysis of canister and basket for PWR configurations for tip-over event).

11.2.15.1.2 Parametric Study of Support Disk Evaluation for Maine Yankee Consolidated Fuel

A parametric study is performed to show that the PWR basket loaded with a Maine Yankee consolidated fuel lattice is bounded by the PWR basket design basis loading for a side impact condition. Only one consolidated fuel lattice, in a Maine Yankee Fuel Can, will be loaded in any single Transportable Storage Canister. However, Maine Yankee Fuel Cans holding other intact or damaged fuel can be loaded in the other three corner positions of the basket. (Maine Yankee Fuel

Cans may be loaded only in the four corner positions of the basket. See Figure 11.2.15.1.2-2 for corner positions. Therefore, the bounding case for Maine Yankee is the basket configuration with twenty (20) Maine Yankee fuel assemblies, three (3) fuel cans containing spent fuel, and one (1) fuel can containing consolidated fuel.

A two-dimensional ANSYS model is employed for the parametric study as shown in Figure 11.2.15.1.2-1. The load from a PWR fuel assembly is modeled as a pressure load at the inner surface of each support disk slot opening. The design basis fuel pressure loading (1g) is 12.26 psi. Based on the same design parameters (slot size = 9.272 in., disk thickness = 0.5 inch, and the number of disks = 30), the pressure load corresponding to a Maine Yankee standard CE 14×14 fuel assembly is 10.3 psi. The pressure load is 11.3 psi for a Maine Yankee fuel can holding an intact or damaged fuel assembly. For a Maine Yankee fuel can holding consolidated fuel the pressure load is 17.0 psi.

This study considers a 60g side impact condition for four different basket orientations: 0°, 18.22°, 26.28° and 45°, as shown in Figure 11.2.15.1.2-2. The 60g bounds the g-load for the PWR support disks (40g) due to the Vertical Concrete Cask tip-over accident as shown in Section 11.2.12.

A total of five cases are considered in the study. Inertial loads are applied to the support disk in all cases. The base case considers that all 24 fuel positions hold design basis PWR fuel assemblies. The other four cases (Cases 1 through 4) represent four possible load combinations for the placement of four Maine Yankee fuel cans in the corner positions, one of which holds consolidated fuel. The remaining twenty basket positions hold Maine Yankee standard 14×14 fuel assemblies. The basket loading positions are shown in Figure 11.2.15.1.2-2. The load combinations evaluated in the four Maine Yankee fuel can loading cases are:

Case	Basket Position 1	Basket Position 2	Basket Position 3	Basket Position 4
1	Consolidated	Damaged	Damaged	Damaged
2	Damaged	Consolidated	Damaged	Damaged
3	Damaged	Damaged	Damaged	Consolidated
4	Damaged	Damaged	Consolidated	Damaged

Table 11.2.15.1.2-1 provides a parametric comparison between the Base Case and the four cases evaluated, based on the maximum sectional stress in the support disk. As shown in the table, the maximum stress in the PWR basket support disk loaded with 20 standard fuel assemblies and four Maine Yankee fuel cans, including one holding consolidated fuel, is bounded by that for the support disk loaded with the design basis PWR fuel.

11.2.15.1.5 Buckling Evaluation for Maine Yankee High Burnup Fuel Rods

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

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

80 Micron Oxide Layer Thickness Evaluation

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

The buckling mode for the fuel rod is governed by the boundary conditions. For this configuration, the grids provide a lateral support, but no vertical support. The only vertical restraint is considered to be at the point of contact of the fuel rod and the base of the assembly. The weight of the fuel rod pellets and cladding is assumed to be uniformly distributed along the length of the fuel rod. In the end drop, this results in the maximum compressive load occurring at the base of the fuel rod. The first buckling mode shape corresponding to these conditions is computed as shown in Figure 11.2.15.1.5-2.

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

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

where:

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

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

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

where:

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

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

This DLF is applied to the end drop acceleration of 60g, which is the bounding load to potentially result in the buckling of the fuel rod. The product of 60g × DLF (= 14.6g) 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 ($\beta = 8.44$).

120 Micron Oxide Layer Thickness Evaluation

The buckling calculation used the same model employed for the mode shape calculation. The load that would potentially buckle the fuel rod in the end drop is due to the deceleration of the rod. This loading was implemented by applying a 1g acceleration in the direction that would result in compressive loading of the fuel rod. The acceleration required to buckle the fuel rod is computed to be 37.3g.

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

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

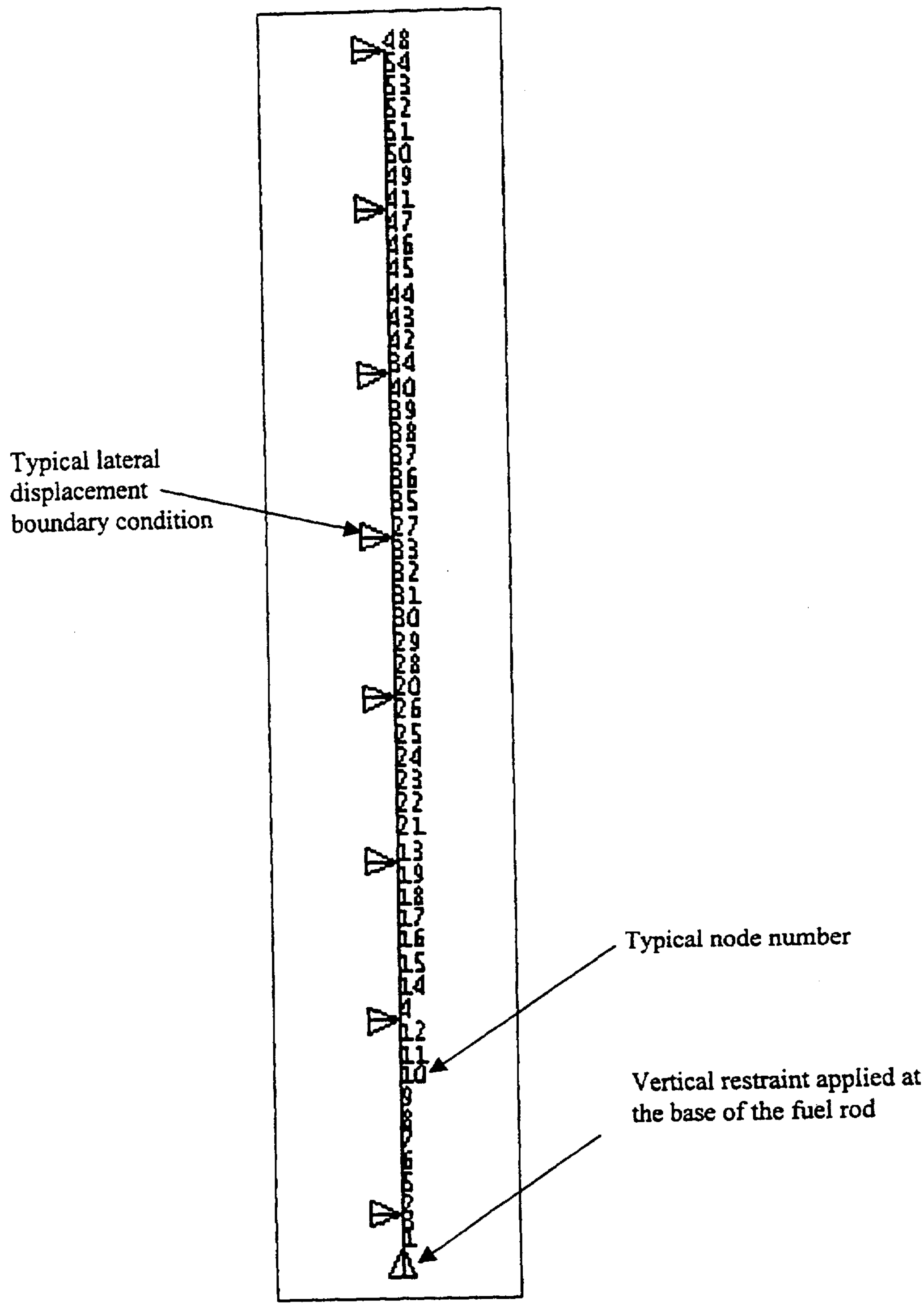
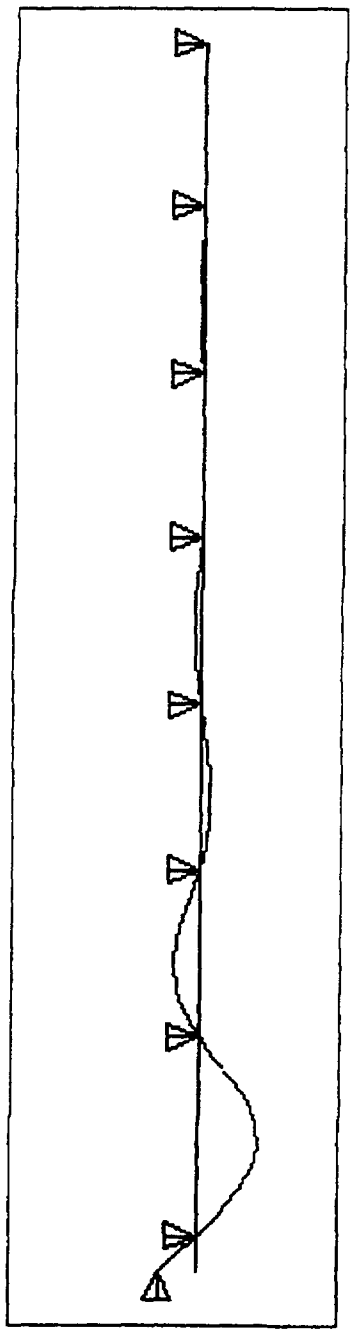
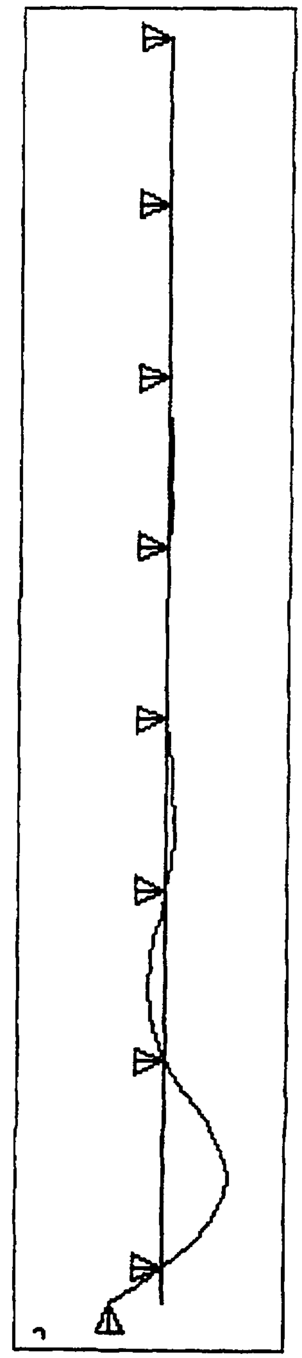


Figure 11.2.15.1.5-2 Mode Shape and First Buckling Shape for the Maine Yankee Fuel Rod

First Lateral Dynamic
Mode Shape at 25.9 Hz



First Buckling
Shape at 37.9g



11.2.15.1.6 Buckling Evaluation for High Burnup Fuel with Mechanical Damage

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

End Drop Evaluation

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

The case of the missing grid is evaluated using the methodology presented in Section 11.2.15.1.5 for the fuel assembly with 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 $0.072 \times 60 = 4.3$ g. Using the same method to compute the acceleration at which buckling occurs, the lowest buckling acceleration is 14.4 g, which is significantly greater than 4.3 g. Therefore, the fuel rod does not buckle during an end drop. Figures 11.2.16-1 and 11.2.16-2 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:

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

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

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

$$(17\text{-}1) \times (0.82\text{-}0.43) = 6.18\ \text{in.}$$

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

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

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

The cladding bending stress is given by the equation:

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

Inserting the equation for 'g':

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

where:

$c = 0.217$ inch distance from center of fuel rod to extreme outer fiber

$L = 60$ inches (the unsupported fuel rod length)

$\Delta = 6.18$ inches (the maximum deflection)

The bending stress in the fuel rod is:

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

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

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

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

where:

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

The total stress is:

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

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

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

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

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

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

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

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

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

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

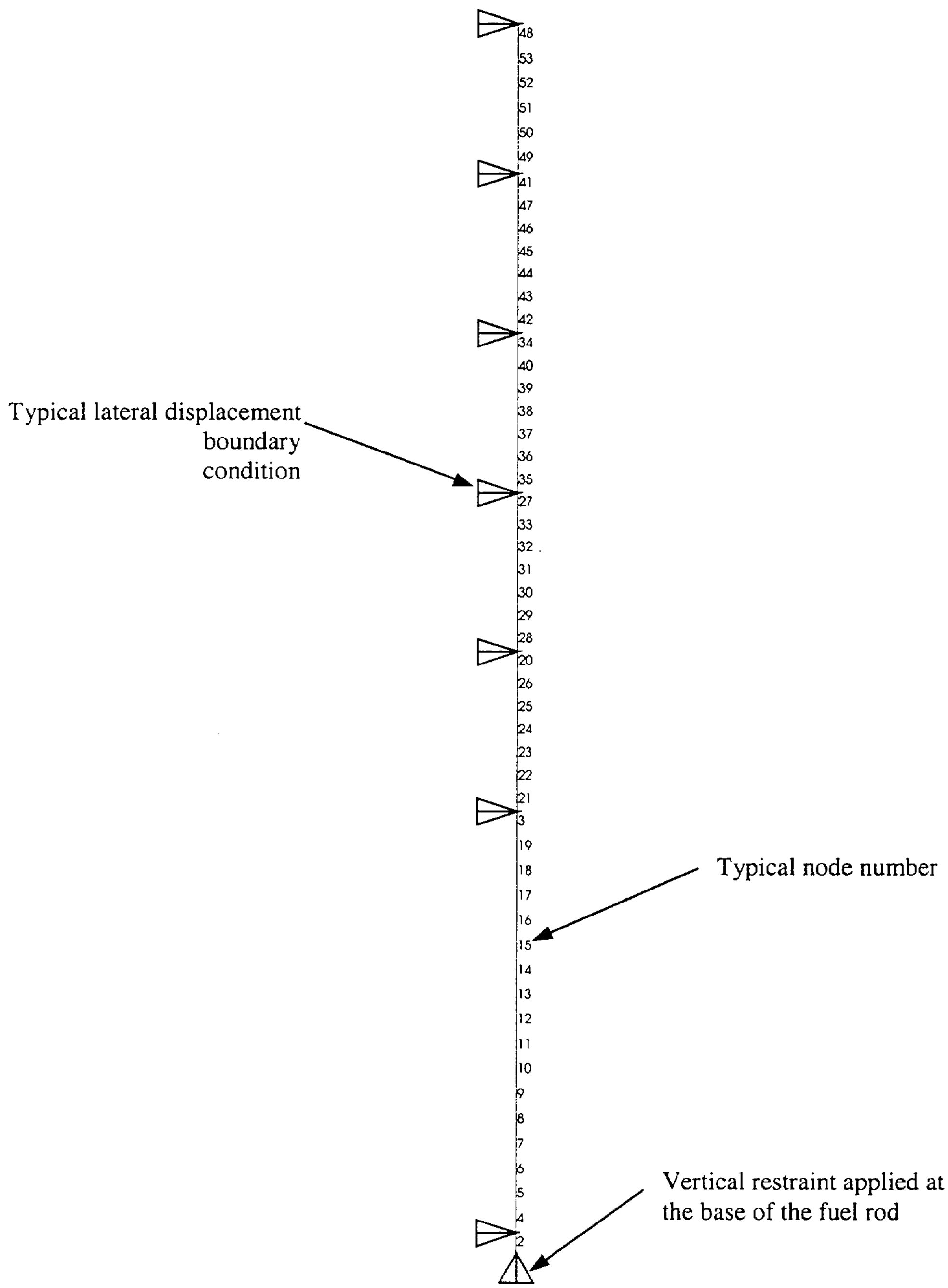
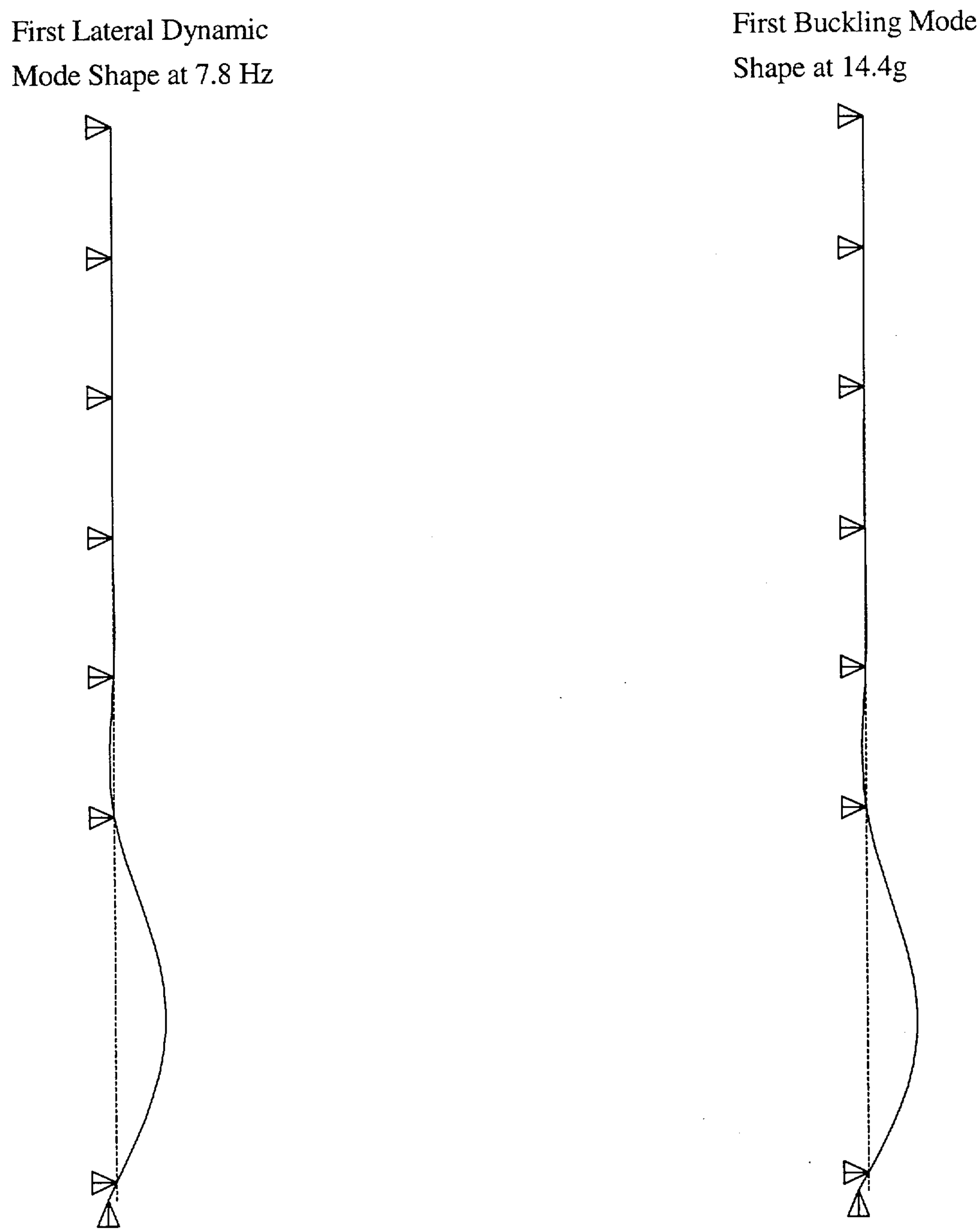


Figure 11.2.15.1.6-2 Modal Shape and First Buckling Mode Shape for a Fuel Rod with a Missing Grid



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Chapter 12

Spent fuel having a burnup from 45,000 to 50,000 MWD/MTU is assigned to peripheral locations, and based on a cladding oxide layer thickness determination, may require loading in a Maine Yankee fuel can. The interior locations must be loaded with fuel that has lower burnup and/or longer cool times in order to maintain the design basis heat load and component temperature limits for the basket and canister.

The Fuel Assembly Limits for the Maine Yankee SITE SPECIFIC FUEL are shown in Table 12B2-7. Part A of the table lists the STANDARD, INTACT FUEL ASSEMBLY and SITE SPECIFIC FUEL that does not require preferential loading except as required by Section B 2.1.2 to assure that short-term fuel cladding temperature limits are not exceeded.

Part B of the table lists the SITE SPECIFIC FUEL configurations that require preferential loading due to the criticality, shielding or thermal evaluation. The loading pattern for Maine Yankee SITE SPECIFIC FUEL that must be preferentially loaded is presented in Section B 2.1.3. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation. The preferential loading required by Part B must also consider the preferential loading requirements of Section B 2.1.2 for short-term cladding temperature limits.

Fuel assemblies with a Control Element Assembly (CEA) or a CEA plug inserted are loaded in a Class 2 canister and basket due to the increased length of the assembly with either of these components installed. However, these assemblies are not restricted as to loading position within the basket.

The Transportable Storage Canister loading procedures for Maine Yankee SITE SPECIFIC FUEL will indicate that the loading of a fuel configuration with removed fuel or poison rods, or a MAINE YANKEE FUEL CAN, or fuel with burnup between 45,000 MWD/MTU and 50,000 MWD/MTU, is administratively controlled in accordance with the requirements of Section B 2.1.3.

Table 12-1 NAC-UMS® System Controls and Limits

Control or Limit	Applicable Technical Specification	Condition or Item Controlled
1. Fuel Characteristics	Table 12B2-1	Type and Condition
	Table 12B2-2	Class, Dimensions and Weight for PWR
	Table 12B2-3	Class, Dimensions and Weight for BWR
	Table 12B2-4	Minimum Cooling Time for PWR Fuel
	Table 12B2-5	Minimum Cooling Time for BWR Fuel
	Table 12B2-7	Maine Yankee SITE SPECIFIC Loading
	Table 12B2-8	Minimum Cooling Time for Maine Yankee Fuel – No CEA
	Table 12B2-9	Minimum Cooling Time for Maine Yankee Fuel – With CEA
2. Canister	LCO 3.1.4	Time in Transfer Cask (fuel loading)
Fuel Loading	Table 12B2-1	Weight and Number of Assemblies
	Table 12B2-7	Maine Yankee Site Specific Fuel Loading
	Table 12B2-4	Minimum Cooling Time for PWR Fuel
	Table 12B2-5	Minimum Cooling Time for BWR Fuel
Drying	LCO 3.1.2	Vacuum Drying Pressure
Backfilling	LCO 3.1.3	Helium Backfill Pressure
Sealing	LCO 3.1.5	Helium Leak Rate
Vacuum	LCO 3.1.1	Time in Vacuum Drying
External Surface	LCO 3.2.1	Level of Contamination
Unloading	Note 1	Fuel Cooldown Requirement
3. Concrete Cask	LCO 3.2.2	Surface Dose Rates
	Note 1	Cask Spacing
	Note 2	Cask Handling Height
4. Surveillance	LCO 3.1.6	Heat Removal System
5. Transfer Cask	12B 3.4(8)	Minimum Temperature
	LCO 3.1.7	Canister Removal from the CONCRETE Cask
6. ISFSI Concrete Pad	B3.4.1(6)	Seismic Event Performance
	B3.4.2(7)	

1. Procedure and/or limits are presented in the Operating Procedures of Chapter 8.
2. Lifting height and handling restrictions are provided in Section A5.1.1 of Appendix 12A.

APPENDIX 12A

TECHNICAL SPECIFICATIONS FOR THE NAC-UMS® SYSTEM

AMENDMENT NO. 2

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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 stored as DAMAGED FUEL in a MAINE YANKEE FUEL CAN.
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 × 9 array of tubes in a lattice that has approximately the same dimensions as a standard fuel assembly. FUEL DEBRIS is stored in a MAINE YANKEE FUEL CAN.
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, DAMAGED FUEL or FUEL DEBRIS. The screens preclude the release of gross particulate from the can into the canister cavity. The MAINE YANKEE FUEL CAN may only be loaded in a Class 1 canister.

CANISTER Maximum Time in Vacuum Drying
A 3.1.1

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO time limits not met	A.1 Commence filling CANISTER with helium	2 hours
	AND	
	A.2.1 Submerge TRANSFER CASK with helium filled loaded CANISTER in spent fuel pool.	2 hours
	AND	
	A.2.2 Maintain TRANSFER CASK and CANISTER in spent fuel pool for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS
	OR	
	A.3.1 Commence supplying air to the TRANSFER CASK annulus fill/drain lines at a rate of 375 CFM and a maximum temperature of 75°F	2 hours
	AND	
	A.3.2 Maintain airflow for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Monitor elapsed time from completion of CANISTER draining operations until start of helium backfill	Once after completion of CANISTER draining AND As required to meet time limit.
SR 3.1.1.2	Monitor elapsed time from the end of in-pool cooling or of forced-air cooling until restart of helium backfill	Once at end of in-pool cooling or of forced-air cooling AND As required to meet time limit.

A 3.1 NAC-UMS® SYSTEM Integrity

A 3.1.2 CANISTER Vacuum Drying Pressure

LCO 3.1.2 The CANISTER vacuum drying pressure shall be less than or equal to 3 mm of mercury. Pressure shall be held for not less than 30 minutes.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER vacuum drying pressure limit not met	A.1 Establish CANISTER cavity vacuum drying pressure within limit	25 days
B. Required Action and associated Completion Time not met	B.1 Remove all fuel assemblies from the NAC-UMS® SYSTEM	5 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify CANISTER cavity vacuum drying pressure is within limit	Prior to TRANSPORT OPERATIONS.

A 3.1 NAC-UMS® SYSTEM Integrity

A 3.1.3 CANISTER Helium Backfill Pressure

LCO 3.1.3 The CANISTER helium backfill pressure shall be 0 (+1, -0) psig.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

---NOTE---

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER helium backfill pressure limit not met	A.1 Establish CANISTER helium backfill pressure within limit	25 days
B. Required Action and associated Completion Time not met	B.1 Remove all fuel assemblies from the NAC-UMS® SYSTEM	5 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify CANISTER helium backfill pressure is within limit	Prior to TRANSPORT OPERATIONS.

A 3.1 NAC-UMS® SYSTEM Integrity

A 3.1.4 CANISTER Maximum Time in TRANSFER CASK

LCO 3.1.4 The following limits for CANISTER time in TRANSFER CASK shall be met, as appropriate:

1. The time duration from completion of backfilling the CANISTER with helium through completion of the CANISTER transfer operation from the TRANSFER CASK to the CONCRETE CASK shall not exceed 24 hours for the design basis BWR heat load of 23 kW or the time shown below for a specific PWR heat load:

Total PWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	16
$17.6 < L \leq 20$	20
$14 < L \leq 17.6$	48
$L \leq 14$	Not Limited

2. The time duration from completion of in-pool or external forced air cooling of the CANISTER through completion of the CANISTER transfer operation from the TRANSFER CASK to the CONCRETE CASK shall not exceed 15 hours for the BWR configuration or the time shown below for a specific PWR heat load:

Total PWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	6
$17.6 < L \leq 20$	16
$14 < L \leq 17.6$	20
$L \leq 14$	Not Limited

The LCO time limits are also applicable if SR 3.1.5.1 was not met during vacuum drying operations.

APPLICABILITY: During LOADING OPERATIONS

(continued)

A 3.1 NAC-UMS® SYSTEM

A 3.1.6 <u>CONCRETE CASK Heat Removal System</u>

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

APPLICABILITY: During STORAGE OPERATIONS

ACTIONS

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

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

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

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS

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

A 3.2 NAC-UMS® SYSTEM Radiation Protection

A 3.2.2 CONCRETE CASK Average Surface Dose Rates

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

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

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

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

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

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

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

(continued)

A 5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

A 5.1 Training Program

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

A 5.2 Pre-Operational Testing and Training Exercises

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

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

(continued)

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

k. CONCRETE CASK shield plug and lid installation

l. Transport of the CONCRETE CASK to the ISFSI

m. CANISTER unloading, including reflooding and weld removal or cutting

n. CANISTER removal from the CONCRETE CASK

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

A 5.3 Special Requirements for the First System Placed in Service

The heat transfer characteristics and performance of the NAC-UMS® SYSTEM will be recorded by air inlet and outlet temperature measurements of the first system placed in service with a heat load equal to or greater than 10 kW. A letter report summarizing the results of the measurements will be submitted to the NRC in accordance with 10 CFR 72.4 within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® SYSTEM heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® SYSTEMs that are subsequently loaded, provided that the performance of the first system placed in service with a heat load $\geq$ 10 kW, is demonstrated by the comparison of the calculated and measured temperatures.

A 5.4 Surveillance After an Off-Normal, Accident, or Natural Phenomena Event

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

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

(continued)

A 5.5 Radioactive Effluent Control Program

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

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

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

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

A 5.6 NAC-UMS® SYSTEM Transport Evaluation Program

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

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

a. The lift height above the transport surface prescribed in Section B3.4.1(6) of Appendix B to Certificate of Compliance (CoC) No. 1015 shall not exceed the limits in Table 12A5-1. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than those prescribed for the reference pad surface which forms the basis for the values cited in Section B3.4.1(6) of Appendix B to CoC No. 1015.

(continued)

A 5.6 NAC-UMS® SYSTEM Transport Evaluation Program (continued)

b. For site specific transport conditions which are not bounded by the surface characteristics in Section B3.4.1 of Appendix B to CoC No. 1015, the program may evaluate the site specific conditions to ensure that the impact loading due to design basis drop events does not exceed 60g. This alternative analysis shall be commensurate with the drop analyses described in the Safety Analysis Report for the NAC-UMS® SYSTEM. The program shall ensure that these alternative analyses are documented and controlled.

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

A 5.7 Verification of Oxide Layer Thickness on High Burnup Fuel

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

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

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

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

APPENDIX 12B

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

AMENDMENT NO. 2

Appendix 12B
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B 2.0 APPROVED CONTENTS

B 2.1 Fuel Specifications and Loading Conditions

The NAC-UMS® System is designed to provide passive dry storage of canistered PWR and BWR spent fuel. The system requires few operating controls. The principal controls and limits for the NAC-UMS® SYSTEM are satisfied by the selection of fuel for storage that meets the Approved Contents presented in this section and in Tables 12B2-1 through 12B2-5 for the standard NAC-UMS® SYSTEM design basis spent fuels.

This section also permits the loading of fuel assemblies that are unique to specific reactor sites. SITE SPECIFIC FUEL assembly configurations are either shown to be bounded by the analysis of the standard NAC-UMS® System design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

The separate specific evaluation may establish different limits, which are maintained by administrative controls for preferential loading. The preferential loading controls allow the loading of fuel configurations that may have higher burnup, additional hardware material or unique configurations as compared to the standard NAC-UMS® System design basis spent fuels.

Unless specifically excepted, SITE SPECIFIC FUEL must meet all of the controls and limits specified for the NAC-UMS® System.

If any Fuel Specification or Loading Conditions of this section are violated, the following actions shall be completed:

- The affected fuel assemblies shall be placed in a safe condition.
- Within 24 hours, notify the NRC Operations Center.
- Within 30 days, submit a special report that describes the cause of the violation and actions taken to restore or demonstrate compliance and prevent recurrence.

(continued)

B 2.1.1 Fuel to be Stored in the NAC-UMS® SYSTEM

INTACT FUEL ASSEMBLIES meeting the limits specified in Tables 12B2-1 through 12B2-5 may be stored in the NAC-UMS® SYSTEM.

B 2.1.2 Preferential Fuel Loading

The normal temperature distribution in the loaded TRANSPORTABLE STORAGE CANISTER results in the basket having the highest temperature at its center and lowest temperature at the outer edge. Considering this temperature distribution, spent fuel with the shortest cooling time (and, therefore, having a higher allowable cladding temperature) is placed in the center of the basket. Fuel with the longest cooling time (and, therefore, having a lower allowable cladding temperature) is placed in the periphery of the basket.

Using a similar argument, fuel assemblies with cooling times between the highest and lowest cooling times of the designated fuel, are placed in intermediate fuel positions.

Loading of the fuel assemblies designated for a given TRANSPORTABLE STORAGE CANISTER must be administratively controlled to ensure that the dry storage fuel cladding temperature limits are not exceeded for any fuel assembly, unless all of the designated fuel assemblies have a cooling time of 7 years of more.

CANISTERS containing fuel assemblies, all of which have a cooling time of 7 years, or more, do not require preferential loading, because analyses have shown that the fuel cladding temperature limits will always be met for those CANISTERS.

CANISTERS containing fuel assemblies with cooling times from 5 to 7 years must be preferentially loaded based on cooling time. By controlling the placement of the fuel assemblies with the shortest cooling time (thermally hottest), preferential loading ensures that the allowable fuel cladding temperature for a given fuel assembly is not exceeded. The preferential loading of fuel into the CANISTER based on cooling time is described as follows.

(continued)

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

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

(continued)

B 2.1.3 Maine Yankee SITE SPECIFIC FUEL Preferential Loading

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

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

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

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

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

(continued)

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

Figure 12B2-1 PWR Basket Fuel Loading Positions

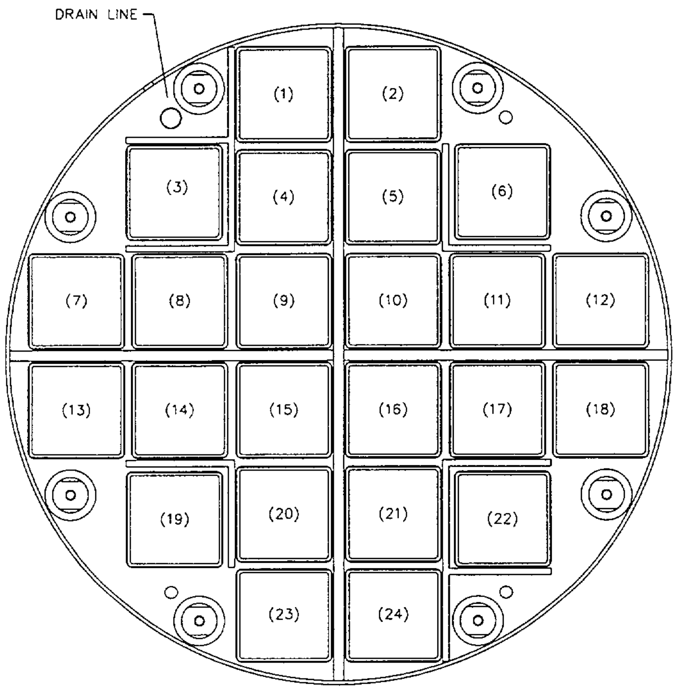

Figure 12B2-2 BWR Basket Fuel Loading Positions

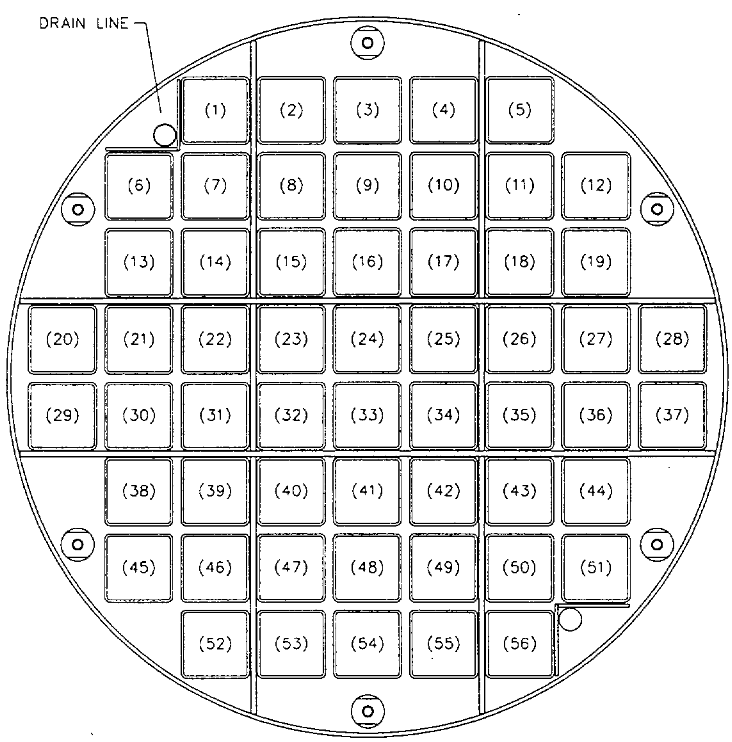

Table 12B2-1
Fuel Assembly Limits

I. NAC-UMS® CANISTER: PWR FUEL

A. Allowable Contents

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

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

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

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

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

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

F. Unenriched fuel assemblies are not authorized for loading.

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

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

Table 12B2-1
Fuel Assembly Limits (continued)

II. NAC-UMS® CANISTER: BWR FUEL

A. Allowable Contents

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

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

Table 12B2-1

Fuel Assembly Limits (continued)

B. Quantity per CANISTER: Up to 56 BWR INTACT FUEL ASSEMBLIES

C. BWR INTACT FUEL ASSEMBLIES can be unchanneled or channeled with Zircaloy channels.

D. BWR INTACT FUEL ASSEMBLIES with stainless steel channels shall not be loaded.

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

F. Unenriched fuel assemblies are not authorized for loading.

G. The minimum length of the BWR INTACT FUEL ASSEMBLY internal structure and bottom end fitting and/or spacers shall ensure that the minimum distance to the fuel region from the base of the CANISTER is 6.2 inches.

Table 12B2-2 PWR Fuel Assembly Characteristics

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

Note: Parameters shown are nominal pre-irradiation values.

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

Table 12B2-3 BWR Fuel Assembly Characteristics

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

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

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

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

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

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

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]			30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]		
	7×7	8×8	9×9	7×7	8×8	9×9
$1.9 \leq E < 2.1$	5	5	5	8	7	7
$2.1 \leq E < 2.3$	5	5	5	6	6	6
$2.3 \leq E < 2.5$	5	5	5	5	5	5
$2.5 \leq E < 2.7$	5	5	5	5	5	5
$2.7 \leq E < 2.9$	5	5	5	5	5	5
$2.9 \leq E < 3.1$	5	5	5	5	5	5
$3.1 \leq E < 3.3$	5	5	5	5	5	5
$3.3 \leq E < 3.5$	5	5	5	5	5	5
$3.5 \leq E < 3.7$	5	5	5	5	5	5
$3.7 \leq E \leq 4.0$	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]		
	7×7	8×8	9×9	7×7	8×8	9×9
$1.9 \leq E < 2.1$	16	14	15	26	24	25
$2.1 \leq E < 2.3$	13	12	12	23	21	22
$2.3 \leq E < 2.5$	9	8	8	18	16	17
$2.5 \leq E < 2.7$	8	7	7	15	14	14
$2.7 \leq E < 2.9$	7	6	6	13	11	12
$2.9 \leq E < 3.1$	6	6	6	11	10	10
$3.1 \leq E < 3.3$	6	5	6	9	8	9
$3.3 \leq E < 3.5$	6	5	6	8	7	8
$3.5 \leq E < 3.7$	6	5	6	7	7	7
$3.7 \leq E \leq 4.0$	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**
Total Number of Fuel Assemblies[3]	1,434	----
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 Start-up Source	4	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 these site specific spent fuel configurations and standard fuel assemblies. Standard fuel assemblies may be loaded in any canister position.
4. Basket corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1. Corner positions are also periphery positions.
5. Basket 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 × 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. CEAs or CEA Plugs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or CEA Plug inserted must be loaded in a Class 2 CANISTER and cannot be loaded in a Class 1 CANISTER. Fuel assemblies without an inserted CEA or CEA Plug, including those with inserted ICI Thimbles, must 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 meeting the requirements of Section A 5.7(1).
  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) start-up source may be loaded in any fuel assembly or any CANISTER.
5. PWR INTACT FUEL ASSEMBLIES with CEA ends (finger tips) and/or ICI segment inserted in corner guide tube positions. The assembly must also have a CEA 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.

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

e) CONSOLIDATED FUEL lattice structure with a 17 × 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight ≤ 2,100 pounds. A CONSOLIDATED FUEL lattice cannot have an inserted CEA or other non-fuel component. Only one CONSOLIDATED FUEL lattice may be stored in any CANISTER.

f) HIGH BURNUP FUEL assemblies not meeting the criteria of Section A 5.7(1).

C. Unenriched fuel assemblies are not authorized for loading.

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

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

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

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

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

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

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

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

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

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

ACTIONS (continued)

A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonable, based on the time required to reflood the CANISTER, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK into the spent fuel pool, and remove the CANISTER shield lid in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time. A low vacuum pressure is an indication that the cavity is dry. The surveillance must be performed prior to TRANSPORT OPERATIONS. This allows sufficient time to backfill the CANISTER cavity with helium, while minimizing the time the fuel is in the CANISTER without water or the assumed inert atmosphere in the cavity.

REFERENCES

1. SAR Sections 4.4, 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.3 CANISTER Helium Backfill Pressure

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. Backfilling of the CANISTER cavity with helium promotes heat transfer from the spent fuel to the CANISTER structure and the inert atmosphere protects the fuel cladding. Providing a helium pressure equal to atmospheric pressure ensures that there will be no in-leakage of air over the life of the CANISTER, which might be harmful to the heat transfer features of the NAC-UMS® SYSTEM and harmful to the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on the ability of the NAC-UMS® SYSTEM to remove heat from the CANISTER and reject it to the

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APPLICABLE SAFETY ANALYSIS (continued)	environment. This is accomplished by removing water from the CANISTER cavity and backfilling the cavity with an inert gas. The heat-up of the CANISTER and contents will continue following backfilling with helium, but is controlled by LCO 3.1.4. The thermal analyses of the CANISTER assume that the CANISTER cavity is dried and filled with dry helium.
LCO	Backfilling the CANISTER cavity with helium at a pressure equal to atmospheric pressure ensures that there is no air in-leakage into the CANISTER, which could decrease the heat transfer properties and result in increased cladding temperatures and damage to the fuel cladding over the storage period. The helium backfill pressure specified in Table 12A3-1 was selected based on a minimum helium purity of 99.9% to ensure that the CANISTER internal pressure and heat transfer from the CANISTER to the environment are maintained consistent with the design and analysis basis of the CANISTER.
APPLICABILITY	Helium backfill is performed during LOADING OPERATIONS, before the TRANSFER CASK and CANISTER are moved to the CONCRETE CASK for transfer of the CANISTER. Therefore, the backfill pressure requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent condition entry and application of associated Required Actions. <u>A.1</u> If the backfill pressure cannot be established within limits, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which would prevent backfilling of the CANISTER cavity with helium. These actions include identification and repair of helium leak paths or replacement of the helium backfill equipment.

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ACTIONS (continued)

B.1

If the CANISTER cavity cannot be backfilled with helium to the specified pressure, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by reperforming A.1. The Completion Time is reasonable based on the time required to re-flood the CANISTER, perform cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert atmosphere and maintenance of adequate heat transfer mechanisms. Filling the CANISTER cavity with helium at a pressure within the range specified in Table 12A3-1 will ensure that there will be no air in-leakage, which could potentially damage the fuel. This pressure of helium gas is sufficient to maintain fuel cladding temperatures within acceptable levels.

Backfilling of the CANISTER cavity must be performed successfully on each CANISTER before placing it in storage. The Surveillance must be performed prior to TRANSPORT OPERATIONS. This allows sufficient time to backfill the cavity with helium, while minimizing the time the loaded CANISTER is in the TRANSFER CASK without the assumed inert atmosphere in the cavity.

REFERENCES

1. SAR Sections 4.4, 7.1 and 8.1.

LCO (continued)	Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environment at a sufficient rate to maintain fuel cladding and CANISTER component temperatures within design limits.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once a CONCRETE CASK containing a CANISTER loaded with spent fuel has been placed in storage, the heat removal system must be OPERABLE to ensure adequate heat transfer of the decay heat away from the fuel assemblies.
ACTIONS	A note has been added to ACTIONS which states that, for this LCO, separate Condition entry is allowed for each CONCRETE CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent CONCRETE CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the heat removal system has been determined to be inoperable, it must be restored to OPERABLE status within 8 hours. Eight hours is reasonable based on the accident analysis which shows that the limiting CONCRETE CASK component temperatures will not reach their temperature limits for 24 hours after a complete blockage of all inlet air ducts. <u>B.1</u> SR 3.1.6.1 is performed to document the continuing status of the operability of the CONCRETE CASK Heat Removal System. <u>B.2.1</u> Efforts must continue to restore the heat removal system to OPERABLE status by removing the air flow obstruction(s) unless optional Required Action B.2.2 is being implemented.

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ACTIONS (continued)

B.2.1 (continued)

This Required Action must be completed in 12 hours. The Completion Time reflects a conservative total time period without any cooling of 24 hours. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short-term temperature limit for more than 24 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlets and outlets immediately after the last successful Surveillance.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The long-term integrity of the stored fuel is dependent on the ability of the CONCRETE CASK to reject heat from the CANISTER to the environment. The temperature rise between ambient and the CONCRETE CASK air outlets shall be monitored to verify operability of the heat removal system. Blocked air inlets or outlets will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the CANISTER. Based on the analyses, provided the air temperature rise is less than the limits stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long-term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency of 24 hours is reasonable based on the time necessary for CONCRETE CASK components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of the blockage of the air inlets and outlets.

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REFERENCES	1.	SAR Chapter 4 and Chapter 11, Section 11.2.13.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.7 CANISTER Removal from the CONCRETE CASK

BASES

BACKGROUND

A loaded CANISTER is removed from a CONCRETE CASK using the TRANSFER CASK, so that the CANISTER may be transferred to another CONCRETE CASK or transferred to a TRANSPORT CASK for purposes of transport. The CANISTER is removed from the CONCRETE CASK using the procedure provided in Section 8.2. Once in the TRANSFER CASK, the CANISTER begins to heat up due to the decay heat of the contents and the reduced heat transfer provided by the TRANSFER CASK compared to the CONCRETE CASK.

The CANISTER time in the TRANSFER CASK is limited when forced air cooling is not used to ensure that the short-term temperature limits established in the Safety Analysis Report for the spent fuel cladding and CANISTER materials are not exceeded.

If forced air cooling is maintained, then the CANISTER time in the TRANSFER CASK is not limited, since the short-term temperature limits of the spent fuel cladding and of the CANISTER components are not exceeded.

APPLICABLE SAFETY ANALYSIS

Limiting the total time that a loaded CANISTER backfilled with helium may be in the TRANSFER CASK, prior to unloading the CANISTER from the TRANSFER CASK, ensures that the short-term temperature limits for the spent fuel cladding and CANISTER materials are not exceeded. Upon placement of the loaded CANISTER in the CONCRETE CASK or TRANSPORT CASK, the temperatures of the

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