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Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks

Oak Ridge National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing the technical specifications for spent fuel storage casks in an effort to develop standard technical specifications (STS) that define the allowable spent nuclear fuel (SNF) contents. One of the objectives of the review is to minimize the level of detail in the STS that define the acceptable fuel types. To support this initiative, this study has been performed to identify potential fuel specification parameters needed for criticality safety and radiation shielding analysis and rank their importance relative to a potential compromise of the margin of safety.

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1 INTRODUCTION

1.1 Background

Under the current licensing procedures for spent nuclear fuel (SNF) dry storage casks, vendors must identify all fuel assembly types that may potentially be stored in their casks. The acceptable cask contents are limited to the fuel specifications (i.e., dimensions or ranges of dimensions) as detailed in the Technical Specifications (TS). Each time a candidate fuel assembly type or specification falls outside the range identified in the TS, a Part 72 license amendment request must be submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. This practice results in frequent license amendments that create an unnecessary burden on the NRC and industry.

Recently, Section 72.48 in Title 10 of the Code of Federal Regulations (CFR) was revised, effective April 5, 2001, to better define the changes in the SNF storage cask design or procedures that can be made without a license amendment request. In implementing this rule change, some control of the cask contents will be shifted from the TS to the Final Safety Analysis Report (FSAR). The objective is to replace the current detailed TS with more general Standard Technical Specifications (STS) that concentrate control on those fuel parameters that are most important to maintaining safety. The remaining fuel parameters are of lesser importance and would be handled under the Section 72.48 process, which allows the licensees to change those parameters by performing additional safety analyses to update the FSAR. The licensee would notify the NRC of the FSAR updates but no review or approval by NRC would be required.

As a result of these changes, the NRC is working to develop STS that minimize the level of detail that defines the acceptable fuel types that can be loaded in a cask. To support this initiative, a study has been performed to identify and rank potential fuel specification parameters needed for criticality safety and radiation shielding and rank their importance relative to a potential compromise of the margin of safety. Minimizing the level of detail in the STS could allow new fuel types and designs that are not significantly different than the design basis assemblies to be added using a Section 72.48 change to the FSAR.

The following goals have been identified as being important to the development of STS for dry storage casks.

1. The STS should address all fuel design parameters that are important to criticality safety and radiation shielding.

2. The STS, in combination with the Section 72.48 change process, must ensure that the relevant limits for subcriticality and radiation dose are not exceeded.

3. The STS should enable allowance for additional fuel assemblies through the Section 72.48 change process, and thus reduce or eliminate the need for license amendment requests for expanding fuel specifications that do not have a significant impact on the relevant limits for subcriticality and radiation dose.

Section 2 of this report documents a study of the importance of individual fuel design parameters to maintaining subcriticality. Section 3 documents a similar parameter study relative to radiation shielding safety limits. Finally, Section 4 presents a summary of recommendations regarding which parameters should be considered of primary importance to criticality safety and radiation shielding in the STS.

2 REVIEW OF FUEL SPECIFICATIONS FOR CRITICALITY SAFETY

2.1 Review Basis

A large number of fuel assembly design parameters can potentially influence the maximum neutron multiplication factor (k_{eff}) of a spent fuel storage cask. The importance of the parameters can be assessed by determining their impact on the upper subcritical limit (USL), that calculated k_{eff} value above which loading of the cask would not be allowed. The objective of this section is to determine which fuel assembly design parameters would be included in the STS and which should be specified in the FSAR, based on the importance to criticality safety. Note that no burnup credit is assumed in this study. The USL typically includes 5% margin[1] to assure subcriticality. This margin may potentially be encroached during changes to the FSAR not reviewed by the NRC. Thus, this margin is the basis of examination for compromises to safety.

2.2 Technical Specification Parameters

The following candidate fuel specification parameters for criticality safety analyses were reviewed in the parametric study to determine the sensitivity of k_{eff} to each.

- Enrichment
- Fuel rod pitch
- Fuel pellet outer diameter (OD)
- Cladding thickness
- Cladding OD
- Guide/instrument tube thickness
- Active fuel length
- Fuel stack density

2.3 Analysis Methods

All calculations were performed using the SCALE code system.[2] One-dimensional (1-D) radial CSAS1X/XSDRNPM infinite pin cell lattices and three-dimensional (3-D) CSAS25/KENO V.a infinite fuel assembly array lattices in poisoned cask basket geometries were modeled for Westinghouse pressurized water reactor (PWR) 17 × 17, ABB-Combustion Engineering (ABB-CE) 16 × 16 PWR, and General Electric (GE) boiling water reactor (BWR) 8 × 8 fuel assemblies. The PWR basket model is an infinite radial array of storage cells based on the Holtec MPC-24 basket,[3] and the BWR basket model is based on the Holtec MPC-68 basket.[3] Both baskets use boral absorber plates for criticality control. The nominal design specifications for each of the fuel assembly types are listed in Table 1.

Table 1 Design specifications for selected fuel assembly types

Design parameter	Westinghouse 17 × 17 OFA	ABB CE 16 × 16	GE BWR 8 × 8
Number of fuel rods	264	236	62
Number of instrument tubes	1	1	0
Number of guide tubes	24	4	2
Number of burnable poison rods	0	0	0
Fuel type	UO_2	UO_2	UO_2
Nominal enrichment (wt % ^{235}U)	4.0	4.6	.0
Fuel density (g/cm^3)	10.412	10.412	10.5216
Clad type	Zircaloy	Zircaloy	Zircaloy
Fuel rod outer diameter (cm)	0.7975	0.8262	1.0414
Clad inner diameter (cm)	0.8225	0.8432	1.0642
Clad outer diameter (cm)	0.9385	0.9702	1.1176
Fuel rod pitch (cm)	1.2675	1.2852	1.6256
Guide tube data			
Inner radius (cm)	0.56130	1.143	0.6570
Outer radius (cm)	0.59686	1.2446	0.7080
Guide tube material	Zircaloy	Zircaloy	Zircaloy

2.4 Results

2.4.1 Principal Fuel Specification Parameters

The change, in k_{eff} over the range of each parameter for different fuel assembly types, has been used to determine which design parameters have the greatest impact on criticality safety. This approach assumes that the **fuel assembly type** (i.e., assemblies with the same array size, number and cladding material of fuel rods, and number and material of guide tubes) will be specified in the STS, because sensitivity of k_{eff} to the design parameters is dependent on the fuel assembly design. The cladding and guide tube materials are considered important to the definition of the fuel assembly type because of the potential impact on reactivity and neutron spectrum in switching from one type of material to another (e.g., from stainless steel to Zircaloy). Loading a fuel assembly type not specified in the STS would require a licensing amendment.

Where possible, the parameter ranges for the Westinghouse and GE assemblies were based on actual fuel dimensions.[3] For enrichment, fuel stack density, and fuel length, the parameter ranges were defined using an arbitrary percentage change from the nominal value. Because ranges in actual fuel dimensions were not available for the ABB-CE fuel, the percentage changes from nominal based on the Westinghouse fuel were applied. Note the Westinghouse 17 × 17 study covers the full parameter range for both Standard and Optimized Fuel Assembly (OFA) fuel assembly designs. Tables 2 through 4 show the calculated results and Δk values for the Westinghouse 17 × 17, ABB-CE 16 × 16, and GE 8 × 8 fuel assemblies, respectively. It is noted in the final column of each table whether the minimum or maximum value of each parameter produces the limiting k_{eff}. The most significant parameter, based on the change in k_{eff} over the entire range of parameter variation, is **enrichment** (approximately 5% Δk/wt % ^{235}U). Due to the large sensitivity to enrichment, it is recommended that maximum enrichment be required in the STS.

2.4.2 Secondary Fuel Specification Parameters

Other parameters in Tables 2 through 4 that cause changes in the calculated k_{eff} value of > 1% over the entire range of parameter variation have been flagged as significant. These parameters include **fuel rod pitch, fuel pellet OD,** and **cladding thickness**. Note that the classification of significant parameters would not change notably if the criterion were reduced from 1% to 0.5%. It is recommended that changes in these parameters would require an update to the FSAR via the Section 72.48 process. Based on the calculated results listed in Tables 2 through 4, the maximum anticipated change in k_{eff} due to changes in all these parameters should be no more than 1.5 to 2%.

Note that the assumption of no clad is worth approximately 5% Δk. The assumption of no clad carries a very high degree of conservatism. It is unlikely to be used by applicants, but it is presented for informational purposes. However, as noted above, within the anticipated range of clad thickness, the worth is ≤ 1.2%.

2.4.3 Less Significant Fuel Specification Parameters

Cladding OD (assuming constant clad thickness), **guide or instrument tube/water rod thickness, active fuel length,** and **fuel stack density** are shown to have relatively minor impact on k_{eff} (approximately 0.5% or less). The worth of the guide and instrument tubes for PWRs and water rods and channels for BWRs is very small, approximately 0.5%. For this reason, an applicant may find it desirable to eliminate them from consideration in the FSAR. Because replacing them with water in the criticality safety model is conservative, their presence may be neglected entirely in the criticality safety analysis. The additional conservatism associated with these model simplifications has a minimal impact on the USL. If the licensee does include them in the FSAR analysis, these parameters must be included in the Section 72.48 process.

Table 2 Westinghouse 17 × 17 fuel assembly parameter study results

Parameter	Minimum	Nominal	Maximum	1-D k-eff min. parameter	1-D k-eff max. parameter	3-D k-eff min. parameter[1]	3-D k-eff max. parameter[1]	% Δk 1-D	Significant	% Δk 3-D	Significant	Limiting parameter value
Enrichment	3.500	4.000	4.500	1.42800	1.47885	0.9358	0.9862	**5.09**	Yes	**5.04**	Yes	Maximum
Pitch	1.2570	1.2675	1.2780	1.45315	1.45890	0.9581	0.9666	**0.58**		**0.85**		Maximum
Pellet OD	0.7680	0.7975	0.8270	1.46370	1.44587	0.9674	0.9603	**-1.78**	Yes	**-0.71**		Minimum
Clad thickness	0.0520	0.0580	0.0640	1.45884	1.45343	0.9694	0.9574	**-0.54**		**-1.20**	Yes	Minimum
No clad	0.0000	0.0580	0.0640	1.48094	1.45343	1.0086	0.9574	**-2.75**	Yes	**-5.12**	Yes	Minimum
Clad OD[3]	0.9140	0.9385	0.9630	1.45740	1.45498	0.9644	0.9629	**-0.24**		**-0.15**		Minimum
GT thickness	0.0320	0.0356	0.0391			0.9622	0.9620	**N/A**		**-0.02**		Minimum
No guide tube	0.0000	0.0356	0.0391			0.9681	0.9620	**N/A**		**-0.61**		Minimum
Fuel length	351.00	366.00	381.00			0.9632	0.9626	**N/A**		**-0.06**		Negligible
Fuel stack density (% TD)	93.00	95.00	97.00	1.45718	1.45506	0.9606	0.9661	**-0.21**		**0.55**		Maximum[4]

Notes:
1. The statistical uncertainties associated with the 3-D k-eff calculations are less than 0.1%.
2. All dimensions are in cm.
3. A fixed nominal clad thickness was maintained in the clad OD calculations.
4. Limiting value based on 3-D results (see Section 2.4.4.2 for discussion).

Table 3 ABB-CE 16 × 16 fuel assembly parameter study results

Parameter	Minimum	Nominal	Maximum	1-D k-eff min. parameter	1-D k-eff max. parameter	3-D k-eff min. parameter[1]	3-D k-eff max. parameter[1]	% Δk 1-D	Significant	% Δk 3-D	Significant	Limiting parameter value
Enrichment	4.025	4.600	5.175	1.44901	1.49365	0.9300	0.9738	**4.46**	Yes	**4.38**	Yes	Maximum
Pitch	1.2746	1.2852	1.2958	1.46985	1.47738	0.9499	0.9606	**0.75**		**1.07**	Yes	Maximum
Pellet OD	0.7956	0.8262	0.8432	1.48449	1.45968	0.9589	0.9467	**-1.77**	Yes	**-1.22**	Yes	Minimum
Clad thickness	0.0569	0.0635	0.0701	1.47718	1.47030	0.9607	0.9492	**-0.69**		**-1.15**	Yes	Minimum
No clad	0.0000	0.0635	0.0701	1.50518	1.47030	1.0056	0.9492	**-3.49**	Yes	**-5.64**	Yes	Minimum
Clad OD[3]	0.9449	0.9702	0.9955	1.47735	1.47227	0.9590	0.9532	**-0.51**		**-0.58**		Minimum
GT thickness	0.0914	0.1016	0.1118			0.9548	0.9535	N/A		**-0.13**		Minimum
No guide tube	0.0000	0.1016	0.1118			0.9591	0.9535	**N/A**		**-0.56**		Minimum
Fuel length	351.00	366.00	381.00			0.9531	0.9539	**N/A**		**0.08**		Negligible
Fuel stack density (%TD)	93.00	95.00	97.00	1.47525	1.47221	0.9520	0.9566	**-0.30**		**0.46**		Maximum[4]

Notes:
1. The statistical uncertainties associated with the 3-D k-eff calculations are less than 0.1%.
2. All dimensions are in cm.
3. A fixed nominal clad thickness was maintained in the clad OD calculations.
4. Limiting value based on 3-D results (see Section 2.4.4.2 for discussion).

Table 4 GE 8 × 8 fuel assembly parameter study results

Parameter	Minimum	Nominal	Maximum	1-D k-eff min. parameter	1-D k-eff max. parameter	3-D k-eff min. parameter[1]	3-D k-eff max. parameter[1]	% Δk 1-D	Significant	% Δk 3-D	Significant	Limiting parameter value
Enrichment	3.675	4.200	4.725	1.43951	1.48767	0.9213	0.9787	**4.82**	**Yes**	**5.74**	**Yes**	Maximum
Pitch	1.6154	1.6256	1.6282	1.46365	1.46677	0.9470	0.9531	**0.31**		**0.61**		Maximum
Pellet OD	1.0288	1.0414	1.0566	1.46920	1.46213	0.9519	0.9531	**-0.71**		**0.12**		Negligible
Clad thickness	0.0729	0.0813	0.0897	1.46919	1.46313	0.9561	0.9480	**-0.61**		**-0.81**		Minimum
No Clad	0.0000	0.0813	0.0897	1.49408	1.46313	0.9932	0.9480	**-3.10**	**Yes**	**-4.52**	**Yes**	Minimum
Clad OD[3]	1.2046	1.2268	1.2522	1.46729	1.46495	0.9543	0.9505	**-0.23**		**-0.38**		Minimum
Water rod thickness	0.0635	0.0762	0.1016			0.9523	0.9521	**0**		**-0.02**		Minimum
No water rod	0.0000	0.0762	0.1016			0.9553	0.9521	**0**		**-0.32**		Minimum
Fuel stack density (%TD)	94.00	96.00	98.00	1.46738	1.46491	0.9499	0.9555	**-0.25**		**0.56**		Maximum[4]

Notes:
1. The statistical uncertainties associated with the 3-D k-eff calculations are less than 0.1%.
2. All dimensions are in cm.
3. A fixed nominal clad thickness was maintained in the clad OD calculations
5. Limiting value based on 3-D results (see Section 2.4.4.2 for discussion).

2.4.4 Examination of Anomalous or Unexpected Results

For fuel parameters where 1-D calculations are possible, the 1-D and 3-D results show generally good agreement. However, this observation is not true for the fuel stack density. In addition, the fuel pellet OD results contradicted the expected results. The variation of k_{eff} versus these parameters was studied in more detail, as presented below. Because the fuel pellet OD is a more significant parameter, it is examined first.

2.4.4.1 Fuel Pellet OD

The minimum pellet OD was identified as the limiting value for criticality based on the results in Tables 2 and 3 and assuming nominal values for all other parameters, including enrichment. Because the maximum pellet OD was expected to be the limiting value for k_{eff}, additional calculations were performed with both CSAS1X/XSDRNPM and CSAS25/KENO V.a for the Westinghouse 17 × 17 fuel assembly design at various pellet ODs and uranium enrichments to better understand the impact of the pellet OD on k_{eff}. The results of these calculations are plotted as normalized k_{eff} versus percent of nominal pellet OD in Figures 1 and 2. The maximum pellet OD is limited by the nominal inside diameter (ID) of the clad. The calculated k_{eff} values have been normalized to the calculated k_{eff} value at the nominal pellet OD for the same uranium enrichment. This normalization allows the trends in results to be easily compared in a single figure. The XSDRNPM 1-D infinite pin cell lattice results and the KENO V.a fuel assembly cask basket results are fairly consistent. They show that k_{eff} does increase as previously expected with increasing fuel pellet OD for low enrichments (e.g., 2 wt %), so the optimum k_{eff} occurs at the maximum pellet OD. As the enrichment increases, the optimum point shifts toward smaller pellet ODs. The optimum k_{eff} value for the 3.5 wt % results occurs in the middle of the range, and at 5 wt %, the optimum occurs near the low end of the range.

The reason for this phenomenon may be understood by examination of Figure 3, which plots the normalized k_{eff} values as a function of the spectral index, the energy of the average lethargy causing fission (EALF). The results for all three enrichments have an optimum k_{eff} for an EALF of approximately 0.16 eV. As the size of the pellet increases, the spectrum hardens. If the EALF is less than the optimum, increasing the pellet OD results in an increased k_{eff} as shown in Figure 3. However, if the EALF is equal to or greater than the optimum, increasing the OD results in a decreased k_{eff} value due to the associated spectral hardening. For cask analysis performed at enrichments between 3.5 and 5 wt %, it appears that the optimum pellet OD will occur somewhere between the nominal and the minimum OD; tending toward minimum with increasing enrichment.

2.4.4.2 Fuel Stack Density

The 1-D deterministic results for fuel stack density are inconsistent with the 3-D Monte Carlo results. Although the change in k_{eff} over the range of fuel stack density is small, in both the 1-D and 3-D results, it seemed prudent to understand the apparent discrepancy in results and to identify the optimum fuel density. As was done for the pellet OD cases, additional calculations were performed with both XSDRNPM and KENO V.a for the Westinghouse 17 × 17 fuel at various fuel stack densities and uranium enrichments to better understand the impact of the fuel density on k_{eff}. The results of the 1-D pin cell calculations are presented in Figure 4. For low enrichment (e.g., 2 wt % ^{235}U) the reactivity increases with increasing density. However, as the enrichment increases, the reactivity of the fuel decreases with increasing fuel density. This behavior was not observed with the 3-D basket cell calculations, as shown in Figure 5. To compare the 1-D and 3-D methods directly, infinite pin cell calculations were performed at the minimum and maximum densities with KENO V.a for the minimum and maximum enrichments studied (2 wt % and 5 wt %). The 1-D and 3-D results are compared in Figure 6.

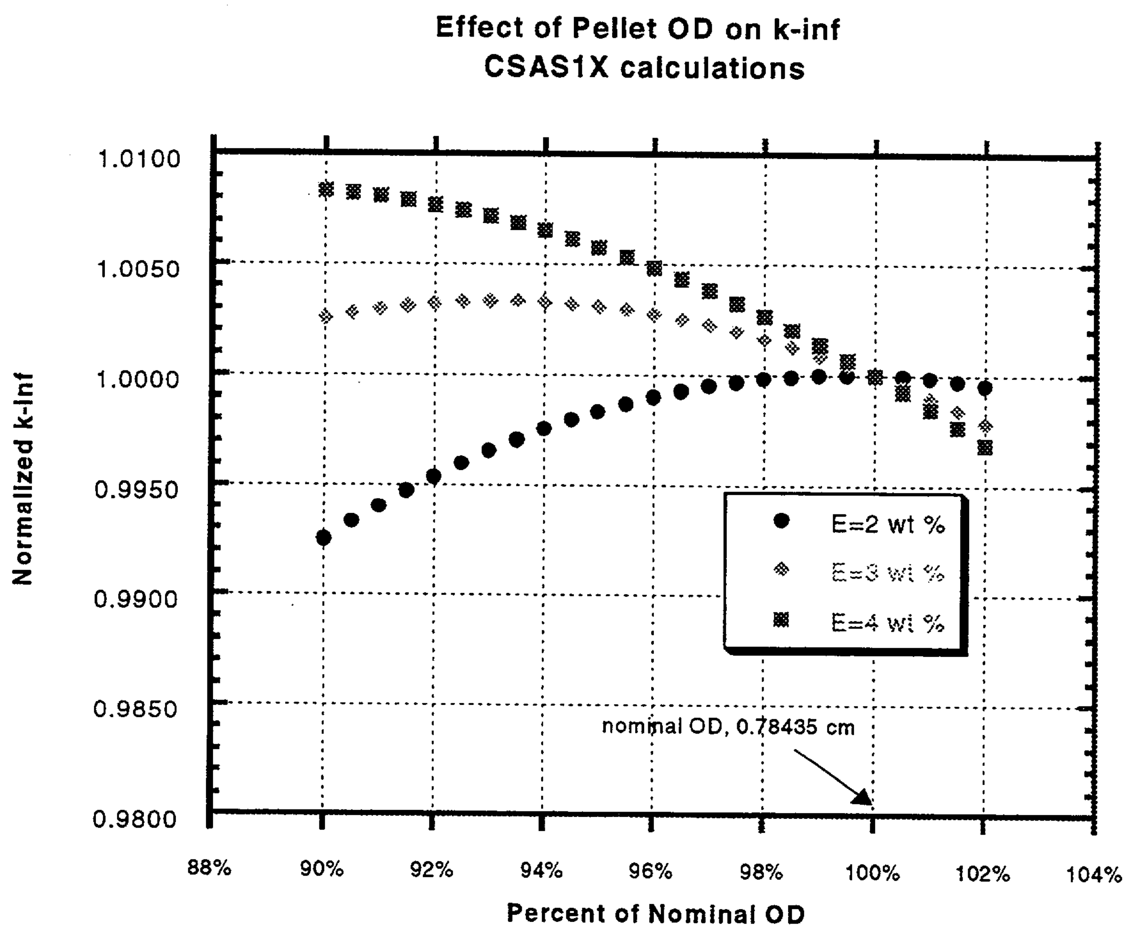


Figure 1 Normalized k-inf versus pellet OD for an infinite pin cell lattice

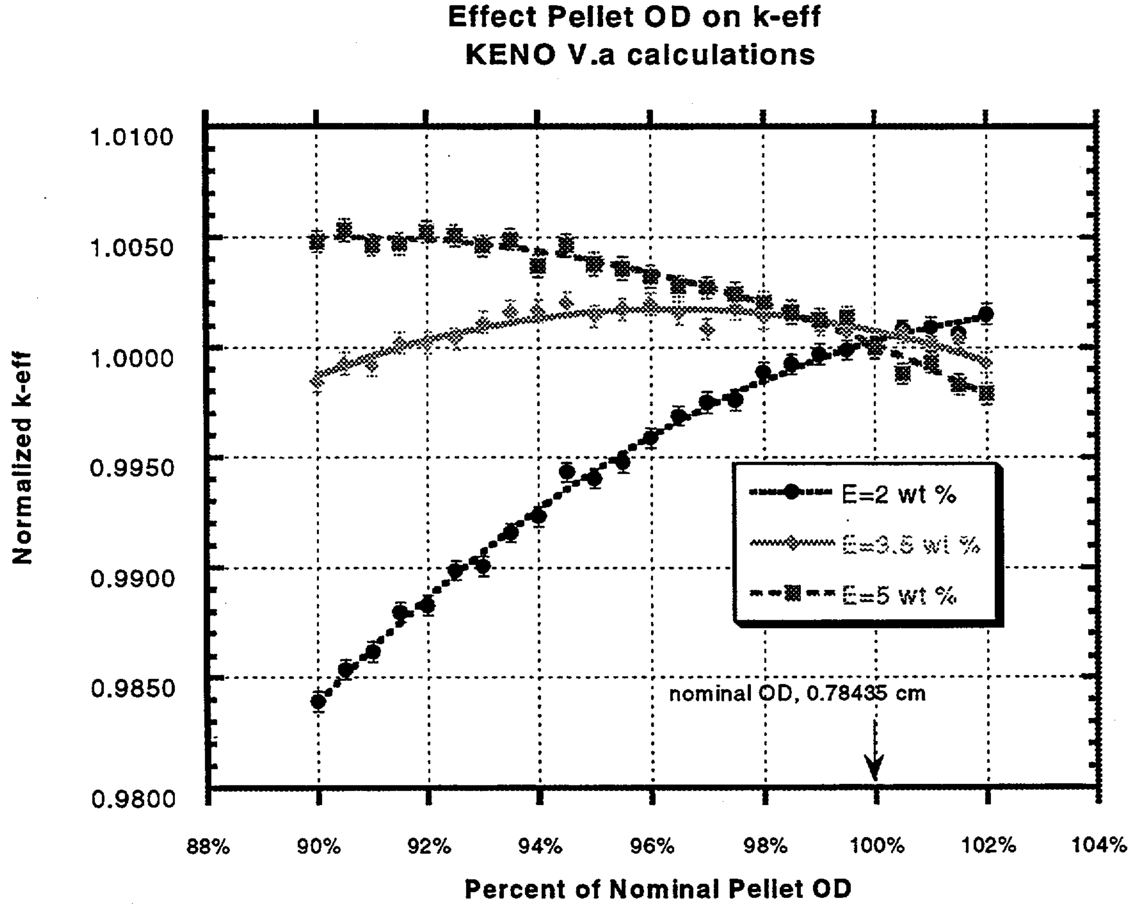


Figure 2 Normalized k-eff versus pellet OD for an infinite radial array of fuel assemblies in cask basket

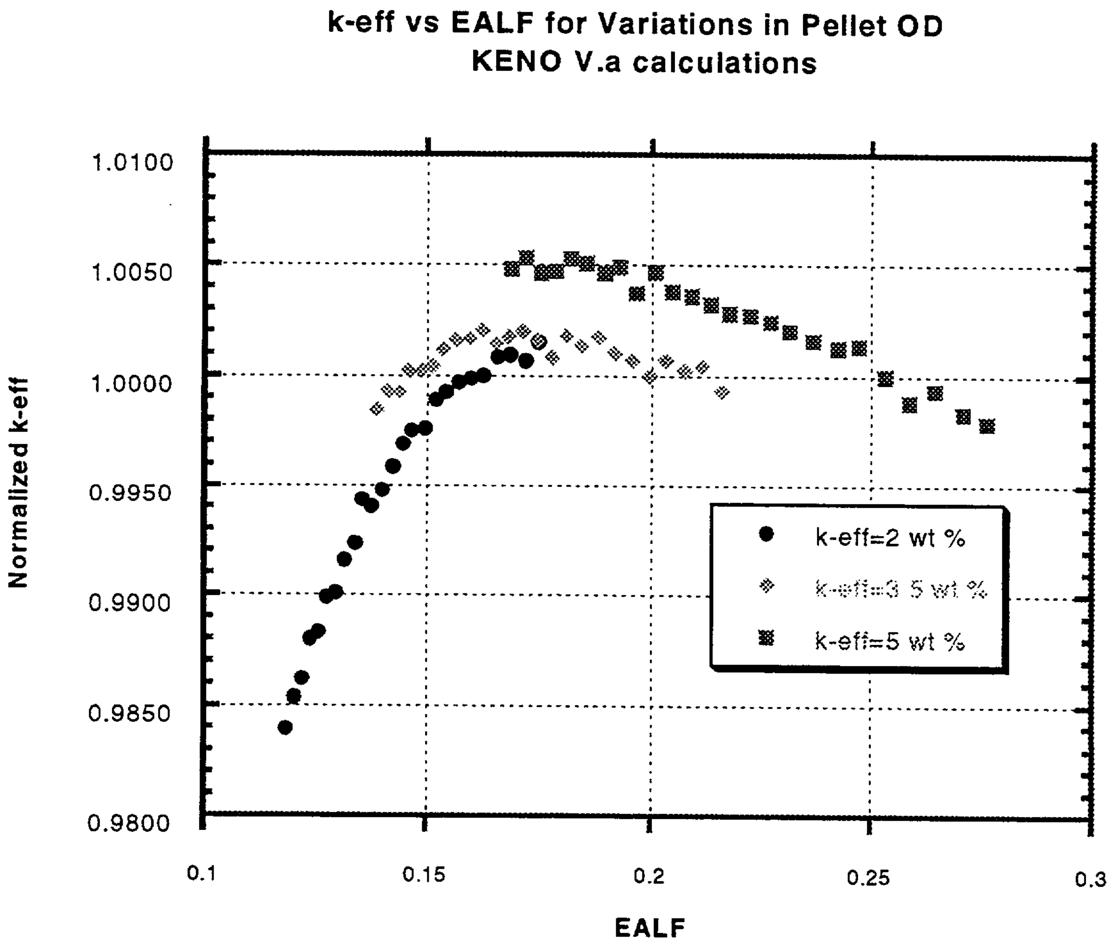


Figure 3 Normalized k-eff versus EALF for KENO V.a pellet OD calculations

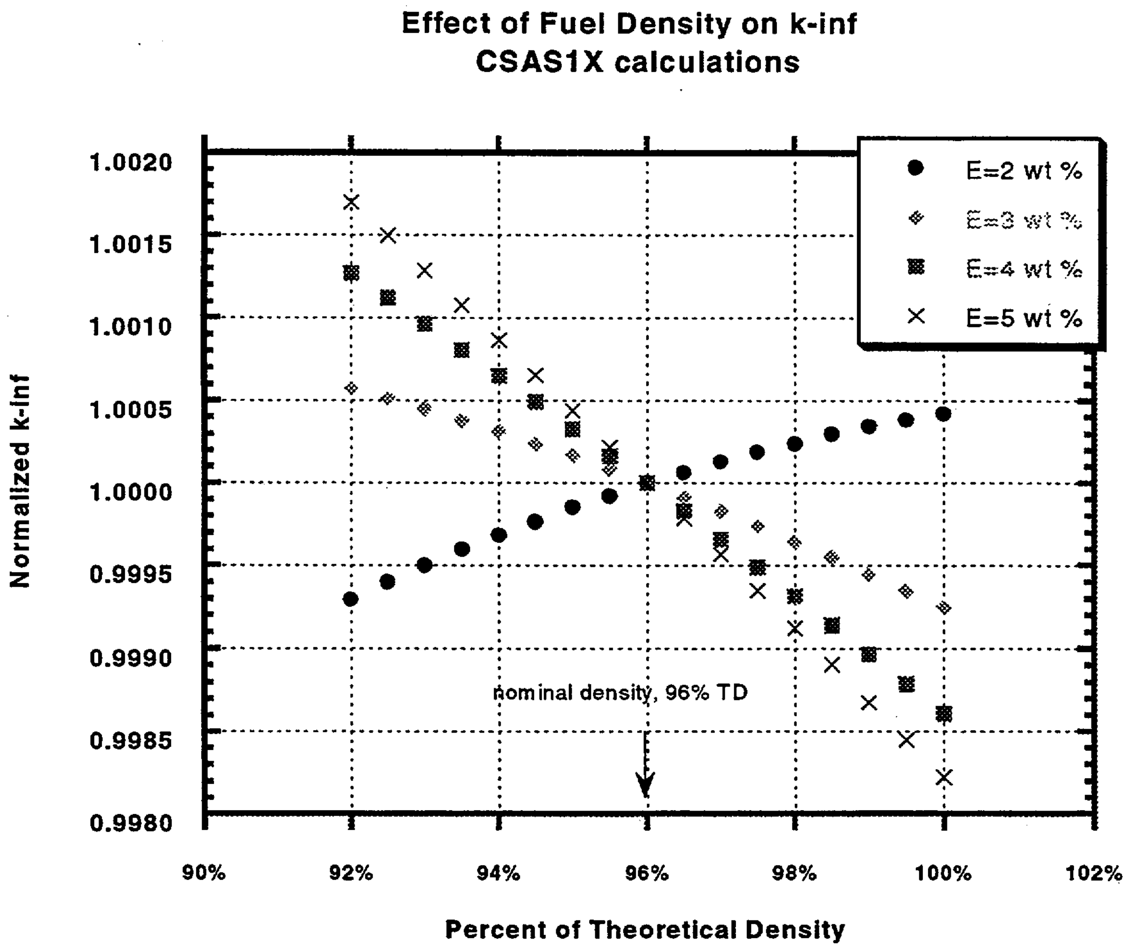


Figure 4 Normalized 1-D k-inf versus fuel density for an infinite pin cell lattice at various enrichments

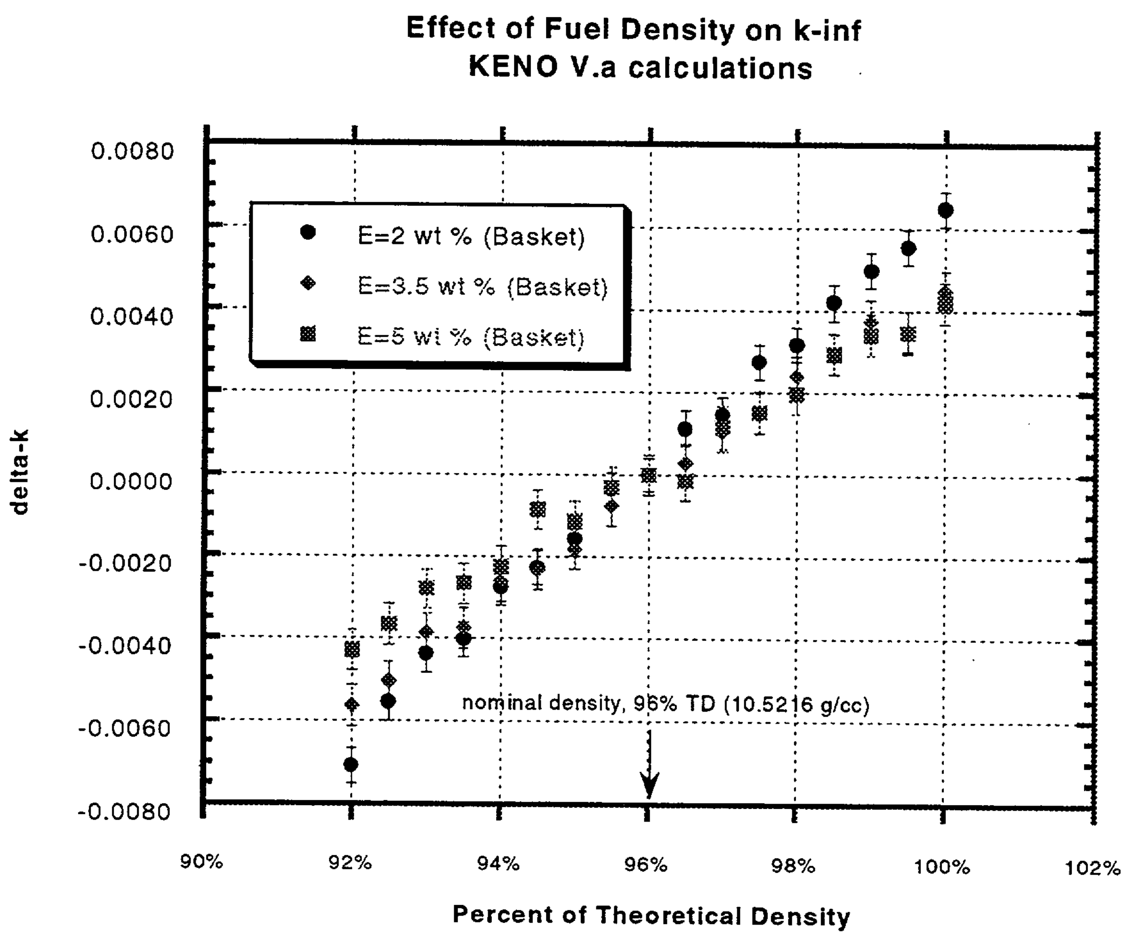


Figure 5 Normalized 3-D k-eff versus fuel density results at various enrichments

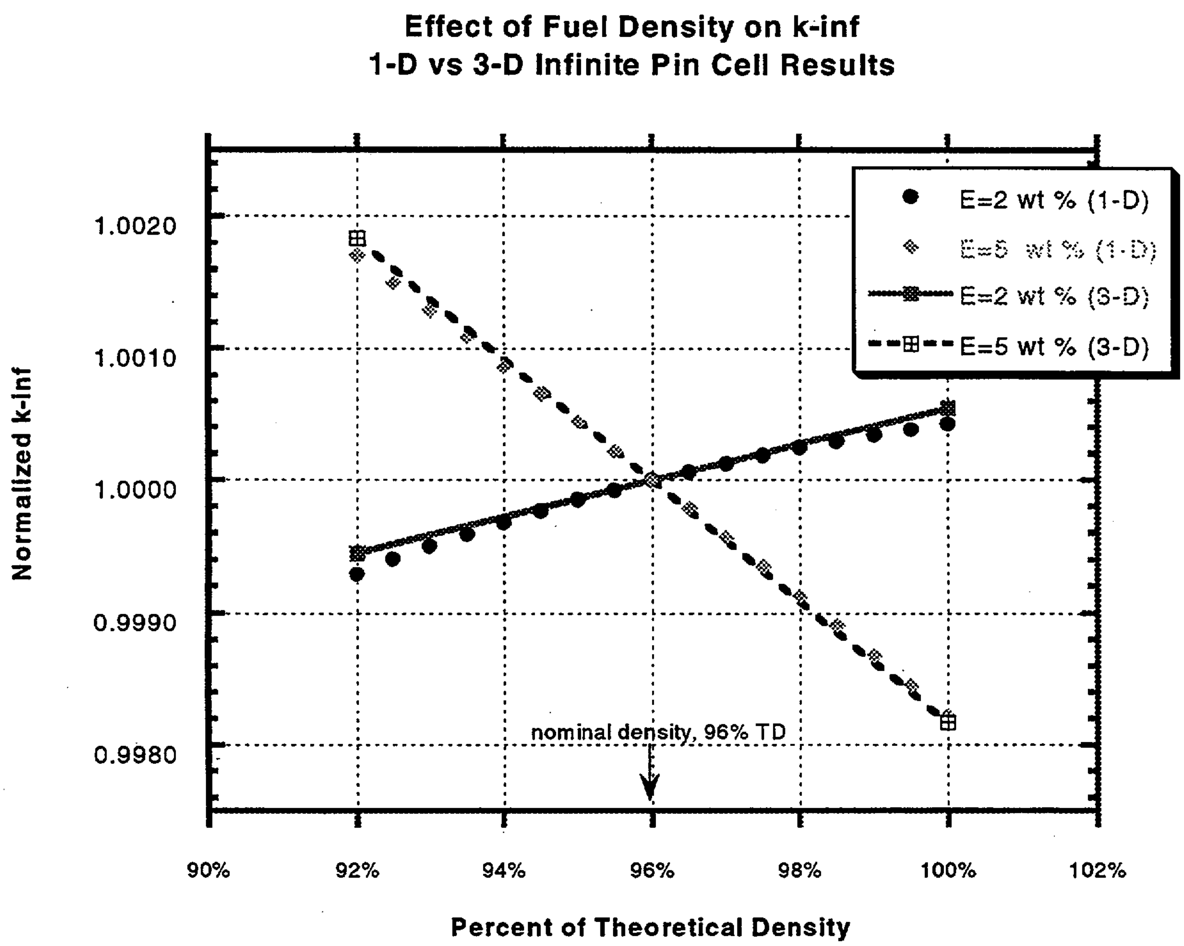


Figure 6 Comparison of 1-D and 3-D infinite pin cell results for normalized k-inf versus fuel density

Assuming that the 3-D results vary linearly, the 3-D results show very good agreement with the 1-D results. Thus, Figure 6 indicates that the differences between 1-D and 3-D results shown in Tables 2–4 are due to spectral effects caused by geometrical differences between the infinite pin cell and the infinite cask basket models, not differences between XSDRNPM and KENO V.a.

This conclusion was confirmed by performing KENO V.a calculations at 2 and 5 wt % for three different geometry models based on the Westinghouse 17 × 17 fuel assembly design: (1) infinite pin cell lattice, (2) infinite array of 17 × 17 fuel assemblies with a water gap between assemblies, and (3) infinite array of 17 × 17 fuel assemblies in poisoned cask baskets. The results are shown in Figure 7. The calculations were performed at 0.5% increments in fuel density for the infinite cask basket model. Since the results were approximately linear, calculations were only performed at the minimum and maximum densities for the infinite pin cell and assembly models. The slope of the line for k_{inf} versus fuel density decreases with increasing enrichment. This behavior is consistent with that observed in Figures 4–6. Likewise, comparing the three geometry models, the slope decreases from the cask basket model to the infinite assembly array to the infinite pin cell array. This effect can be seen more clearly in Figure 8, which shows the results for a single enrichment. The results in Figure 6 confirm that, for fuel up to 5 wt % enriched in a cask basket, the maximum fuel stack density is the optimum for criticality.

2.5 Recommendations

Based on the calculated parameter sensitivities over the range from minimum to maximum for each parameter, the following criticality safety parameters are recommended for inclusion in the STS:

1. Fuel type

   a. Array size, number of fuel rods, including number of partial length rods (where applicable), and cladding material

   b. Number and material of guide and instrument tubes for PWRs and water rods for BWRs

2. Enrichment

Other parameters considered to be significant in criticality safety licensing calculations that should require updates to the FSAR via the Section 72.48 process are listed below. Note that these are recommended parameters based on their influence on k_{eff}. Applicants could be permitted to eliminate any of these parameters with appropriate justification. For example, neglecting their presence in the safety analysis could eliminate guide/instrument tube material.

3. Fuel rod pitch

4. Fuel pellet OD

5. Cladding thickness

6. Cladding OD

7. PWR guide/instrument tube thickness
8. Active fuel length
9. Fuel stack density

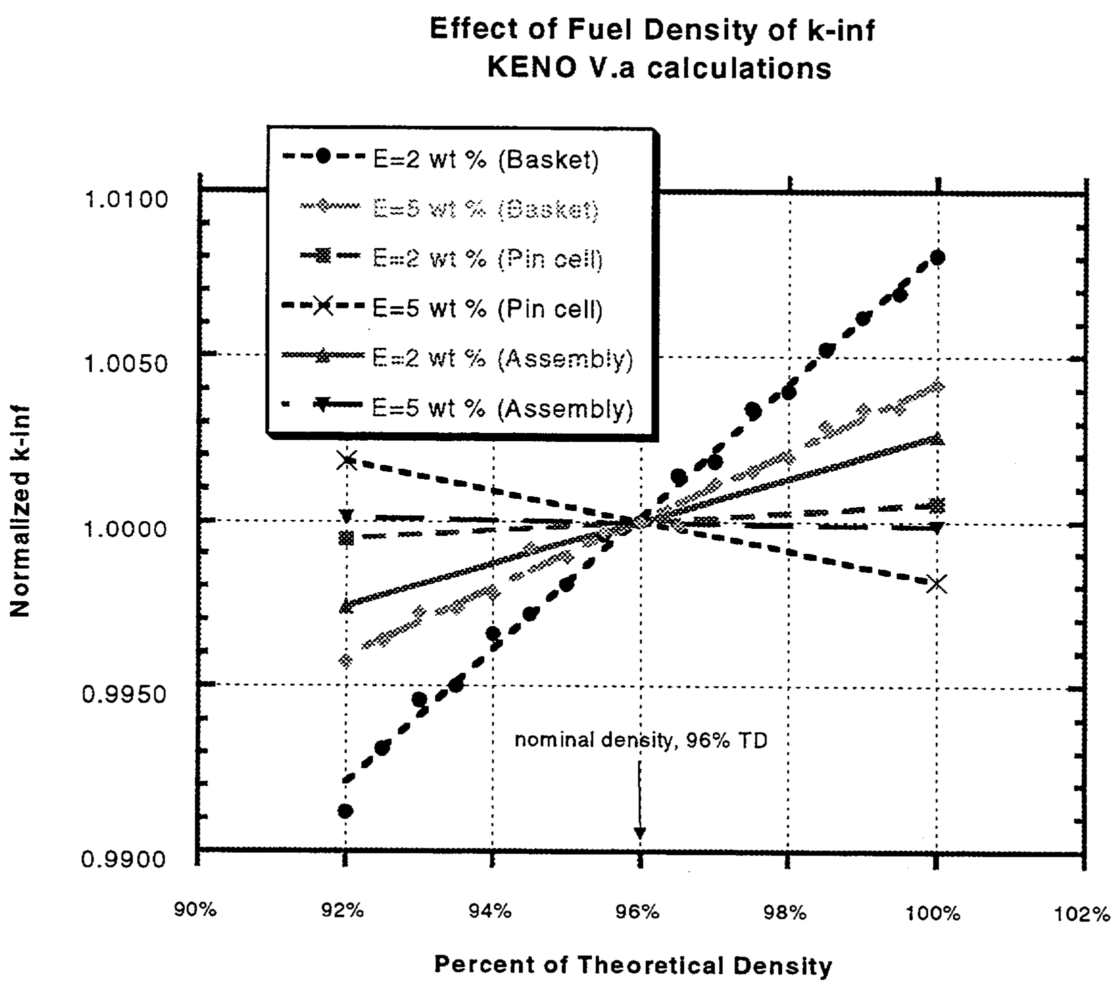


Figure 7 Normalized 3-D k-inf versus fuel density for the 3 geometric models

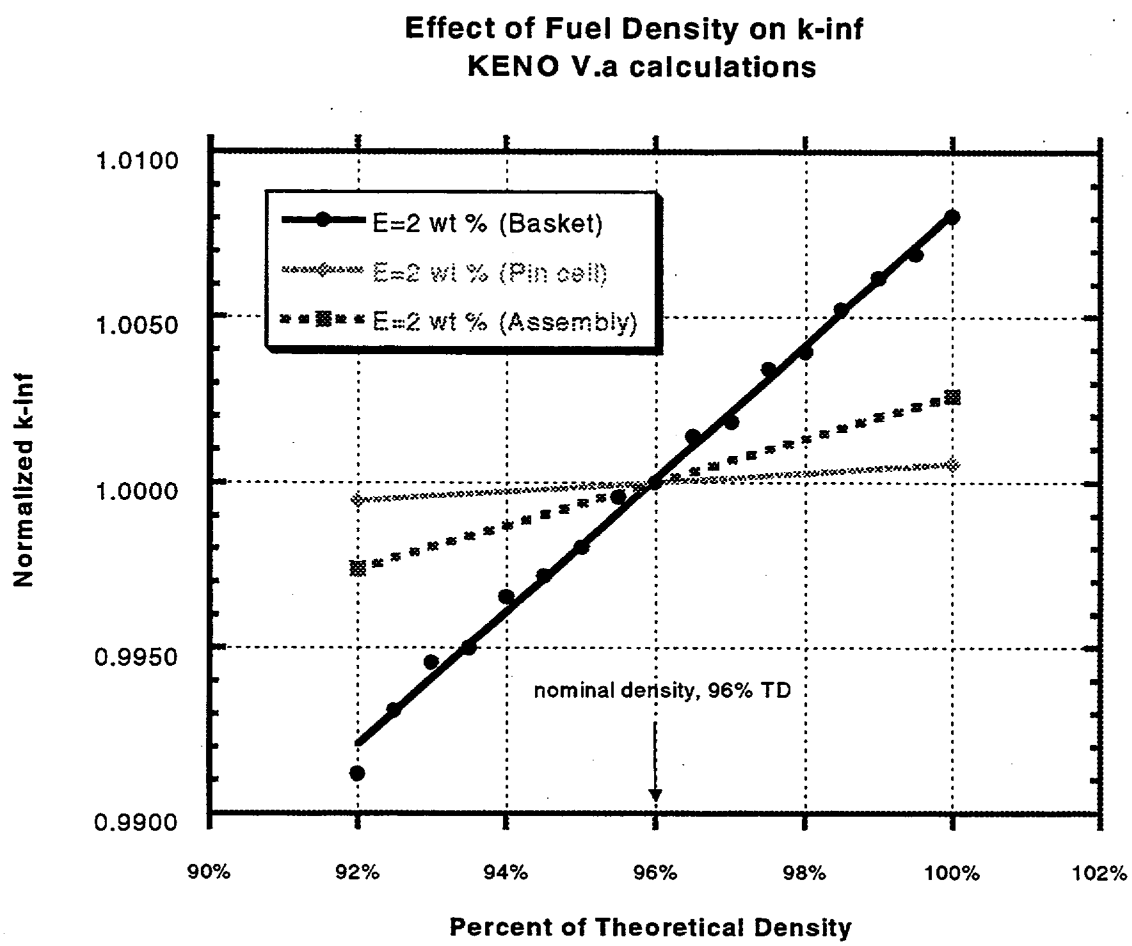


Figure 8 Normalized 3-D k-inf versus fuel density for the 3 geometric models at 2 wt %

3 REVIEW OF FUEL SPECIFICATIONS FOR SHIELDING

3.1 Review Basis

A large number of fuel assembly design, cask design, and operating parameters can potentially influence the radiation dose rates for a spent fuel storage cask. The basis for establishing a minimum set of specifications requires criteria against which the importance of the parameters can be assessed. Unlike criticality, which has a relatively well-defined criterion for safety, the basis for establishing parameters of importance to cask radiation dose rates is less straightforward. Because the acceptability criteria for a storage cask are based on the dose rate at the public boundary of a licensed storage facility (i.e., no specific local limits on the dose rate for the cask), it may be acceptable to allow larger variations in the specifications related to dose assessment than might be acceptable for criticality safety.

Because the dose rate is dependent on the SNF compositions, the specifications may include irradiation and decay parameters such as burnup and cooling time, in addition to assembly design parameters. The total dose rate includes both gamma and neutron components, and each component exhibits very different behavior/sensitivity in different spent fuel regimes. Thus, the cask design may become a key factor due to its influence on the relative importance of the neutron and gamma components.

3.2 Technical Specification Parameters

The candidate technical specification parameters for shielding analyses that were reviewed in this study are listed in Table 5. Each of the parameters identified in the table has been evaluated to determine the influence of typical variations in the parameters on the shielded cask dose rate.

The parameters commonly used to define radiation source levels in SNF are burnup and cooling time. These parameters can vary over a wide range for spent fuel storage applications and are clearly important in determining the dose rates. The initial fuel enrichment may also have a large effect on cask dose rates.

The other fuel specifications considered in this study include assembly uranium mass, fuel assembly type, burnable poison rod exposure, integral burnup poison assemblies, assembly hardware and cladding type, and moderator density and specific power during fuel irradiation. As noted in Section 2, the fuel assembly type may be defined for groups of assembly designs with common lattice types (e.g., 14 × 14, 15 × 15, etc.) and common characteristics (e.g., number and location of fuel rods, water holes, guide tubes, etc.). The fuel assembly type (beyond being either a PWR- or BWR-assembly type) is typically not included as shielding technical specifications. Similarly, exposure to burnable poison rods, moderator density, and specific power has typically not been specified in the TS. In general, the effect of variations in these parameters has been addressed by assuming bounding values in the source term analysis. These parameters are included in this study to provide a wide range of different candidate parameters, and serve as a baseline against which parameter importance can be judged.

Table 5 Fuel technical specification parameters and common restrictions

Technical specifications (candidate)	Commonly used restrictions
Cooling time (years)	Minimum cooling time
Assembly burnup (MWd/t)	Maximum burnup
Initial enrichment (wt % ^{235}U)	Minimum enrichment
Assembly/cask uranium mass (kg U)	Maximum uranium content
Fuel assembly type (14 × 14, 15 × 15, 17 ×17, etc.)	Unrestricted
Integral burnable poison rods (IBAs)	Unrestricted
Burnable poison rods (BPRs)	Unrestricted
Assembly hardware: cladding type and structural impurity levels	Unrestricted
Moderator density (g/cm^3)	Unrestricted
Specific power (MW/t)	Unrestricted

3.3 Analysis Methods and Models

Storage cask surface dose rates were calculated with the SCALE 1-D radial shielding sequence SAS1 that uses the XSDRNPM transport code. Cross sections from the SCALE coupled 27-neutron/18-gamma group transport library were used. Spent fuel isotopic compositions and the associated neutron and gamma-ray source terms were generated using the ORIGEN-ARP methodology, which includes burnup-dependent cross sections created for several assembly designs that are distributed with SCALE. The SAS2 depletion analysis sequence of SCALE was used to create additional libraries for assembly designs that did not have ARP libraries already available. The XSDRNPM transport calculations applied the neutron and gamma fixed sources calculated by ORIGEN-S with the fission option in XSDRNPM enabled in order to calculate secondary fissions from subcritical neutron multiplication.

The principal storage cask design used in this study was based on the HI-STORM (Holtec International) concrete cask design[3] constructed with approximately 68 cm of normal concrete and 5 cm of steel. The inner diameter of the cask model was about 187 cm. The internal canister contained Boral absorber plates. Several calculations were also performed using a carbon steel/resin type cask model[4] similar to the TN24 cask to assess the impact of using higher Z shielding materials.

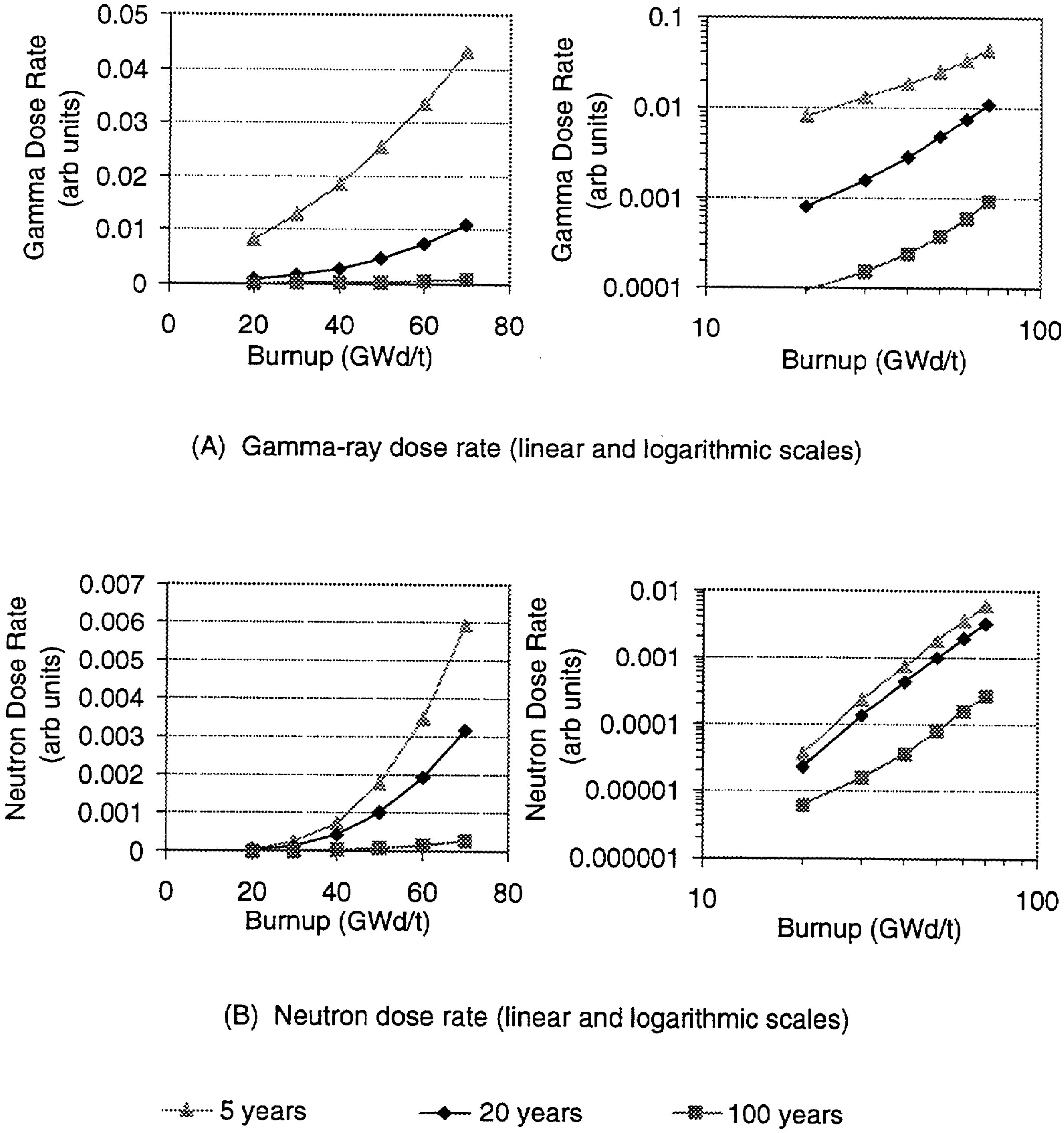
The neutron and gamma-ray components of the dose rate were calculated separately, since the behavior of the respective dose rate components can be significantly different for many of the parameters investigated. The relative importance of the neutron and gamma-ray components is also influenced by the cask design. The individual dose rate components are presented to provide a means of estimating the parameter importance for cask designs that differ from those used in this study and allow the bounding variations (for any cask design) to be estimated.

3.4 Results

3.4.1 Principal Fuel Specification Parameters

3.4.1.1 Burnup

Previous studies of the radionuclide importance to shielding[5,6] have demonstrated that the dominant effect of increasing burnup is the dramatic increase in the spontaneous fission neutron source, primarily from ^{244}Cm. The neutron dose rate (D_n) has been observed to increase approximately as the burnup (B) to the power of four, i.e., $D_n \propto B^4$. The gamma dose rate increases nearly linearly with burnup. The variation in the neutron and gamma dose rate on the external surface of the concrete cask model, as a function of burnup, is illustrated in Figure 9 for 3 wt % fuel. The slope, of the profiles (m) from the log-log plots yields the power relationship of the dose rates to changing burnup (e.g., $D \propto B^m$). The gamma ray dose rate exhibits some nonlinear behavior, which is due entirely to the contribution of secondary gamma rays resulting from the neutron sources. The steel cask results were also investigated and found to be similar to those of the concrete cask. Therefore, only the concrete cask results are presented. It can be seen that neutrons become an increasingly larger component of the total dose rate with increasing burnup. The variation in the total dose rate for both the concrete and steel casks with burnup is illustrated in Figure 10 assuming a constant enrichment of 3 wt %. The variation in the total dose rate with burnup is slightly larger for the steel cask because neutrons (and secondary gamma rays) contribute a larger fraction of the total dose rate than for the concrete cask.



(A) Gamma-ray dose rate (linear and logarithmic scales)

(B) Neutron dose rate (linear and logarithmic scales)

Figure 9 Variation of the neutron and gamma dose rates as a function of burnup, for cooling times of 5, 20, and 100 years. Calculated for a concrete storage cask with 3 wt % fuel.

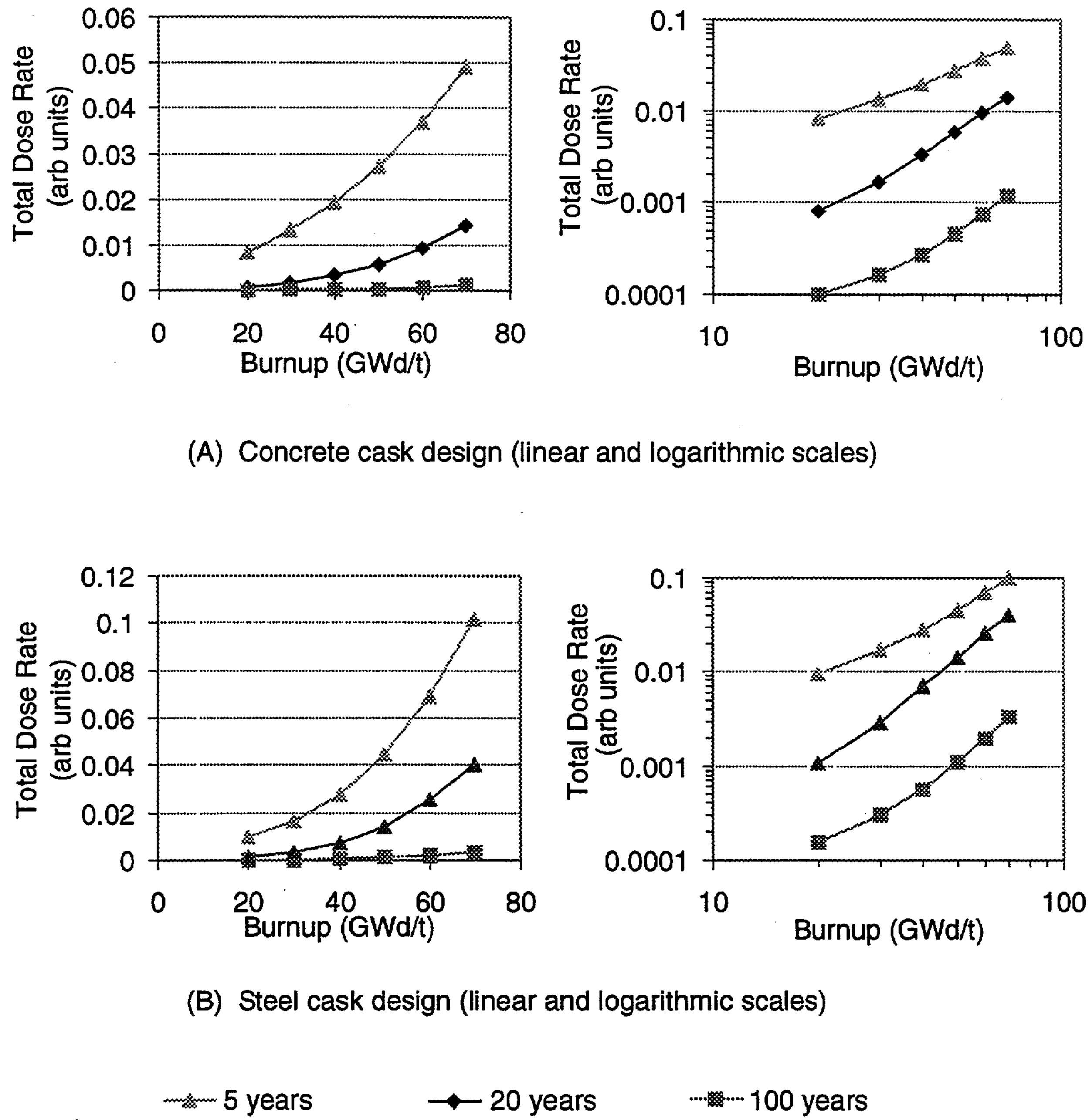


Figure 10 Variation of the total dose rates as a function of burnup, for cooling times of 5, 20, and 100 years for the concrete (A) and steel (B) cask designs. Calculated assuming 3 wt % fuel.

The relative sensitivity, or importance, of burnup to the dose rate is dependent on many factors; the cooling time, enrichment, fraction of the neutron dose rate component (cask design), and the burnup level (i.e., the rate of increase is significantly greater at higher burnup than lower burnup). Therefore, it is difficult to characterize the sensitivity with a single value that captures the importance of the parameters in all the different regimes.

3.4.1.2 Enrichment

The variation in the gamma and neutron dose rate with changing enrichment (assuming constant burnup) is shown in Figure 11 (5-year cooling) and Figure 12 (100-year cooling) for the concrete cask. The gamma-ray dose rate is seen to exhibit a lower sensitivity to enrichment variations than the neutron dose rate, and the gamma contribution is seen to be effectively independent of enrichment at low burnup values. At higher burnups, the gamma dose rate increases as the enrichment decreases because of the contribution of secondary gamma rays caused by neutron interactions in the shielding material. The variation of the neutron dose rate is significantly greater than for gamma rays. The neutron dose rate increases by more than a factor of two for a reduction in enrichment from 5 to 2.5 wt %, assuming a constant burnup of 60 GWd/t.

The enrichment effect is caused by the fact that, as the enrichment decreases, the fuel must be exposed to a larger neutron fluence (typically longer irradiation times) to achieve the same burnup. Consequently, the neutron and secondary gamma ray dose rate contribution will increase due to a larger actinide content, and, therefore, a larger neutron source term. The change in the total dose rate for the concrete and steel cask designs is illustrated in Figure 13.

3.4.1.3 Cooling Time

Cooling time is a primary fuel specification parameter that has significant influence on the neutron and gamma source term and the spectra of the gamma source term. For safety analyses, a minimum cooling time (sometimes based on the burnup) is typically used to define acceptable spent fuel assemblies for loading.

The variation in the dose rate with cooling time has been studied extensively (Ref. 5). The variation for the individual neutron and gamma dose rate components is illustrated in Figure 14 for an enrichment of 3.5 wt % and burnups of 20, 40 and 60 GWd/t, for the concrete cask design. The neutron dose rate decreases at a near-constant exponential rate for all cooling times. The gamma dose rate decreases very rapidly between 5 and 20 years cooling as the short-lived fission products decay. After 20 years cooling the dose rate decreases exponentially. The dose rate decreases by roughly two orders of magnitude from 5 to 100-years cooling.

3.4.2 Secondary Fuel Specification Parameters

3.4.2.1 Fuel Assembly Type

The importance of the fuel assembly type on the shielded neutron and gamma cask dose rates was investigated for three PWR assembly designs: ABB-CE 14 × 14, Westinghouse 15 × 15, and Westinghouse 17 × 17. A summary of the assembly design specifications is given in Table 6. These designs were used to estimate the importance of the fuel assembly type on the cask dose rates. These designs do not fully represent the range of commercial assemblies in use, but are intended only to provide a general estimate of the level of importance of the fuel assembly type.

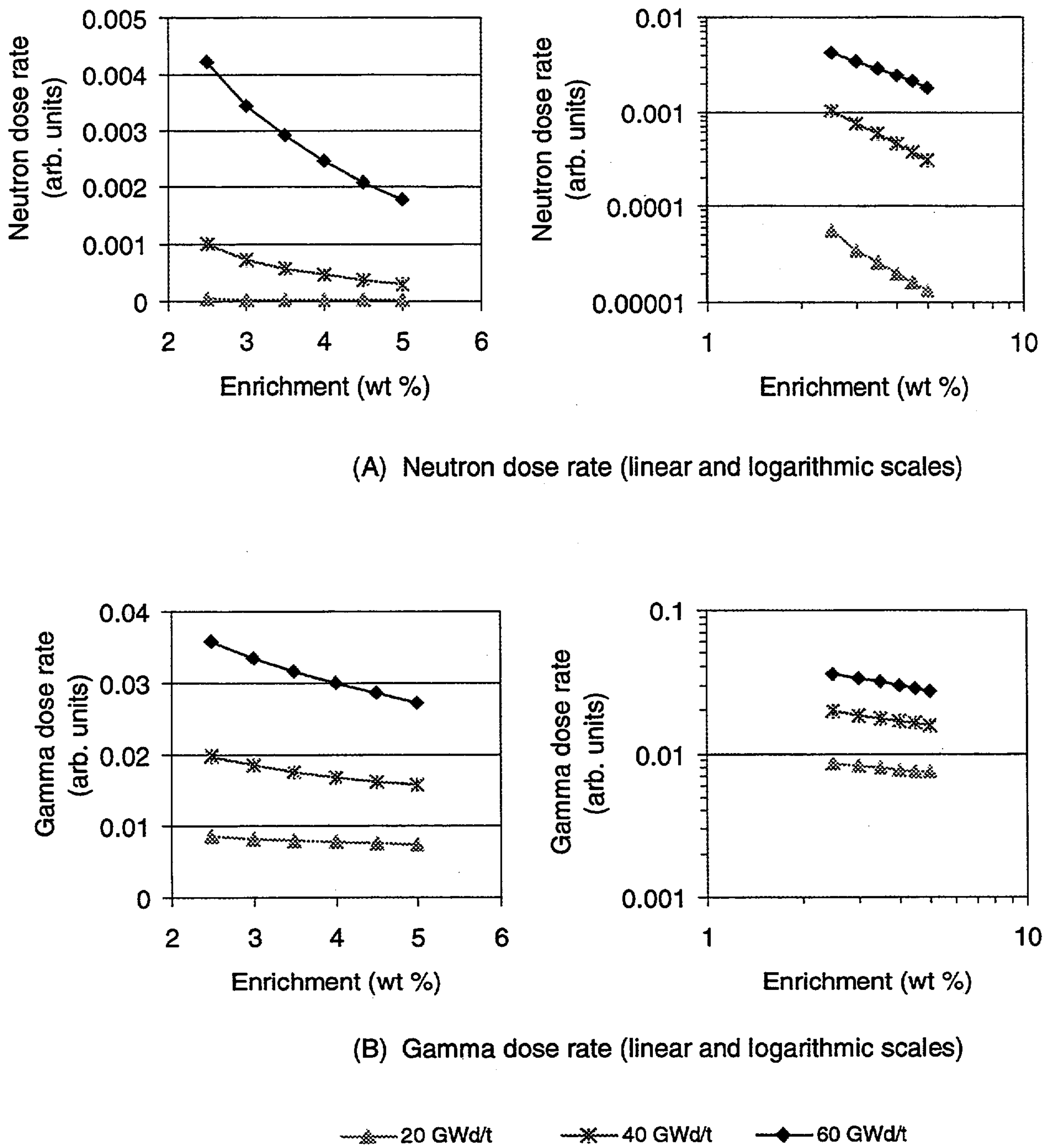
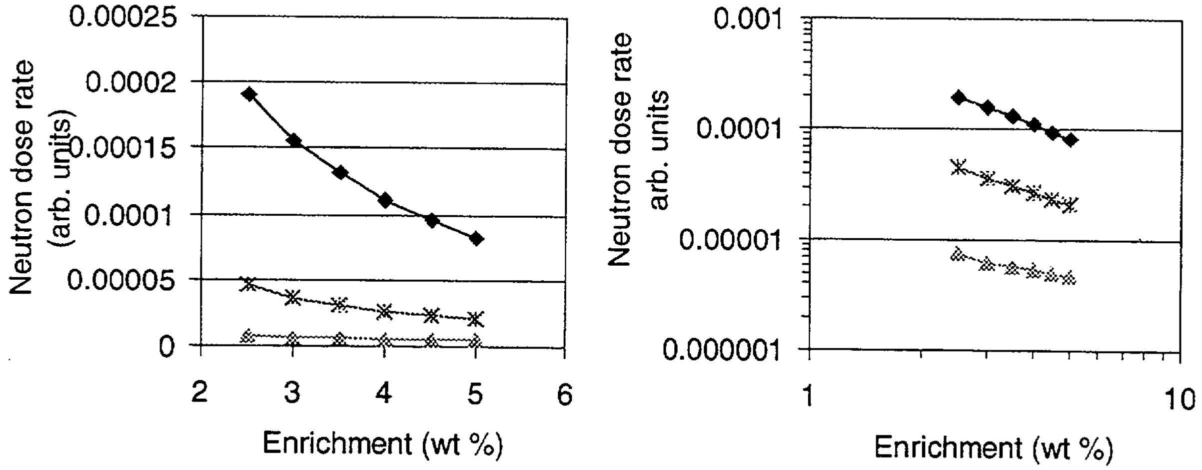
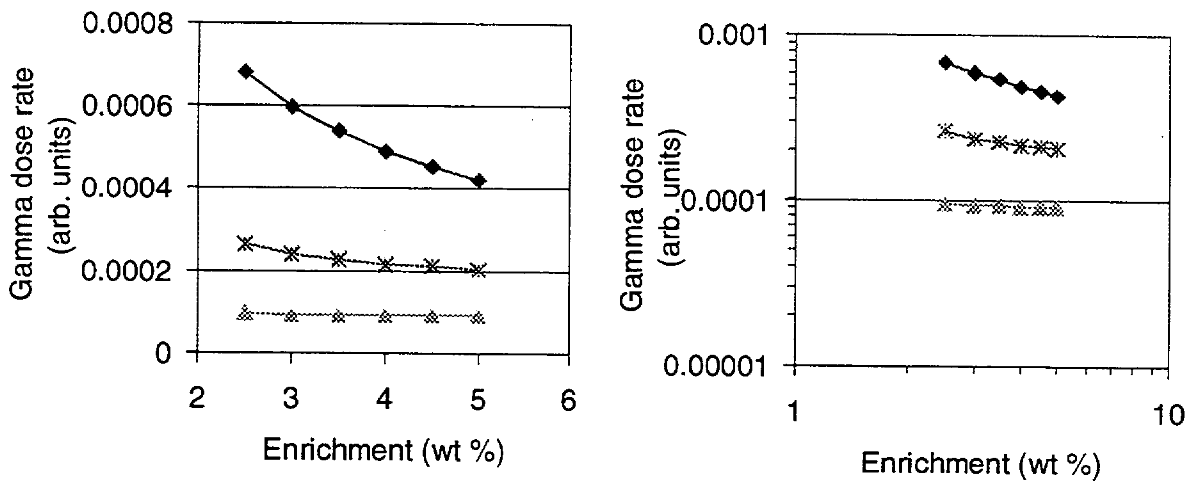


Figure 11 Variation of the neutron (A) and gamma (B) dose rate as a function of enrichment, for burnups of 20, 40, and 60 GWd/t for the concrete cask design, assuming 3 wt % enrichment and 5 year cooling.



(A) Neutron dose rate (linear and logarithmic scales)



(B) Gamma dose rate (linear and logarithmic scales)

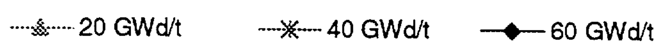


Figure 12 Variation of the neutron (A) and gamma (B) dose rate as a function of enrichment, for burnups of 20, 40, and 60 GWd/t for the concrete cask design, assuming 3 wt % enrichment and 100 years cooling.

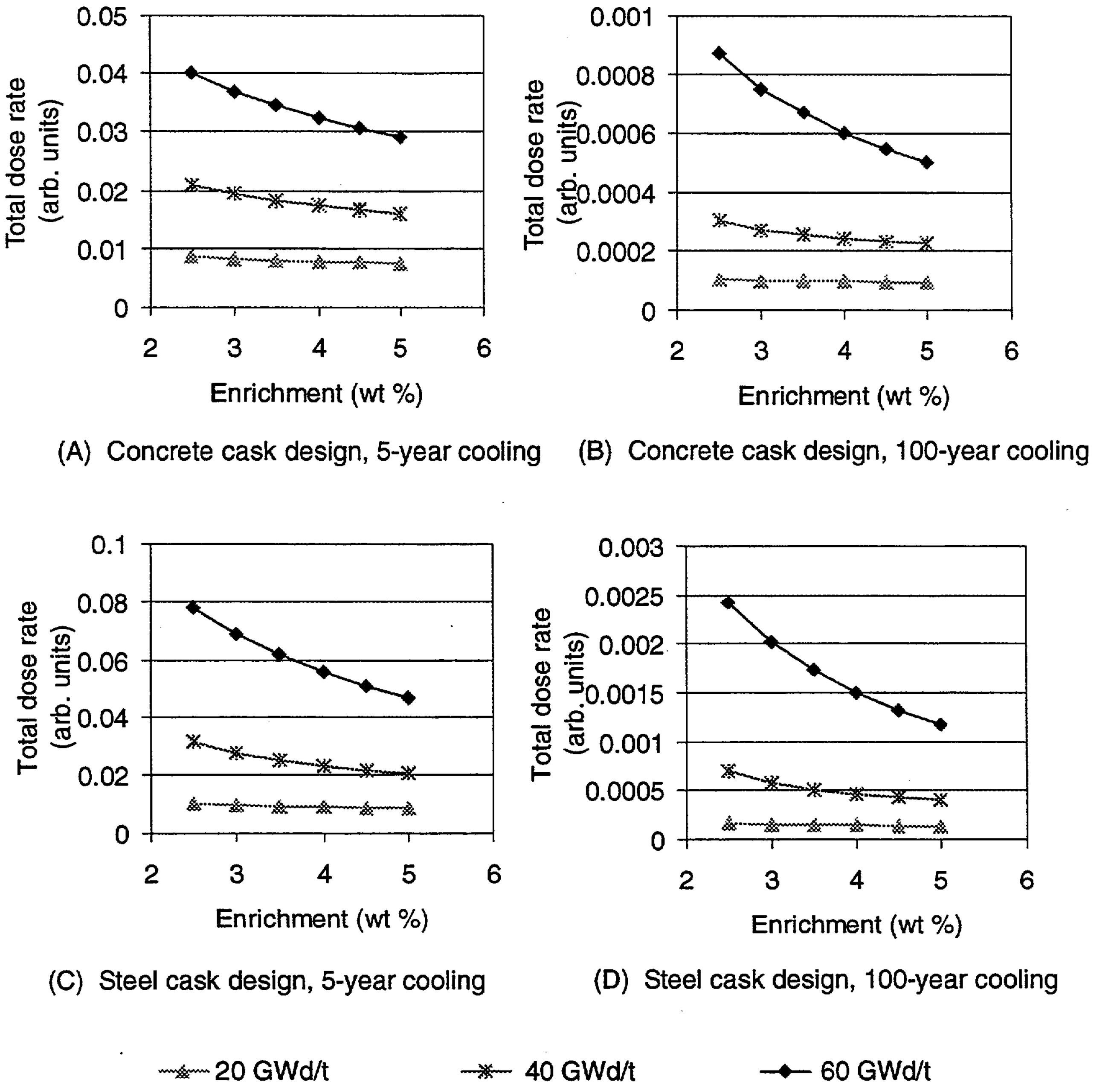
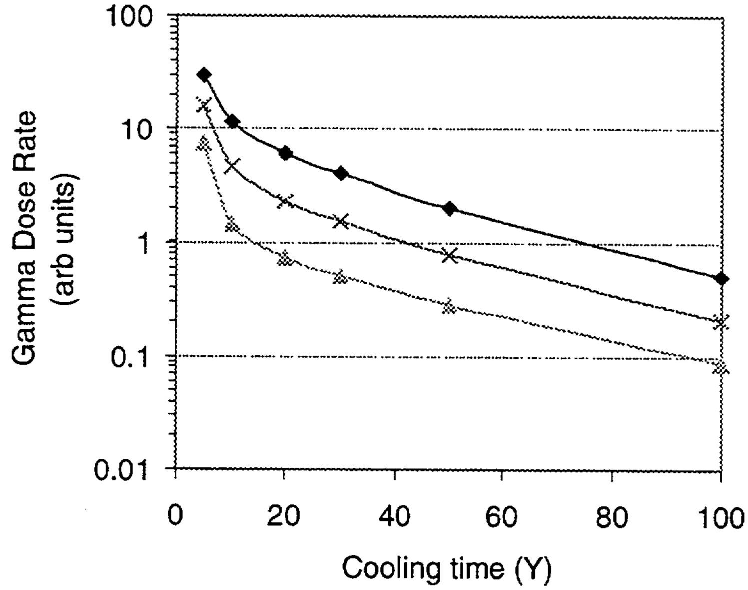
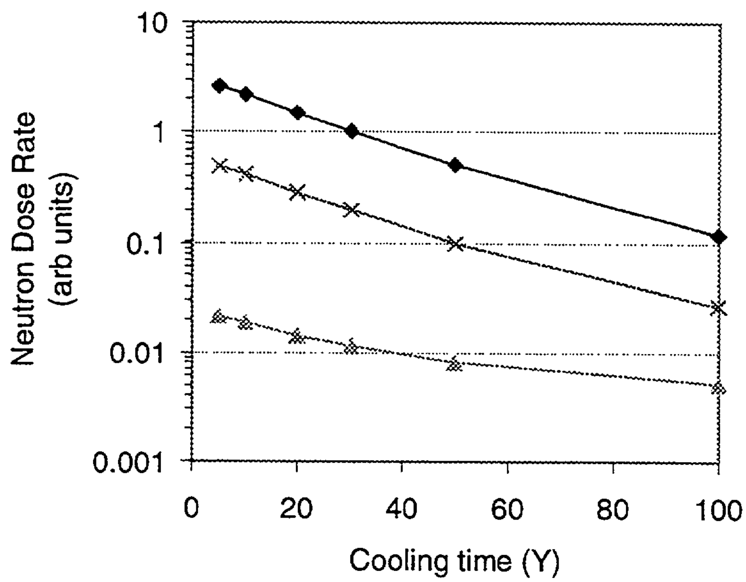


Figure 13 Variation of the total dose rate as a function of enrichment, for burnup of 20, 40, and 60 GWd/t for concrete and steel cask designs (5 year and 100 year cooling times shown)



(A) Gamma dose rate



(B) Neutron dose rate

60 GWd/t 40 GWd/t 20 GWd/t

Figure 14 Variation of the gamma (A) and neutron (B) dose rates as a function of cooling time for an enrichment of 3.5 wt % and burnups of 20, 40, and 60 GWd/t, for the concrete cask design.

Table 6 Design specifications and operational conditions for selected fuel assembly types (no burnable poisons)

Design parameter	ABB-CE 14 × 14	Westinghouse 15 × 15	Westinghouse 17 × 17
Assembly pitch (cm)	20.780	21.4503	21.4173
Number of fuel rods	176	204	264
Number of instrument tubes	0	1	1
Number of guide tubes	20	20	24
Number of burnable poison rods	0	0	0
Fuel type	UO_2	UO_2	UO_2
Enrichment range (wt % ^{235}U)	2.5 – 5.0	2.5 – 5.0	2.5 – 5.0
Fuel density (g/cm^3)	10.045	9.44	10.32
Fuel temperature (K)	873	923	811
Clad type	Zircaloy	Zircaloy	Zircaloy-4
Clad temperature (K)	620	595	620
Fuel rod data			
Fuel rod outer diameter (cm)	0.9563	0.9294	0.81915
Gap outer diameter (cm)	0.9855	0.9484	0.83566
Clad outer diameter (cm)	1.1176	1.0719	0.94966
Fuel rod pitch (cm)	1.4732	1.430	1.25984
Guide tube data			
Inner radius (cm)	0.6570	0.6502	0.57150
Outer radius (cm)	0.7080	0.6934	0.61214
Guide tube material	Zircaloy	Zircaloy	Zircaloy-4
Moderator data			
Average density (g/cm^3)	0.7332	0.7135	0.7295
Average boron concentration (ppm)	331	653	550
Moderator temperature (K)	570	579	570

The total mass of uranium in the assemblies (and cask) was assumed to be the same for all fuel assembly types. Therefore, the variations in the dose rate are due only to the changes in actinide and fission product inventories attributed to the different assembly designs (i.e., spectral differences in fuel region during irradiation). These assemblies did not use burnable poison rods, which are addressed as a separate fuel parameter.

The neutron and gamma dose rates for each fuel assembly type were calculated for initial ^{235}U enrichments of 2.5 to 5 wt % and burnups of 20, 40 and 60 GWd/t. Assembly cooling times of 5 years and 100 years were evaluated. The total dose rate exhibited a maximum difference of about 6% between the ABB-CE 14 × 14 and W 17 × 17 designs. Significantly less difference was observed between the W 15 × 15 and W 17 × 17 designs. The gamma component exhibited less variability than the neutron component. The maximum variation in the neutron dose rate was about 20% for the different assembly designs considered. For both the steel and concrete cask designs studied, neutrons were generally a smaller component of the total dose rate than gamma rays and, consequently, the variation of the gamma dose rate dominated the variation in the total dose rate. For cask designs and spent fuel regimes with a larger neutron dose component, the effect of assembly type could be larger than observed here, but in any case the differences are likely to be < 10% for any cask design.

3.4.2.2 Burnable Poison Assemblies

The effects of exposure to burnable poisons rods (BPRs) and integral burnable absorbers (IBAs) were investigated to determine: (1) the effect on the spent fuel compositions and, consequently, the cask dose rates caused by the shift in the neutron spectrum during irradiation for the fuel associated with the presence of neutron poisons; and (2) the potential direct contribution from activated hardware components associated with the BPRs to the dose rate, when irradiated BPRs are loaded in the cask assemblies.

Two removable BPR designs were considered: (1) a B_4C-Al absorber design, and (2) a steel and borosilicate glass burnable poison rod design. The spectral effects of the B_4C absorbers and borosilicate glass absorbers are expected to be similar, because both designs displace moderator in the guide tubes, which has been demonstrated to be the dominant spectral factor affecting the spent fuel compositions.[7] However, the borosilicate glass design uses a significant quantity of stainless steel, which will present a large potential activation product (cobalt) source. Therefore, only the borosilicate glass BPR design was evaluated.

Two IBA designs were considered. Cask shielding evaluations were performed for a ABB-CE 16 × 16 design using 8 integral Gd_2O_3 burnable poison rods, and an ABB-CE 14 × 14 design with 60 Er_2O_3 burnable poison rods. These were only intended to represent some typical assembly burnable poison absorber designs to estimate the potential impact of integral absorbers on shielded cask dose rates. The assumed ^{235}U enrichment for these studies was 4 wt % (uniform assembly enrichment). The poison loadings were 5 wt % Gd_2O_3 and 2 wt % Er_2O_3, respectively. The IBA assembly design specifications used for these studies are listed in Table 7.

Table 7 Design specifications for integral burnable poison rod fuel assemblies

Design parameter	ABB-CE 16 × 16	ABB-CE 14 × 14
Assembly pitch (cm)	20.78	20.8
Number of fuel rods	228	116
Number of water holes	5	5
Number of burnable poison rods	8	60
Burnable poison	5 wt % Gd_2O_3	2 wt % Er_2O_3
Fuel type	UO_2	UO_2
Enrichment (wt % ^{235}U)	4.0	4.0
Fuel density (g/cm^3)	10.44	10.44
Fuel temperature (K)	1000	1000
Clad type	Zircaloy	Zircaloy
Clad temperature (K)	620	620
<u>Fuel rod data</u>		
Fuel rod outer diameter (cm)	0.82550	0.95631
Gap outer diameter (cm)	0.84328	0.97536
Clad outer diameter (cm)	0.97028	1.1176
Fuel rod pitch (cm)	1.285	1.470
Guide tube data		
Inner radius (cm)	1.1430	1.1532
Outer radius (cm)	1.2446	1.2040
Guide tube material	Zircaloy	Zircaloy
<u>Moderator data</u>		
Average density (g/cm^3)	0.710	0.71
Average boron concentration (ppm)	650	650
Moderator temperature (K)	600	600

The borosilicate glass absorber rod design used in the study was constructed with two concentric stainless steel tubes with the inner region containing inert gas and the interstitial region containing the borosilicate glass matrix. The nominal BPR design was obtained from Ref. 8 and used to derive representative absorber and hardware masses for an assembly. This design was used with Westinghouse 17 × 17 assemblies in the North Anna Unit 1 reactor. The assembly lattice contains 264 fuel rods and 24 guide tubes, and 1 instrument tube. The different BPR clusters used in this reactor during cycle 5 contained 12, 16, or 24 rods. The shielding assessment was performed assuming the assembly nominally contained 12 BPRs. The potential number of BPR rods could range up to 24 rods per assembly. Therefore, this component of the dose rate evaluation will be highly dependent on the number of rods assumed to be present in the assemblies, and the number of assemblies containing irradiated BPR clusters. This analysis assumed that all assemblies in the storage cask contained BPRs with 12 rods per cluster. The mass of stainless steel per MTU is about 5.6 kg. However, the thermal neutron flux level in the absorber region is about 70% that of the fuel region. Consequently, the effective mass of stainless steel used in the activation calculations was correspondingly reduced to 4 kg to account for the flux difference between the regions. The stainless steel in the BPRs was assumed to have a nominal cobalt impurity level of 800 ppm, a value associated with older assembly designs. The cobalt mass was therefore equal to 3.2 g cobalt per MTU of fuel (0.0032 g/kg U). This value is actually less than the estimated 0.075 g/kg U cobalt level associated with older PWR assemblies from inconel grid spacers and other assembly hardware.

The neutron and gamma dose rates for the concrete cask design were calculated using the different burnable poison assembly models for a uniform fuel enrichment of 4 wt %, burnups of 20 and 60 GWd/t, and cooling times of 5 and 100 years. These results are compared with the dose rates for a reference fuel design with no burnable poisons (results for the Westinghouse 17 × 17 design listed in Table 6) in Figure 15. All calculations assumed a uniform uranium mass per assembly. The variations in the dose rate are due entirely to the spectral effects caused by burnable poison exposure during the irradiation, with the exception of the borosilicate glass BPRs which are calculated with and without the contribution of activated cobalt to the gamma dose rate. These latter calculations indicate that the level of cobalt associated with the borosilicate glass rods does not have a significant effect on the gamma dose rates. However, the results will be strongly dependent on the cobalt level in activated hardware and the number of rods stored in the cask assemblies.

Figure 15 plots the ratio of the dose rates for assemblies with burnable poisons to the reference assembly without burnable poisons. The results indicate that the impact of the IBAs is greatest on the neutron dose rate. This effect is most pronounced at the lower burnup. However, at low burnup the relative importance of the neutron dose rate component is typically small. At higher burnup, the effects of the assembly poison are negligible and the neutron dose rates are very similar to the reference assembly with no exposure to burnable poisons. The total dose rate, which is typically dominated by gamma rays, is about 2% higher for the IBA assemblies compared to the reference fuel with no burnable poisons. The BPRs have a slightly larger effect, particularly at longer cooling time. In all cases, the dose rates increased by no more than about 10% for any of the assemblies considered.

3.4.2.3 Uranium Mass

The total uranium mass of a storage cask depends on the number of assemblies that can be loaded into a particular cask and the uranium mass of each assembly. Variations in the uranium mass for a given cask design may result from the variations in the uranium mass for the range of assembly types approved for the cask. An increase in the uranium mass per assembly results in a larger radiation source term. This is offset to some extent by greater self-attenuation of the source due to the larger amount of uranium that acts as a self-shield. Increasing the uranium mass also has a small effect on the neutron subcritical multiplication factor of the cask that may increase the secondary fission source.

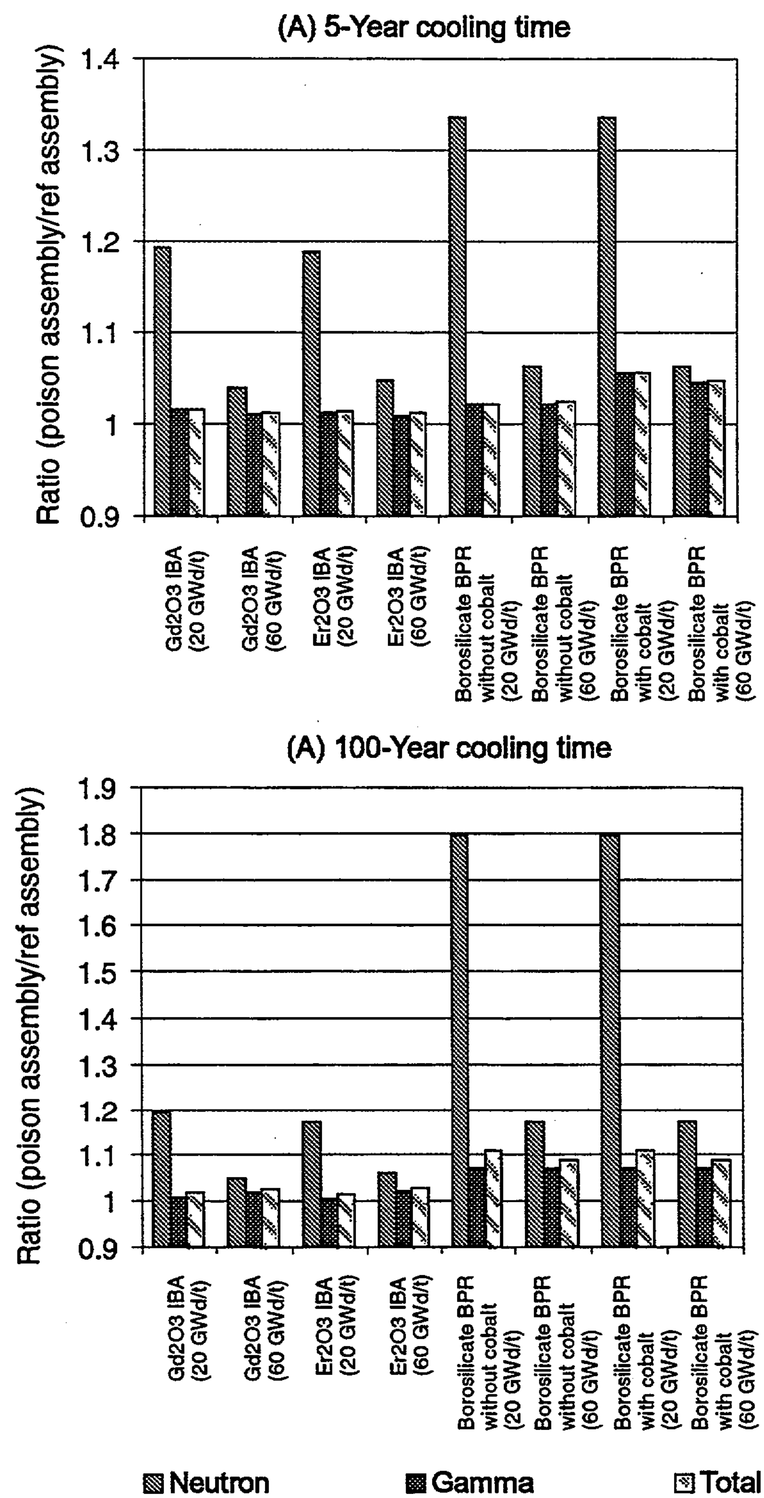


Figure 15 Ratio of the neutron, gamma, and total dose rate for assemblies with and without burnable poisons

The impact of variations in assembly uranium mass on the radiation dose rates was studied for the concrete and steel type casks. The variation in uranium mass was obtained from a U.S. Department of Energy (DOE) database[9] containing assembly design and irradiation history information for all spent fuel assemblies discharged from commercial operating reactors in the United States through 1998. The database contains the uranium mass for over 130,000 individual assemblies (approximately 76,000 BWR assemblies and 56,000 PWR assemblies) currently in storage.

The distribution of uranium mass for PWR and BWR assemblies is illustrated in Figure 16. Note that although the variation in mass is quite large when all assemblies are considered, most assemblies reside in a relatively well-defined mass range. The PWR assembly outliers include assemblies from Yankee Rowe (most assemblies below 300 kg U) and South Texas Units 1 and 2 (assemblies greater than 500 kg U) reactors. For the BWR fuel assemblies, the outliers (below 150 kg U) include mainly assemblies from Dresden 1, La Crosse, and Big Rock Point reactors.

The uranium mass variation for the large majority of assemblies is approximately ±20% (about the mean) for the PWR assemblies and approximately ±10% for the BWR assemblies currently in storage. The variation is much larger if the outlier assemblies were to be included in the distribution. However, the only group of assemblies that have a significantly larger mass than the mean, and hence could result in dose rates well above the mean, are roughly 800 assemblies from the South Texas PWR reactors.

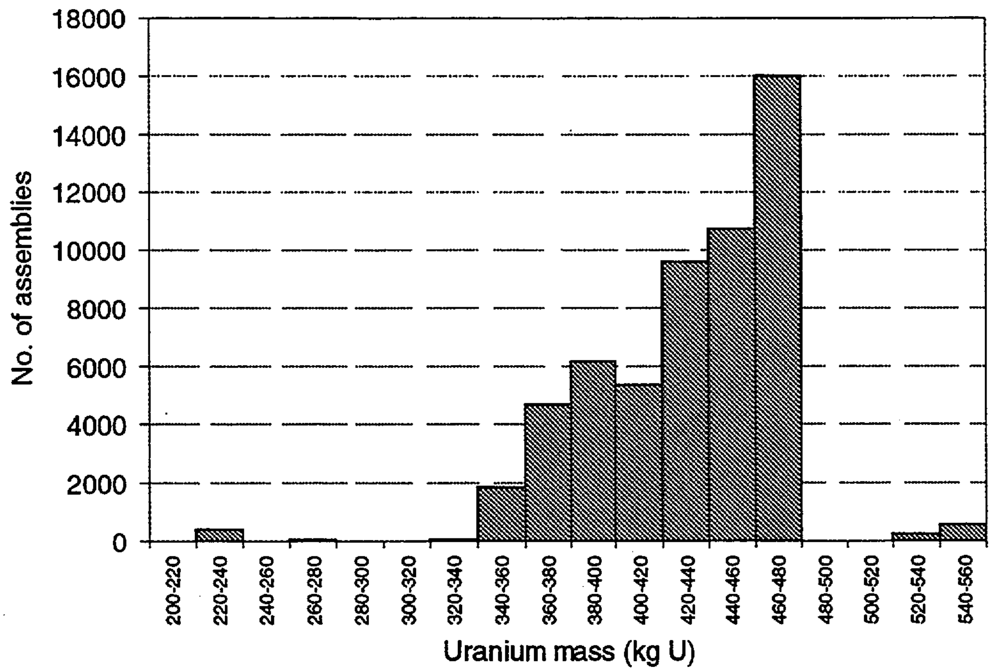
The variation in the dose rate, as a function of the uranium mass in the casks, was calculated for the neutron and gamma dose rate components by increasing the reference uranium mass by 10%. The maximum increase in the neutron dose rate for both the concrete and steel casks was about 6%, while the gamma ray dose rate increased by about 2%. These values were largely independent of enrichment and burnup. The maximum change in the total dose rate for either cask design was only about 3% due to domination of the gamma dose rate for the cask designs considered.

When these results are combined with the typical variation in the uranium mass for different assembly designs (maximum of ±20% for PWR assemblies), the expected variation in the cask surface dose rate is less than ±10% for any cask design. For BWR assembly designs, which exhibit lower relative variability, the dose rate variation will be lower. For concrete storage casks, which typically have a lower neutron dose rate component compared to a steel case design, the variation will also be lower (< ±5%) since the gamma dose rate has a lower sensitivity to uranium mass than the neutron dose rate.

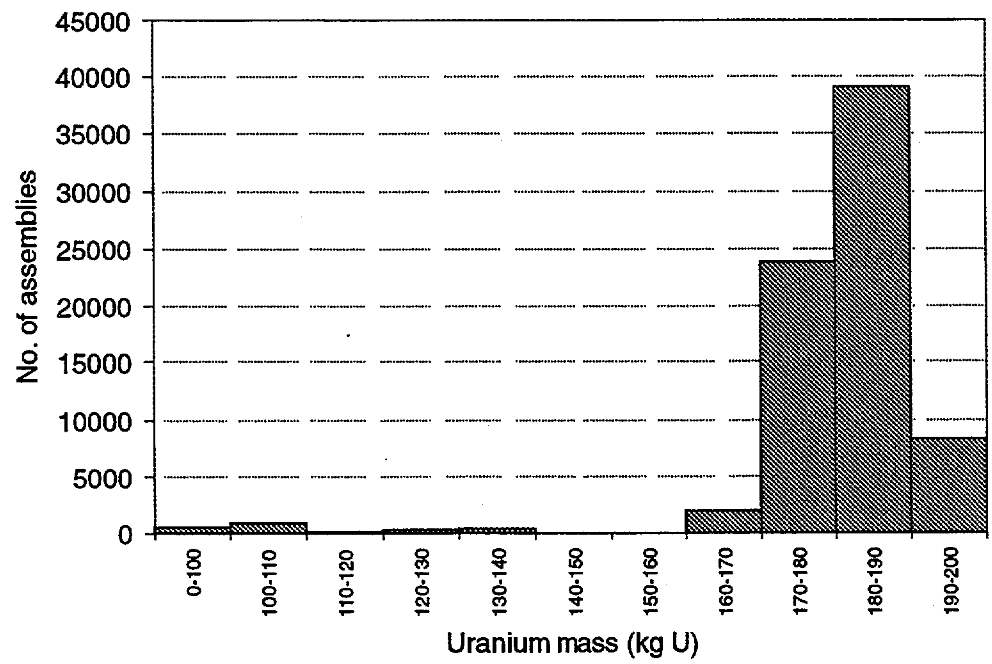
3.4.2.4 Specific Power

The specific power of the fuel during irradiation is typically not included in fuel specifications. The predominant effect of the specific power (particularly near the end of life) is on the short-lived fission product inventory, which affects the radiation source term and decay heat power. Variations in specific power may occur as a result of many factors such as reactor operating history, changing assembly position in the core, and axial power variations in an assembly.

The effect of specific power on the cask dose rates was investigated for 3.5 wt % PWR fuel irradiated to 40 GWd/t using specific powers of 20, 30, and 40 MW/t. The neutron and gamma dose rates were compared at cooling times from 5 to 100 years. The results confirmed that the neutron dose is largely unaffected by the specific power level. The gamma dose rate increases with higher specific power, with the effect most pronounced at short cooling times. For 5-years cooling, the gamma dose rate increased by about 30% using a specific power of 40 MW/t compared to 20 MW/t. After a 10-year cooling time the increase was 7%, and after 20 years the effect decreased to only 3%.



(A) PWR Assemblies



(B) BWR Assemblies

Figure 16 Distribution of uranium mass for currently discharged PWR (A) and BWR (B) fuel assemblies. The distributions reflect the actual inventory of spent fuel assemblies through 1998 from commercial reactor operation.

3.4.2.5 Moderator Density

Reactor moderator density is another parameter not typically included as a shielding technical specification. The relative importance of the moderator density in hardening the neutron spectrum and influencing the SNF inventories was assessed for a range of moderator densities associated with axial variations in BWR assemblies (typically 0.3 – 0.7 g/cm^3). The calculations assumed fuel with 4 wt % and 40 GWd/t. The neutron and gamma dose rates were observed to increase with decreasing water moderator density. The neutron dose rate was found to vary by more than 30% over the moderator density range, while the gamma dose rate varied by about 10%. The net impact of moderator density on cask shielding is expected to be low for PWR fuels. However, the axial variation in moderator density in BWR reactors can have a measurable effect on the axial dose rate profile on a cask, and increase the dose rate near the top of the assemblies where the moderator density is lowest.

3.4.2.6 Fuel Cladding Material

The impact on the dose rates from loading assemblies with stainless steel fuel cladding was estimated by scaling the results for the assemblies containing borosilicate glass and stainless steel BPRs (12 BPR rods per assembly). The amount of stainless steel associated with the BPR rod design studied is about one half that of a fuel rod with steel cladding. Therefore, the dose rate results for the borosilicate glass absorbers were divided by the ratio of the number of fuel rods to BPR rods (12) and scaled by a factor of two. For example, in a 16 × 16 design with 228 fuel rods, the dose rates for the assembly containing 12 BPR rods (with stainless steel) were scaled by a factor of roughly 40 to estimate the dose rate for an assembly containing 228 fuel rods clad with stainless steel. The potential impact on the gamma dose rate would be very large for the cooling times during which ^{60}Co is the dominant gamma ray source (up to about 50 years). The steel clad fuel potentially increases the cask dose rate by more than an order of magnitude over that from conventional Zircaloy clad fuel.

3.4.3 Combined Burnup, Enrichment, and Cooling Time Effects

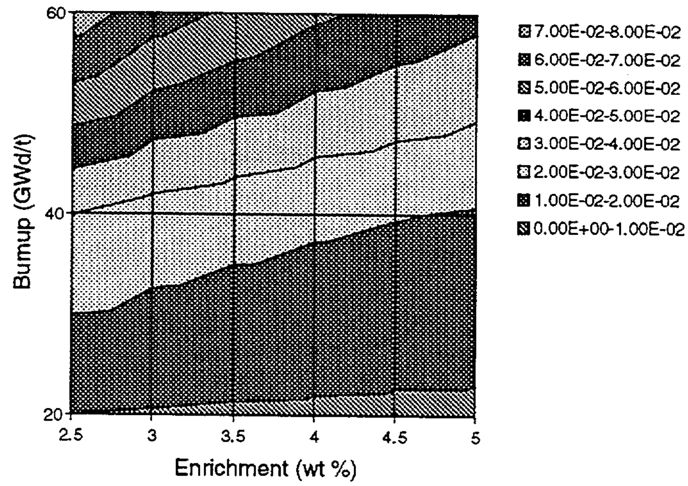
The discharge burnup and cooling time are clearly dominant parameters for shielding, and established combinations of burnup and cooling time for acceptable spent fuel assemblies have been used previously in technical specifications. The neutron source exhibits a rapid increase with burnup, which is most pronounced when fuel is irradiated well beyond a typical discharge burnup as determined by the initial enrichment. To limit the increase in the neutron dose rate, technical specifications have sometimes imposed a minimum fuel enrichment.

The variation of the neutron and gamma dose rate for combinations of burnup *and* enrichment was investigated for a concrete storage cask and is shown in Figure 17. The contour plot lines indicate the regimes (enrichment/burnup combinations) of constant dose rate. For example, the neutron dose rate for spent fuel with 4.5 wt % and 50 GWd/t is seen to be nearly equal to 2.5 wt % and 40 GWd/t. The figure illustrates the rapid increase in the neutron dose component in the overburned (low enrichment and high burnup) region. The large gradient in the high burnup region (rapid rate of change) indicates that the sensitivity of the dose rate to changes in enrichment will be significantly greater for high burnup fuel than low burnup fuel. That is, a given change in burnup, for example, will have a much larger affect on the dose rate for high burnup fuel than for low burnup fuel. The figure also illustrates that combining enrichment and burnup as a single parameter may provide a basis for a less restrictive and more general technical specification involving the correlated parameters of enrichment, burnup, and cooling time. Such an approach would be similar to the loading curve proposed for burnup credit whereby only spent fuel with a specified enrichment *and* burnup would be acceptable for loading.

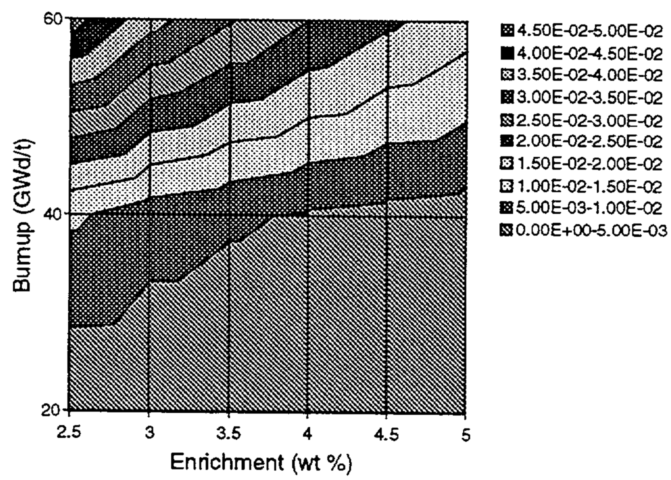
As an example, loading could be restricted to spent fuel with a minimum enrichment-to-burnup ratio instead of imposing a minimum enrichment alone. This might permit low enrichment fuel to be loaded in the cask if there was a commensurate reduction in the burnup of the assembly. To illustrate such an approach, consider a concrete storage cask licensed for 5-year-cooled fuel (Figure 17). If the shielding analysis for the cask demonstrated spent fuel with 5 wt % and 60 GWd/t was acceptable for loading, it can be seen that any fuel with an enrichment-to-burnup ratio greater than or equal to the licensing basis fuel (e.g., 3 wt % up to 36 GWd/t) would generate a dose rate less than the safety analysis value.

A less restrictive prescription that provided a more realistic representation of the dose rate dependence would allow a greater range of assemblies to be safely loaded in a storage cask. Such a loading curve, however, would need to consider the specific cask design since the neutron and gamma dose rate behavior is significantly different (e.g., the gamma dose rate varies more slowly than neutron dose rate with changes in enrichment).

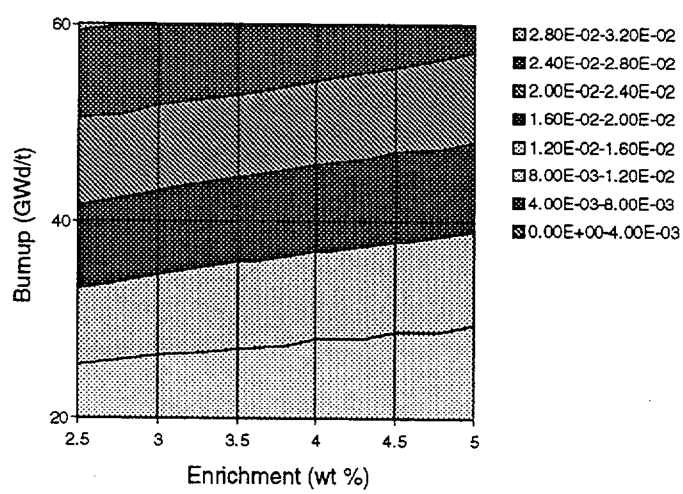
The changing neutron and gamma dose rates for the concrete storage cask for different burnup and cooling time combinations are illustrated in Figure 18. The total dose rate for the cask is dominated by gamma rays (i.e., neutrons contribute less than 25% of the total for high burnup fuel and long cooling times) and; therefore, the total dose rate profile is dominated by the gamma ray behavior.



(A) Total Dose Rate (arb. units)

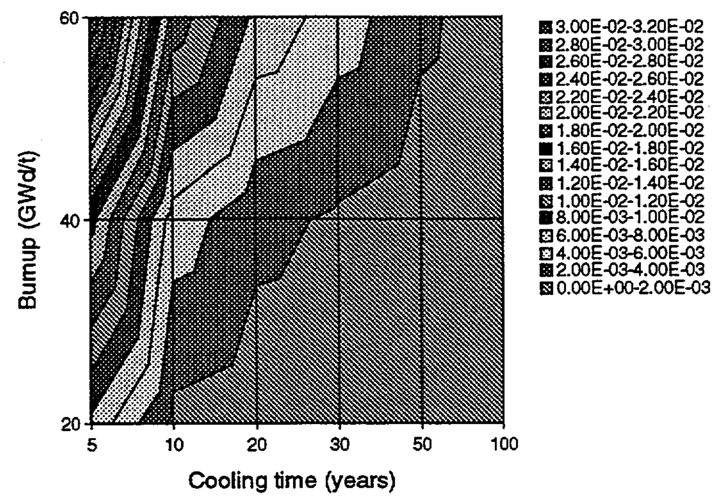


(B) Neutron Dose Rate (arb. units)

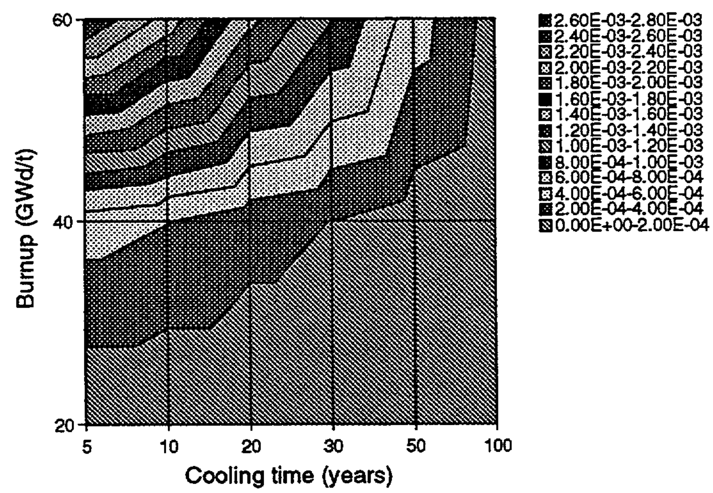


(C) Gamma Dose Rate (arb. units)

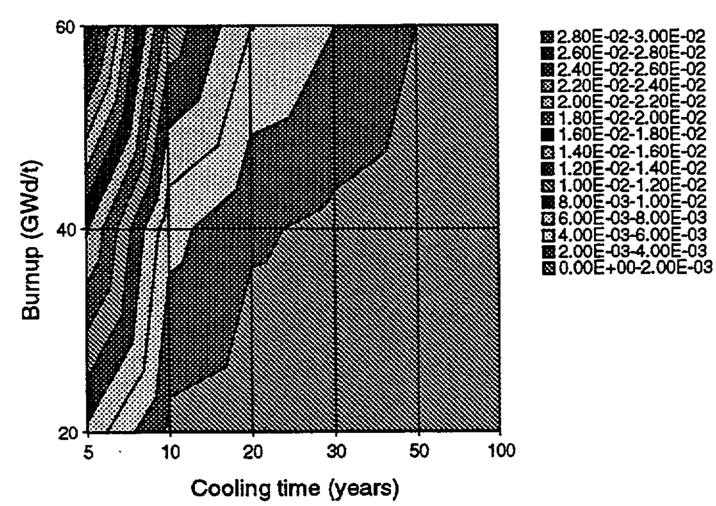
Figure 17 Enrichment-burnup dose rate contour plots for concrete storage cask, 5-y cooling



(A) Total Dose Rate (arb. units)



(B) Neutron Dose Rate (arb. units)



(C) Gamma Dose Rate (arb. units)

Figure 18 Burnup-cooling time dose rate contour plots for concrete storage cask

3.5 Recommendations

A summary of findings is given in Table 8. The reported variations in the table are presented for a concrete storage cask design, and may be somewhat different for alternate cask designs that different shielding materials and/or have a larger relative neutron dose rate component. The variations in the dose rates reflect the limited range of parameter variations and fuel assembly designs considered here. They are not intended to cover the full range of spent fuel and assembly designs used in commercial PWR and BWR reactors. These results are intended only to provide a rough guide to the importance of the respective candidate fuel specifications as they affect typical storage cask dose rates to allow judgments to be made on the relative importance of the parameters.

The parameters having the largest impact on the dose rate are burnup and cooling time. In addition, assemblies containing significant quantities of stainless steel (e.g., steel cladding or BPR rods containing steel) may also have a very large effect on the dose rates. The enrichment level is also observed to be important. The magnitude of the enrichment effect, like many of the parameter effects, is dependent on the relative neutron-to-gamma ray dose rate for the cask, since the gamma ray dose rate exhibits a smaller sensitivity to changes in enrichment.

Assemblies with integral burnable absorbers were found to have dose rates similar to those for nonpoison assemblies. Larger effects were observed for assemblies exposed to burnable poison rods during irradiation, caused primarily by the spectral effects of displacing moderator in the assembly. However, these effects were most pronounced for the neutron-dose rate, and had only small effects on the total shielded dose rate for the cask. The largest potential effect from assemblies residing in a cask that contains irradiated BPR clusters is from activated component hardware (mainly activated cobalt in steel). For BPR designs containing stainless steel, the impact on the gamma dose rate can be large.

The uranium mass was found to be of intermediate importance to shielding. The variation in the mass for different assembly designs is typically less than ±20% and results in a maximum variation in the cask exterior surface dose rate of less than 10%. The variation in the dose rate due to typical variations in fuel assembly type, various integral burnable poison designs, BPR exposure, moderator density, and specific power were all found to be less than about 10% over most of the parameter range studied.

Table 8 Summary of fuel specification study finding

Technical specifications (candidate)	Parameter range studied	Observed variation[a]	Bounding value
Initial enrichment (wt % ^{235}U)	2.5 – 5.0	100%	Minimum enrichment
Cooling time (years)	5 – 100	> 100%	Minimum cooling time
Assembly burnup (MWd/t)	20 – 60	> 100%	Maximum burnup
Assembly/cask uranium mass (kg U)	mean ± 20 %	20%	Maximum mass
Fuel assembly type (no burnable poisons)	ABB-CE 14 × 14, W 15 × 15, W 17 × 17	5%	17 × 17 design
Integral burnable poison rods (IBAs)	5 wt % Gd_2O_3, 2 wt % Er_2O_3	5%	Maximum poison loading
Burnable poison rods (BPRs)	12 borosilicate glass and stainless steel rods	10%	Maximum poison loading and cobalt level (if applicable)
Assembly structural materials	stainless steel cladding	> 100%	Maximum cobalt level
Moderator density (g/cm^3) – BWR	0.3 – 0.7	10%	Minimum moderator density
Specific power (MW/t)	20 – 40	10% after 5 years < 5% after 10 years	Maximum specific power

[a] Approximate maximum variation in total dose rate expressed as percentage difference over full range = (max/min – 1) × 100, concrete storage cask design

4 SUMMARY

The use of STS for spent fuel storage casks would benefit both the NRC and licensees because of the reduced personnel time and costs associated with the reduction in license amendment submittals. Applicants should perform safety analysis for unique fuel types (i.e., array size, number of fuel rods, cladding material) and develop bounding specifications based on the safety analysis that include all parameters important to safety. These specifications should be defined in the FSAR. The safety analysis should include justification for the exclusion of any parameters from the FSAR. Fuel assemblies that satisfy the STS, but are not explicitly covered in the FSAR (e.g., one or more parameters are outside of the defined range in the FSAR), could be added using the Section 72.48 process.

As stated, this type of approach does not enable the addition of new *fuel types* (i.e., assemblies with differing array sizes, numbers of fuel rods, or number and location of guide tubes) or increases in enrichment limits. Consequently, the burden still rests on the applicants to provide general analyses that consider the unique fuel types that may potentially be stored in their cask(s).

This report has studied the important parameters that influence criticality safety and radiation shielding doses. The following parameters are recommended for inclusion in the STS.

1. Fuel type
    a. Array size, number of fuel rods, including number of partial length rods (where applicable), and cladding type
    b. Number and material of guide and instrument tubes;
2. Enrichment (maximum for criticality safety, minimum for radiation shielding);
3. Maximum burnup;
4. Minimum cooling time;
5. Maximum uranium mass; and
6. Maximum cobalt level.

Other criticality safety parameters (pitch, pellet OD, clad thickness, clad OD, guide tube and water rod thickness, and fuel stack density) could be updated via the Section 72.48 process. Likewise, radiation shielding parameters (e.g., maximum poison loading, minimum moderator density (BWR), and maximum specific power) could be updated via the Section 72.48 process.

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11. ABSTRACT *(200 words or less)*

The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing the technical specifications for spent fuel storage casks in an effort to develop standard technical specifications (STS) that define the allowable spent nuclear fuel (SNF) contents. One of the objectives of the review is to minimize the level of detail in the STS that define the acceptable fuel types. To support this initiative, this study has been performed to identify potential fuel specification parameters needed for criticality safety and radiation shielding analysis and rank their importance relative to a potential compromise of the margin of safety.

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