Criticality Safety Computations for Spent CANDU Fuel in a Deep Geologic Repository

NWMO TR-2014-08

February 2014

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This report has been prepared under contract to NWMO. The report has been reviewed by NWMO, but the views and conclusions are those of the authors and do not necessarily represent those of the NWMO.

Document History

Title:	Criticality Safety Computations for Spent CANDU Fuel in a Deep Geologic Repository		
Report Number:	NWMO TR-2014-08		
Revision:	R000	Date:	February 2014
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Accepted by:	N. Hunt		

ABSTRACT

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Date: February 2014

Abstract
The purpose of this study is to conduct a literature search, define a series of criticality scenarios (based on the literature search results), and calculate the safety margin to criticality for bounding operational and postclosure scenarios and configurations, relevant to surface handling and storage of spent CANDU fuel in deep geologic repository (DGR).

Literature review and scenario development produced 5 bounding scenarios encompassing a range of different container counts; fuel and container conditions; and materials inside and outside of the container. Bounding scenarios include:

1. A single intact container with intact fuel geometry, bentonite-shielded, filled with water (flooded), and surrounded by rock.
2. A single intact container, with degraded fuel geometry, bentonite-shielded, filled with water (flooded), and surrounded by rock.
3. A single degraded container, with degraded fuel geometry, bentonite-shielded, where radionuclides have been released into the bentonite and rock surrounding the bentonite.
4. Radionuclides are released from multiple degraded containers (with degraded fuel geometries) into the surrounding rock (far field).
5. Calculation of critical volumes and masses for mixtures of plutonium in water. This scenario assesses plutonium criticality when the fissile materials are released from a container, mix with water, migrate, and potentially accumulate.

Conservative burnup and cooling (decay) times were determined from results of the literature review (where possible) and confirmed by initial benchmark criticality calculations using MCNP. Criticality calculations were completed for intact-container and degraded-container scenarios, corresponding to the 5 scenarios, and conservative K_{eff} values were obtained. To assess scenarios involving radionuclides released into surrounding bentonite or rock, minimum mass and minimum volume spheres were back-calculated for varying densities and masses. Criticality volumes were determined, as a function of density, for crystalline and sedimentary rock types.

Overall, for intact or failed containers, it was found that criticality is not possible. For very unlikely scenarios in which plutonium is released from container(s) and assumed to accumulate within bentonite or within void space in rocks, the amounts required to reach critical mass were calculated. Multiple containers must fail, releasing plutonium, which then must migrate to the same region and accumulate without other nuclides, in order to reach critical mass.

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1. INTRODUCTION

1.1 BACKGROUND AND PURPOSE

Nuclear criticality requires a sufficient concentration and critical mass of fissile isotopes, the presence of moderators in a favorable geometry and the lack of neutron absorbers.

Due to its lack of enrichment, criticality of standard CANDU fuel cannot be achieved outside of a nuclear reactor where it is maintained in a defined configuration surrounded by heavy water coolant and heavy water moderator. Post discharge, inadvertent criticality is similarly not anticipated due to the lack of heavy water, depletion of fissile isotopes and the accumulation of neutron absorbing fission products and actinides.

The purpose of this study is to investigate the potential for criticality for a set of bounding operational and postclosure scenarios and configurations relevant to a deep geologic repository (DGR) for spent CANDU fuel. Consistentency with the requirements of CNSC RD-327 Nuclear Criticality Safety (CNSC 2010) is discussed in Section 4.1.

1.2 LIMITS AND APPLICABILITY

The results of this report are based on material characteristics and design information provided by NWMO. The calculations performed here are specific to the container designs and storage configurations proposed by NWMO and are subject to change as the repository design evolves.

2. LITERATURE REVIEW: INTERNATIONAL EXPERIENCE WITH CRITICALITY FOR NUCLEAR WASTE MANAGEMENT PROGRAMS

As part of the Criticality Safety Computations Project, a literature review was conducted using readily-available information on criticality-related design basis conditions and accidents considered in other international waste management programs. The review encompasses topics such as above-ground on-site transport and storage, operations within the packaging plant, transfer and placement with the repository, and the postclosure period.

It is understood that the lack of enrichment in standard CANDU fuel results in a much lower (or zero) risk of criticality (lower reactivity), and therefore, the literature search focuses particularly on scenarios that may be relevant for natural, non-enriched fuel. However, many circumstances that are typically being analyzed for enriched fuels may still be relevant to the project, as small amounts of enriched fuels e.g., from Canadian research programs, may be handled by the NWMO.

A list of reference documents reviewed during the literature search – along with their summaries or abstracts where available – is provided in Appendix C.

The issue of potential criticality in high level waste disposal sites has been the subject of several studies in the past few decades, and there are numerous reports on criticality safety of spent nuclear fuel in storage. Following Fukushima, there has also been increased interest in beyond-design-basis accidents involving spent fuel. However, a large portion of the information deals with enriched fuel and/or criticality assessment under normal operating conditions and design-basis accidents.

Studies available in literature were found to cover a range of topics including, for example:

- In-container criticality;
- Onsite container criticality;
- Near-field and far-field criticality;
- Container design;
- Burnup credit;
- Geochemical aspects;
- Plutonium criticality; and,
- Uranium criticality.

Summaries of information and case studies from international radioactive waste management programs are provided in the following subsections.

2.1 ATOMIC ENERGY OF CANADA LIMITED

The most relevant report for the current project regarding CANDU fuel is the 1994 report by Atomic Energy of Canada Limited (AECL) titled *The Disposal of Canada's Nuclear Fuel Waste: The Vault Model for Postclosure Assessment (AECL 1994)*. One of the supporting studies used in the AECL report, was a study by McCamis (1992) which considered criticality calculations for CANDU fuel.

The McCamis (1992) study included calculations for:

- A spent fuel container with intact fuel bundles;
- A spent fuel container with distributed fissile materials;
- The critical radius of spherical plutonium (Pu) solutions as a function of concentrations.

These calculations provided the starting point for a criticality assessment of:

- A flooded container;
- A container failure with material released inside the container;
- Release of Pu-239 from a failed container; and
- Release of Pu-239 from multiple failed containers (AECL 1994).

The McCamis (1992) calculations concluded that criticality of used CANDU fuel was not an issue with undue risk.

2.2 SKB (SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT COMPANY)

SKB (1996) Criticality in a high level waste repository

SKB (1996), *Criticality in a high level waste repository,* investigates the conditions and scenarios that might allow for criticality of spent nuclear fuel in waste repository, with a focus on geochemistry. SKB (1996) notes that based on a 12-bundle spent fuel canister arrangement, for spent fuel with 35 MWd/kgU or more, criticality cannot be achieved inside the canister even if the void space is filled with water (a worst-case scenario). For any in-repository criticality scenario to be possible, the canister must be breached in a way that allows water to enter and flood the canister.

The SKB (1996) study focuses on 4 scenarios, all of which involve flooding of the container. It was found that:

- Criticality due to Pu inside the canister is not possible, due to lack of sufficient Pu within a single container, and due to lack of space to allow for accumulation of water to moderate neutron energies.
- Criticality due to Pu outside the container is not possible, due to lack of mechanism to dissolve and transport the Pu out of the container and also accumulate the Pu in the necessary geometry under reasonable repository conditions.
- Criticality due to U inside the container is not possible, due to the use of low solubility materials to fill the space inside the container and therefore exclude water from entering and moderating neutron energies.
- Criticality due to U outside the container would require dissolution and transportation under oxidizing conditions, followed by deposition of U under reducing conditions. There is no credible mechanism to achieve both of these conditions in the near-field host rock.

SKB (1999) Postclosure Safety Assessment

In 1999, the Swedish Nuclear Fuel and Waste Management Company (SKB) released Technical Report TR-99-06 (Volumes I and II), a post-closure safety assessment for a proposed deep geologic repository for spent nuclear fuel. Volume 1 notes that SVEA-64 (PWR) and FA17x17 (BWR) spent nuclear fuel are the most unfavourable with regard to criticality, and are dealt with in Volume II which provides mostly qualitative investigations of criticality for BWR fuel (SVEA-64 with a mean enrichment of 3.6%) and PWR fuel (FA17x17 with a mean enrichment of 4.5%). Investigations focus on postclosure with the following assumptions:

- Spent fuel is assumed to be placed in disposal canisters;
- The disposal canisters are assumed to be placed in the DGR;
- The disposal canisters are assumed to be flooded with water; and,
- The disposal canisters are assumed to be surrounded by bentonite and then rock.

The influence of burn-up and fuel enrichment percentage is investigated using a limit curve and fuel data for both PWR and BWR fuel. The graphs show that current (as of November 1999) combinations of burn-up and fuel enrichment produce Keff values that are all below 0.95. From this, it is concluded that criticality is unlikely.

Note that a more recent SKB criticality assessment is available for these fuel types (SKB 2002; TR-17-02). A summary of SKB (2002) is provided later in this section.

Over the long-term, SKB (1999) notes that although reactivity rises due to the decay of actinides, reactivity in the repository is never greater than it is for the fuel composition within 40 days after operation. As such, all estimates are based on inventories at that time.

SKB (1999) also investigates the long-term effects of corrosion on reactivity. It was found that with time, corrosion of the canister insert could reduce the dimensions of the fuel channels due to buildup of corrosion products. Reactivity was found to decreases sharply if the channels are filled with corrosion products and the fuel is intact. Based on the most reactive geometry, the resulting Keff values are 0.7 for BWR and 0.65 for PWR.

Finally, the possibility of local accumulation of fissile material is briefly discussed. SKB (1999) concludes that the probability of a local accumulation of critical mass is low, and that even if one should form, the consequences would be small.

SKB (2002) Criticality Safety Calculations of Storage Canisters

SKB (2002), *Criticality Safety Calculations of Storage Canisters (SR97),* considers criticality of SVEA-64 (PWR) and FA17x17 (BWR) spent nuclear fuel. These fuel types are indicated in the SKB (2006) Safety Assessment (see above) as being the most unfavourable with regard to criticality.

Variants of canister design, in-container configurations, fuel types, canister location, and temperatures were investigated. Criticality calculations were performed using Scale CSAS25 with depletion calculations using Scale SAS2H. All calculations are initially conducted for fresh fuel, with select cases re-calculated to take credit for the burnup of the fuel. Findings were as follows:

- **Canister Design:**
  - BWR: 12-compartment arrangement
  - PWR: 4-compartment arrangement
  - Canisters are made of cast iron with an outer copper shell.
- **Fuel Assembly Type:**
  - BWR – Svea 64
  - PWR – F17x17
- **Underground Storage:**
  - Worst Case: assuming canisters are leaking and become filled with water. With the canister filled with water, Keff values of 1.055 and 0.905 were found for PWR and BWR, respectively.
- **Dry Storage:**
  - Worst Case: assuming canisters are filled with water, but the space between the canisters consists of air, yields Keff values of 1.0868 and 0.9242 for PWR and BWR, respectively.
- **In-Canister Arrangement:**
  - Worst Case: all fuel assemblies are located in a radial arrangement toward the centre of the canister. See Figure 2-1 below.

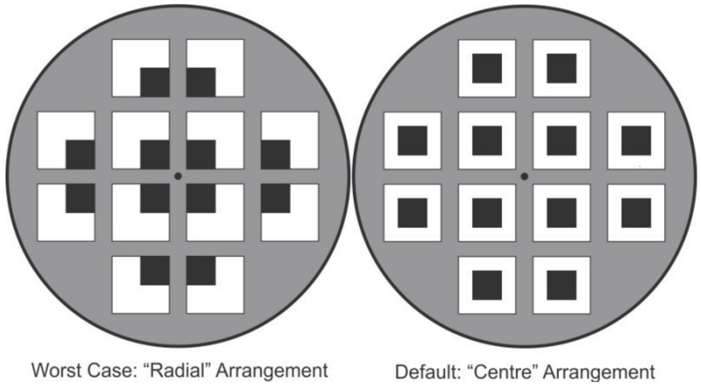

Figure 2-1: Fuel Assembly Arrangement

SKB (2010) Criticality Safety Calculations of Disposal Canisters

SKB (2010), *Criticality Safety Calculations of Disposal Canisters,* considered criticality of spent nuclear fuel in disposal containers of cast iron and copper. Variants of in-container configurations, fuel types, material compositions, and temperatures were investigated. Calculations were performed with fresh fuel with an initial enrichment of 5% U-235. Key findings were as follows:

- **Reactivity related to location of canister**
  SKB 2010 conducted scoping calculations assuming 5% fuel enrichment, corresponding to the maximum existing and planned enrichment encountered within the Swedish program. The results are as follows:
  - Encapsulation Plant:
    - Higher Keff values [1.0872 (PWR) and 0.9942 (BWR)] are seen if the canister is filled with water, and, room is filled with water; as opposed to if only the canister is filled with water [1.0860 (PWR) and 0.9926 (BWR)].
  - Storage Room:
    - Disposal canisters (dry, argon-filled) placed inside of transport casks (dry). Keff is less than 0.4 for PWR and BWR for an infinite number of canisters in a transport cask.
  - Transport:
    - Worst case for reactivity: cask is damaged in an accident, cask is submerged in water, and both the cask and the disposal canister are filled with water [1.0888 (PWR) and 0.9951 (BWR)].
  - Disposal:
    - Canister surrounded by 35 cm bentonite, placed in rock. Assuming the canister has become filled with water: 1.0888 (PWR) and 0.9951 (BWR).
      Note:
      From this, calculations were repeated using decreasing enrichment until the resulting Keff values were below the 0.95 criterion (with allowance for calculation uncertainties). This was achieved at an enrichment of 2.4% for PWR and 3.5% for BWR. These correspond to the highest enrichment that can be stored without burnup credit.
    - Investigation of the interactions between 2 canisters both placed in rock shows insignificant interaction.

2.3 POSIVA OY (FINNISH NUCLEAR WASTE MANAGEMENT AUTHORITY)

Posiva Oy (POSIVA) (2005) investigates the potential for criticality to occur among disposal canisters of spent fuel. Disposal canisters are comprised of cast iron and copper. Three container types are considered in the assessment: VVER 12-canister design, BWR 12-canister design, and EPR-type 4-canister design. Calculations are performed for canisters filled with air (void) or water, and surrounded by air or water. The overall findings show that the worst case arrangement is a canister filled with water, surrounded by air, with a resulting Keff value of 0.9441. Canisters filled with water and also surrounded by water (submerged) are slightly less reactive, with a resulting Keff value of 0.9112. The lowest reactivity is shown by canisters filled with air yet surrounded by water, with a resulting Keff value of 0.2535. These trends hold true for all 3 container types, though individual Keff values vary by container type, with the EPR container producing the highest Keff values (1.0215, before burnup credit is applied). Burnup calculations were performed for the EPR container determining that a minimum burnup of 20 MWd/kgU is necessary in order to meet the criticality safety criterion.

POSIVA (1999 and 1996) also outline criticality calculations for different disposal canister designs containing spent fuel. Calculations are performed for individual (isolated) canisters filled with air (empty) or water, and for an infinite lattice. Boundary conditions were based on containers (or single container, depending on the scenario) in a vacuum, without any reflector. The overall findings show that the worst case arrangement is a canister filled with water as opposed to being filled with air (empty).

2.4 UNITED STATES DEPARTMENT OF ENERGY (US DOE)

Scaglione & Wagner 2011: Review of Yucca Mountain Disposal Criticality Studies

Scaglione and Wagner (through Oak Ridge National Laboratory [ORNL], under contract to the U.S. Department of Energy [US DOE]) completed a criticality study related to the Yucca Mountain Project (YMP). The United States (US NRC) regulation 10 CFR 63 (2004) requires criticality at a disposal site to have a probability of occurrence of less than 1-in-10,000 within 10,000 years of disposal. Scaglione & Wagner (2011) calculated the total probability of a criticality event during the disposal time period and compared it against the regulatory criterion in 10 CFR 63. The total probability of criticality included contributions associated with both internal (within the waste package) and external (external to the waste package) criticality. Despite numerous and significant conservative analysis assumptions in the event sequences requisite to enabling criticality, the probability of nuclear criticality during the post closure performance period was estimated at 4.4×10^{-5} per 10,000 years of operation.

The Scaglione & Wagner (2011) study assumes that for in-package criticality to be possible, all of the following conditions must be met:

- Waste package damage (barriers breached);
- Presence of a moderator (i.e., water); and,
- Materials inside the package must degrade and/or reconfigure (e.g., separation of fissionable material from the neutron absorber material, or lack of absorber material).

Scaglione & Wagner (2011) also assumes that external criticality requires:

- The same conditions for in-package criticality; plus,
- Sufficient accumulation of fissile material in a critical configuration (critical mass).

Bechtel (2004;2008): Screening Analysis & Preclosure Criticality Analysis Reports

In two separate studies prepared for the US DOE for the YMP, Bechtel (2004; 2008) conducted a comprehensive assessment of criticality for the following scenarios:

- In-package criticality (intact configuration);
- In-package criticality (degraded configuration);
- Near-field criticality;
- Far-field criticality;
- In-package criticality resulting from a seismic event (intact configuration);
- In-package criticality resulting from a seismic event (degraded configuration);
- Near-field criticality resulting from a seismic event;
- Far -field criticality resulting from a seismic event;
- In-package criticality resulting from rock fall (intact configuration);
- In-package criticality resulting from rock fall (degraded configuration);
- Near-field criticality resulting from rock fall;
- Far -field criticality resulting from rock fall;
- In-package criticality resulting from an igneous event (intact configuration);
- In-package criticality resulting from an igneous event (degraded configuration);
- Near-field criticality resulting from an igneous event; and,
- Far -field criticality resulting from an igneous event.

The Bechtel (2004) assessment follows the U.S. methodology where criticality scenarios are defined, their probabilities of occurrence are determined and compared to a screening probability criterion, and criticality calculations are performed only for those that exceed the screening criterion. The Bechtel (2004) study presents a comprehensive evaluation of the conditions and probabilities associated with each scenario, and concluded that no scenario exceeds the probability criterion, and therefore no criticality calculations were necessary. As such, the report contains no actual criticality calculations.

Bechtel (2008) provides a high-level overview of the Preclosure Criticality Analysis Process methodology, developed to evaluate the preclosure period of the Monitored Geologic Repository at Yucca Mountain, Nevada. It also outlines regulatory and other requirements from the project. The Bechtel (2008) process essentially involves 5 stages: initiating event identification; event sequence analysis; consequence analysis; criticality analysis for select events; and, recommendation of safety controls to minimize risks. The report does not contain the application of this process (i.e., the actual criticality safety calculations).

Rechard et al. (1996): Consideration of Criticality when Directly Disposing Highly Enriched Spent Nuclear Fuel in Unsaturated Tuff: Bounding Estimates

A criticality study by Rechard et al. (1996), of Sandia National Laboratories' Nuclear Waste Management Programs Center, conducted calculations including the probability and consequences of a series of bounding criticality scenarios, and concluded that concerns about a criticality on the surface where humans can be exposed either directly from the event or indirectly from cleaning up contaminated material do not apply to conditions in a deep, closed, geologic repository. The Rechard et al. (1996) work was based in part on a study commissioned by the US DOE to guide the development of spent fuel technology at the Idaho National Engineering Laboratory; though both the YMP and WIPP projects are mentioned, the work is not specific to either. A later study of the YMP by Rechard *et al. (*2003) also concluded that criticality is unlikely to cause undue risk.

2.5 SWEDISH NUCLEAR POWER INSPECTORATE (SKI)

Hicks & Prescott (2000) conducted a study for criticality in a Spent Nuclear Fuel Repository (SNFR) based on its canister designs for Boiling-Water Reactor (BWR) and Pressurized-Water Reactor (PWR) spent fuel assemblies. The study makes extensive reference to the analyses conducted by Behrenz & Hannerz (1978) and by Oversby (1996) on the risk of criticality as a result of redistribution of material. Analyses by Hicks & Prescott (2000) involved determining the neutron multiplication factor for various disposal configurations, depending on the type of canister and fuel assemblies, the initial fuel enrichment, the amount of fuel burn-up, and the amount of burnable poison present. The study considered the following groups of the criticality scenarios:

- Plutonium criticality inside a canister
- Plutonium criticality outside a canister;
- Autocatalytic criticality;
- Uranium criticality in a tunnel;
- Uranium criticality in a deposition hole.

Overall, Hicks & Prescott (2000) reported that for criticality to occur following disposal of canisters containing typical irradiated BWR and PWR fuel, fissile material would need to become concentrated in a moderating environment in the repository and the possibility of criticality occurring anywhere in the repository would be low and that, even if criticality did occur, the consequences on safety would be insignificant.

Hicks & Prescott (2000) also concluded that for the criticality scenarios addressed, based on the generally large margins of sub-criticality determined in the analyses, it is unlikely that modifications to canister design (changes in the arrangements, dimensions, and numbers of spent fuel rods to be placed in each canister) will have a substantial effect on the results of the criticality assessment.

2.6 SWISS NATIONAL COOPERATIVE FOR THE DISPOSAL OF RADIOACTIVE WASTE (NAGRA)

The Switzerland National Cooperative for the Disposal of Radioactive Waste (NAGRA) released a technical report in 2005 titled *Project Opalinus Clay – Safety Report*. The report presents a comprehensive description of the post-closure radiological safety assessment of a repository for spent nuclear fuel, vitrified high level waste, and long-lived intermediate-level waste.

Ensuring sub-criticality through selection of appropriate materials, preventative designs, segregation of wastes, and administrative measures is one of the key principles of the methodology.

More specifically, sub-criticality is primarily achieved by applying minimum burn-up criteria to all wastes accepted by the facility (15 GWd/t_{IHM}, or, 22 GWd/t_{IHM} for canisters with both MOX and UO_2 spent fuel), and by designing waste canisters and arrays to use geometries which reduce the likelihood of criticality. NAGRA (2005) notes that when burn-up criteria are enforced, if waste canisters are fully loaded with UO_2 spent fuel they will be sub-critical. Furthermore, so long as the burn-up criteria have been enforced, canisters loaded with UO_2 spent fuel will remain sub-critical both when intact, and after failure (when the void space in the canister becomes flooded).

NAGRA (2005) states that for the long term, changes in geometry such as diffusion of large quantities of uranium into the bentonite, followed by precipitation within pore spaces, would not result in criticality.

2.7 JAPAN NUCLEAR CYCLE DEVELOPMENT INSTITUTE (JNC)

In 2000 the Japan Nuclear Cycle Development Institute (JNC) authored a report titled *H12: Project to Establish the Scientific and Technical Basis for HLW Disposal in Japan*. The report compiles the results of many years of R&D studies, and overall, outlines the underlying scientific basis in support of deep geologic disposal of HLW in Japan.

It is important to understand that the JNC (2000) report deals with the disposal of vitrified waste, as opposed to spent nuclear fuel.

It is also important to understand that the geologic environment of Japan is complex, with far more seismic and volcanic activity than locations such as Canada, Sweden, or Finland. As a result, the report places a large emphasis on identifying and understanding the geologic environment (and its related phenomena) and their potential effects on a DGR. Geology, hydrogeology and geochemistry (and related mechanisms such as precipitation, dissolution, migration, and corrosion) are discussed in great detail.

Criticality is discussed only briefly. JNC (2000) notes that the only credible scenario that could give rise to the conditions necessary for criticality is for the uranium nuclides of all canisters (40,000) of vitrified waste to be concentrated at one point (assuming realistic impediments by natural and artificial barriers). This further requires a porosity of 30% or more in the host rock, and a uranium enrichment of 12% or more. Overall, JNC (2000) concludes that since the probability of achieving the conditions for criticality is negligibly small – and that even if criticality were to occur the consequences would be limited – it follows that criticality does not require further consideration in safety analyses.

2.8 RESULTS OF THE LITERATURE SEARCH REVIEW

The results of the literature review are used to develop an informed list of proposed criticality scenarios that will undergo further quantitative assessment. The proposed scenarios are representative of a deep geologic repository for used CANDU fuel.

The literature review yielded a number of important points that require consideration when developing bounding scenarios. This information is useful in not only developing scenarios but in ranking them in order to decide which scenarios are truly bounding. Table 2-1 outlines the chosen scenarios. Further discussion on the scenarios is presented following the table.

Table 2-1: Case Matrix & Reference Information

Bounding Scenario No.	# of Containers	Source	Condition of Fuel	Inside Container	Outside Container	Bounding? (Y/N)	Why?	Represents	Reference Studies	Reference Information
-	1	In-container	Intact	Air	Air	N	Lower Keff value than in rock (SKB 2010). Bounded by Scn #1.	Preclosure period (transfer, handling, & placement activities).	SKB 2010 SKB 2002	Keff values for dry containers surrounded by air are the lowest of combinations between flooded and non-flooded, surrounded by air versus water.
-	1	In-container	Intact	Flooded	Water	N*	Lower Keff value than in rock (SKB 2010). Bounded by Scn #1.	Preclosure period (transfer, handling, & placement activities).	SKB 2010 SKB 2002	Keff values for containers filled with water and surrounded by air are lower than if filled and surrounded by water.
-	1	In-container	Intact	Flooded	Air	N*	Lower Keff value than in rock (SKB 2010). Bounded by Scn #1.	Preclosure period (transfer, handling, & placement activities).	SKB 2010 SKB 2002	Keff values for containers filled with water and surrounded by air are lower than if filled and surrounded by water.
1	1	In-container	Intact	Flooded	Bentonite Shield and Rock	Y	Higher Keff value (SKB 2010). Bounding.	Postclosure period.	SKB 2010 SKB 2002 SKB 1999	Keff values for containers filled with water and surrounded by rock are greater than if filled with water and surrounded by water (SKB 2010; 2002). Over the long-term, reactivity is greatest within 40 days (SKB 1999).
-	Several	In-container	Intact	Flooded	Bentonite Shield and Rock	N	Keff value not influenced by increasing container number (SKB 2010). Bounded by Scn #1.	Postclosure period.	SKB 2010	Increase in Keff value for multiple containers is negligible compared to Keff value for 1 container (SKB 2010).
2	1	In-container	Degraded	Flooded	Bentonite Shield and Rock	Y	Not explicitly addressed in other studies.	Postclosure period: Allows for assessment of a case where a container is degraded and also flooded. The fuel geometry is collapsed.	-	Most studies do not consider 'degraded' fuel bundles, only degraded containers. Consideration is not explicitly given to a scenario where the container is flooded, and the dissolved radionuclide mixture is retained within the container. SKB (1999) investigates the long-term effects of corrosion on reactivity. It was found that with time, corrosion of the canister insert could reduce the dimensions of the fuel channels due to buildup of corrosion products. Reactivity was found to decreases sharply if the channels are filled with corrosion products. However, these evaluations are based on the assumption that the fuel is intact (i.e. *not* degraded).
-	Several	In-container	Degraded	Flooded	Bentonite Shield and Rock	N	Keff value not influenced by increasing container number (SKB 2010). Bounded by Scn #2.	Postclosure period: Allows for assessment of a case where many containers are degraded and also flooded.	SKB 2010	Increase in Keff value for multiple containers is negligible compared to Keff value for 1 container (SKB 2010).
3	1	Radionuclides released into bentonite surrounding a degraded container *(near field)*	Degraded	-	Rock	Y	-	Postclosure period: Allows for assessment of a case where a container is degraded, flooded, and radionuclides are released into the surrounding bentonite.	NAGRA 2005	Diffusion of large quantities of uranium into surrounding bentonite, followed by precipitation within pore spaces, would not result in criticality (NAGRA 2005).

Table 2-1 Case Matrix & Reference Information (Cont'd)

Bounding Scenario No.	# of Containers	Source	Condition of Fuel	Inside Container	Outside Container	Bounding? (Y/N)	Why?	Represents	Reference Studies	Reference Information
-	Several	Radionuclides released into bentonite surrounding a degraded container *(near field)*	Degraded	-	Rock	N	Keff value not influenced by increasing container number (SKB 2010). Bounded by Scn #3.	Postclosure period: Allows for assessment of a case where many containers are degraded, flooded, and radionuclides are released into the surrounding bentonite.	SKB 2010 SKB 1999 JNC 2000 Hicks & Prescott 2000	Increase in Keff value for multiple containers is negligible compared to Keff value for 1 container (SKB 2010). The probability of localized accumulation of material achieving critical mass material is negligible (SKB 1999). The probability of multiple containers leaking, followed by localized accumulation of all leaked material thereby achieving critical mass is negligible (JNC 2000; Hicks & Prescott 2000).
4	Several	Radionuclides released into rock surrounding a degraded container *(far field)*	Degraded	-	Rock	Y	-	Postclosure period: Allows for assessment of a case where many containers are degraded, flooded, and radionuclides are released into the surrounding rock at distance.	-	While many references mention the low probability of a leak followed by localization and accumulation of fissile material, quantitative Keff evaluations of such a scenario were not found.
5	Several	Dissolved Pu solution	-	-	Rock	Y	Cases involving dissolution of plutonium are recognized and assessed by AECL (1994). Rock is a bounding surrounding material (SKB 2010).	Postclosure period: Critical radius of spherical plutonium solutions as a function of concentration. Assesses plutonium criticality when the fissile materials are released from a container, dissolved in water, and migrate.	AECL 1994	A similar calculation could be useful today. The Pu solubility may need to be reconsidered if the redox conditions due to radiolysis are now interpreted differently.

Note:
'Rock' surrounding material includes sedimentary rock (limestone) as well as granite.

2.9 BOUNDING CONDITIONS AND SCENARIOS

Table 2-2 below presents a summary of criticality results calculated for different arrangements of fuel bundles, container types, and surrounding materials from SKB (2010).

Table 2-2: Comparison of Criticality Results from SKB (2010)

No.	Characteristics	Resulting PWR K_{eff} Value	Resulting BWR K_{eff} Value
1	*Bentonite Shield* *Rock Surrounding* *Water Filled Canister*	1.0888	0.9959
2	*No Bentonite Shield* *Air Surrounding* *Water Filled Canister*	1.0860	0.9926
3	*No Bentonite Shield* *Water Surrounding* *Water Filled Canister*	1.0872	0.9942

A review of Table 2-2 yields the following important general conclusions:

- When a container is submerged in water and also becomes filled with water, the Keff value increases;
- When a container becomes filled with water but is surrounded by air (i.e. not submerged), the Keff value increases, but not as much as were the container to be water-filled and also submerged;
- Keff increases further when the material surrounding the canister is rock (as opposed to air or water); and,
- Addition of bentonite as a surrounding buffer material slightly increases the Keff value (supported by benchmarking calculations used to distinguish the influence of bentonite versus water, see Section 6).

These conclusions are used in developing the scenarios shown in Table 2-1. It follows that intact containers surrounded by a bentonite shield provide slightly higher K_{eff} results in comparison to containers without bentonite shielding. Similarly, intact containers filled with water provide higher K_{eff} results in comparison to containers that are not filled with water (i.e. are filled with air). Furthermore, it is inferred that rock (as a surrounding material) provides the highest K_{eff} results in comparison to air or water surrounding materials. As such, a scenario with a flooded, bentonite-shielded, intact container, surrounded by rock, is bounding over all other configurations of surrounding material, fill material (i.e. flooded or not flooded), and surrounding material (for intact containers). These different variants are represented by the first four rows of Table 2-1. Of these four rows, the actual bounding scenario is the fourth row. It becomes **Bounding Scenario #1** and will undergo further investigation.

It is interesting to note that the findings of POSIVA (2005) are somewhat different for one particular configuration: POSIVA (2005) shows that the K_{eff} value for a flooded container surrounded by air is **greater** than the K_{eff} value for a flooded container surrounded by water. SKB (2010) shows the opposite: that the K_{eff} value for a flooded container surrounded by air is actually less than the K_{eff} value for a flooded container surrounded by water.

Scenarios: Effects of Single or Multiple Intact Containers

SKB (2010) studied the effect of interaction between intact containers deposited in a repository by modelling a single intact container scenario and a scenario with an infinite number of intact containers. SKB (2010) found that the interaction between intact containers is insignificant. In other words, if more intact containers are added to the placement room in repository, they will not influence each other in terms of criticality and their K_{eff} values will not increase significantly. These findings are represented by the fifth row in Table 2-1, which shows a “several-container” scenario being bounded by Bounding Scenario #1.

Scenarios: Intact Versus Degraded Containers

After the containers are deposited, over time the container and the fuel bundle will degrade. Few studies consider degraded containers, and even fewer consider degraded fuel bundles. For these conditions, the literature review identified only the McCamis (1992) study as having applicable quantitative information. Due to the lack of scientific literature on these conditions, scenarios have been developed for further investigation. These scenarios include various configurations of ‘degraded’ containers and fuel bundles. In Table 2-1, these correspond to the scenarios in the last six rows, where the condition of fuel is indicated as ‘degraded’.

The first row of the ‘degraded’ variants in Table 2-1 uses the same bounding conditions as Bounding Scenario #1 for intact containers (i.e. flooded, bentonite-shielded, and surrounded by rock). The key differences being that the fuel geometry is assumed to have collapsed and the dissolved radionuclide mixture is assumed to be retained within the container. Among the ‘degraded’ variants (rows) this scenario is bounding. It becomes **Bounding Scenario #2** and will undergo further investigation.

The second row of the degraded variants follows the assumption that multiple-container scenarios are bounded by single-container scenarios, and is therefore considered to be bounded by Bounding Scenario #2.

The third row of the degraded variants allows for assessment of a case where a container is degraded, the fuel geometry is collapsed, flooded, and radionuclides are released into the surrounding *bentonite*. This represents a unique scenario and warrants further investigation. As such, it becomes **Bounding Scenario #3**.

The fourth row of the degraded variants follows the assumption that multiple-container scenarios are bounded by single-container scenarios, and is therefore considered to be bounded by Bounding Scenario #3.

The fifth row of the degraded variants represents a case where many containers are degraded, flooded, and radionuclides are released into the surrounding *rock (at distance)*. While many references mention the low probability of a leak with localization and accumulation of fissile material, quantitative Keff evaluations of such scenarios were not found. Therefore this represents a unique scenario and warrants further investigation. As such, it becomes **Bounding Scenario #4**.

The sixth and final row of the degraded variants involves determining the critical radius of spherical plutonium solutions as a function of concentration. This scenario assesses plutonium criticality when the fissile materials are released from a container, dissolved in water, migrate, and potentially accumulate. Again, this represents a unique concept and warrants further investigation. As such, it becomes **Bounding Scenario #5**.

Burnup

Although many of the studies do not share the same type of spent fuel, results from McCamis (1992), SKB (2010), and Hicks & Prescott (2000) all indicate that, in general, reactivity decreases as burnup increases. From this, it is inferred that the lowest value would likely result in the highest reactivity (i.e. lowest burnup is most conservative for criticality calculations). Of the burnup values for which NWMO has detailed fuel inventory data (i.e. 220, 280, and 320 MWh/kg U), it is inferred that 220 MW/kg U is likely to be the most conservative. To confirm this prediction, benchmark criticality calculations are conducted using varying burnup values (see Section 6).

Cooling Time/Decay

The most conservative cooling time (decay time) has been determined using benchmark criticality calculations (see Section 5 and Section 6) which calculate criticality across a range of cooling times in order to identify the most conservative value.

3. COMPUTER CODES USED IN THE ANALYSIS AND CALCULATIONS

The 3-D transport calculations are performed using the Monte Carlo code, MCNP v5. MCNP v5 (build 1.4) (LANL 2003) is a general-purpose Monte Carlo N-Particle transport code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. MCNP was developed by the Los Alamos National Laboratory, Transport Methods Group, to solve a wide variety of transport problems.

MCNP was installed and verified as demonstrated in Revolinski (2007).

MCNP models a physical system with a three-dimensional configuration of geometric cells bounded by first and second-degree surfaces and fourth-degree elliptical tori. Each geometric cell contains a material or void as specified by the user to model the physical system. Material characteristics (i.e., cross sections) are represented by point-wise continuous cross-section data. For neutrons, all reactions given in a particular cross-section library (such as ENDF/B-VI) are taken into account. Thermal neutrons are described by the free gas and S(α, β) models. The MCNP neutron data library based on Evaluated Neutron Data File B-VI (ENDF/B-VI) is the default for continuous energy neutron transport.

The specific elements used in this evaluation are:

^{235}U	Al	^{53}Cr
^{236}U	Si	^{54}Cr
^{238}U	P	Mn
^{239}Pu	S	^{54}Fe
^{240}Pu	Cl	^{56}Fe
H	Ar	^{57}Fe
C	K	^{58}Fe
N	Ca	Cu
O	Ti	Zr
Na	^{50}Cr	Sn
Mg	^{52}Cr	

The light water S(α,β) correction (lwtr.60t) is used for water.

4. ASSUMPTIONS, BIASES AND UNCERTAINTIES, AND APPLICABILITY

4.1 ASSUMPTIONS REGARDING CNSC RD-327 (2010)

The criticality safety of storage of CANDU irradiated fuel within the NWMO containers has been evaluated in this report. The resulting k_{eff} for "worst case" degraded fuel conditions is less than 0.70 (as presented in later sections). There are no moderators more effective than light water present in the container design or upset conditions. Because of the low k_{eff}, the long-term storage of CANDU fuel does not require the implementation of a full criticality safety program as per CNSC (2010) RD-327.

The code validation portion of RD-327, Section 2.3.4, *was* followed in this report as recommended since criticality safety calculations were performed.

4.2 BIASES AND UNCERTAINTIES

The MCNP code Upper Subcritical Limit (USL) for homogeneous plutonium systems was determined in Newmyer (2014) (Appendix A), and is shown in Equation 1 below.

Equation 1

$$USL=0.9814-MoS$$

where,
MoS = Margin of Subcriticality.

For this study, no MoS was applied to the USL; that is, MoS is assumed to be equal to 0.00 Δk. This is not to say that MoS is - or should be - disregarded, rather, the judgement of an adequate MoS should be made later when the results of this report are applied. At that time the source data will need to be reviewed along with the desired goal, and a judgement can be made as to what level of conservatism is necessary to meet the desired goal. Regarding consistency with CNSC (2010), the validation report (Newmyer 2014) (Appendix A) provides additional supporting information on USL calculations, in addition to the discussion outlined in Section 4.1.

Therefore the USL is equal to 0.9814.

For an acceptable result, the MCNP $k_{eff} + 2\sigma$ must be less than the USL value.

4.3 AREA OF APPLICABILITY

The Area of Applicability (AoA) derived in Newmyer (2014) (see Appendix A) includes most of the materials in this study. The materials which are not explicitly included in the validation benchmarks have small number densities in the modeled materials and are judged to be insignificant in the determination of k_{eff}. However they are still included in the material specification for completeness.

Table 4-1 presents an AoA summary. Calculations were completed in two stages: first, initial benchmark calculations were completed in order to determine or confirm bounding conditions; then the final criticality calculations were completed based on the benchmark calculation results. In Table 4-1 the "Combined AoA" column lists the materials used in the benchmark calculations conducted to determine bounding conditions. The "Calculations" column lists those used in the final criticality calculations. The discussions and calculations presented in the Validation Report (Appendix A – Section 3.1) support this table, they include oxide, solution and metal forms, as well as the thermal, intermediate and fast energy ranges. From these investigations it is concluded that the k_{eff} data are not correlated with any of the Area Of Applicability (AoA) parameters, and therefore there is no bias as a function of the evaluated parameters.

The MCNP container models with intact and degraded fuel conditions include both plutonium and uranium isotopes. The benchmark AoA only includes plutonium. However, there is sufficient margin present in the k_{eff} results to ensure subcriticality even without accounting for the addition of uranium isotopes in the benchmarks. The calculations using only ^{239}Pu have characteristics that are well within the AoA boundaries of Table 4-1. Therefore, it is judged that the models of this study are within the AoA of the code validation.

Table 4-1: Area of Applicability Summary (See Appendix A – 3.1)

Parameter	**Combined AoA** *(See Appendix A – 3.1)*	**Calculations**
Fissile Material	Pu metal, PuO_2, $PuO_2(NO_3)_2$	Spent CANDU fuel, including U-235, U-236, U-238, Pu-239, Pu-240
Fissile Material Form	Plutonium Solids and Solutions	Pu and U solids and solutions
H/^{239}Pu ratio	0.0 ≤ H/^{239}Pu ≤ 1061	0.0 ≤ H/^{239}Pu ≤ 661
Average Neutron Energy Causing Fission (MeV)	0.005 < ANECF < 1.91	0.005 < ANECF < 2.47*
wt% ^{240}Pu	0.84 to 18.35 wt% ^{240}Pu	0 to 41 wt% ^{240}Pu#
Moderating Materials	Polystyrene, Graphite, Water, HNO_3	Water
Reflecting Materials	Water, Plexiglas, Polyethylene, Unreflected	Bentonite, Water, and Rock
Absorber Materials	Concrete, Aluminum, Steel	Bentonite and Container materials
Geometry	Sphere, Cylinder and Cuboid Arrays	Sphere, Cylinder and Cuboid Arrays

* Calculations performed at the USL are within the actual AoA range. See Appendix A: Validation Report (Newmyer 2014) for more information.

Calculations performed at the USL consider only ^{239}Pu which is very conservative since it is virtually impossible to have ^{239}Pu without some amount of ^{240}Pu.

5. MATERIAL SPECIFICATIONS & CALCULATIONS

5.1 FUEL ISOTOPICS

Spent fuel isotopics were obtained from Tait *et al*. (2000). The data are provided for power levels from 220 MWh/kgU to 320 MWh/kgU and radionuclide inventories are specified in g/kgU.

For the intact and degraded fuel condition models, only uranium and plutonium isotopes were considered. All other actinide and fission products were ignored (being either non-fissile species or neutron absorbers or in isotropic concentrations so low as to have no impact on Keff), therefore producing conservative results. The amount of Pu-241 is less than 0.6% of the total Pu content after 50 years of decay. This is considered to be an essentially negligible amount and has been excluded. Similarly, the amount of U-233 never exceeds 0.0014% of the total U content, and it is therefore also excluded. Overall, for uranium, the three largest isotopic mass values were modeled. These were U-235, U-236 and U-238. Overall, for plutonium, the two largest isotopic mass values were modeled. These were Pu-239 and Pu-240.

The original CANDU fuel has the chemical form UO_2. The density of the fuel is 10.6 g/cm^3. The oxygen content in the fuel needs to be calculated. The oxygen concentration does not undergo significant change during irradiation so it only needs to be calculated for fresh fuel.

The calculation of the oxygen content is straightforward. Natural uranium has a molecular weight of 238 g/mol. Therefore an initial mass of 1 kg U has 4.201171 moles of U and 8.402341 moles of O. Oxygen has a molecular weight of 15.99492 g/mol and therefore 134.3947 g per kg U.

So, the final MCNP input specification for the fuel can simply use the mass values for each isotope (including oxygen) as weight fractions of the fuel material. The material density is 10.6 g/cm^3. Burnup/decay data from Tait *et al*. (2000) are arranged into time-steps (configurations) shown in Table 5-1.

Table 5-1: Burnup/Decay Data Sets

Set Number	Burnup (MWh/kgU)	Decay time (years)	^{235}U g/kgU	^{236}U g/kgU	^{238}U g/kgU	^{239}Pu g/kgU	^{240}Pu g/kgU	O g/kgU
1	220	0	1.67	0.8223	981.8	2.605	1.285	134.3947
2		10	1.671	0.8236	981.8	2.684	1.284	134.3947
3		50	1.674	0.8289	981.8	2.681	1.279	134.3947
4		100	1.678	0.8356	981.8	2.678	1.272	134.3947
5		200	1.685	0.8487	981.8	2.67	1.259	134.3947
6		500	1.708	0.8873	981.8	2.647	1.219	134.3947
7		1000	1.745	0.949	981.8	2.61	1.157	134.3947
8		1E+04	2.33	1.646	981.8	2.017	0.4471	134.3947
9		1E+05	4.167	2.081	981.8	0.1519	3.32E-05	134.3947
10		1E+06	4.313	2.026	981.7	3.85E-12	4.23E-15	134.3947
11		1E+07	4.275	1.552	980.3	2.01E-12	4.5E-15	134.3947
12	280	0	1.095	0.9022	979.1	2.672	1.633	134.3947
13		10	1.096	0.9039	979.1	2.754	1.632	134.3947
14		50	1.099	0.9107	979.1	2.751	1.626	134.3947
15		100	1.103	0.9191	979.1	2.747	1.618	134.3947
16	320	0	0.8213	0.938	977.3	2.678	1.829	134.3947
17		10	0.8221	0.9399	977.3	2.76	1.828	134.3947
18		50	0.8252	0.9475	977.3	2.757	1.821	134.3947
19		100	0.8291	0.9569	977.3	2.753	1.812	134.3947

5.2 INTACT SPENT CANDU FUEL CONCEPT

Intact spent fuel scenarios are straightforward, and involve MCNP models developed from information provided by NWMO for the small (4L012, Mark II) and large (IV17, Mark I) container types. The MCNP container models used for criticality calculations are consistent with those developed as part of previous MCNP projects (NWMO 2013a). See Section 5.8 for MCNP model design information.

5.3 DEGRADED SPENT CANDU FUEL CONCEPT

Separate MCNP models were created for calculations involving degraded fuel configurations. The degraded fuel condition assumes that the water, fuel and zircaloy cladding are all homogeneously mixed together, but the fuel is not separated from the zircaloy cladding within the container. Therefore the volume of fuel and cladding for each container type must be determined.

The large container (IV17) holds 288 fuel bundles. Each bundle has 37 elements. So the entire container holds 10656 pins.

The pin has a fuel radius of 0.6116 cm and a clad radius of 0.6535 cm. No gap is present between the fuel and clad. Each fuel bundle has an axial height of 52.45 cm and there are 8 axial layers of fuel bundles. The total height of the fuel in the container is 419.6 cm.

Based on the above dimensions, the total volume of fuel in the container is 0.657 m^3. The total volume of clad in the container is 0.0931 m^3. The container radius is 39.45 cm. The total volume of the container (in which fuel is present) is 2.05 m^3.

If all the fuel and cladding degrades and drops to the bottom of the large container, it will occupy a height of 153.37 cm.

The smaller container (4L012) holds 48 fuel bundles. Each bundle has 37 elements. So the entire container holds 1776 pins.

The total height of the fuel in the container is 208.74 cm which represents 4 layers of bundles.

Based on the above dimensions, the total volume of fuel in the container is 0.109 m^3. The total volume of clad in the container is 0.0154 m^3. The container radius is 23.8125 cm. The total volume of the container (in which fuel is present) is 0.372 m^3.

If the fuel and cladding degrades and falls to the bottom of the small container, it will occupy a height of 69.8 cm.

When water is included in the fuel/clad mixture, the total volume will increase. This will cause an increase in the height of the resulting fuel/clad/water mixture. The water to fuel ratio was varied to find the optimum water content to yield the highest k_{eff} value.

The following calculations were used to determine the height of the degraded fuel/clad/water mixture.

First, the volume fraction of water was calculated using Equation 2:

Equation 2

$$vf\_h20 = \frac{w\_fmixr}{w\_fmixr + 1}$$

where,

vf_h20 = volume fraction of water
w_fmix = water-to-fuel mix ratio

Next the fractional height of the container volume was calculated using Equation 3:

Equation 3

$$frac\_height = (w\_fmixr + 1) \times \frac{(fuel\_mix\ vol)}{(total\ vol)}$$

where,

frac_height = fraction height of the container volume
w_fmix = water-to-fuel mix ratio
fuel_mix vol = total volume of both fuel and cladding
total vol = total container volume

The fractional height (frac_height) is multiplied by the total height (419.6 cm) to set the height of the fuel/clad/water mixture.

Next, the fuel/clad/water mixture MCNP material specification also needs to be calculated. This involves calculating weight fractions for the fuel, cladding, and water portions of the mixture. The weight fractions of the fuel isotopics can be determined using the burnup gram data (from the intact fuel models). The clad material has intact fuel weight fractions which are already determined. The water also has known weight fractions. From this, Equation 4 shows the calculation of the fuel mixture density, whereas Equations 5 to 7 show calculation of the actual weight fractions for the fuel, cladding, and water.

Equation 4

$$fuelmix\_dens = [10.6\times(vf\_fuelmix)\times(vf\_fuel)]+[6.55\times(vf\_fuelmix)\times(vf\_clad)] + [0.9982\times(vf\_h20)]$$

where,

fuelmix_dens	= density of the fuel/clad/water mixture
10.6	= fuel oxide density
vf_fuelmix	= volume fraction of the fuel mixture *(from 1 – (vf_h2o); using Equation 2)*
vf_fuel	= volume fraction of fuel: 0.875878 (based on pin dimensions)
6.55	= cladding material density
vf_clad	= volume fraction of cladding: 0.124122 (based on pin dimensions)
0.9982	= assumed water density
vf_h20	= volume fraction of water (from Equation 2)

Equation 5

$$wf\_fuel = (fuel\_dens) \times (vf\_fuelmix) \times \frac{(vf\_fuel)}{(fuelmix\_dens)}$$

where,

wf_fuel	= weight fraction for all fuel isotopes
fuel_dens	= fuel oxide density: 10.6
vf_fuelmix	= volume fraction of the fuel mixture *(obtained from 1 – (vf_h2o); calculated previously using Equation 2)*
vf_fuel	= volume fraction of fuel: 0.875878 *(based on pin dimensions)*
fuelmix_dens	= from Equation 4.

Equation 6

$$wf\_clad = 6.55 \times (vf\_fuelmix) \times \frac{(vf\_clad)}{(fuelmix\_dens)}$$

where,

wf_clad	= weight fraction for the cladding material
6.55	= cladding material density
vf_fuelmix	= volume fraction of the fuel mixture *(from 1 – (vf_h2o); using Equation 2)*
vf_clad	= 0.124122 (based on pin dimensions)
fuelmix_dens	= from Equation 4.

Equation 7

$$wf\_h20 = 0.9982 \times \frac{(vf\_h20)}{(fuelmix\_dens)}$$

where,

wf_h20	= weight fraction for the water
0.9982	= assumed water density
vf_h20	= from Equation 2
fuelmix_dens	= from Equation 4.

Each of the three separate weight fractions (fuel, clad and water) are multiplied by the isotopic weight fractions making up each component. In the case of oxygen, each component is calculated and added together to arrive at a total oxygen weight fraction.

5.4 PLUTONIUM/WATER MIXTURES

Some calculations were performed with mixtures of plutonium and water. For these mixtures, the density of plutonium (modeled as ^{239}Pu) was specified and the volume fraction was determined based on a plutonium metal density of 19.816 g/cc. The remaining volume was assumed to be occupied by water with a density of 0.9982 g/cc.

5.5 BENTONITE COMPOSITION

Table 5-2 shows the composition of bentonite based on information provided by NWMO (Karnland 2010).

Table 5-2: Bentonite Composition

	Dry (0% saturation)	As placed (65% saturation)	Saturated (100% saturation)
Composition	%wt	%wt	%wt
SiO_2	65.9	56.48	52.45
Al_2O_3	21.5	18.43	17.11
Fe_2O_3	4.46	3.82	3.55
MgO	2.82	2.42	2.24
CaO	1.63	1.40	1.30
Na_2O	2.69	2.31	2.14
K_2O	0.56	0.48	0.45
TiO_2	0.24	0.21	0.19
P_2O_5	0.06	0.05	0.05
C	0.51	0.44	0.41
S	0.37	0.32	0.29
H_2O	0	14.40	20.57
Sum	100.74	100.74	100.74

The bentonite density (without water) is specified as 1.61 g/cm^3. The water density in saturated bentonite is 0.413 g/cc based on a water density of 0.9982 g/cm^3 and a bentonite porosity of 41.37%. The remaining bentonite clay volume has a calculated density of 2.7462 g/ cm^3. The bentonite mineral composition weight fractions are based on the Table 5-2 normalized to 100%. All calculations with Bentonite include water saturation. For calculations where a plutonium/water mixture is modeled in the bentonite, the water saturation is replaced with the plutonium/water mixture.

5.6 GRANITE COMPOSITION

Table 5-3 shows the composition of granite (NWMO 2012a). Granite compositions are based on a density of 2.7 g/cm^3. The weight % of water is approximately 0.11% (i.e. 3 kg of water per m^3 of granite).

Table 5-3: Granite Composition

Compound	% Weight
SiO_2	71.9
Al_2O_3	14.4
K_2O	1.2
Na_2O	0.7
CaO	1.8
FeO	3.7
Fe_2O_3	4.1
MgO	0.3
TiO_2	1.7
P_2O_5	0.1
MnO	0.02
H_2O	0.11

5.7 SEDIMENTARY ROCK COMPOSITION

Table 5-4 presents sedimentary rock composition information (Jackson and Murphy, 2011; Wingston and Jackson, 2010a; Wingston and Jackson, 2010b). All properties are for the Cobourg limestone layer (the layer in which the repository is assumed to be located). A Cobourg limestone has a density of 2710 kg/m^3 and a porosity of 0.015.

The empirical formula for each mineral was determined using the website www.webmineral.com. Webpages are available for each mineral revealing its exact molecular formula. The empirical formulas are shown below. These were used to determine each element's atomic density.

Table 5-4: Cobourg Rock Composition

Average	Empirical Formula	%wt
Albite	$Na_{0.95}Ca_{0.05}Al_{1.05}Si_{2.95}O_8$	0.13
Ankerite	$CaFe_{0.6}Mg_{0.3}Mn_{0.1}(CO_3)_2$	3.78
Anorthite	$Na_{0.05}Ca_{0.95}Al_{1.95}Si_{2.05}O_8$	0.08
Calcite	$CaCO_3$	80.25
Clinochlore	$Mg_{3.75}Fe_{1.25}Si_3Al_2O_{10}(OH)_8$	0.14
Dolomite	$CaMg(CO_3)_2$	3.07
Halite	NaCl	0.08
Orthoclase	$KAlSi_3O_8$	3.53
Pyrite	FeS_2	0.13
Quartz	SiO_2	3.95
Rozenite	$Fe(SO_4)\cdot 4(H_2O)$	0.05
Sheet silicates	$Mg_3(OH)_2(Si_2O_5)$	4.92

The overall number densities for sedimentary rock were calculated by specifying how many moles of each element is present in a molecule of a given mineral. The total atomic mass for each mineral was then determined. The gram density of each mineral was determined by multiplying the total density by the weight fraction. This gram density was converted to mole density and then to atom density for each element in each mineral. The atomic density for each element was then summed over all minerals to provide the atomic densities for each element in sedimentary rock.

5.8 MODELS

5.8.1 Intact Fuel in Container Models

MCNP models were developed by SENES based on container design information provided by the NWMO (NWMO 2012b, NWMO 2013b). These models were used in all criticality safety models generated in this report. For **Bounding Scenario #1,** modelling considers both water and air inside an intact container, and water, air, and bentonite outside the intact container. It was determined that rock is far enough away as to be neglected in these calculations. Burnup values for all provided power levels and decay values were used in the models. Figure 5-1 to Figure 5-4 present the containers as modelled using MCNP.

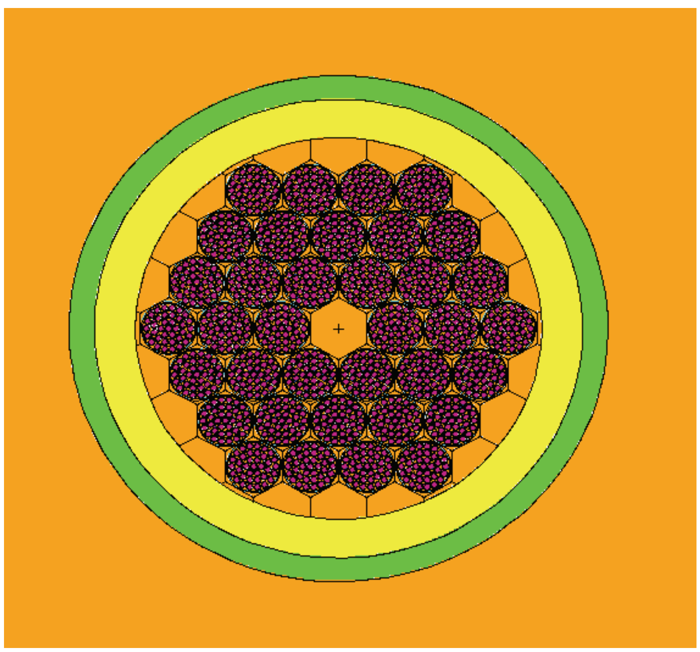

Figure 5-1: IV17 MCNP Model: Z Plane Slice

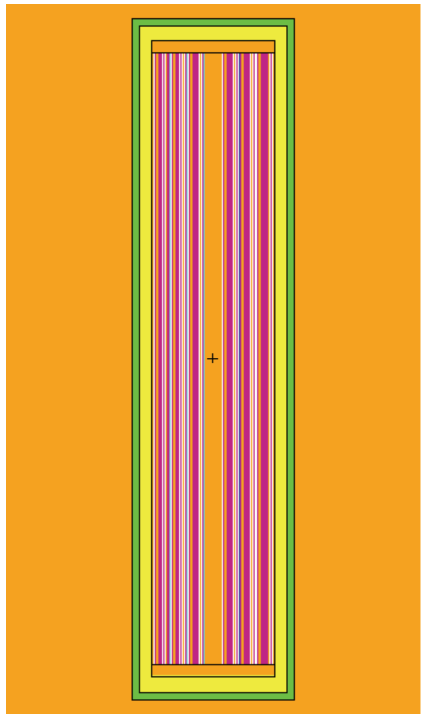

Figure 5-2: IV17 MCNP Model: Y Plane Slice

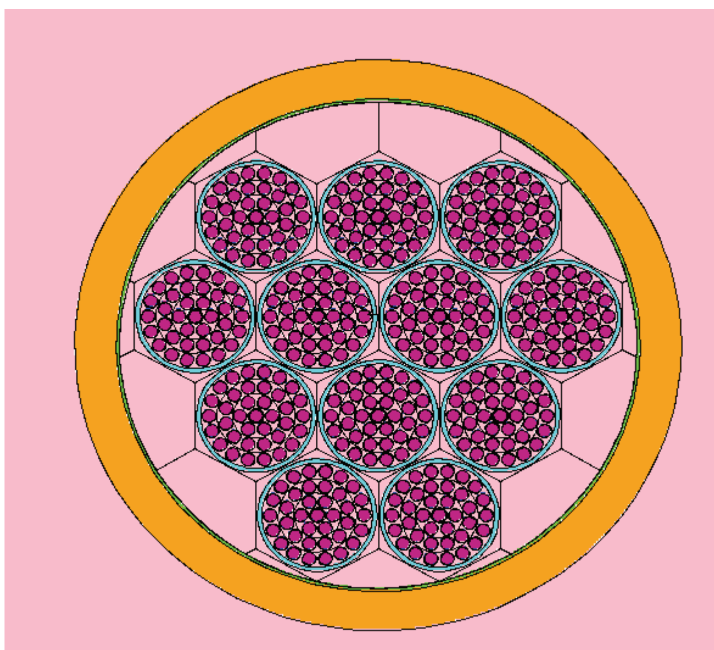

Figure 5-3: 4L012 MCNP Model: Z Plane Slice

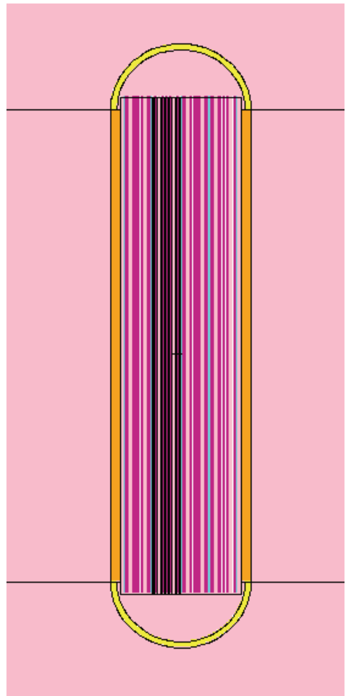

Figure 5-4: 4L012 MCNP Model: Y Plane Slice

5.8.2 Degraded Fuel in Container Models

For **Bounding Scenario #2**, the intact fuel container models were modified to remove all fuel bundle structure. Only the outer shielding of the container remained, fuel and clad are homogenized. The height of the fuel/clad/water mixture was calculated as previously described in Section 5.2. The space above the fuel/clad/water mixture is modeled as water. Both water and bentonite were modeled outside the container because they are similar in terms of limiting conditions; the benchmark calculations help to clarify which truly represents the limiting condition. Air was not considered outside the container since the Intact Fuel Container Models showed it was not the limiting condition. Also, only the lowest power level burnup values were used since the previous intact fuel container models showed the higher power level burnup values are not limiting.

When placed in the final storage configuration, the material outside the container is different for the large and the small container. The material outside the container is based on Mark I and Mark II design concepts as discussed in Garisto *et al*. 2012 and Gobien *et al*. 2013. For the large container in-room placement (Mark I Design Concept), the reports show the containers surrounded by a Bentonite cylinder with a 87.5 cm outer radius (1.750 m outer diameter). This is then inside a square of diameter 210 cm of granite rock. A reflective boundary condition was used on all surfaces. For the small container, (Mark II Design Concept) the reports show the container completely surrounded by Bentonite: approximately 50 cm diameter in both the x and y dimensions, and approximately 140 cm for the z direction. A reflective boundary condition was used in all six directions.

5.8.3 Plutonium in Bentonite Model

A model was created to examine the reactivity of fuel leaking from the container into the surrounding Bentonite. Bentonite has a high porosity (41.37%). A sphere of Bentonite was mixed with a plutonium/water mixture of varying density (see Section 5.2 and 5.4) to determine the minimum critical mass and volume. For the volume calculations, the minimum critical volume was reported as a function of Pu density. Bentonite, with a thickness of 120 cm, was used as a reflector. The volume required for criticality keeps the fissile material within the bentonite, therefore it is not necessary to model the surrounding rock. This model configuration produces results relevant to **Bounding Scenarios #3.**

5.8.4 Plutonium Reflected by Rock Model

A model was created to examine the reactivity of fuel leaking into the surrounding rock structure. There are two types of rock that the containers may be placed into: granite and limestone. For these models, a sphere of plutonium/water mixture was surrounded by 120 cm of each rock type. The minimum critical volume was calculated at varying Pu densities. This model configuration produces results relevant to **Bounding Scenarios #4 and #5.**

6. EVALUATIONS, ANALYSIS, AND DETAILED CALCULATIONS

The results of the MCNP calculations are presented in the following subsections.

6.1 INTACT FUEL IN CONTAINER RESULTS

For **Bounding Scenario #1**, the large and small container designs were modeled in MCNP as described in Section 5.8.1. Water and/or air was modeled inside and outside the containers. All 19 burnup/decay data sets were used. Detailed results are presented in Appendix B, where each burnup/decay configuration is assigned a set number for easy reporting.

Results for the large and small container types show that the large container is more reactive than the small container. The worst case interior material is water and the worst case exterior material is Bentonite; which due to the density and material composition is a better reflector than water. The maximum keff for the intact fuel large container model is 0.58857 which occurs at burnup/decay set six (i.e. 220 MWh/kgU and 500 years decay). The intact fuel small container model is 0.48390 and occurs at burnup/decay set 4 (i.e. 220 MWh/kgU burnup and 100 years decay).

6.2 DEGRADED FUEL IN CONTAINER RESULTS

For **Bounding Scenario #2**, the large and small container designs were modeled in MCNP with degraded fuel conditions as described in Section 5.8.2. The material outside the container is Bentonite which was shown to be the worst case from the intact fuel in the container calculations. Detailed results are presented in Appendix B, where each burnup/decay configuration is assigned a set number for easy reporting.

The maximum k_{eff} for the degraded fuel in the large container is 0.69943, which occurs at burnup/decay set eight (i.e. 220 MWh/kgU and 10,000 years decay). The maximum k_{eff} for the degraded fuel in the small container is 0.62752, which also occurs at burnup/decay set eight (i.e. 220 MWh/kgU and 10,000 years decay). The results show that the degraded fuel model yields higher k_{eff} values than the intact fuel in the container model.

The analysis examined a range of water/fuel ratios and determined the worst case conditions indicated above. The k_{eff} of the system is not expected to exceed these bounding k_{eff} values with varying fuel arrangements. In other words, even with more compact or less compact fuel arrangements, the resulting k_{eff} values would still be lower than these worst-case values.

6.3 PLUTONIUM IN BENTONITE

Relevant to **Bounding Scenario #3**, calculations were performed with spheres of plutonium/water mixed with Bentonite. The minimum critical volume and minimum critical mass were determined based on the optimum density of plutonium. Results are shown in Table 6-1 for minimum critical mass and in Table 6-2 for minimum critical volume.

Table 6-1: Minimum Mass Spheres with Plutonium/Water/Bentonite Mixture MCNP Results

Pu density (g/cc)	Pu Mass (g)	k_{eff}	σ	$k_{eff}+2\sigma$
0.04		0.96885	0.00111	0.97107
0.05	1200	0.97269	0.00119	0.97507
0.06		0.96652	0.00119	0.96890
0.04		0.97247	0.00111	0.97469
0.05	1225	0.97449	0.00115	0.97679
0.06		0.97157	0.00121	0.97399
0.04		0.97884	0.00108	0.98100
0.05	1250	0.98108	0.00114	0.98336
0.06		0.97638	0.00116	0.97870
0.04		0.98281	0.00115	0.98511
0.05	1275	0.98243	0.00113	0.98469
0.06		0.98040	0.00123	0.98286
0.04		0.98515	0.00106	0.98727
0.05	1300	0.99107	0.00116	0.99339
0.06		0.98369	0.00116	0.98601

'σ' denotes the calculated uncertainty value from the MCNP calculations

Table 6-2: Minimum Volume Spheres with Plutonium/Water/Bentonite Mixture MCNP Results

Pu density (g/cc)	Volume (cm^3)	k_{eff}	σ	$k_{eff}+2\sigma$
0.1	37000	0.97780	0.00122	0.98024
	37250	0.97896	0.00126	0.98148
	37500	0.98140	0.00128	0.98396
	37750	0.98327	0.00130	0.98587
	38000	0.98427	0.00127	0.98681
1	19000	0.96968	0.00135	0.97238
	19250	0.97877	0.00127	0.98131
	19500	0.98100	0.00140	0.98380
	19750	0.98186	0.00129	0.98444
	20000	0.98580	0.00130	0.98840
10	4000	0.95889	0.00128	0.96145
	4125	0.96561	0.00122	0.96805
	4250	0.97452	0.00121	0.97694
	4375	0.98148	0.00123	0.98394
	4500	0.98880	0.00123	0.99126

'σ' denotes the calculated uncertainty value from the MCNP calculations

The results of Table 6-1 show the peak k_{eff} occurs at a Pu density of 0.05 g/cc. Interpolation of the k_{eff} values to the USL value of 0.9814 yields a **critical mass of 1240 grams Pu**. Interpolation of the results in Table 6-2 to the USL value of 0.9814 yield the following critical volumes as a function of plutonium density (see Section 7 for further discussion):

0.1 g/cm^3: 37,234 cm^3
1 g/cm^3: 19,259 cm^3
10 g/cm^3: 4,330 cm^3

Calculations were done to determine the critical plutonium density. The model was a 200 cm per-side cube with reflective boundary condition on all sides. The results are shown in Table 6-3.

Table 6-3: Infinite Density for a Plutonium/Water/Bentonite Mixture MCNP Results

Pu density (g/cc)	k_{eff}	σ	$k_{eff}+2\sigma$
0.0100	0.94190	0.00030	0.94250
0.0105	0.96650	0.00030	0.96710
0.0110	0.99040	0.00030	0.99100
0.0115	1.01330	0.00030	1.01390
0.0120	1.03436	0.00034	1.03504

'σ' denotes the calculated uncertainty value from the MCNP calculations

Interpolation of the results in Table 6-3 to the USL value yield a critical concentration of 0.0108 g/cc. This value reflects the concentration below which the plutonium/water/Bentonite mixture will remain subcritical. Further discussion related to spent fuel inventory is provided in Section 7.

6.4 PLUTONIUM REFLECTED BY ROCK

To investigate **Bounding Scenario #4 and #5**, minimum volume calculations were performed with a plutonium/water mixture surrounded by each type of rock: granite and sedimentary. The results from MCNP are provided in Table 6-4 and Table 6-5.

Table 6-4: Minimum Volume Spheres with Plutonium/Water Mixture Surrounded by Granite Rock MCNP Results

Pu density (g/cc)	Volume (cm^3)	k_{eff}	σ	$k_{eff}+2\sigma$
	1.30E+05	0.97737	0.00064	0.97865
	1.33E+05	0.97753	0.00063	0.97879
0.01	1.35E+05	0.98120	0.00067	0.98254
	1.38E+05	0.98294	0.00068	0.98430
	1.40E+05	0.98391	0.00064	0.98519
	5900	0.97403	0.00133	0.97669
	5925	0.97567	0.00125	0.97817
0.1	5950	0.97745	0.00130	0.98005
	5975	0.97982	0.00137	0.98256
	6000	0.98034	0.00138	0.98310
1	3000	0.92149	0.00132	0.92413

'σ' denotes the calculated uncertainty value from the MCNP calculations

Table 6-4 Minimum Volume Spheres with Plutonium/Water Mixture Surrounded by Granite Rock MCNP Results (Cont'd)

Case	Pu density (g/cc)	Volume (cm^3)	k_{eff}	σ	$k_{eff}+2\sigma$
pu_granite_refl_vol_1_3250		3250	0.94274	0.00134	0.94542
pu_granite_refl_vol_1_3500		3500	0.96747	0.00132	0.97011
pu_granite_refl_vol_1_3750		3750	0.98694	0.00133	0.98960
pu_granite_refl_vol_1_4000		4000	1.00457	0.00138	1.00733
pu_granite_refl_vol_10_700		700	0.95488	0.00107	0.95702
pu_granite_refl_vol_10_725		725	0.96434	0.00114	0.96662
pu_granite_refl_vol_10_750	10	750	0.97413	0.00116	0.97645
pu_granite_refl_vol_10_775		775	0.98678	0.00117	0.98912
pu_granite_refl_vol_10_800		800	0.99580	0.00110	0.99800

'σ' denotes the calculated uncertainty value from the MCNP calculations

Table 6-5: Minimum Volume Spheres with Plutonium/Water Mixture Surrounded by Sedimentary Rock MCNP Results

Case	Pu density (g/cc)	Volume (cm^3)	k_{eff}	σ	$k_{eff}+2σ$
pu_sedim_refl_vol_0.01_1.3e+05		1.30E+05	0.97790	0.00061	0.97912
pu_sedim_refl_vol_0.01_1.325e+05		1.33E+05	0.97920	0.00068	0.98056
pu_sedim_refl_vol_0.01_1.35e+05	0.01	1.35E+05	0.98150	0.00063	0.98276
pu_sedim_refl_vol_0.01_1.375e+05		1.38E+05	0.98443	0.00064	0.98571
pu_sedim_refl_vol_0.01_1.4e+05		1.40E+05	0.98519	0.00066	0.98651
pu_sedim_refl_vol_0.1_5700		5700	0.97469	0.00130	0.97729
pu_sedim_refl_vol_0.1_5725		5725	0.97596	0.00125	0.97846
pu_sedim_refl_vol_0.1_5750	0.1	5750	0.97686	0.00128	0.97942
pu_sedim_refl_vol_0.1_5775		5775	0.98077	0.00134	0.98345
pu_sedim_refl_vol_0.1_5800		5800	0.97970	0.00130	0.98230
pu_sedim_refl_vol_1_3500		3500	0.97929	0.00132	0.98193
pu_sedim_refl_vol_1_3525		3525	0.97581	0.00131	0.97843
pu_sedim_refl_vol_1_3550	1	3550	0.98288	0.00131	0.98550
pu_sedim_refl_vol_1_3575		3575	0.98211	0.00133	0.98477
pu_sedim_refl_vol_1_3600		3600	0.98447	0.00129	0.98705
pu_sedim_refl_vol_10_700		700	0.96325	0.00114	0.96553
pu_sedim_refl_vol_10_725		725	0.97126	0.00114	0.97354
pu_sedim_refl_vol_10_750	10	750	0.98264	0.00110	0.98484
pu_sedim_refl_vol_10_775		775	0.99104	0.00115	0.99334
pu_sedim_refl_vol_10_800		800	1.00471	0.00110	1.00691

'σ' denotes the calculated uncertainty value from the MCNP calculations

Interpolation of the results in Table 6-4 and Table 6-5 to the USL value of 0.9814 yield the following critical volumes as a function of plutonium density:

Granite

0.01 g/cm^3: 134,240 cm^3
0.1 g/cm^3: 5,963 cm^3
1 g/cm^3: 3,536 cm^3
10 g/cm^3: 760 cm^3

Sedimentary

0.01 g/cm^3: 133,455 cm^3
0.1 g/cm^3: 5,762 cm^3
1 g/cm^3: 3,5636 cm^3
10 g/cm^3: 742 cm^3

Further discussion related to spent fuel inventory is provided in Section 7.

7. CONCLUSIONS

The following results were determined in this report.

Containers

First, with respect to used fuel in containers, all scenarios were significantly subcritical:

- Intact Fuel in Containers
  - Large container (IV17): keff + 2σ = 0.58857
  - Small container (4L012): k_{eff} + 2σ = 0.48390

- Degraded Fuel in Containers
  - Large container (IV17): k_{eff} + 2σ = 0.69943
  - Small container (4L012): k_{eff} + 2σ = 0.62752

Burnup & Decay

Appendix B, Tables B-1 to B-5, present the resulting k_{eff} values for Scenarios 1 and 2, with varying burnup (220, 280, and 320 MWh/kgU) and decay time (0 to 10 Million years). For each of the scenarios, the combination producing the highest k_{eff} value is based on a burnup of 220 MWh/kgU. 220 MWh/kgU is therefore the most conservative burnup condition, as it consistently produces the highest k_{eff} values. These findings are consistent with the conclusions of other criticality studies as discussed in Section 2.9.

For intact container scenarios, Figure 7-1 graphs the criticality results from the most conservative combination of burnup (220 MWh/kgU), container type (large container), interior material (flooded, water), and surrounding material (bentonite) starting at 10-year decay time.

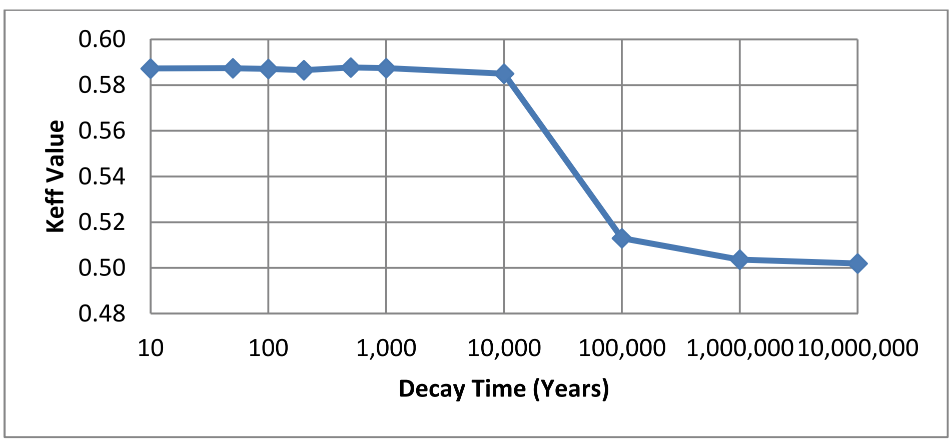

Figure 7-1: Decay Time Versus K_{eff} – Intact-Container (log scale)

Figure 7-1 shows that there is variation in k_{eff} values for decay times up to 1,000 years (high at 50 years, but low at 200 years), with the 500-year decay time producing the maximum k_{eff} result (0.58857). A noticeable decrease is seen beyond 10,000 years decay time. Overall, the variation observed within the first 1,000 years is small when compared to the range of k_{eff} results produced over the 10 Million year period.

For perspective, McCamis (1992) examined k_{eff} values for intact containers as a function of cooling times from 1 to 10,000 years (based on a burnup of 220 GJ/kgU [~61 MWh/kgU]). McCamis (1992) identified that k_{eff} decreases gradually as cooling time increases, and concluded that "while reactivity does decrease slightly with increasing cooling time, because of the differing decay rates for the various isotopes considered in the calculations, the dependence of k_{eff} on cooling time was minimal".

Overall, given the McCamis (1992) findings and the results shown in Figure 7-1, it is reasonable to conclude that the most conservative decay time would occur within the first 500 years for in-container scenarios.

Figure 7-2 similarly graphs the criticality results for the degraded-container scenario producing the highest k_{eff} value (from Appendix B, Table B-4, 1:3 water-fuel ratio).

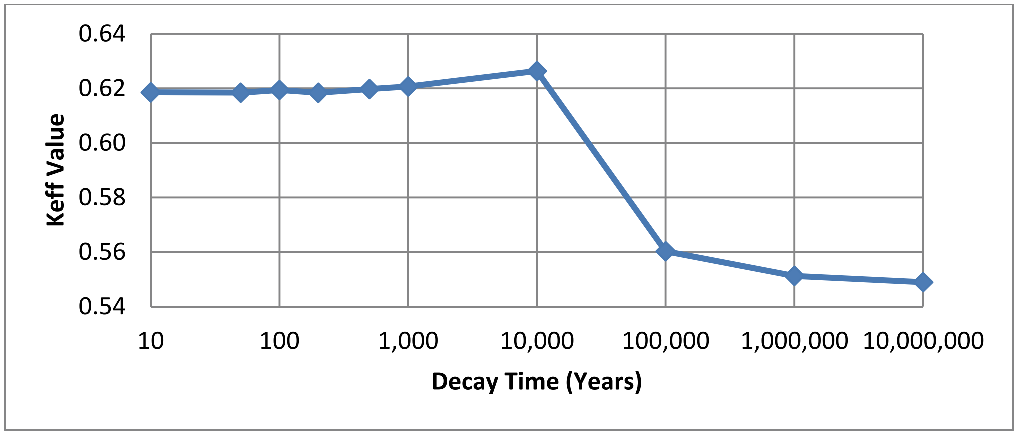

Figure 7-2: Decay Time Versus K_{eff} – Degraded-Container (log scale)

Figure 7-2 clearly shows that the maximum value occurs at the 10,000-year decay time. Therefore, when evaluating degraded-container scenarios and large time periods, the most conservative cooling time to select would be 10,000 years (based on conservative 220 MWh/kgU burnup).

Plutonium in Bentonite

With respect to the potential scenario where plutonium is released from containers and concentrates in solution within the bentonite clay:

- Plutonium in Bentonite:
  - Minimum critical mass: 1240 g Pu
  - Minimum critical volume:
    - 0.1 g/cm^3: 37,234 cm^3
    - 1 g/cm^3: 19,259 cm^3
    - 10 g/cm^3: 4,330 cm^3
  - Minimum critical concentration: 0.0108 g/cm3.

The NWMO fuel dissolution model (NWMO, 2012c) was used to estimate the total amount of Pu released per container. The ability for Pu to migrate out of the container is limited by:

- the low UO_2 fuel dissolution rate;
- solubility limit of Pu in the container;
- size and extent of the container failure; and
- transport properties of the engineered sealing materials.

The fuel dissolution model shows the 1240 g critical mass is larger than the total amount of Pu released per container over 1 million years for crystalline and sedimentary rock. This means that criticality is not possible for release from one container. Multiple containers would have to fail and migrate to the same region to reach the critical mass in bentonite.

In the unlikely event that multiple containers fail, there are numerous limiting factors that will prevent the Pu from forming a critical mass. These include:

- Pu-239 must preferentially separate from the other Pu isotopes and other chemically similar species;
- the Pu must not migrate away (i.e. diffuse) but rather preferentially migrate to a single location such as a fracture or void; and
- Pu-239 must distribute in the 0.05 g/cc arrangement at which the peak critical mass and k_{eff} occurs;

The potential consequences of multiple container failure are dependent on the solubility of the plutonium. In lower pH groundwater systems, the plutonium is likely to be soluble. In this case, it is likely that the Pu would disperse, and unlikely that it would be able to accumulate in a single location outside the container and form a critical mass. At higher pH, the plutonium is largely insoluble. This restricts the release rate from the containers.

For illustration, the fuel dissolution model (NWMO 2012c) was used to estimate the amount of Pu that dissolves and leaves a container for a repository in an environment with a low Pu solubility. It is assumed that the dissolved Pu from all the containers could precipitate in a single nearby location without much dispersion. The results for the Fourth Case Study crystalline geosphere were used (Garisto et al., 2012). The integrated amount of Pu (including decay) that leaves the container is summarized in Table 7-1. This information is combined with the minimum critical mass results for bentonite to determine the number of containers needed to provide enough Pu-239 that, when released outside of the containers, could reach critical mass.

Table 7-1: Critical Mass Versus Container Count and Pu-239 Release Over Time

Time [a]	Integrated Pu-239[a] [g]	Integrated Pu[a,b] [g]	Fraction Pu-239/Pu [-]	Critical Pu-239 Mass[c] [g]	# Containers Required [-]	# Containers in Repository[d]
1	0	0	N/C	1240	N/C	15,973 (Mark I)
10	1.39×10^{-5}	2.10×10^{-5}	6.60×10^{-1}	1240	89,362,015	
100	1.53×10^{-4}	2.31×10^{-4}	6.62×10^{-1}	1240	8,106,456	
1000	1.52×10^{-3}	2.23×10^{-3}	6.82×10^{-1}	1240	815,623	
10000	1.28×10^{-2}	1.55×10^{-2}	8.26×10^{-1}	1240	96,791	
100000	1.06×10^{-2}	3.99×10^{-2}	2.66×10^{-1}	1240	116,850	
1000000	9.07×10^{-14}	3.25×10^{-1}	2.79×10^{-13}	1240	1.37×10^{16}	

Notes:
[a]Based on model assumptions and data from Garisto et al. (2012) most notably that Pu is largely insoluble in the crystalline groundwater and the Pu release from the container is solubility limited.
[b]Includes Pu-242, Pu-240, and Pu-239.
[c]Lowest critical mass needed to achieve criticality based on Table 6-1, Table 6-2, and related discussions.
[d]Based on an inventory of 4.6×10^6 CANDU fuel bundles.
N/C – Not Calculated

Overall, given these container count results and the very specific assumptions that must be met, criticality is very unlikely.

<u>*Plutonium Surrounded by Rock*</u>

For the bounding scenario where Pu accumulates in specific fractures or other voids in the rock, it is not clear what the effective density of Pu would be when deposited in the rock formation. Analysis of such behaviour was not performed due to the very small percentage of pore space, and, because it is a more credible assumption to assume a void in the rock where Pu/water would accumulate in sufficient volume to cause a criticality.

Therefore, a range of densities and their critical masses have been calculated:

- Minimum critical volume (granite):
  - 0.01 g/cm^3: 134,240 cm^3
  - 0.1 g/cm^3: 5,963 cm^3
  - 1 g/cm^3: 3,645 cm^3
  - 10 g/cm^3: 760 cm^3

- Minimum critical volume (sedimentary):
  - 0.01 g/cm^3: 133,455 cm^3
  - 0.1 g/cm^3: 5,762 cm^3
  - 1 g/cm^3: 3,5636 cm^3
  - 10 g/cm^3: 742 cm^3

The potential accumulation of Pu in rock is different from that in bentonite because of the very low porosity of rock compared to bentonite. Therefore, the focus of the criticality calculations in rock is on the availability of sufficient void space that can accommodate Pu in different densities, sufficient to support criticality. These minimum requirements for void space and Pu concentrations can be compared to the Pu released into the rock mass from the bentonite barrier as a function of time after disposal.

Additional Considerations & Conservative Assumptions

It is important to note that Pu shares some similar chemical properties with other species, and as such, these other species have the potential to migrate along with the Pu and interfere with the conditions required to reach criticality. However it is difficult to identify which isotopes could potentially be involved and the quantities that could be carried along with Pu. Overall, the results presented in this report are conservative in this respect since they neglect the potential interference caused by other species. In addition, the calculations contained in this report assume that groundwater consists of pure water, whereas in reality, groundwater encountered at depth is quite saline. Sodium is a fairly strong neutron absorber; it would reduce the keff and therefore increase the critical volume or mass needed to reach criticality. By assuming pure water as groundwater, the results in this report are conservative.

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APPENDIX A: VALIDATION REPORT (NEWMYER 2014)

CONTENTS

Determination of Bias for MCNP 5 and the ENDF/B-VI Cross Section Library for the Nuclear Criticality Safety Calculations to Support Long-Term Disposal of Spent CANDU Nuclear Fuel

Revision A

January 2014

NSA-TR-SENES-14-01

A.1. INTRODUCTION

A.1.1 Background/Purpose

SENES Consultants (SENES) and Nuclear Safety Associates (NSA) have been contracted by the Nuclear Waste Management Organization (NWMO) to perform nuclear criticality safety calculations on the long-term disposal of spent CANDU nuclear fuel.

The purpose of these calculations is to investigate the potential for criticality for a set of bounding operational and postclosure scenarios and configurations relevant to a deep geologic repository (DGR).

The purpose of this report is to document the determination of the code bias and bias uncertainties per Section 2.3.4 of CNSC (2010). The results presented herein are based on NSA proprietary technical reports (Revolinski 2008a,b). The results within this report represent a non-proprietary summary of the referenced technical reports.

A.1.2 Limits of Applicability

The results summarized in this report are based on NSA proprietary technical reports. The results presented here should only be used directly to support the NCS calculations performed in the main report (of which this report is an Appendix).

A.2. COMPUTER CODES USED IN CALCULATION

The MCNP 5 code (LANL 2003) provides a method of analysis for criticality and shielding analysis on workstations or personal computers (PCs). MCNP 5 is one of the codes chosen for criticality safety use. The MCNP 5 (build 1.40) code is executed on the NSA servers using the Linux operating system identified in Revolinski (2007).

The default library is used for the critical experiment calculations. This is primarily the ENDF/B-VI library which contains data for all nuclides (more than 300). Table 2-1 lists the specific elements used in this evaluation. Where the default library does not contain a “natural” mixture of isotopes, the isotopic fractions are included. The light water (lwtr.60t) and poly (poly.60t) s(α,β) correction are used for water and polyethylene materials.

Table A2-1: Default Library Definitions for Various Elements

Element	ZAID	Isotopic Fraction	Element	ZAID	Isotopic Fraction
Hydrogen	1001		Nickel	28058	0.682737
Carbon	6000			28060	0.261053
Nitrogen	7014			28061	0.011263
Oxygen	8016			28062	0.035895
Sodium	11023			28064	0.009053
Magnesium	12000				
Aluminum	13027		Copper	29063	0.6917
Silicon	14000			29065	0.3083
Phosphorus	15031				
Sulfur	16032		Molybdenum	42000	
Chlorine	17000				
Potassium	19000		Plutonium	94238	
Calcium	20000			94239	
Titanium	22000			94240	
Chromium	24050	0.043474		94242	
	24052	0.837895			
	24053	0.095000	Americium	95241	
	24054	0.023632			
Manganese	25055				
Iron	26054	0.059006			
	26056	0.917181			
	26057	0.021007			
	26058	0.002806			

A.3. CALCULATIONS

A.3.1 Method Discussion

It is desirable to utilize the MCNP 5 code to model systems with homogeneous plutonium materials. These systems include homogeneous materials such as PuO_2 and PuO_2NO_3 solutions, and solid forms of Pu dispersed in hydrocarbons. This benchmark utilizes selected experiments from OECD (2006). The experiments involve water and hydrocarbon moderated plutonium with water, concrete and hydrocarbon reflectors. A variety of other neutron absorbing materials are included. The wt. % ^{240}Pu enrichment ranges from 0.84 to 18.35 wt.%. The $H/^{239}Pu$ ratios vary from 0 to 1061. The experiments selected offer a wide range of material properties. A summary of the experiment information is listed in Table 3-1.

Unmoderated Pu Metal Button Array (PU-MET-FAST-003)

Between 1965 and 1969 at the Lawrence Livermore Laboratory, plutonium metal "parts" weighing 3 kg and 6 kg were used to form reflected and unreflected arrays, of various sizes, on an aluminum table. An array was formed with half of the units on each side of a split table. A center-to-center spacing in both the vertical and lateral dimensions was chosen, and then the table was remotely pushed together.

Pu-Graphite Cylinder with Steel Reflector (PU-MET-INTER-002)

The Pu/C/SST Benchmark Assembly was part of the Diagnostic Cores Program planned for the Argonne National Laboratory (ANL) ZPR-6 and ZPR-9 critical facilities. The Pu/C/SST Benchmark Assembly had a very uniform core assembled entirely from a single core unit cell loaded into stainless steel drawers that were then loaded into the ZPR-6 matrix. The inclusion of graphite in the core unit cell moderates the spectrum more than the traditional ZPR fast reactor cores. This experiment calculates a k_{eff} significantly higher than the experimental value of 0.9868, as do all but two the OECD (2006) sample calculations. The discussion in OECD (2007) indicated that this was due to incomplete treatment of the unresolved resonances. However, the calculation herein uses the continuous-energy ENDF/B-VI which should not have the problem. Additionally, the calculation herein is consistent with other intermediate energy experiments (PU-COMP-MIX-001 and -002). Thus, the high calculation herein is unexplained and the benchmark is not considered an outlier that can be removed.

Polystyrene-Moderated PuO_2 (PU-COMP-MIX-001 and -002)

Thirty-four critical experiments involving unreflected and Plexiglas-reflected arrays of plutoniumoxide - polystyrene cubes (compacts) are reported in OECD (2007). The five unreflected experiments are evaluated in PU-COMP-MIX-001 with the 29 plexiglas reflected experiments are evaluated in PU-COMP-MIX-002. Experimental arrays were constructed from PuO2-polystyrene cubes with H/Pu ratios of 0.04, 5, 15, and 49.6.

Water-Reflected Spheres of Pu NO3 (PU-SOL-THERM-001)

This benchmark consists of six experiments with stainless steel spherical shell, 11.5 inches in diameter, surrounded by an effectively infinite water reflector. The solution was plutonium nitrate with the plutonium having a ^{240}Pu weight percent of 4.57.

Water-Reflected Spheres of Pu NO3 (PU-SOL-THERM-004)

This benchmark consists of 13 experiments with stainless steel spherical shell, 14 inches in diameter, surrounded by an effectively infinite water reflector. The solution was plutonium nitrate with the plutonium having a ^{240}Pu weight percent of 0.54 to 3.43.

Water-Reflected Spheres of Pu NO3 (PU-SOL-THERM-006)

This benchmark consists of three experiments with stainless steel spherical shell, 15 inches in diameter, surrounded by an effectively infinite water reflector. The solution was plutonium nitrate with the plutonium having a ^{240}Pu weight percent of 3.12.

Water-Reflected Spheres of Pu NO3 (PU-SOL-THERM-007)

This benchmark consists of 10 experiments with stainless steel spherical shell, 11.5 inches in diameter, surrounded by an effectively infinite water reflector. The solution was plutonium nitrate with the plutonium having a ^{240}Pu weight percent of 4.57.

Table A3-1: Homogeneous Plutonium Benchmark Critical Experiment Summary

Case	^{240}Pu wt. %	Chemical Form	Geometry	Moderator / Reflector	$H/^{239}Pu$	Other Materials	k_{exp}	σ_{exp}
PU-MET-FAST-003								
101	5.97	Metal	Array of Cylinders	None /None	0.0	Al, Steel	1.0000	0.0030
102				None /Poly			1.0000	0.0030
103				None /None			1.0000	0.0030
104				None /Poly			1.0000	0.0030
105				None /None			1.0000	0.0030
PU-MET-INTER-002								
01	4.7	Metal	Cylinder	None / Graphite	0.0	Al, Steel	0.9868	0.0026
PU-COMP-MIX-001								
01	18.35	PuO_2	Arrays of Cubes	Poly / None	0.04	Concrete	0.9986	0.0041
02	11.46				5.0		1.0000	0.0068
03	2.20				15.0		0.9990	0.0067
04	8.06				15.0		1.0000	0.0066
05	18.35				50.0		0.9989	0.0072
PU-COMP-MIX-002								
01	18.35	PuO_2	Arrays of Cubes	Poly / Plexiglas	0.0	None	0.9990	0.0046
02	18.35				0.0		0.9990	0.0046
03	18.35				0.0		0.9990	0.0046
04	18.35				0.0		0.9990	0.0046
05	18.35				0.0		0.9990	0.0046
06	11.46				5.0		1.0000	0.0075
07	11.46				5.0		1.0000	0.0075
08	11.46				5.0		1.0000	0.0075
09	11.46				5.0		1.0000	0.0075

Case	^{240}Pu wt. %	Chemical Form	Geometry	Moderator / Reflector	H/^{239}Pu	Other Materials	k_{exp}	σ_{exp}
10	2.2				15.0		1.0000	0.0073
11	2.2				15.0		1.0000	0.0073
12	2.2				15.0		1.0000	0.0073
13	2.2				15.0		1.0000	0.0073
14	2.2				15.0		1.0000	0.0073
15	2.2				15.0		1.0000	0.0073
16	2.2				15.0		1.0000	0.0073
17	8.06				15.0		0.9988	0.0055
18	8.06				15.0		0.9988	0.0055
19	8.06				15.0		0.9988	0.0055
20	8.06				15.0		0.9988	0.0055
21	8.06				15.0		0.9988	0.0055
22	8.06				15.0		0.9988	0.0055
23	18.35				50.0		1.0000	0.0068
24	18.35				50.0		1.0000	0.0068
25	18.35				50.0		1.0000	0.0068
26	18.35				50.0		1.0000	0.0068
27	18.35				50.0		1.0000	0.0068
28	18.35				50.0		1.0000	0.0068
29	18.35				50.0		1.0000	0.0068
PU-SOL-THERM-001								
1	4.57	$PuNO_3$	Sphere	Water / Water	371.0	Steel	1.0000	0.0050
2					272.0		1.0000	0.0050
3					216.0		1.0000	0.0050
4					190.0		1.0000	0.0050
5					180.0		1.0000	0.0050
6					91.0		1.0000	0.0050
PU-SOL-THERM-004								
01	0.54	$PuNO_3$	Sphere	Water / Water	987.0	Steel	1.0000	0.0047
02	0.54				977.0		1.0000	0.0047
03	0.54				935.0		1.0000	0.0047
04	0.54				889.0		1.0000	0.0047
05	0.54				942.0		1.0000	0.0047
06	1.76				927.0		1.0000	0.0047

Case	^{240}Pu wt. %	Chemical Form	Geometry	Moderator / Reflector	H/^{239}Pu	Other Materials	k_{exp}	σ_{exp}
07	3.12				892.0		1.0000	0.0047
08	3.21				869.0		1.0000	0.0047
09	3.21				805.0		1.0000	0.0047
10	3.12				689.0		1.0000	0.0047
11	3.12				592.0		1.0000	0.0047
12	3.12				893.0		1.0000	0.0047
13	3.34				903.0		1.0000	0.0047
PU-SOL-THERM-006								
1	3.12	$PuNO_3$	Sphere	Water / Water	1061.0	Steel	1.0000	0.0035
2					1018.0		1.0000	0.0035
3					940.0		1.0000	0.0035
PU-SOL-THERM-007								
2	4.57	$PuNO_3$	Sphere	Water / Water	110.0	Steel	1.0000	0.0047
3					114.0		1.0000	0.0047
5					268.0		1.0000	0.0047
6					262.0		1.0000	0.0047
7					266.0		1.0000	0.0047
8					259.0		1.0000	0.0047
9					260.0		1.0000	0.0047
10					285.0		1.0000	0.0047

A.3.2 Input

All input is obtained from OECD (2006).

A.3.3 Evaluations and Analysis of the Calculations

The MCNP 5 (build 1.40) code is executed on the NSA servers using the Linux operating system identified in Revolinski (2007). Each case was run for sufficient neutron generations and neutrons per generation to achieve a calculation uncertainty (σ_{calc}) about 0.001. The calculation results are recorded in Revolinski (2008a). The data were plotted as k_{normal} vs H/^{239}Pu (solutions and oxides only), k_{normal} vs ANECF (average neutron energy causing fission), and k_{normal} vs ^{240}Pu wt.% in Figures 1, 2 and 3 of Revolinski (2008a). The data are grouped as oxides, solutions and metals. Revolinski (2008a) concluded that the data are not correlated to any parameter. Therefore there is no bias as a function of the evaluated parameters.

A.3.4 Determination of Bias and Bias Uncertainty

RD-327 (CNSC 2010) requires that calculational methods used for nuclear criticality safety (e.g., determining k_{eff} of a system or deriving subcritical limits) be validated to determine the appropriate biases and uncertainties for the areas of applicability. The bias and uncertainty represent the numerical difference between the results of modeling critical benchmark experiments with a computer code and a k_{eff} of 1.0. These biases may result in either under- or over-predictions of criticality. The bias may be reported as either a positive or negative bias. A positive bias occurs when the computations tend to report a higher k_{eff} than the benchmark experiments (i.e., k_{eff} > 1.0). A negative bias occurs when the calculated results tend to report a lower k_{eff} than the benchmark experiments (i.e., k_{eff} < 1.0).

Biases, and associated uncertainties, are determined through statistical treatment of the criticality benchmark experiment calculated results. If the benchmark results are normally distributed, then a technique such as the single sided lower tolerance limit may be used. Several tests are performed on the benchmark data to confirm they are from a normal distribution with a high degree of probability. The Shapiro-Wilk, modified Chi Square, Kolmogorov-Smirnov, and Lilliefors tests for normality are performed for all critical experiment cases investigated. The Shapiro-Wilk test is the preferred test for data set populations of 50 or less with the other test being used for larger populations.

If the benchmark experiment results are verified to be part of a normal distribution, a weighted, single sided lower tolerance limit (LTL) technique may be used to construct an Upper Subcritical Limit for criticality. The weighted, single sided lower tolerance limit is calculated with a 95% confidence that 95% of the benchmark data lies above it. Thus a calculation involving a subcritical system would have a 95% confidence that 95% of all calculations performed on it would yield a result less than the tolerance limit. The weighted, single sided lower tolerance limit is calculated using the method presented in NUREG/CR-6698 (Dean & Tayloe 2001). The weighted, single sided lower tolerance limit is adjusted by applying a margin of subcriticality to define the USL.

If the benchmark critical experiment data are not from a normal distribution, the Upper Subcritical Limit must be determined using non-parametric techniques. The method used for this evaluation is discussed in Revolinski (2008b). The approach is more conservative than other non-parametric techniques available to determine distribution-free confidence interval. This method results in a determination of the degree of confidence that a fraction of the true population of data lies above the smallest observed value.

The above statistical methods were applied to all benchmark experiment calculation results in Revolinski (2008b). In addition, three subsets of experimental calculation results were examined; plutonium solutions, plutonium oxides, and plutonium metals.

Revolinski (2008b) concluded that the bias and bias uncertainty for the plutonium oxide subset was larger than the bias and bias uncertainty for all experimental results combined. Therefore bias and bias uncertainty determined for this subset will be used for the entire homogeneous plutonium experiment set. The bias and bias uncertainty, as determined in Revolinski (2008b), is show in Section 4.2.

A.4. CONCLUSIONS

Homogeneous plutonium benchmark critical cases were modeled using MCNP 5 with the default cross section library. Revolinski 2008a documents the methodology and results. The experiments, discussed previously in Appendix A Section 3.1, include oxide, solution and metal forms, and include the thermal, intermediate and fast energy ranges. It is concluded that the k_{eff}

data are not correlated with any of the Area of Applicability (AoA) parameters. The AoA parameters are listed in Table 4-1.

The bias and bias uncertainty for homogeneous plutonium systems modeled using MCNP 5 and the default cross section library was determined in Revolinski (2008b) as:

$$b + \sigma_b = 0.0186$$

The Upper Subcritical Limit (USL) for specific applications shall include a Margin of Subcriticality (MoS) appropriate for the application. The analyzed system k_{eff} is determined to be acceptable if:

$$k_s + 2\sigma_s \leq 0.9814 - MoS$$

where:

k_s = k_{eff} from MCNP 5, and
σ_s = the standard deviation in k_{eff} from MCNP 5

Table A4-1: Area of Applicability Summary

Parameter	Combined AoA
Fissile Material	Spent CANDU Fuel (including U-235, U-236, U-238, Pu-239, Pu-240); Pu metal, PuO_2, $PuO_2(NO_3)_2$
Fissile Material Form	Plutonium Solids and Solutions.
$H/^{235}U$ ratio	$0.0 \leq H/^{239}Pu \leq 1061$
Average Neutron Energy Causing Fission (MeV)	0.005 < ANECF < 1.91
wt% ^{240}Pu	0.84 to 18.35 wt% ^{240}Pu
Moderating Materials	Polystyrene, Graphite, Water, HNO_3
Reflecting Materials	Water, Plexiglas, Polyethylene, Unreflected
Absorber Materials	Concrete, Aluminum, Steel
Geometry	Sphere, Cylinder and Cuboid Arrays

APPENDIX A REFERENCES

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APPENDIX B: DETAILED MCNP RESULTS

Table B-1: MCNP Calculation Configuration/Set Numbers and their Parameter Values

Set Number	Burnup (MWh/kgU)	Decay time (years)	^{235}U g/kgU	^{236}U g/kgU	^{238}U g/kgU	^{239}Pu g/kgU	^{240}Pu g/kgU	O g/kgU
1	220	0	1.67	0.8223	981.8	2.605	1.285	134.3947
2		10	1.671	0.8236	981.8	2.684	1.284	134.3947
3		50	1.674	0.8289	981.8	2.681	1.279	134.3947
4		100	1.678	0.8356	981.8	2.678	1.272	134.3947
5		200	1.685	0.8487	981.8	2.67	1.259	134.3947
6		500	1.708	0.8873	981.8	2.647	1.219	134.3947
7		1000	1.745	0.949	981.8	2.61	1.157	134.3947
8		1E+04	2.33	1.646	981.8	2.017	0.4471	134.3947
9		1E+05	4.167	2.081	981.8	0.1519	3.32E-05	134.3947
10		1E+06	4.313	2.026	981.7	3.85E-12	4.23E-15	134.3947
11		1E+07	4.275	1.552	980.3	2.01E-12	4.5E-15	134.3947
12	280	0	1.095	0.9022	979.1	2.672	1.633	134.3947
13		10	1.096	0.9039	979.1	2.754	1.632	134.3947
14		50	1.099	0.9107	979.1	2.751	1.626	134.3947
15		100	1.103	0.9191	979.1	2.747	1.618	134.3947
16	320	0	0.8213	0.938	977.3	2.678	1.829	134.3947
17		10	0.8221	0.9399	977.3	2.76	1.828	134.3947
18		50	0.8252	0.9475	977.3	2.757	1.821	134.3947
19		100	0.8291	0.9569	977.3	2.753	1.812	134.3947

Table B-2: Intact Fuel in Large Container MCNP Results

Case	Material Inside	Material Outside	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
IV17_intact_h2oair_sing_1_1_1	Air	Air	1	0.20901	0.00021	0.20943
IV17_intact_h2oair_sing_1_1_2			2	0.20946	0.00022	0.20990
IV17_intact_h2oair_sing_1_1_3			3	0.20958	0.00021	0.21000
IV17_intact_h2oair_sing_1_1_4			4	0.20982	0.00022	0.21026
IV17_intact_h2oair_sing_1_1_5			5	0.20949	0.00021	0.20991
IV17_intact_h2oair_sing_1_1_6			6	0.20971	0.00022	0.21015
IV17_intact_h2oair_sing_1_1_7			7	0.20967	0.00021	0.21009
IV17_intact_h2oair_sing_1_1_8			8	0.20793	0.00022	0.20837
IV17_intact_h2oair_sing_1_1_9			9	0.20513	0.00021	0.20555
IV17_intact_h2oair_sing_1_1_10			10	0.20447	0.00021	0.20489
IV17_intact_h2oair_sing_1_1_11			11	0.20404	0.00022	0.20448
IV17_intact_h2oair_sing_1_1_12			12	0.20650	0.00023	0.20696
IV17_intact_h2oair_sing_1_1_13			13	0.20716	0.00022	0.20760
IV17_intact_h2oair_sing_1_1_14			14	0.20672	0.00023	0.20718
IV17_intact_h2oair_sing_1_1_15			15	0.20713	0.00022	0.20757
IV17_intact_h2oair_sing_1_1_16			16	0.20483	0.00021	0.20525
IV17_intact_h2oair_sing_1_1_17			17	0.20559	0.00023	0.20605
IV17_intact_h2oair_sing_1_1_18			18	0.20584	0.00021	0.20626
IV17_intact_h2oair_sing_1_1_19			19	0.20560	0.00022	0.20604
IV17_intact_h2oair_sing_1_2_1		Water	1	0.21008	0.00022	0.21052
IV17_intact_h2oair_sing_1_2_2			2	0.21038	0.00021	0.21080
IV17_intact_h2oair_sing_1_2_3			3	0.21077	0.00020	0.21117
IV17_intact_h2oair_sing_1_2_4			4	0.21085	0.00022	0.21129
IV17_intact_h2oair_sing_1_2_5			5	0.21094	0.00022	0.21138

Case	Material Inside	Material Outside	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
IV17_intact_h2oair_sing_1_2_6			6	0.21044	0.00021	0.21086
IV17_intact_h2oair_sing_1_2_7			7	0.21091	0.00020	0.21131
IV17_intact_h2oair_sing_1_2_8			8	0.20902	0.00023	0.20948
IV17_intact_h2oair_sing_1_2_9			9	0.20603	0.00021	0.20645
IV17_intact_h2oair_sing_1_2_10			10	0.20662	0.00021	0.20704
IV17_intact_h2oair_sing_1_2_11			11	0.20554	0.00022	0.20598
IV17_intact_h2oair_sing_1_2_12			12	0.20780	0.00022	0.20824
IV17_intact_h2oair_sing_1_2_13			13	0.20803	0.00021	0.20845
IV17_intact_h2oair_sing_1_2_14			14	0.20827	0.00022	0.20871
IV17_intact_h2oair_sing_1_2_15			15	0.20800	0.00021	0.20842
IV17_intact_h2oair_sing_1_2_16			16	0.20626	0.00022	0.20670
IV17_intact_h2oair_sing_1_2_17			17	0.20692	0.00021	0.20734
IV17_intact_h2oair_sing_1_2_18			18	0.20703	0.00023	0.20749
IV17_intact_h2oair_sing_1_2_19			19	0.20705	0.00022	0.20749
IV17_intact_h2oair_sing_2_1_1	Water	Air	1	0.58074	0.00040	0.58154
IV17_intact_h2oair_sing_2_1_2			2	0.58694	0.00041	0.58776
IV17_intact_h2oair_sing_2_1_3			3	0.58640	0.00041	0.58722
IV17_intact_h2oair_sing_2_1_4			4	0.58652	0.00040	0.58732
IV17_intact_h2oair_sing_2_1_5			5	0.58586	0.00040	0.58666
IV17_intact_h2oair_sing_2_1_6			6	0.58593	0.00038	0.58669
IV17_intact_h2oair_sing_2_1_7			7	0.58696	0.00040	0.58776
IV17_intact_h2oair_sing_2_1_8			8	0.58368	0.00042	0.58452
IV17_intact_h2oair_sing_2_1_9			9	0.51237	0.00035	0.51307
IV17_intact_h2oair_sing_2_1_10			10	0.50264	0.00035	0.50334
IV17_intact_h2oair_sing_2_1_11			11	0.50095	0.00034	0.50163
IV17_intact_h2oair_sing_2_1_12			12	0.55411	0.00038	0.55487
IV17_intact_h2oair_sing_2_1_13			13	0.56028	0.00041	0.56110
IV17_intact_h2oair_sing_2_1_14			14	0.56064	0.00041	0.56146
IV17_intact_h2oair_sing_2_1_15			15	0.56132	0.00042	0.56216
IV17_intact_h2oair_sing_2_1_16			16	0.53936	0.00039	0.54014
IV17_intact_h2oair_sing_2_1_17			17	0.54623	0.00039	0.54701
IV17_intact_h2oair_sing_2_1_18			18	0.54548	0.00039	0.54626
IV17_intact_h2oair_sing_2_1_19			19	0.54518	0.00040	0.54598
IV17_intact_h2oair_sing_2_2_1		Water	1	0.58129	0.00039	0.58207
IV17_intact_h2oair_sing_2_2_2			2	0.58682	0.00039	0.58760
IV17_intact_h2oair_sing_2_2_3			3	0.58704	0.00043	0.58790
IV17_intact_h2oair_sing_2_2_4			4	0.58625	0.00041	0.58707
IV17_intact_h2oair_sing_2_2_5			5	0.58680	0.00045	0.58770
IV17_intact_h2oair_sing_2_2_6			6	0.58688	0.00042	0.58772
IV17_intact_h2oair_sing_2_2_7			7	0.58700	0.00043	0.58786
IV17_intact_h2oair_sing_2_2_8			8	0.58461	0.00041	0.58543
IV17_intact_h2oair_sing_2_2_9			9	0.51309	0.00034	0.51377
IV17_intact_h2oair_sing_2_2_10			10	0.50256	0.00032	0.50320
IV17_intact_h2oair_sing_2_2_11			11	0.50176	0.00032	0.50240
IV17_intact_h2oair_sing_2_2_12			12	0.55541	0.00037	0.55615
IV17_intact_h2oair_sing_2_2_13			13	0.56080	0.00040	0.56160
IV17_intact_h2oair_sing_2_2_14			14	0.56182	0.00042	0.56266
IV17_intact_h2oair_sing_2_2_15			15	0.56096	0.00042	0.56180
IV17_intact_h2oair_sing_2_2_16			16	0.53980	0.00038	0.54056
IV17_intact_h2oair_sing_2_2_17			17	0.54585	0.00041	0.54667
IV17_intact_h2oair_sing_2_2_18			18	0.54676	0.00041	0.54758

Case	Material Inside	Material Outside	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
IV17_intact_h2oair_sing_2_2_19			19	0.54625	0.00039	0.54703
IV17_intact_bent_sing_2_3_1		Bentonite	1	0.58219	0.00042	0.58303
IV17_intact_bent_sing_2_3_2			2	0.58725	0.00041	0.58807
IV17_intact_bent_sing_2_3_3			3	0.58739	0.00043	0.58825
IV17_intact_bent_sing_2_3_4			4	0.58708	0.00042	0.58792
IV17_intact_bent_sing_2_3_5			5	0.58656	0.00040	0.58736
IV17_intact_bent_sing_2_3_6			6	0.58771	0.00043	0.58857
IV17_intact_bent_sing_2_3_7			7	0.58741	0.00041	0.58823
IV17_intact_bent_sing_2_3_8			8	0.58501	0.00041	0.58583
IV17_intact_bent_sing_2_3_9			9	0.51302	0.00034	0.51370
IV17_intact_bent_sing_2_3_10			10	0.50366	0.00032	0.50430
IV17_intact_bent_sing_2_3_11			11	0.50196	0.00034	0.50264

Note: Shading indicates highest Keff value.
'σ' denotes the calculated uncertainty value from the MCNP calculations

Table B-3: Intact Fuel in Small Container MCNP Results

Case	Material Inside	Material Outside	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
C4L012_intact_h2oair_sing_2_1_1	Water	Air	1	0.46463	0.00039	0.46541
C4L012_intact_h2oair_sing_2_1_2			2	0.46973	0.00040	0.47053
C4L012_intact_h2oair_sing_2_1_3			3	0.47030	0.00038	0.47106
C4L012_intact_h2oair_sing_2_1_4			4	0.46989	0.00040	0.47069
C4L012_intact_h2oair_sing_2_1_5			5	0.47033	0.00042	0.47117
C4L012_intact_h2oair_sing_2_1_6			6	0.47032	0.00040	0.47112
C4L012_intact_h2oair_sing_2_1_7			7	0.46929	0.00041	0.47011
C4L012_intact_h2oair_sing_2_1_8			8	0.46606	0.00038	0.46682
C4L012_intact_h2oair_sing_2_1_9			9	0.40573	0.00033	0.40639
C4L012_intact_h2oair_sing_2_1_10			10	0.39711	0.00033	0.39777
C4L012_intact_h2oair_sing_2_1_11			11	0.39644	0.00032	0.39708
C4L012_intact_h2oair_sing_2_2_1		Water	1	0.46796	0.00039	0.46874
C4L012_intact_h2oair_sing_2_2_2			2	0.47269	0.00041	0.47351
C4L012_intact_h2oair_sing_2_2_3			3	0.47249	0.00038	0.47325
C4L012_intact_h2oair_sing_2_2_4			4	0.47286	0.00039	0.47364
C4L012_intact_h2oair_sing_2_2_5			5	0.47294	0.00042	0.47378
C4L012_intact_h2oair_sing_2_2_6			6	0.47249	0.00039	0.47327
C4L012_intact_h2oair_sing_2_2_7			7	0.47257	0.00041	0.47339
C4L012_intact_h2oair_sing_2_2_8			8	0.46825	0.00043	0.46911
C4L012_intact_h2oair_sing_2_2_9			9	0.40741	0.00034	0.40809
C4L012_intact_h2oair_sing_2_2_10			10	0.39912	0.00030	0.39972
C4L012_intact_h2oair_sing_2_2_11			11	0.39867	0.00033	0.39933
C4L012_intact_bent_sing_2_3_1		Bentonite	1	0.47712	0.00038	0.47788
C4L012_intact_bent_sing_2_3_2			2	0.48221	0.00040	0.48301
C4L012_intact_bent_sing_2_3_3			3	0.48308	0.00038	0.48384
C4L012_intact_bent_sing_2_3_4			4	0.48310	0.00040	0.48390
C4L012_intact_bent_sing_2_3_5			5	0.48236	0.00038	0.48312
C4L012_intact_bent_sing_2_3_6			6	0.48208	0.00040	0.48288
C4L012_intact_bent_sing_2_3_7			7	0.48196	0.00042	0.48280

Case	Material Inside	Material Outside	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
C4L012_intact_bent_sing_2_3_8			8	0.47872	0.00043	0.47958
C4L012_intact_bent_sing_2_3_9			9	0.41607	0.00033	0.41673
C4L012_intact_bent_sing_2_3_10			10	0.40708	0.00034	0.40776
C4L012_intact_bent_sing_2_3_11			11	0.40650	0.00031	0.40712

Note: Shading indicates highest Keff value.
'σ' denotes the calculated uncertainty value from the MCNP calculations

Table B-4: Degraded Fuel in Large Container MCNP Results

Case	Water/Fuel Ratio	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
IV17_degrade_bent_sing_1_1	1	1	0.67549	0.00048	0.67645
IV17_degrade_bent_sing_2_1		2	0.68173	0.00047	0.68267
IV17_degrade_bent_sing_3_1		3	0.68070	0.00046	0.68162
IV17_degrade_bent_sing_4_1		4	0.68215	0.00048	0.68311
IV17_degrade_bent_sing_5_1		5	0.68135	0.00045	0.68225
IV17_degrade_bent_sing_6_1		6	0.68266	0.00046	0.68358
IV17_degrade_bent_sing_7_1		7	0.68437	0.00045	0.68527
IV17_degrade_bent_sing_8_1		8	0.69489	0.00046	0.69581
IV17_degrade_bent_sing_9_1		9	0.63009	0.00039	0.63087
IV17_degrade_bent_sing_10_1		10	0.62043	0.00036	0.62115
IV17_degrade_bent_sing_11_1		11	0.61878	0.00037	0.61952
IV17_degrade_bent_sing_1_1.1	1.1	1	0.67834	0.00046	0.67926
IV17_degrade_bent_sing_2_1.1		2	0.68627	0.00046	0.68719
IV17_degrade_bent_sing_3_1.1		3	0.68497	0.00046	0.68589
IV17_degrade_bent_sing_4_1.1		4	0.68527	0.00043	0.68613
IV17_degrade_bent_sing_5_1.1		5	0.68586	0.00047	0.68680
IV17_degrade_bent_sing_6_1.1		6	0.68562	0.00050	0.68662
IV17_degrade_bent_sing_7_1.1		7	0.68805	0.00041	0.68887
IV17_degrade_bent_sing_8_1.1		8	0.69735	0.00042	0.69819
IV17_degrade_bent_sing_9_1.1		9	0.63028	0.00036	0.63100
IV17_degrade_bent_sing_10_1.1		10	0.61975	0.00037	0.62049
IV17_degrade_bent_sing_11_1.1		11	0.61921	0.00037	0.61995
IV17_degrade_bent_sing_1_1.2	1.2	1	0.68064	0.00043	0.68150
IV17_degrade_bent_sing_2_1.2		2	0.68824	0.00044	0.68912
IV17_degrade_bent_sing_3_1.2		3	0.68762	0.00043	0.68848
IV17_degrade_bent_sing_4_1.2		4	0.68747	0.00045	0.68837
IV17_degrade_bent_sing_5_1.2		5	0.68793	0.00044	0.68881
IV17_degrade_bent_sing_6_1.2		6	0.68786	0.00043	0.68872
IV17_degrade_bent_sing_7_1.2		7	0.68949	0.00045	0.69039
IV17_degrade_bent_sing_8_1.2		8	0.69851	0.00046	0.69943
IV17_degrade_bent_sing_9_1.2		9	0.62927	0.00037	0.63001
IV17_degrade_bent_sing_10_1.2		10	0.61931	0.00036	0.62003
IV17_degrade_bent_sing_11_1.2		11	0.61813	0.00039	0.61891
IV17_degrade_bent_sing_1_1.3	1.3	1	0.68083	0.00042	0.68167
IV17_degrade_bent_sing_2_1.3		2	0.68726	0.00043	0.68812
IV17_degrade_bent_sing_3_1.3		3	0.68815	0.00043	0.68901
IV17_degrade_bent_sing_4_1.3		4	0.68847	0.00046	0.68939

Case	Water/Fuel Ratio	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
IV17_degrade_bent_sing_5_1.3		5	0.68948	0.00044	0.69036
IV17_degrade_bent_sing_6_1.3		6	0.68898	0.00043	0.68984
IV17_degrade_bent_sing_7_1.3		7	0.68993	0.00048	0.69089
IV17_degrade_bent_sing_8_1.3		8	0.69827	0.00042	0.69911
IV17_degrade_bent_sing_9_1.3		9	0.62736	0.00038	0.62812
IV17_degrade_bent_sing_10_1.3		10	0.61858	0.00035	0.61928
IV17_degrade_bent_sing_11_1.3		11	0.61727	0.00036	0.61799
IV17_degrade_bent_sing_1_1.4	1.4	1	0.68069	0.00045	0.68159
IV17_degrade_bent_sing_2_1.4		2	0.68773	0.00041	0.68855
IV17_degrade_bent_sing_3_1.4		3	0.68769	0.00045	0.68859
IV17_degrade_bent_sing_4_1.4		4	0.68833	0.00044	0.68921
IV17_degrade_bent_sing_5_1.4		5	0.68878	0.00046	0.68970
IV17_degrade_bent_sing_6_1.4		6	0.68917	0.00041	0.68999
IV17_degrade_bent_sing_7_1.4		7	0.69097	0.00045	0.69187
IV17_degrade_bent_sing_8_1.4		8	0.69728	0.00044	0.69816
IV17_degrade_bent_sing_9_1.4		9	0.62519	0.00037	0.62593
IV17_degrade_bent_sing_10_1.4		10	0.61604	0.00039	0.61682
IV17_degrade_bent_sing_11_1.4		11	0.61422	0.00034	0.61490
IV17_degrade_bent_sing_1_1.5	1.5	1	0.68084	0.00042	0.68168
IV17_degrade_bent_sing_2_1.5		2	0.68681	0.00045	0.68771
IV17_degrade_bent_sing_3_1.5		3	0.68710	0.00044	0.68798
IV17_degrade_bent_sing_4_1.5		4	0.68657	0.00045	0.68747
IV17_degrade_bent_sing_5_1.5		5	0.68759	0.00044	0.68847
IV17_degrade_bent_sing_6_1.5		6	0.68854	0.00045	0.68944
IV17_degrade_bent_sing_7_1.5		7	0.68938	0.00043	0.69024
IV17_degrade_bent_sing_8_1.5		8	0.69578	0.00041	0.69660
IV17_degrade_bent_sing_9_1.5		9	0.62195	0.00036	0.62267
IV17_degrade_bent_sing_10_1.5		10	0.61187	0.00037	0.61261
IV17_degrade_bent_sing_11_1.5		11	0.61065	0.00035	0.61135

Note: Shading indicates highest Keff value.
'σ' denotes the calculated uncertainty value from the MCNP calculations

Table B-5: Degraded Fuel in Small Container MCNP Results

Case	Water/Fuel Ratio	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
C4L012_degrad_bent_sing_1_1	1	1	0.60411	0.00060	0.60531
C4L012_degrad_bent_sing_2_1		2	0.60930	0.00061	0.61052
C4L012_degrad_bent_sing_3_1		3	0.60977	0.00061	0.61099
C4L012_degrad_bent_sing_4_1		4	0.60960	0.00060	0.61080
C4L012_degrad_bent_sing_5_1		5	0.60924	0.00063	0.61050
C4L012_degrad_bent_sing_6_1		6	0.61117	0.00063	0.61243
C4L012_degrad_bent_sing_7_1		7	0.61175	0.00057	0.61289
C4L012_degrad_bent_sing_8_1		8	0.61947	0.00057	0.62061
C4L012_degrad_bent_sing_9_1		9	0.55882	0.00052	0.55986
C4L012_degrad_bent_sing_10_1		10	0.55034	0.00051	0.55136

Case	Water/Fuel Ratio	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
C4L012_degrad_bent_sing_11_1		11	0.54834	0.00050	0.54934
C4L012_degrad_bent_sing_1_1.1	1.1	1	0.60699	0.00058	0.60815
C4L012_degrad_bent_sing_2_1.1		2	0.61274	0.00058	0.61390
C4L012_degrad_bent_sing_3_1.1		3	0.61491	0.00063	0.61617
C4L012_degrad_bent_sing_4_1.1		4	0.61493	0.00058	0.61609
C4L012_degrad_bent_sing_5_1.1		5	0.61554	0.00059	0.61672
C4L012_degrad_bent_sing_6_1.1		6	0.61483	0.00060	0.61603
C4L012_degrad_bent_sing_7_1.1		7	0.61576	0.00059	0.61694
C4L012_degrad_bent_sing_8_1.1		8	0.62162	0.00056	0.62274
C4L012_degrad_bent_sing_9_1.1		9	0.55996	0.00052	0.56100
C4L012_degrad_bent_sing_10_1.1		10	0.55143	0.00047	0.55237
C4L012_degrad_bent_sing_11_1.1		11	0.54974	0.00050	0.55074
C4L012_degrad_bent_sing_1_1.2	1.2	1	0.60980	0.00057	0.61094
C4L012_degrad_bent_sing_2_1.2		2	0.61758	0.00060	0.61878
C4L012_degrad_bent_sing_3_1.2		3	0.61694	0.00062	0.61818
C4L012_degrad_bent_sing_4_1.2		4	0.61621	0.00062	0.61745
C4L012_degrad_bent_sing_5_1.2		5	0.61742	0.00060	0.61862
C4L012_degrad_bent_sing_6_1.2		6	0.61752	0.00059	0.61870
C4L012_degrad_bent_sing_7_1.2		7	0.61869	0.00063	0.61995
C4L012_degrad_bent_sing_8_1.2		8	0.62451	0.00059	0.62569
C4L012_degrad_bent_sing_9_1.2		9	0.55965	0.00051	0.56067
C4L012_degrad_bent_sing_10_1.2		10	0.55027	0.00048	0.55123
C4L012_degrad_bent_sing_11_1.2		11	0.54942	0.00049	0.55040
C4L012_degrad_bent_sing_1_1.3	1.3	1	0.61388	0.00061	0.61510
C4L012_degrad_bent_sing_2_1.3		2	0.61853	0.00061	0.61975
C4L012_degrad_bent_sing_3_1.3		3	0.61839	0.00057	0.61953
C4L012_degrad_bent_sing_4_1.3		4	0.61939	0.00056	0.62051
C4L012_degrad_bent_sing_5_1.3		5	0.61836	0.00060	0.61956
C4L012_degrad_bent_sing_6_1.3		6	0.61974	0.00061	0.62096
C4L012_degrad_bent_sing_7_1.3		7	0.62069	0.00060	0.62189
C4L012_degrad_bent_sing_8_1.3		8	0.62634	0.00059	0.62752
C4L012_degrad_bent_sing_9_1.3		9	0.56026	0.00052	0.56130
C4L012_degrad_bent_sing_10_1.3		10	0.55125	0.00050	0.55225
C4L012_degrad_bent_sing_11_1.3		11	0.54893	0.00049	0.54991
C4L012_degrad_bent_sing_1_1.4	1.4	1	0.61267	0.00054	0.61375
C4L012_degrad_bent_sing_2_1.4		2	0.61902	0.00059	0.62020
C4L012_degrad_bent_sing_3_1.4		3	0.61864	0.00058	0.61980
C4L012_degrad_bent_sing_4_1.4		4	0.61885	0.00059	0.62003
C4L012_degrad_bent_sing_5_1.4		5	0.62066	0.00056	0.62178
C4L012_degrad_bent_sing_6_1.4		6	0.62060	0.00059	0.62178
C4L012_degrad_bent_sing_7_1.4		7	0.62000	0.00057	0.62114
C4L012_degrad_bent_sing_8_1.4		8	0.62614	0.00057	0.62728
C4L012_degrad_bent_sing_9_1.4		9	0.55870	0.00051	0.55972
C4L012_degrad_bent_sing_10_1.4		10	0.54946	0.00051	0.55048
C4L012_degrad_bent_sing_11_1.4		11	0.54790	0.00047	0.54884
C4L012_degrad_bent_sing_1_1.5	1.5	1	0.61201	0.00057	0.61315
C4L012_degrad_bent_sing_2_1.5		2	0.61894	0.00057	0.62008
C4L012_degrad_bent_sing_3_1.5		3	0.62042	0.00058	0.62158

Case	Water/Fuel Ratio	Burnup data set *(Table C-1)*	k_{eff}	σ	$k_{eff}+2\sigma$
C4L012_degrad_bent_sing_4_1.5		4	0.61924	0.00057	0.62038
C4L012_degrad_bent_sing_5_1.5		5	0.61943	0.00058	0.62059
C4L012_degrad_bent_sing_6_1.5		6	0.61995	0.00056	0.62107
C4L012_degrad_bent_sing_7_1.5		7	0.62146	0.00058	0.62262
C4L012_degrad_bent_sing_8_1.5		8	0.62355	0.00055	0.62465
C4L012_degrad_bent_sing_9_1.5		9	0.55589	0.00049	0.55687
C4L012_degrad_bent_sing_10_1.5		10	0.54694	0.00045	0.54784
C4L012_degrad_bent_sing_11_1.5		11	0.54518	0.00046	0.54610

Note: Shading indicates highest Keff value.
'σ' denotes the calculated uncertainty value from the MCNP calculations

APPENDIX C: LITERATURE REVIEW

Complete Literature Search Results:

The following is an expanded list of reference material obtained through the literature search. Where available, brief descriptive text – abstracts or executive summaries – is provided. Descriptive text is in the form of the authors original words, without additional conclusions or revisions.

(1985). Criticality Safety Considerations in the Storage of Nuclear Material Throughout the Fuel Cycle. Proceedings of a Topical Meeting, La Grange Park, IL, USA, ANS.

The following topics were dealt with: criticality safety during manufacturing of fuel; storage problems and experience of solutions, fresh and spent fuel; vault and transport storage; calculations and benchmark problems. Abstracts of individual papers can be found under the relevant classification codes in this or other issues.

(1997). Proceedings of the Topical Meeting on Criticality Safety Challenges in the Next Decade, La Grange Park, IL, USA, American Nucl Soc.

This conference highlights one of the major trends in criticality safety that of changing from nuclear material production and interim storage to long-term storage of spent fuel and management of waste byproducts contaminated with fissile material. These new activities, especially with materials now classified as waste, present considerable challenges. Whereas fuel assemblies and most other fabricated and laboratory nuclear materials are well characterized, nuclear wastes might not be well characterized, making precise conclusions impossible.

Anderson, W.J., P.M. O'Leary, *et al*. (2000). Selection of reactor criticals as benchmarks for spent nuclear fuels, USA, ANS.

With increased interest in the use of burnup credit for spent nuclear fuel (SNF) storage, transportation, and disposal, the scarcity of SNF critical experiments is a pressing concern in the nuclear industry. Commercial reactor criticals (CRCs) offer an immediate solution to this problem. A CRC is a zero-power critical measurement, which is the point of criticality in a reactor prior to the addition of sensible heat. Each CRC is a critical condition defined by the measured reactor core conditions. The CRC data are actual reactor operating history data and represent the exact reactor conditions. Historically, laboratory critical experiments (LCEs) have been used for criticality benchmarks. However, for SNF applications, no directly applicable LCEs are available. For burnup credit applications, CRCs offer the necessary SNF benchmarks. Commercial reactors offer an excellent and inexhaustible source of critical configurations against which criticality analyses can be validated for spent-fuel configurations. However, recent proposals for using CRCs as benchmarks have met with opposition from research and regulatory agencies. This opposition points to the fact that not all CRC state points are applicable as benchmarks for SNF. To ensure the suitability of CRCs as benchmarks, the proposed CRCs must be compared against established criteria.

Anno, J. and J. Krebs (1991). Safety margin estimation effects of six fission products in spent fuel transport and storage in water, Dorchester, UK, AEA Technol.

The effect of the various fission products existing in spent fuel, has been already studied, compared to the assumption of fresh fuel. The authors selected only six FP, responsible of about the half of the whole FP absorption : 149Sm 152Sm, 143Nd, 103Rh, 133Cs, 155Eu. In spite of validations, of cross-section libraries and of computer codes, this reduced choice decreases the remaining uncertainties in use of FP and still preserves a safety margin. The authors performed calculations on spent fuel, at initial enrichment greater (or equal) than 3.5 % 235U, by the APOLLO-MORET scheme, in practical underwater cases (storage, transportation). For fuels initially enriched at 3.5% 235U, irradiated until 10 GWd/tonU, the safety margin in 0.036 and 0.072 at 30 GWd/tonU.

Apostolov, T.G., M.A. Manolova, *et al*. (1999). WWER spent fuel criticality, depletion and shielding studies, Vienna, Austria, Int. Atomic Energy Agency.

The purpose of this paper is to present the results of WWER spent fuel inventory studies, in applying well-known depletion codes, as well as a criticality evaluation of the WWER spent fuel casks by Monte Carlo codes. Some results of WWER spent fuel shielding calculations are given too.

Ashe, K.L., R.G. Eble, *et al*. (1992). Shielding and criticality at the MRS facility, New York, NY, USA, American Soc. Civil Eng.

The Nuclear Waste Policy Act (NWPA) requires that the Department of Energy (DOE) provide a Monitored Retrieval Storage Facility (MRS) for interim storage and subsequent retrieval of spent nuclear fuel. This facility provides for the shipment of spent nuclear fuel for permanent disposal at a federal repository. The mission of this facility is accomplished in a manner that protects the health and safety of personnel and the public while maintaining the quality of the environment.

Aziz, M. and K. Andrzejewski (2000). "Criticality calculations for the spent fuel storage pools for Etrr 1 and Etrr 2 reactors." Nukleonika 45(2): 141-4.

A criticality analysis of two spent fuel storage pools for Etrr 1 and Etrr 2 research reactors was performed. The multiplication factor for the pools was calculated as a function of relevant lattice physics parameters. The Monte Carlo code MCNP-4A was used in the criticality calculations. The results were compared with those given by CITATION code and the results obtained formerly during the design phase of the pools with the MONK 6.3 code. Safety of the pools was confirmed.

Best, R.E., S.J. Maheras, *et al*. (2003). "Radiation doses to the public from the transport of spent-nuclear fuel." JNMM 31(3): 12-18.

Spent-nuclear fuel or high-level radioactive waste will be shipped in casks certified by the NRC in accordance with the requirements in Title 10, Part 71 of the Code of Federal Regulations. This means that the casks will meet a suite of performance requirements that have been selected to protect public health and safety. It includes requirements limiting radiation dose external to the cask for normal conditions of transport and following accidents. In addition, the cask must include features that prevent the occurrence of nuclear criticality under normal and accident conditions, and it must contain its radioactive material contents when subjected to a sequence of drop, puncture, fire, and immersion accident tests.

Bostian, R.W. (1980). Spent Fuel Storage. A Status Report, Springfield, VA, USA, Nat. Tech. Inf. Service.

Assesses the status of spent fuel storage in the United States with reference to pending legislation, national storage status and the experience of Duke Power Company.

Brice, D.J. and T.S. Palmer (1998). MCNP criticality calculations in support of the Trojan independent spent-fuel storage installation, USA, ANS.

Sierra Nuclear Corporation's Dry Transtor basket and storage overpack have been chosen to accommodate spent fuel at Trojan nuclear plant's independent spent-fuel storage installation (ISFSI). The purpose of this project is to create a computational model specific to this ISFSI to allow Trojan engineers to perform accurate criticality calculations of individual casks loaded with specific fuel assemblies. The model has also been used to compare MCNP results against those produced by a Sierra Nuclear KENO model of the same system.

Broadhead, B., B. Rearden, *et al*. (2004). "Sensitivity-and uncertainty-based criticality safety validation techniques." Nuclear science and engineering 146(3): 340-366.

The theoretical basis for the application of sensitivity and uncertainty (S/U) analysis methods to the validation of benchmark data sets for use in criticality safety applications is developed. Sensitivity analyses produce energy-dependent sensitivity coefficients that give the relative change in the system multiplication factor keff value as a function of relative changes in the cross-section data by isotope, reaction, and energy. Integral indices are then developed that utilize the sensitivity information to quantify similarities between pairs of systems, typically a benchmark experiment and design system. Uncertainty analyses provide an estimate of the uncertainties in the calculated values of the system keff due to crosssection uncertainties, as well as correlation in the keff uncertainties between systems. These uncertainty correlations provide an additional measure of system similarity. The use of the similarity measures from both S/U analyses in the formal determination of areas of applicability for benchmark experiments is developed. Furthermore, the use of these similarity measures as a trending parameter for the estimation of the computational bias and uncertainty is explored. The S/U analysis results, along with the calculated and measured keff values and estimates of uncertainties in the measurements, were used in this work to demonstrate application of the generalized linear-least-squares methodology (GLLSM) to data validation for criticality safety studies. An illustrative example is used to demonstrate the application of these S/U analysis procedures to actual criticality safety problems. Computational biases, uncertainties, and the upper subcritical limit for the example applications are determined with the new methods and compared to those obtained through traditional criticality safety analysis validation techniques. The GLLSM procedure is also applied to determine cutoff values for the similarity indices such that applicability of a benchmark experiment to a criticality safety design system can be assured. Additionally, the GLLSM procedure is used to determine how many applicable benchmark experiments exceeding a certain degree of similarity are necessary for an accurate assessment of the computational bias.

Broulik, J. (1992). Spent VVER fuel storage subcriticality investigations, Bury St Edmonds, UK, Mech. Eng. Publications.

The method used on the LR-O critical assembly to estimate the subcriticality of VVER-type spent fuel compact storage configurations is described. The results of the experiments and calculations are presented. Agreement between experimental and calculated values of the multiplication factor defining the subcriticality is shown to be good.

Brown, C.O. and L.E. Hansen (1978). Methods of criticality analysis for spent-fuel storage facilities, USA.

Due to delays in reprocessing or disposal of spent fuel, increased on-site spent-fuel storage capacity is being installed at many reactors. In such high-capacity storage arrays where fuel assemblies are no longer neutronically isolated by water, the accuracy of criticality calculations becomes of primary importance. The methods of analysis and typical design-base assumptions employed by the Exxon Nuclear Company to assess the criticality safety of high-density spent-fuel storage arrays are summarized and the results of some benchmark calculations that verify the adequacy of the analytical methods, are presented.

Carlson, R.W. and L.E. Fischer (1990). Safety implications of burnup effects in criticality safety margins for spent pressurized water reactor fuel transport and storage: Medium: X; Size: Pages: (28 p).

Criticality safety margins must be based upon the combination of the best available prediction of the margin and all uncertainties in the prediction. Inclusion of the effects of burnup in the evaluation of spent fuel shipping or storage casks must be based upon a thorough understanding of the prediction of the effects of burnup and the uncertainties in the measurements (or predictions) of burnup and predictions of the effects. This report presents a preliminary estimate of the effects of burnup and its uncertainties. This will serve as the first step in the effort to develop acceptance criteria that assure public safety. An assembly average burnup of 20,000 MWD/MTU represents an increase in the criticality safety margin of about 20% ({delta}k/k), and the current estimate of the uncertainty in this value is close to 4% ({delta}k/k). The uncertainties in the components of the effects of burnup were based upon relevant literature citations and -- where no other information was available -- upon estimates. Consequently, the margins and uncertainties in the margin presented here should be considered as initial estimates upon which more refined analyses should build to develop a defensible basis for predicting and reviewing the criticality safety margins which include the effects of burnup.

Chang, J.K. (1992). Applications of probabilistic risk analysis in nuclear criticality safety design.

Many documents have been prepared that try to define the scope of the criticality analysis and that suggest adding probabilistic risk analysis (PRA) to the deterministic safety analysis. The report of the US Department of Energy (DOE) AL 5481.1B suggested that an accident is credible if the occurrence probability is >1 x $10^{-6/}$yr. The draft DOE 5480 safety analysis report suggested that safety analyses should include the application of methods such as deterministic safety analysis, risk assessment, reliability engineering, common-cause failure analysis, human reliability analysis, and human factor safety analysis techniques. The US Nuclear Regulatory Commission (NRC) report NRC SG830.110 suggested that major safety analysis methods should include but not be limited to risk assessment, reliability engineering, and human factor safety analysis. All of these suggestions have recommended including PRA in the traditional criticality analysis.

Chong Chul, Y. (1979). "Criticality safety determination of spent fuel storage vault." Radiation Protection 4(1): 1-4.

The effective multiplication factor has been calculated for one PWR fresh fuel assembly immersed in a spent fuel storage vault, on the basis of the neutron transport theory. A numerical calculation has been carried out by means of Sn approximation. The method employed is that the energy domain is broken into 16 groups, the angular variable is divided into four discrete directions, i.e., S4, and the spatial variable which is divided into fine meshes at the interface between different materials is discretized into 27 mesh points. The calculated Keff value of 0.6145 seems to be very small in comparison with the value obtained previously for an infinite array of fuel assemblies.

Clemson, P.D. and P.R. Thorne (1989). The criticality implications of taking credit for fuel burn-up, Oak Ridge, TN, USA, Oak Ridge Nat. Lab.

Given the high safety standards demanded in the nuclear industry, a naturally cautious view is taken and pessimistic assumptions of parameter values are therefore made. This explains why assessments of spent fuel transport and storage assume unirradiated fissile compositions (the `fresh fuel' assumption). The challenge is to develop ways of taking credit for fuel burn-up with no reduction in safety. This raises various issues which are discussed briefly, together with a scoping study to investigate the scale of reactivity change with burn-up for an existing BNFL cask design.

DeHart, M. (1998). "An advanced deterministic method for spent-fuel criticality safety analysis." Transactions of the American Nuclear Society 78(CONF-980606--).

Over the past two decades, criticality safety analysts have come to rely to a large extent on Monte Carlo methods for criticality calculations. Monte Carlo has become popular because of its capability to model complex, nonorthogonal configurations or fissile materials, typical of real-world problems. In the last few years, however, interest in determinist transport methods has been revived, due to shortcomings in the stochastic nature of Monte Carlo approaches for certain types of analyses. Specifically, deterministic methods are superior to stochastic methods for calculations requiring accurate neutron density distributions or differential fluxes. Although Monte Carlo methods are well suited for eigenvalue calculations, they lack the localized detail necessary to assess uncertainties and sensitivities important in determining a range of applicability. Monte Carlo methods are also inefficient as a transport solution for multiple-pin depletion methods. Discrete ordinates methods have long been recognized as one of the most rigorous and accurate approximations used to solve the transport equation. However, until recently, geometric constrains in finite differencing schemes have made discrete ordinates methods impractical for nonorthogonal configurations such as reactor fuel assemblies. The development of an extended step characteristic (ESC) technique removes the grid structure limitation of traditional discrete ordinates methods. The NEWT computer code, a discrete ordinates code built on the ESC formalism, is being developed as part of the SCALE code system. This paper demonstrates the power, versatility, and applicability of NEWT as a state-of-the-art solution for current computational needs.

Denver, D.J., B.F. Momsen, *et al*. (1975). Criticality calculations on Maine Yankee spent-fuel racks containing Boral, USA.

Due to the delays in the availability of reprocessing many nuclear power plant operators have had to contend with a shortage of on-site storage space for spent-fuel assemblies. The authors describe the criticality calculations that were performed in support of the licensing effort for the increased capacity Maine Yankee spent-fuel storage racks which utilize Boral as a fixed absorber, thereby reducing the nominal assembly pitch from 20.5 to 12.0 in. and increasing the available storage space by a factor of 3.

Eggers, P.E. (1983). Storage and transportation of spent fuel and high level waste using dry storage casks, La Grange Park, IL, USA, ANS.

The storage of spent fuel and high level waste in high capacity spent fuel storage casks is described including onsite handling, transfer and storage operations. The influence of such spent fuel characteristics as enrichment, burnup and post-irradiation time on dry storage cask operating characteristics is also described. The fuel assembly decay heat, peak cladding temperature, criticality, and surface dose rates are presented as a function of the spent fuel characteristics. A transportation system is also described which allows the dry storage cask to be transported offsite by rail, barge or heavy haul truck by employing reusable impact limiters. The dry storage cask transportation system described has been designed to meet all 10CFR 71 licensing requirements. The status of dry storage cask licensing, manufacture and demonstration is also reviewed as well as the benefits and limitations of dry storage casks as compared with conventional spent fuel storage methods.

El-Kady, A., N. Ashoub, *et al*. (1995). A study for providing additional storage spaces to ET-RR-1 spent fuel, Ljubljana, Slovenia, Nucl. Soc. Slovenia.

The ET-RR-1 reactor spent fuel storage pool is a trapezoidal aluminum tank concrete shield and of capacity 10 m3. It can hold up to 60 fuel assemblies. The long operation history of the ET-RR-1 reactor resulted in a partially filled spent fuel storage with the remaining spaces not enough to host a complete load from the reactor. This work have been initiated to evaluate possible alternative solutions for providing additional storage spaces to host the available EK-10 fuel elements after irradiation and any foreseen fuel in case of reactor upgrading. Several alternate solutions have been reviewed and decision on the most suitable one is under study. These studies include a criticality calculation of some suggested alternatives like reracking the present spent fuel storage pool and double tiering by the addition of a second level storage rack above the existing rack. The two levels may have a different densification factor. A criticality calculation of a possible double tiering accident was also studied.

Evans, D. and B. Palmer (1994). Repository Criticality Safety for the DOE Spent Nuclear Fuel Program. High Level Radioactive Waste Management 1994, ASCE.

Evans, M.C., T.H. Jones, *et al*. (1983). Arrangements for the detection of criticality incidents at BNFL Sellafield, USA.

Sellafield has facilities for the interim storage of spent nuclear fuel, the separation of uranium and plutonium, the treatment and storage of wastes arising, and the fabrication of fast reactor fuel. Accidental criticality is an acknowledged hazard associated with these operations; the authors describe the criticality detection and alarm systems.

Glumac, B., M. Ravnik, *et al*. (1997). "Criticality safety assessment of a TRIGA reactor spent-fuel pool under accident conditions." Nuclear Technology 117(2): 248-54.

Additional criticality safety analysis of a pool-type storage for TRIGA spent fuel at the Jozef Stefan Institute in Ljubljana, Slovenia, is presented. Previous results have shown that subcriticality is not guaranteed for some postulated accidents (earthquake with subsequent fuel rack disintegration resulting in contact fuel pitch) under the assumption that the fuel rack is loaded with fresh 12 wt% standard fuel. To mitigate this deficiency, a study was done on replacing a certain number of fuel elements in the rack with cadmium-loaded absorbed rods. The Monte Carlo computer code MCNP4A with an ENDF/B-V library and detailed three-dimensional geometrical model of the spent-fuel rack was used for this purpose. First, a minimum critical number of fuel elements was determined for contact pitch, and two possible geometries of rack disintegration were considered. Next, it was shown that subcriticality can be ensured when pitch is decreased from a rack design pitch of 8 cm to contact, if a certain number of fuel elements (8 to 20 out of 70) are replaced by absorber rods, which are uniformly mixed into the lattice. To account for the possibility that random mixing of fuel elements and absorber rods can occur during rack disintegration and result in a supercritical configuration, a probabilistic study was made to sample the probability density functions for random absorber rod lattice loadings. Results of the calculations show that reasonably low probabilities for supercriticality can be achieved (down to 10-6 per severe earthquake, which would result in rack disintegration and subsequent maximum possible pitch decrease) even in the case where fresh 12 wt% standard TRIGA fuel would be stored in the spent-fuel pool.

Grahn, P.H. and M. Wikstrom (1999). Experiences from the operation of the Swedish central interim storage facility for spent fuel, CLAB, Vienna, Austria, Int. Atomic Energy Agency.

Today about 50% of the electric power in Sweden is generated by means of nuclear power. The Swedish nuclear programme comprises 12 plants. According to political decisions no more nuclear power plants will be built and the existing plants shall not be operated beyond the year 2010. The programme will give rise to not more than 7800 metric tonnes (U) of spent fuel, which will be directly disposed of in the crystalline bedrock without reprocessing. A keystone in the spent fuel management strategy is the central interim storage facility, CLAB. After an intensive preproject work the licensing of CLAB according to the Building Act, Environment Protection Act and Atomic Energy Act took place in 1978-1979. After a total licensing time of about 20 months the last permit was obtained in August 1979. By June 1998, CLAB had received and unloaded some 1000 fuel transport casks corresponding to about 2752 tonnes U and 81 casks containing highly active core components. The performance of the plant has been very satisfactory and with increasing experiences it has been possible to reduce the operating and maintenance costs. The extensive efforts during the design phase have resulted in a collective dose of 25-30% of the dose calculated in the final safety report. Due to a low activity release from the fuel and an optimised management of the used water filtering agents the number of waste packages emanating from CLAB has been less than 10% of what was originally expected. The activity release to air and water from the facility during the five first years of operation has been around 0.01% of the permissible release. In order to postpone the building of additional storage pools, new storage canisters have been developed which has increased the storage capacity from 3000 to 5000 tonnes U.

Greene, N.M. and R.M. Westfall (1997). Cross sections for criticality safety applications, USA, ANS.

Whether for a reactor, a spent-fuel shipping cask, a storage facility for nuclear material, a nuclear weapon, or any radioactive system, a complete analysis will include calculations to study the transport of nuclear particles in the system. These calculations require parameters that describe how the nuclear particles interact with the materials in the system and what happens as a result of an interaction. These parameters are called cross sections. This paper will limit itself to discussing neutron interactions since this is of primary importance for a great number of real situations.

Harms, G.A., F.J. Davis, *et al*. (1995). The spent fuel safety experiment.

The Department of Energy is conducting an ongoing investigation of the consequences of taking fuel burnup into account in the design of spent fuel transportation packages. A series of experiments, collectively called the Spent Fuel Safety Experiment (SFSX), has been devised to provide integral benchmarks for testing computer-generated predictions of spent fuel behavior. A set of experiments is planned in which sections of unirradiated fuel rods are interchanged with similar sections of spent PWR fuel rods in a critical assembly. By determining the critical size of the arrays, one can obtain benchmark data for comparison with criticality safety calculations. The SFSX provides a direct measurement of the reactivity effects of spent PWR fuel using a well-characterized, spent fuel sample. The SFSX also provides an experimental measurement of the end-effect, i.e., the reactivity effect of the variation of the burnup profile at the ends of PWR fuel rods. The design of the SFSX is optimized to yield accurate benchmark measurements of the effects of interest, well above experimental uncertainties.

Harris, D.R. (1987). Criticality of spent reactor fuel, USA.
Discusses the criticality safety of spent reactor fuel in a storage pool in the case of a water-spill accident. The LEOPARD code is utilised.

Hicks, T. and A. Prescott (2000). A Study of Criticality in a Spent Fuel Repository Based on Current Canister Designs. Prepared for the Swedish Nuclear Power Inspectorate. SKI Report 00:13. January.

Hopper, C.M. (1994). DOE spent nuclear fuel -- Nuclear criticality safety challenges and safeguards initiatives.
The field of nuclear criticality safety is confronted with growing technical challenges and the need for forward-thinking initiatives to address and resolve issues surrounding economic, safe and secure packaging, transport, interim storage, and long-term disposal of spent nuclear fuel. These challenges are reflected in multiparameter problems involving optimization of packaging designs for maximizing the density of material per package while ensuring subcriticality and safety under variable normal and hypothetical transport and storage conditions and for minimizing costs. Historic and recently revealed uncertainties in basic data used for performing nuclear subcriticality evaluations and safety analyses highlight the need to be vigilant in assessing the validity and range of applicability of calculational evaluations that represent extrapolations from ``benchmark`` data. Examples of these uncertainties are provided. Additionally, uncertainties resulting from the safeguarding of various forms of fissionable materials in transit and storage are discussed.

Hutson, D.L., T.A. Keys, *et al*. (1988). Criticality analysis performed at Tennessee Valley Authority (TVA), La Grange Park, IL, USA, ANS.
In order to increase the current enrichment limit of the high density spent fuel storage racks at Sequoyah Nuclear Plant, TVA has performed a critically analysis using the SCALE-3 system of computer codes developed at the Oak Ridge National Laboratory. SCALE-3 (KENO-Va) has been used to show that under normal storage conditions, there is 95 percent probability, at 95 percent confidence level, that the effective multiplication factor (Keff) of the stored fuel array will be less than or equal to 0.95, including all uncertainties. In order to increase the enrichment limit to 4.5 wt. percent U235, this analysis has taken credit for burnup using equivalent enrichment data obtained from CASMO3 calculations. This analysis is the first coupling of CASMO3 and KENO.Va for a criticality analysis of a high density spent fuel rack.

Itoh, C., O. Katoh, *et al*. (2003). "Long term containment performance test for spent fuel transport/storage casks." Transactions of the Atomic Energy Society of Japan 2(2): 158-62.
The use of transport/storage cask for spent fuel storage is considered to be rational and economical. Since the storage duration may continue for 40 years or so, the function of sealing radioactive materials in the casks must be reliable for long-term. Long-term containment test of full-scale spent fuel transport/storage cask models have been in progress since 1990 in CRIEPI, Japan. It has been 11 years since it started. The results so far demonstrate and confirm very reliable containment performance of the cask lid structure with metal gaskets. Using the test data it is predicted by Larson-Miller Parameter (LMP) method that the containment system will keep its integrity at least for 40 years.

Jor-Shan, C., C. Lee, *et al*. (2007). Application of neutron-absorbing structural-amorphous metal (SAM) coatings for spent nuclear fuel (SNF) container to enhance criticality safety controls, Warrendale, PA, USA, Materials Research Society.

Spent nuclear fuel contains fissionable materials (235U, 239Pu, 241Pu, etc.). To prevent nuclear criticality in spent fuel storage, transportation, and during disposal, neutron-absorbing materials (or neutron poisons, such as borated stainless steel, Boral, Metamic, Ni-Gd, and others) would have to be applied. The success in demonstrating that the high-performance corrosion-resistant material (HPCRM) can be thermally applied as coating onto base metal to provide for corrosion resistance for many naval applications raises the interest in applying the HPCRM to USDOE/OCRWM spent fuel management program. The fact that the HPCRM relies on the high content of boron to make the material amorphous -an essential property for corrosion resistance - and that the boron has to be homogenously distributed in the HPCRM qualify the material to be a neutron poison.

Kaisavelu, A. (2009). Criticality analyses of the used and spent fuel storage facility of the 400 MWth PBMR plant. Nuclear Engineering. Potchefstroom, North-West University. M.Sc. Nuclear Engineering (Thesis).

The development of the Pebble Bed Modular Reactor entails the design of numerous systems for various purposes. One such system of significant importance is the Sphere Storage System (a subsystem of the Fuel Handling and Storage system) where fuel spheres that are unloaded from the core will be stored until approximately eighty years after the power plant has been decommissioned. Over and above the normal conventional safety analyses that one expects to be performed for any new system being designed, in the case of the Sphere Storage System a detailed Criticality Safety Analysis must be performed. The universally accepted Effective Neutron Multiplication Factor, keff, was used to indicate the margins of subcriticality for all the conditions modelled. Since this Used and Spent Fuel storage facility is a Critical Safety-relevant system that will store nuclear fuel for a long time, it is required by regulation that the Criticality Safety Analyses be performed to verify whether this system will always remain “critical safe” (keff > 0.95) under all plausible conditions. This study covers a variety of tasks, from the modelling of a single fuel sphere to modelling of the entire Sphere Storage System for the normal and various off-normal conditions, and for the determination of keff values for the system under these conditions. Additional models were also created to investigate the phenomena of clustering of low burnup fuel spheres and the effects of graphite spheres being mixed with the fuel spheres in the storage containers. The entire study was done using the SCALE 5.1 computer code package. SCALE 5.1 is licensed by the United States Nuclear Regulatory Commission (US NRC) and is a package that is widely used in the US and around the world to perform criticality safety analyses as well as other nuclear-related calculations. For this study the control module CSAS6 was specifically used to develop the appropriate models because of its suitability for the modelling of pebble fuel and its advanced geometric modelling capabilities. It also automatically invokes the specific functional modules using the sequence CSAS26 in order to obtain the appropriate information as required by another functional module KENO-VI, which calculates keff for the specified input models. 3 The results from the models for the various scenarios representing normal and off-normal conditions show that the design of the proposed current design of the Sphere Storage System remains critical safe (keff < 0.95) for all the plausible scenarios considered. Any change to the current design requires new Criticality Safety Analyses to be performed.

However, the methodology developed in this study can be used as a guide for future studies.

Ketzlach, N. (1983). Nuclear criticality safety in LWR fuel assembly storage, USA.
The author considers the dry storage of fresh LWR fuel assemblies in spent fuel storage pools. Their nuclear criticality safety has not yet been demonstrated; Keff calculations using nuclear reactor design codes are shown to be inadequate.

Las Vegas, N. (1995). "Review of Criticality Safety and Shielding Analysis Issues for Transportation Packages C. V. Parks and B. L. Broadhead Computational Physics and Engineering Division Oak Ridge National Laboratory."
The staff of the Nuclear Engineering Applications Section (NEAS) at Oak Ridge National Laboratory (ORNL) have been involved for over 25 years with the development and application of computational tools for use in analyzing the criticality safety and shielding features of transportation packages carrying radioactive material (RAM). The majority of the computational tools developed by ORNL/NEAS have been included within the SCmE modular code system (SCALE 1995). This code system has been used throughout the world for the evaluation of nuclear facility and package designs. With this development and application experience as a basis, this paper will present a perspective on important issues related to nuclear safety analyses for a package design.

Lipner, M.H. and J.M. Ravets (1980). Criticality analyses for safety evaluation of dry storage of spent-fuel assemblies, USA.
Presents the results of the nuclear analyses utilized in the criticality safety evaluation for the spent-fuel handling and packaging program demonstration at the Nevada Test Site. This demonstration is being conducted at the Engine Maintenance Assembly and Disassembly (EMAD) Facility in Area 25, which is operated by the Westinghouse Advanced Energy Systems Division for the Nevada Operations Office of the Department of Energy (DOE). The configurations that were considered are associated with encapsulating the spent-fuel assemblies and subsequently storing them in surface and near-surface dry storage configurations.

Logar, M., B. Glumac, *et al*. (1997). Some aspects of accidental criticality safety of TRIGA reactor spent fuel pool, Berne, Switzerland, Eur. Nucl. Soc.
Criticality safety analysis of a pool type storage for TRIGA spent fuel at Josef Stefan Institute in Ljubljana, Slovenia, is presented. Previous results have shown that subcriticality is not guaranteed for some postulated accidents. To mitigate this deficiency, a study was made about replacing a certain number of fuel elements in the rack with absorber rods. For this purpose, the Monte Carlo computer code MCNP4A with ENDF-B/V library and the detailed three dimensional fuel rack model was used. At first the analysis was done about the number of uniformly mixed absorber rods in the lattice needed to sustain the subcriticality of the storage when pitch is decreased from rack design pitch of 8 cm to contact, assuming that the absorbers retain their proper positions. Because of supercriticality possibility due to random mixing of the absorber rods among the fuel elements during lattice compaction, a probabilistic study was made, sampling the probability density functions for such random lattice loadings. The results show reasonably low probabilities for supercriticality even for fresh standard TRIGA fuel containing 12 wt% uranium stored in the pool.

Marshall, W. and J.C. Wagner (2012). Impact of Fuel Failure on Criticality Safety of Used Nuclear Fuel, Oak Ridge National Laboratory (ORNL).

Commercial used nuclear fuel (UNF) in the United States is expected to remain in storage for considerably longer periods than originally intended (e.g., <40 years). Extended storage (ES) time and irradiation of nuclear fuel to high-burnup values (>45 GWd/t) may increase the potential for fuel failure during normal and accident conditions involving storage and transportation. Fuel failure, depending on the severity, can result in changes to the geometric configuration of the fuel, which has safety and regulatory implications. The likelihood and extent of fuel reconfiguration and its impact on the safety of the UNF is not well understood. The objective of this work is to assess and quantify the impact of fuel reconfiguration due to fuel failure on criticality safety of UNF in storage and transportation casks. This effort is primarily motivated by concerns related to the potential for fuel degradation during ES periods and transportation following ES. The criticality analyses consider representative UNF designs and cask systems and a range of fuel enrichments, burnups, and cooling times. The various failed-fuel configurations considered are designed to bound the anticipated effects of individual rod and general cladding failure, fuel rod deformation, loss of neutron absorber materials, degradation of canister internals, and gross assembly failure. The results quantify the potential impact on criticality safety associated with fuel reconfiguration and may be used to guide future research, design, and regulatory activities. Although it can be concluded that the criticality safety impacts of fuel reconfiguration during transportation subsequent to ES are manageable, the results indicate that certain configurations can result in a large increase in the effective neutron multiplication factor, keff. Future work to inform decision making relative to which configurations are credible, and therefore need to be considered in a safety evaluation, is recommended.

Massoud, E., O.H. Sallam, *et al*. (2001). "Criticality safety of the ET-RR-1 new spent fuel storage pool." Annals of Nuclear Energy 28(4): 375-83.

A new ET-RR-I spent fuel storage pool is now under construction on the reactor site at Inshass. In addition, the pool is designed to accommodate spent fuel of MTR type as well. Criticality safety of this pool for the different fuel types has been evaluated as a function of U235 loading. The effect of fuel element separation (rows and columns) on the eigenvalue has been studied. As a conservative assumption, the pool is assumed to be filled with fresh fuel. The eigenvalue considering a realistic degree of fuel burn-up was determined in order to determine the safety margin. The calculations have been carried out using the code packages of the National Center for Nuclear Safety and Radiation Control.

Maucec, M. and B. Glumac (2005). "Criticality safety and sensitivity analyses of PWR spent nuclear fuel repository facilities." Nuclear Technology 149(1): 1-13.

Monte Carlo criticality safety and sensitivity calculations of pressurized water reactor (PWR) spent nuclear fuel repository facilities for the Slovenian nuclear power plant Krsko are presented. The MCNP4C code was deployed to model and assess the neutron multiplication parameters of pool-based storage and dry transport containers under various loading patterns and moderating conditions. To comply with standard safety requirements, fresh 4.25% enriched nuclear fuel was assumed. The impact of potential optimum moderation due to water steam or foam formation as well as of different interpretations, of neutron multiplication through varying the system boundary conditions was elaborated. The simulations indicate that in the case of compact (all rack locations filled with fresh fuel) single or "double tiering " loading, the supercriticality can occur under the conditions of enhanced neutron moderation, due to accidentally reduced density of cooling water. Under standard operational conditions the effective multiplication factor (keff) of pool-based storage facility remains below the specified

safety limit of 0.95. The nuclear safety requirements are fulfilled even when the fuel elements are arranged at a minimal distance, which can be initiated, for example, by an earthquake. The dry container in its recommended loading scheme with 26 fuel elements represents a safe alternative for the repository of fresh fuel. Even in the case of complete water flooding, the keff remains below the specified safety level of 0.98. The criticality safety limit may however be exceeded with larger amounts of loaded fuel assemblies (i.e., 32). Additional Monte Carlo criticality safety analyses are scheduled to consider the "burnup credit" of PWR spent nuclear fuel, based on the ongoing calculation of typical burnup activities.

Maucec, M., A. Persic, *et al*. (1994). Criticality safety study of NPP Krsko spent fuel pool with Monte Carlo computer code MCNP4A, Ljubljana, Slovenia, Nucl. Soc. Slovenia.

Contribution presents implementation of IBM PC and Digital VAX versions of computer code for Monte Carlo transport calculations MCNP4A, verification of the code by calculation of four benchmarks for fast and thermal critical assemblies and, finally, partial criticality safety study of NPP Krsko spent fuel pool. Geometric model was made for three various loading patterns. We used cross section data from ENDF/B-IV and VI libraries. The last one (reactor materials only) was only recently acquired from LANL, and is considered to be the most updated version. The results gained show satisfactory agreement with experimental data or results from calculations performed at LANL.

Min-Fong, S. and A.H. Wells (1984). Fuel loading effects on spent fuel cask criticality, USA.

Spent fuel shipping and storage casks have traditionally been analyzed for criticality safety as if the entire cask were filled with the highest fresh fuel enrichment (no burnup credit) that maintains the required criticality margin. This formed a simple bounding case, and usually established an enrichment limit higher than most envisioned fuel enrichments, so no penalty in terms of operational flexibility resulted. The cost of cask fuel baskets containing neutron absorbers such as boron or Ag-In-Cd was not a significant problem for shipping casks because the costs could be amortized over the many expected shipments. The lack of away-from-reactor or reprocessing facilities and the resulting introduction of storage casks has significantly altered the economics of cask basket design for nuclear criticality safety. This provides incentive to reduce criticality control costs by analysis of the actual fuel load instead of a hypothetical bounding case and by investigating the effects of varying fuel enrichments.

Morrell, G.A. and J.T. Spencer (1991). Criticality implications of solid waste handling, encapsulation and storage on the Sellafield Site, Dorchester, UK, AEA Technol.

During the handling and reprocessing by BNFL of spent nuclear fuel several solid waste streams are generated. Most of these waste streams will contain some fissile material. The criticality implications of their handling and storage must therefore be considered. It is BNFL's intention to store these wastes in a solidified state pending final decisions being taken on their long term storage and disposal. The authors illustrate the wide range of solid waste streams handled now, or expected in the near future, at the Sellafield Reprocessing Site, and highlights some of the associated criticality implications. The waste streams considered arise from both natural uranium (Magnox) and Oxide reprocessing and include such things as Magnox swarf and associated sludges, oxide fuel cladding (hulls), centrifuge cake slurry, low and medium active concentrates, all encapsulated in grout, and highly active concentrate vitrified in glass.

Napolitano, D.G., F.L. Carpentio, *et al*. (1988). Validation of the YAEC criticality safety methodology, USA.

The Yankee Atomic Electric Company's (YAEC's) criticality safety methodology has evolved over the years to analyze high-density spent fuel rack designs, new fuel vault optimum moderation, burnup credit, pin consolidation, storage rack sensitivities, and large spent fuel rack arrays. The present methodology has three calculational paths: NITAWL-S/KENO V.a Monte Carlo, CASMO-3 integral transport, and CASMO-3/CHART-2/PDQ-7 diffusion theory analysis. The authors have validated these calculational paths by comparison to 21 Babcock and Wilcox (BW) fuel storage criticals. These criticals covered a range of fuel storage conditions in which criticality was maintained by a combination of water height, soluble boron, fixed poison, and array spacing. Statistical analysis of the 21 calculated keff's gives a method bias and 95/95 method uncertainty for each path.

Neuber, J.C. (1999). Criticality analysis of PWR spent fuel storage facilities inside nuclear power plants, Vienna, Austria, Int. Atomic Energy Agency.

This paper describes some of the main features of the actinide plus fission product burnup credit methodology used by Siemens for the criticality safety design analysis of wet PWR storage pools with soluble boron in the pool water. The application of burnup credit requires knowledge of the isotopic inventory of the irradiated fuel for which burnup credit is taken. This knowledge is gained by using depletion codes. The results of the depletion analysis are a necessary input to the criticality analysis. Siemens performs depletion calculations for PWR fuel burnup credit applications with the aid of the Siemens standard design procedure SAV90. The quality of this procedure relies on statistics on the differences between calculation and measurement extracted from in-core measurement data and chemical assay data. Siemens performs criticality safety calculations with the aid of the criticality calculation modules of the SCALE code package. These modules are verified many times with the aid of various kinds of critical experiments and configurations: Application of these modules to spent LWR fuel assembly storage pools was verified by analyzing critical experiments simulating such storage pools. Actinide plus fission product burnup credit applications of these modules were verified by analyzing PWR reactor critical configurations. The result of performing a burnup credit analysis is the determination of a burnup credit loading curve for the spent fuel storage racks designed for burnup credit. This curve specifies the loading criterion by indicating the minimum burnup necessary for the fuel assembly with a specific initial enrichment to be placed in the storage racks designed for burnup credit. The loading of the spent fuel storage racks designed for burnup credit requires the implementation of controls to ensure that the loading curve is met. The controls include the determination of fuel assembly burnup based on reactor records.

Nomura, Y., M.C. Brady, *et al*. (1998). OECD/NEA working party on nuclear criticality safety: Challenge of new realities, Paris, France, SFEN-Soc. Francaise d'Energie Nucl.

New issues in criticality safety continue to emerge as spent fuel storage facilities reach the saturation point, fuel enrichments and burn-ups increase and new types of plutonium-carrying fuels are being developed. The new challenges related to the manipulation, transportation and storage of fuel demand further work to improve models predicting behaviour through new experiments, especially where there is a lack of data in the present databases. This article summarises the activities of the OECD/NEA working groups that co-ordinate and carry out work in the domain of criticality safety. Particular attention is devoted to establishing sound databases required in this area and to addressing issues of high relevance such as burn-up credit. This is aimed toward improving safety and identifying economic solutions to issues concerning the back end of the fuel cycle.

Oversby, V.M. (1998). Criticality in a repository for spent fuel: lessons from Oklo, Warrendale, PA, USA, Mater. Res. Soc.

The conditions that are needed to achieve criticality in a high level waste repository for spent nuclear reactor fuel are reviewed. The effect of initial enrichment of the fuel, burnup, and of mixed oxide fuels on the conditions for criticality are discussed. The situations that produced criticality at Oklo, Gabon, 2000 million years ago are summarized. A model based on the Oklo conditions is presented for estimating the amount of fissile material that must be assembled to create a critical mass in typical granitic rocks. Mechanisms for movement of uranium and plutonium to achieve a critical configuration are discussed and compared to the conditions that are likely to occur in a repository in granite. The sequences of events needed to produce a critical assemblage are shown to be in conflict with the conditions expected in the repository and, in some cases, to require internally inconsistent assumptions to produce the postulated sequence of events. No credible scenario for achieving criticality in a high level waste repository has been found.

Ponomarenko, G.L. (1999). "Analysis of criticality in shipment and storage of fuel at a nuclear power plant with a VVER reactor." Atomic Energy 87(1): 466-71.

The substantiation of nuclear safety during shipment and storage of fresh and spent fuel at nuclear power plants with VVER reactors is examined in the light of the more stringent nuclear safety rules. Possible technical measures for satisfying the safety criterion are examined, for example, the concept of subcritical fresh fuel. An example of the estimation of the probability of the formation of a critical mass as result of fuel assemblies falling randomly out of a container is presented. Certain characteristic features of the calculation of the neutron-physical characteristics of fuel in a cooling pond are presented, for example, the non-conservative nature of a separate analysis in the infinite approximation.

Radulescu, G., D.E. Mueller, *et al*. (2009). "Sensitivity and uncertainty analysis of commercial reactor criticals for burnup credit." Nuclear Technology 167(2): 268-87.

This paper provides insights into the neutronic similarities between a representative high-capacity rail-transport cask containing typical pressurized water reactor (PWR) spent nuclear fuel assemblies and critical reactor state-points, referred to as commercial reactor critical (CRC) state-points. Forty CRC state-points from five PWRs were analyzed, and the characteristics of CRC state-points that may be applicable for validation of burnup-credit criticality safety calculations for spent fuel transport/storage/disposal systems were identified. The study employed cross-section sensitivity and uncertainty analysis methods developed at Oak Ridge National Laboratory and the TSUNAMI set of tools in the SCALE code system as a means to investigate neutronic similarity on an integral and nuclide-reaction-specific level. The results indicate that except for the fresh-fuel-core configuration, all analyzed CRC state-points are either highly similar, similar, or marginally similar to the representative high-capacity cask containing spent nuclear fuel assemblies with burnups ranging from 10 to 60 GWd/t U in terms of their shared uncertainty in keff due to cross-section uncertainties. On a nuclide-reaction-specific level, the CRC state-points provide significant coverage, in terms of neutronic similarity, for most of the actinides and fission products relevant to burnup credit. Hence, in principle, the evaluated CRC state-points could serve as part of a set of benchmark experiments for determining a bias and bias uncertainty to be applied to the calculated keff of a spent fuel transport/storage/disposal system to correct for approximations in computational methods and errors and uncertainties in nuclear data. Note, however, that an evaluation to quantify the uncertainties associated with various CRC modeling parameters (e.g., fuel isotopic compositions, physical characteristics of reactor core components, and reactor operating history information), which has relevance to the use of these critical configurations for bias determination, was not performed as part of this study.

Rathbun, R. and F. Trumble (1995). Criticality safety assessment of foreign research reactor fuel dry storage, USA.

In support of the policy to return spent foreign reactor fuel to the United States for storage, the Westinghouse Savannah River Company Criticality Technology Group addressed criticality safety concerns for the dry vault storage of this fuel. The bulk of this study focused on the storage of multiple assemblies in concrete vaults. Calculations were performed for the canned storage concept, where a 40.6-cm-diam canister is axially loaded with five cans (38.1-cm diameter and 91.4-cm height each) containing four fuel assemblies. A cross-sectional view of this canister arrangement is shown. Parameters varied in this study were water control, vault spacing, and assembly geometry degradation.

Rechard, R.P., M.S. Tierney, *et al.* (1997). "Bounding Estimates for Critical Events When Directly Disposing Highly Enriched Spent Nuclear Fuel in Unsaturated Tuff." Risk Analysis 17(1): 19-35.

This paper examines the possibility of criticality in a nuclear waste repository. The estimated probabilities are rough bounds and do not entirely dismiss the possibility of a critical condition; however, they do point to the difficulty of creating conditions under which a critical mass could be assembled (i.e., corrosion of containers, separation of neutron absorbers from the fissile material, and collapse or precipitation of the fissile material). In addition, should a criticality occur in or near a container, the bounding consequence calculations showed that fissions from one critical event are quite small ($<\tilde{}10^{20}$ fissions, if similar to aqueous and metal accidents and experiments). Furthermore, a reasonable upper bound of total critical events of 10^{28} fissions corresponds to only 0.1% of the number of fissions represented by the spent nuclear fuel inventory in a repository containing 70,000 metric tons of heavy metal (the expected size for the proposed repository at Yucca Mountain, Nevada).

Sandoval, R.P. (1992). Preliminary criticality safety analysis for the preclosure period of the potential Yucca Mountain repository, USA.

Nuclear criticality safety is a basic requirement and consideration in the design and operation of nuclear facilities that contain fissile materials. The potential Yucca Mountain repository will dispose of spent nuclear fuel, vitrified high-level waste (HLW), and a small quantity of other radioactive waste. Spent nuclear fuel, which comes from commercial nuclear power plants, contains significant quantities of fissile materials; the potential for criticality must be considered. The purpose of the preliminary study is to assess the criticality safety of various spent fuel configurations and container designs under consideration for the potential Yucca Mountain repository for normal operating and accident conditions. This study addresses criticality safety for the preclosure period only.

Seifert, E. (1997). Nuclear criticality safety considerations for CASTOR casks with spent fuel of the Rossendorf nuclear devices, Netherlands, Elsevier.

The spent fuel of the shutdown Rossendorf nuclear devices is to be loaded into storage and transport casks of the type CASTOR-MTR-2. According to the variety of different nuclear devices at the Rossendorf site, the Rossendorf fuel is characterized by a great variety with regard to geometry, material, enrichment, and burn-up. According to the special loading conception, the fuel is embedded in aluminium bodies that fill the CASTOR. The void fraction within the CASTOR is very small resulting in a small water fraction if water flooding is assumed. The criticality safety is proved by MCNP and OMEGA calculations. These are independent codes that use a completely different data base. The results of both codes agree very well demonstrating the reliability of the calculations. Apart from the proof of criticality safety, some interesting features were found mainly as a result of the very small water fraction.

Shaukat, S.K. and V.K. Luk (2002). Seismic behavior of spent fuel dry cask storage systems, New York, NY, USA, ASME.

The US Nuclear Regulatory Commission (NRC) is conducting a research program to investigate technical issues concerning the dry cask storage systems of spent nuclear fuel by conducting confirmatory research for establishing criteria and review guidelines for the seismic behavior of these systems. The program focuses on developing 3-D finite element analysis models that address the dynamic coupling of a module/cask, a flexible concrete pad, and an underlying soil/rock foundation, in particular, the soil-structure-interaction. Parametric analyses of the coupled models are performed to include variations in module/cask geometry, site seismicity, underlying soil properties, and cask/pad interface friction. The analyses performed include: 1) a rectangular dry cask module typical of Transnuclear West design at a site in Western USA where high seismicity is expected; 2) a cylindrical dry cask typical of Holtec design at a site in Eastern USA where low seismicity is expected; and 3) a cylindrical dry cask typical of Holtec design at a site in Western USA with medium high seismicity. The paper includes assumptions made in seismic analyses performed, results, and conclusions.

Sheu, R.J., Y.F. Chen, *et al.* (2011). "Dose evaluation for an independent spent-fuel storage installation using MAVRIC." Nuclear Technology 175(1): 335-42.

This study reevaluates the dose rates at the site boundary of an independent spent-fuel storage installation (ISFSI) using the MAVRIC computational sequence in the SCALE6 code package. Based on advanced variance-reduction techniques and powerful geometry modeling capabilities, MAVRIC can tackle this large ISFSI shielding problem by directly simulating the radiation transport in a full-scale model. This study started with a benchmark calculation of a single storage cask and then investigated the impact of a fully loaded ISFSI on the dose rates at the site boundary. Because of the short distance to the nearest site boundary, additional shielding to the cask itself or the site is necessary to meet the stringent design dose limit. Compared to the two-step cask-by-cask approach adopted in the original safety analysis report, this method of analyzing the site boundary doses is straightforward and efficient enough to allow us to evaluate the effect of the cask design modification and to test various options for further improvement.

Szu-Li, C., T. Shi-Ping, *et al*. (1987). "Criticality safety study of consolidated spent fuel." Nuclear Science Journal 24(5): 286-92.

The verified AMPX-KENO code package has been used in the criticality safety study of the storage of consolidated fuel rods in the Boral poisoned rack cells of the spent fuel storage pools at Chinshan and Kuosheng nuclear power plants. In the hypothetical case that the consolidated rods were put in the storage cells in their optimally moderated condition, the rack criticality analyses indicate the multiplication factor will not meet the requirement of being not greater than 0.95 under conditions of optimum moderation.

Thomas, D.A. (1999). Preclosure Criticality Analysis Process Report. Other Information: PBD: 29 Sep 1999; PBD: 29 Sep 1999: Medium: ED; Size: vp.

The design approach for criticality of the disposal container and waste package will be dictated by existing regulatory requirements. This conclusion is based on the fact that preclosure operations and facilities have significant similarities to existing facilities and operations currently regulated by the NRC. The major difference would be the use of a risk-informed approach with burnup credit. This approach could reduce licensing delays and costs of the repository. The probability of success for this proposed seamless licensing strategy is increased, since there is precedence of regulation (10 CFR Part 63 and NUREG 1520) and commercial precedence for allowing burnup credit at sites similar to Yucca Mountain during preclosure. While NUREG 1520 is not directly applicable to a facility for handling spent nuclear fuel, the risk-informed approach to criticality analysis in NUREG 1520 is considered indicative of how the NRC will approach risk-informed criticality analysis at spent fuel facilities in the future. The types of design basis events which must be considered during the criticality safety analysis portion of the Integrated Safety Analysis (ISA) are those events which result in unanticipated moderation, loss of neutron absorber, geometric changes in the critical system, or administrative errors in waste form placement (loading) of the disposal container. The specific events to be considered must be based on the review of the system's design, as discussed in Section 3.2. A transition of licensing approach (e.g., deterministic versus risk-informed, performance-based) is not obvious and will require analysis. For commercial spent nuclear fuel, the probability of interspersed moderation may be low enough to allow nearly the same Critical Limit for both preclosure and postclosure, though an administrative margin will be applied to preclosure and possibly not to postclosure. Similarly the Design Basis Events for the waste package may be incredible and therefore not re quire an administrative margin, or at least one that is less than the one used currently (0.05) for all waste forms (e.g., CRWMS M and O 1999c, criteria 1.2.1.5, p. 10.) In this case, the margin-to-criticality for preclosure and postclosure in the subsurface facility would be closer to that used for postclosure (if any). This would facilitate a seamless transition, including the use of burnup credit.

Tien-Ko, W., C. Szu-Li, *et al*. (1987). Criticality studies on a high density spent fuel storage pool, Paris, France, Soc. Francaise d'Energie Nucl.

The final policy pertaining to spent fuel long-term storage or reprocessing has not yet been resolved in Taiwan (and many other countries as well). This situation mandates the need to enlarge the on-site fuel storage capacities of Taiwan's first two nuclear power plants-Chinshan and Kuosheng, in order to accommodate more spent fuel assemblies. To provide larger spent fuel storage capacity for these two plants, a decision was made to replace the current fuel storage racks with `Boral' poisoned high density fuel storage racks. The design objective of the high density fuel storage rack is to provide underwater storage for spent fuel while maintaining keff = 0.95, with 95% probability at a 95% confidence level (95/95), including an allowance for biases and uncertainties for all normal and abnormal conditions. The authors study the criticality safety, including an uncertainty analysis for the poisoned storage racks. The application of the results presented is applicable to other BWR plants using similar spent fuel storage racks.

Tien-ko, W., C. Szu-li, *et al*. (1988). "Criticality safety evaluation for high-density spent-fuel storage racks." Nuclear Technology 83(1): 5-15.

Using as a starting base the high-density spent-fuel storage racks to be put into the Chinshan and Kuoshang nuclear power plants, a series of criticality analyses with various combinations of fuel assemblies and storage rack designs were performed using an AMPX-KENO/XSDRNPM computer code package. The calculated k value for the storage pools in the two subject plants using Boral (0.013 g/cm2 10B) poisoned rack lattices and 3.2 wt.% enriched fuel assemblies is 0.900 under conservative assumptions. Considering all the calculation biases and statistical and manufacturing uncertainties, the maximum k value is estimated to be 0.929 under normal storage conditions. Variation in water temperature and density or abnormal positioning of fuel assemblies results only in a negative effect on k value.

Toffer, H. (1971). Fuel storage basin criticality safety analysis report and technical basis, Douglas United Nuclear, Inc., Richland, Wash.(USA).

The criticality safety analysis was performed to provide a detailed answer to an AEC-RL inquiry about the feasibility of storing a full N Reactor fuel discharge in the storage basin with fully-loaded cubicles. This would require storage on top of the existing concrete cubicles which was not considered in the previous N fuel basin criticality analyses. No experimental criticality data exist for this particular configuration, so criticality control requirements were established by several independent computational methods. In addition, the storage system studied was too complex to evaluate against current N Reactor criticality criteria; there a set of criteria based on the effective neutron multiplication factor, k/sub eff/, was formulated. The report describes the results of the criticality calculations, related hazards analyses, and provides a technical basis for specifying safe limits for storage of fuel on top of the cubicles in the fuel basin. The discussion includes seismic analysis and a description of modifications to structures which would provide adequate criticality control for the proposed storage method and increase utilization of available storage space. An engineering study is required to finally establish economic feasibility of the proposed storage configuration.

Tomlinson, E.T. and C.L. Brown (1983). "Nuclear criticality safety considerations in design of dry fuel assembly storage arrays." Nuclear Technology 63(2): 347.

The potential effect of low-density water moderation on an Independent Spent Fuel Storage Installation (dry type) was investigated. As a function of water density, k was determined for several water-flooding sequences. Calculations were performed for both unirradiated fuel assemblies and fuel at a burnup of 24 GWd/tonne M.

Wagner, J.C. and C.V. Parks (2001). "A critical review of the practice of equating the reactivity of spent fuel to the fresh fuel in burnup credit criticality safety analyses for PWR spent-fuel pool storage." Nuclear Technology 136(1): 130-40.

This research examines the practice of equating the reactivity of spent fuel to that of fresh fuel for the purpose of performing burnup credit criticality safety analyses for pressurized water reactor (PWR) spent-fuel pool (SFP) storage conditions. The investigation consists of comparing keff estimates based on reactivity equivalent fresh fuel enrichment (REFFE) to keff estimates using the calculated spent-fuel isotopics. Analyses of selected storage configurations common in PWR SFPs show that this practice yields nonconservative results (on the order of a few tenths of a percent) in configurations in which the spent fuel is adjacent to higher-reactivity assemblies (e.g., fresh or lower-burned assemblies) and yields conservative results in configurations in which spent fuel is adjacent to lower-reactivity assemblies (e.g., higher-burned fuel or empty cells). When the REFFE is determined based on unborated water moderation, analyses for storage conditions with soluble boron present reveal significant nonconservative results associated with the use of the REFFE. Finally, it is shown that the practice of equating the reactivity of spent fuel to fresh fuel is acceptable, provided the conditions for which the REFFE was determined remain unchanged.

Wang, T.K. (1991). Criticality analysis for high-density spent-fuel storage racks, Tokyo, Japan, Japan Atomic Ind. Forum.

Using the high-density spent-fuel storage racks to be put into the Chinshan and Kuoshang (BWR) nuclear power plants as a starting base, a series of criticality analyses with various combinations of fuel assemblies and storage rack designs were performed using the AMPX-KENO/XSDRNPM computer code package. The analysis has taken into account all the calculational bias, statistical errors and manufacturing uncertainties as well. The effects of water temperature, water density and abnormal fuel assembly positioning are investigated. The possibility of fuel rod consolidation is also studied. In addition, the k values and the possible uncertainties of several carefully selected combinations of fuel assemblies and rack designs are symmetrically evaluated; the results are tabulated. Based on these tabulated data, interpolations can be used for the estimation of the k values and the associated errors for any particular fuel and rack combination. Thus, the results presented in this paper are applicable to other BWR plants using similar storage racks.

Welfare, F.G. (1980). Criticality safety in the storage of spent power reactor fuel, Springfield, VA, USA, Nat. Tech. Inf. Service.

Under a complete nuclear fuel cycle the criticality safety of spent fuel would be a very simple problem. Under this situation the small number of spent fuel assemblies existing at any given time could be easily handled by providing sufficient storage that interaction between assemblies would be negligible. The large number of fuel assemblies which now have to be stored has given rise to some difficult questions. The assemblies are stored in sufficiently close packed arrays that interaction between assemblies becomes of major importance. The neutron adsorption by structural materials and by specially added neutron poisons is included in the evaluation of the system neutron multiplication. Two sets of critical experiments are available for the benchmarking of criticality calculations. All of the data and techniques are available to allow such high density storage to be safely carried out.

Wells, A.H. (1982). Criticality considerations for consolidated spent fuel, USA.

Fuel assembly consolidation poses several new problems for criticality analyses of spent fuel pools. Light water reactor fuel typically has a `dry', undermoderated rod lattice. The use of gridded or ungridded rod storage containers allows both more fuel in a rack location and a wetter lattice in partially filled containers. Such partially filled containers, if stored in a storage rack designed for standard assemblies, may exceed the regulatory limit for nuclear criticality safety.

Werme, L.O. (1998). Design Premises for Canister for Spent Nuclear Fuel. Technical Report TR-98-08. Swedish Nuclear Fuel and Waste Management Co. (SKB).

Whitesides, G.E. (1996). Criticality safety criteria for the handling, storage, and transportation of LWR fuel outside reactors: ANS-8.17-1984.

The potential for criticality accidents during the handling, storage, and transportation of fuel for nuclear reactors represents a health and safety risk to personnel involved in these activities, as well as to the general public. Appropriate design of equipment and facilities, handling procedures, and personnel training can minimize this risk. Even though the focus of the American National Standard, `Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,` ANSI/ANS-8.1-1983, is general criteria for the ensurance of criticality safety, ANS-8.17-1984, provides additional guidance applicable to handling, storage, and transportation of light-water- reactor (LWR) nuclear fuel units in any phase of the fuel cycle outside the reactor core. ANS-8.17 had its origin in the late 1970s when a work group consisting of representatives from private industry, personnel from government contractor facilities, and scientists and engineers from the national laboratories was established. The work of this group resulted in the issuance of ANSI/ANS-8.17 in January 1984. This document provides a discussion of this standard.

Wilkinson, W.L. (2005). "The relevance of IAEA tests to severe accidents in nuclear fuel cycle transport." Packaging, Transport, Storage and Security of Radioactive Material 16(4): 267-272.

The design and performance standards for packages used for the transport of nuclear fuel cycle materials are defined in the IAEA Regulations for the Safe Transport of Radioactive Materials, TS-R-1, in order to ensure safety under both normal and accident conditions of transport. The underlying philosophy is that safety is vested principally in the package and the design and performance criteria are related to the potential hazard. Type B packages are high-duty packages which are used for the transport of the more radioactive materials, notably spent fuel and vitrified high-level waste (VHLW). Tests are specified in the IAEA regulations to ensure the integrity of these packages in potential transport accidents involving impacts, fires or immersion in water. The mechanical tests for Type B packages include drop tests onto an unyielding surface without giving rise to a significant release of radioactivity. The objects which could impact upon a package in real-life transport accidents, such as concrete roads, bridge abutments and piers, will yield to some extent and absorb some of the energy of the moving package. Impact tests onto an unyielding surface are therefore relevant to impacts onto real-life objects at much higher speeds. The thermal test specifies that Type B packages must be able to withstand a fully engulfing fire of 800°C for 30 min without significant release of radioactivity, and this has to be demonstrated, for example, by analytical studies backed up by experimental tests. The regulations also specify immersion tests for Type B packages of 15 m for 8 h without significant release of radioactivity; and in addition for spent fuel and VHLW packages, 200 m for 1 h without rupture of the containment. There is a large body of evidence to show that the current IAEA Type B test requirements are severe and cover all the situations which can be realistically envisaged in the transport of spent fuel, VHLW and other fuel cycle materials. Any proposals for more severe tests, which have little technical justification, should therefore be treated with caution since this could result in a loss of public confidence in the current regulations, and the ratcheting up of design requirements which could not be justified on quantitative safety grounds.

Young Shin, L., K. Hyun Soo, *et al*. (2004). "Effect of irradiation on the impact and seismic response of a spent fuel storage and transport cask." Nuclear Engineering and Design 232(2): 123-9.

The spent fuel storage and transport cask must withstand various accident conditions such as fire, free drop and puncture in accordance with the requirement of the IAEA and

domestic regulations. The spent fuel storage and transport cask should maintain the structural safety not to release radioactive material in any condition. And also the effects of the irradiation should be considered because the spent fuels stored in the cask for a long time and be possible to change the mechanical properties of the cask. In this study, the changed mechanical properties of the cask after irradiation for the 30 years storage periods are assumed and applied to the impact analysis using ABAQUS/Explicit code and seismic analysis using ANSYS code. The stress intensity on each part of the cask is calculated and the effects of irradiation are studied and structural integrity of the package is evaluated.

Zaker, M. and H. Azimi (1991). Criticality calculations of spent fuel storage racks, Dorchester, UK, AEA Technol.

The main objective is the criticality study of spent fuel storage racks of the Tehran Research Reactor. The study has been performed for HEU (Highly Enriched Uranium 93%) and LEU (Low Enriched Uranium 20%) fuel elements. Criticality accident analysis shows that if the spent fuel storage racks which are hanged from the wall of the pool and each of them contain 6 standard fuel elements, fall down in the bottom of the pool in such a way that they are positioned on top of each other, the effective multiplication factor for HEU and LEU will be 0.74397 and 0.77595 respectively.

APPENDIX C REFERENCES

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