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MONTE CARLO CRITICALITY CALCULATIONS FOR THE CANDU SPENT FUEL

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-Research report-

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Abstract

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In order to perform a safety analysis criticality calculations are needed, that means determination of the neutron multiplication factor for the system.

In the paper, for criticality calculations the KENO VI Monte Carlo code, integrated in SCALE programs package, was used. The neutron effective multiplication factor was calculated both in normal conditions and accidental scenarios for intermediate storage of the CANDU spent fuel bundles.

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In order to perform a safety analysis criticality calculations are needed, that means determination of the neutron multiplication factor for the system.

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1. Introduction

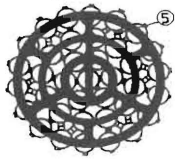
In order to know the excess reactivity in the reactor core fuel bundles criticality calculations are needed. The spent fuel contains plutonium, who can fission, so the reactivity problem we must looking for, related to the personnel protection during the spent fuel manipulation and storage.

The CANDU - 600MWd reactor is a PHWR type reactor, cooled with pressurized heavy water, moderated with heavy water, and witch use natural Uranium as a fuel.

The heavy water moderator is contained in a horizontal stainless steel cylindrical shell (the calandria) crossed by 380 pressure tubes. Pressurized heavy water coolant is pumped through the pressure tubes, cooling fuel and conveying heat from the fuel to the steam generators. Each pressure tube is isolated and insulated from the heavy water moderator by a concentric calandria tube. The annular space between the pressure and calandria tubes (gap) is filled with pressurized gas that acts like a thermal insulator. The pressure tube together with the calandria tube composes a technological channel. The 380 technological channels form the reactor core, /1/.

Inside the pressure tubes are placed the 4560 (= 380 × 12) fuel elements assemblies (fuel bundles). CANDU fuel bundle used for the Cernavoda nuclear reactor is composed by 37 zircalloy rods filled with natural UO_2 pellets, /1/. The geometrical arrangement consists in 3 concentrical rings (6, 12 and 18 rods, respectively) and 1 rod in the middle (Figure 1).

1 zircalloy bearing pads
2 zircalloy sheath
3 zircalloy end plate
4 UO_2 pellets
5 zircalloy spacers
6 zircalloy end cap



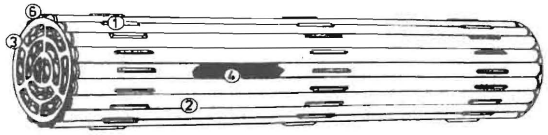


Figure 1 CANDU fuel bundle with 37 rods

One of the basic steps in the spent fuel management is the spent fuel intermediate storage, that follows to impose technical operations of fuel bundles download from the reactor, and also the decision for spent fuel final storage or fuel reprocessing.

Irradiated fuel bundles transport rise up many manipulation, biological protection and cooling problems. Therefore, the spent fuel bundles must be kept in the nuclear power plant storage pools, at least 6 months, before been send to final storage or reprocessing. The major task for this storage period is to ensure the appropriate time range for deactivation and reduction of the spent nuclear fuel radioactivity. In these terms, the basic demand consists in maintaining the fuel bundles sheath integrity in order to eliminate the uncontrolled escape of fission products into the environment.

2. Spent fuel intermediate storage facilities

The concrete cask storage concept represents a passive cooling storage system, with no needs for equipment under normal conditions of operation. Starting from this Canadian concept for storage, for Cernavoda nuclear reactor spent fuel intermediate storage consists in concrete monolith modules, one of each with 17,400 fuel bundles storage capability.

Concrete monolith module is a reinforced concrete prismatic construction, empty inside, that contains 29 storage enclosures arranged on 3 levels (Figure 2). The concrete monolith module includes:

- reinforced concrete structure
- storage enclosure
- admission and air evacuation exits
- spent fuel shipping and storage casks.

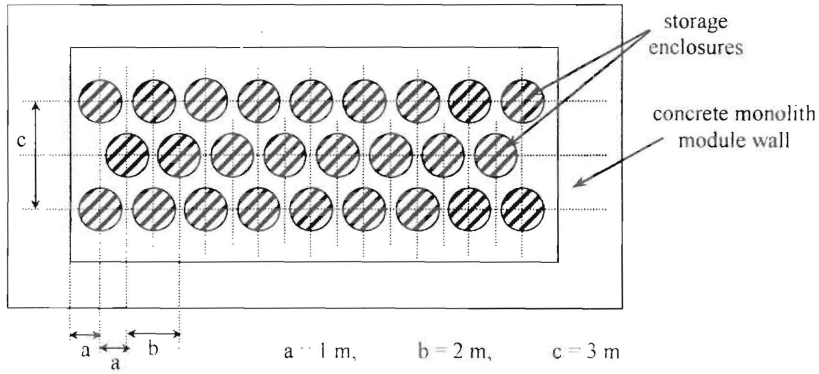


Figure 2 Upside view of concrete monolith module

The storage enclosure has cylindrical shape and is made from galvanized steel clad. In this storage facility can be stored on long term 10 shipping and storage casks each containing 60 spent fuel bundles. At the enclosure bottom there are anti-earthshake supports. To the top the storage enclosure are closed with protection reinforced concrete steel plugs that can be removed in order to allow the insertion of new spent fuel shipping and storage casks.

Spent fuel shipping and storage casks are the active elements of concrete monolith module. In a shipping and storage cask there are 60 CANDU spent fuel bundles, vertically positioned, on 4 concentrically circles (6, 12, 18 and 24 fuel bundles), as it can be seen in Figure 3. Inside these shipping and storage casks residual heat of spent nuclear fuel is generated, together with physical phenomena looking for gamma radiation emission.

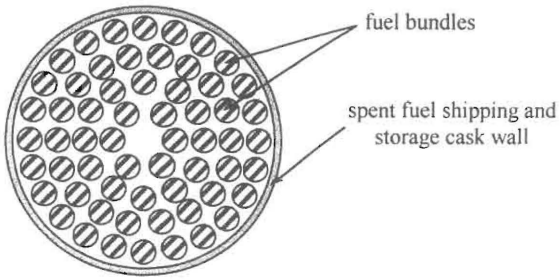


Figure 3 Upside view of the arrangement of the spent fuel bundles in the shipping and storage cask

3. General description of the criticality code

Starting from 1976, as an answer to the NRC (Nuclear Regulatory Commission) demand, a modular package of programs was developed to ORNL (Oak Ridge National Laboratory).

SCALE (Standardized Computer Analyses for Licensing Evaluation) system is an easy to use system for criticality, shielding and thermal analysis of nuclear facility and package designs.

The SCALE system is composed by a number of functional modules for data processing and analyses effectuation in standard sequences:

- **KENO** a 3-D Monte Carlo program optimized for calculating the neutron multiplication factor „k-effective" of a fission system
- **MORSE** 3-D Monte Carlo program designed for tracking particles from a radioactive source to a detector
- **XSDRN** 1-D discrete ordinates radiation transport code
- **ORIGEN-S** point depletion and decay analyses to obtain isotopic concentrations, decay heat sources and radiation source spectra
- **HEATING** 3-D finite volume code for solving steady state and/or transient heat conduction problem.

Standard sequences for criticality calculation used in safety analyses, CSAS (Criticality Safety Analysis Sequences), allow automated processing of cross sections specific to each problem, followed by the system neutronic multiplication factor calculations.

Sequence CSAS6 is adapted to criticality calculations using KENO VI code, an extension of KENO Monte Carlo criticality program, developed for use in the SCALE system. CSAS6 has 2 control modules, CSAS26 and CSAS26X, respectively (Figures 4, 5). Both use BONAMI /2/ and NITAWL-II /3/ functional modules for cross sections processing and KENO VI code for criticality calculation. CSAS26X contains one more functional module, XSDRNPM, whose task is to calculate cell averaged cross sections.

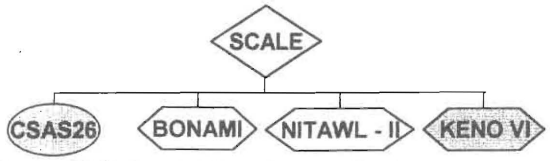


Figure 4 CSAS26 criticality analysis standard sequence structure

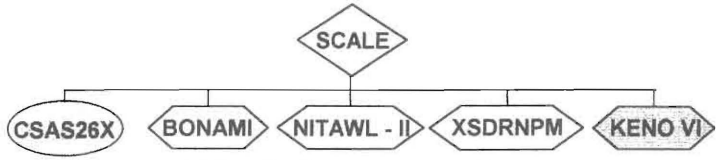


Figure 5 CSAS26X criticality analysis standard sequence structure

KENO VI program /4/ is used in CSAS6 sequences for Monte Carlo multigroup criticality calculations (effective neutronic multiplication factor) of a 3-Dimensional system. In order to obtain the equation that give us the solutions, we start from the Boltzmann neutron transport equation, which may be written as:

$$\frac{1}{v}\frac{\partial\Phi}{\partial t}(X,E,\Omega,t)+\Omega\cdot\nabla\Phi(X,E,\Omega,t)+\Sigma_t(X,E,\Omega,t)\cdot\Phi(X,E,\Omega,t)=S(X,E,\Omega,t)+$$
$$+\int_{E'}\int_{\Omega'}\Sigma_s(X,E'\rightarrow E,\Omega'\rightarrow\Omega,t)\cdot\Phi(X,E',\Omega',t)\,d\Omega'\,dE' \qquad (1)$$

where: $\Phi(X,E,\Omega,t)$ = neutron flux (neutrons/cm^2/sec) per unit energy at energy E, per steradian about direction Ω, at position X and time t, moving with speed v corresponding to E

$\Sigma_t(X,E,\Omega,t)$ = macroscopic total cross section of the media (cm^{-1}) at position X, energy E, about direction Ω and at time t

$\Sigma_s(X,E'\rightarrow E,\Omega'\rightarrow\Omega,t)$ = macroscopic differential cross section of the media (cm^{-1}), per unit energy at energy E', per steradian about direction Ω', at position X and time t, for scattering at energy E and about direction Ω

$S(X,E,\Omega,t)$ = neutron source (neutrons/cm^3/sec born at position X and time t, per energy unit at energy E, per steradian about direction Ω (excludes scattering source).

Defining $q(X,E,\Omega,t)$ as total source of neutrons (sum above all the external sources, scattering, fission and other contributions), assuming that the media is isotropic and ignoring the time dependence of the cross sections, we can rewrite the equation in multigroup form:

$$\frac{1}{v_g}\frac{\partial\Phi_g}{\partial t}(X,\Omega,t)+\Omega\cdot\nabla\Phi_g(X,\Omega,t)+\Sigma_{tg}(X)\cdot\Phi_g(X,\Omega,t)=q_g(X,\Omega,t) \qquad (2)$$

where: g = energy group of interest

v_g = average speed of the neutrons in energy group g

$\Phi_g(X,\Omega,t)$ = angular flux of the neutrons having their energies in group g, at position X and time t

$\Sigma_{tg}(X)$ = macroscopic total cross section of the media at position X for group g corresponding to:

$$\Sigma_{tg}(X) = \frac{\int_{\Delta E_g} \Sigma_t(X,E)\cdot\Phi(X,E,\Omega,t)\,dE}{\int_{\Delta E_g} \Phi(X,E,\Omega,t)\,dE}$$, where ΔE_g defines energy group g

$q_g(X,\Omega,t)$ = total source of neutrons contributing to energy group g at the position X and time t, in direction Ω.

Defining the multiplication factor k as the ratio between the number of neutrons in $(n+1)^{th}$ generation and the number of neutrons in n^{th} generation, the transport equation can be written as follows:

$$\frac{\nu_g(X)\cdot\Sigma_{fg}(X)}{\Sigma_{tg}(X)}\Sigma_{tg}(X)\cdot\Phi_{g,n}(X,\Omega) = \frac{\nu_g(X)\cdot\Sigma_{fg}(X)}{\Sigma_{tg}(X)}\Sigma_{tg}(X)\int_0^\infty dR\, e^{-T(R)}\cdot$$

$$\cdot\left\{\frac{1}{k}\sum_{g'}\int_{\Omega'}\frac{\nu_{g'}(X-R\Omega)\cdot\Sigma_{fg'}(X-R\Omega)}{\Sigma_{tg'}(X-R\Omega)}\chi(X-R\Omega,g'\to g)\cdot\Sigma_{tg'}(X-R\Omega)\cdot\Phi_{g',n-1}(X-R\Omega,\Omega')\cdot\right.$$

$$\left.\cdot\frac{d\Omega'}{4\pi}+\sum_{g'}\int_{\Omega'}\frac{\Sigma_s(X-R\Omega,g'\to g,\Omega'\to\Omega)}{\Sigma_{tg'}(X-R\Omega)}\Sigma_{tg'}(X-R\Omega)\cdot\Phi_{g',n}(X-R\Omega,\Omega')\,d\Omega'\right\} \qquad (3)$$

where: $\nu_g(X)$, $\nu_{g'}(X)$ = number of neutrons resulting from a fission in energy group g and g' respectively, at position X

$\chi(X,g'\to g)$ = fraction of neutrons born in energy group g from fission in energy group g' in the media at position X

$\Sigma_{fg'}(X)$ = macroscopic fission cross section of the material at position X for a neutron from energy group g'.

The above equation is solved in KENO VI by using an iterative procedure.

A collision occurs in a geometrical region when a history exhausts its mean free path length within the boundaries of the region. For each collision, the absorbed weight and the fission weight are tabulated, then the weight is modified by the non-absorption probability. This new weight is checked for splitting and Russian roulette; if it survives, the history is scattered. A new energy group is selected from the cumulative transfer probability distribution. This group-to-group transfer determines an angular scattering distribution, usually expressed as a Legendre expansion of the cross section transfer array. A set of discrete angles and probabilities are generated by a generalized Gaussian quadrature procedure, preserving the moments of the 3 Legendre expansion of the angular scattering distribution.

In order to minimize the statistical deviation of k per unit computer time, KENO VI utilizes weighted k_{eff} tracking rather than analog tracking. Weighted tracking accounts for absorption by reducing the neutron weight, rather than allowing the neutron history to be terminated by absorption.

To prevent expending excessive computer time tracking low-weight neutrons, Russian roulette is played when the weight of the neutron drops below a preset weight, WTLOW. Neutrons that survive Russian roulette are assigned a weight, WTAVG. The value of WTLOW and WTAVG can be assigned as a function of position and energy.

4. Geometrical models used for criticality calculations

The spent fuel shipping and storage cask has a cylindrical shape and contains 60 CANDU spent fuel bundles, vertically positioned, on 4 concentrical rings (6, 12, 18 and 24 fuel bundles), as in Figure3.

A fuel bundle enters in the calandria tube filled with D_2O coolant, concentrically positioned in the pressure tube. Between the calandria tube and the pressure tube there is a gap filled with air that acts like a thermal insulator. 4 cylinders concentrically disposed compose this assembly: 3 for pressure tube, air gap and calandria tube, and 1 for fuel bundle homogenized composition.

The storage enclosure is a steel cylinder that contains maximum 10 spent fuel shipping and storage casks, placed one over the other. Each fuel shipping and storage cask was considered as a cylinder, and all the 10 cylinders were considered inside 2 steel cylinders (the inner and outer wall of the storage enclosure) concentrically disposed.

The storage module is a concrete prismatic box that contains 29 storage enclosures disposed on 3 rows. For geometrical model of the concrete storage module we consider a rectangular parallelepipedic box with 29 cylinders disposed as Figure 2 shows.

5. Results

The paper goal consists in criticality calculations for intermediate storage of CANDU spent fuel bundles both for normal conditions and accidental scenarios.

To execute criticality calculations we used the standard sequence CSAS26 contained by the SCALE system.

The isotopic concentration evolution with burnup degree of the most interesting nuclides in criticality calculation for a CANDU fuel bundle /5/ is given in Figure 6. We used the normalization to U-238 values in order to obtain a more evident evolution.

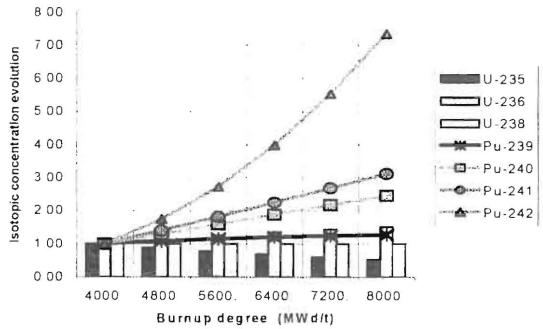


Figure 6 Nuclides isotopic concentrations for an irradiated CANDU fuel bundle

As a preliminary but „covering" analysis for nuclear safety point of view, a criticality calculation for a single fuel bundle burn up to 6400 MWd/tU, discharged from the reactor and placed in air was made. We obtain a neutronic multiplication factor $k_\infty = 0.343$.

In Tabel 1 the isotopic concentrations (in atom-gram/ kg U) of the most interesting actinides contained in the spent fuel (7500MWD/tU burnup degree), for different cooling times in the reactor pool /6/, are given.

Table 1 Isotopic concentrations for the most interesting actinides (atom·g/kgU)

Isotop	Cooling time (years)						
	1	2	3	4	5	6	7
U-233	2.28–09	2.34–09	2.40–09	2.46–09	2.52–09	2.58–09	2.64–09
U-234	1.94–04	1.95–04	1.95–04	1.95–04	1.95–04	1.95–04	1.96–04
U-235	1.17–02	1.17–02	1.17–02	1.17–02	1.17–02	1.17–02	1.17–02
U-236	3.20–03	3.21–03	3.21–03	3.21–03	3.21–03	3.21–03	3.21–03
U-238	4.14+00	4.14+00	4.14+00	4.14+00	4.14+00	4.14+00	4.14+00
Np-237	1.64–04	1.64–04	1.64–04	1.64–04	1.64–04	1.64–04	1.64–04
Np-239	1.51–11	1.51–11	1.51–11	1.51–11	1.51–11	1.51–11	1.51–11
Pu-236	5.25–11	4.11–11	3.22–11	2.53–11	1.98–11	1.55–11	1.22–11
Pu-238	2.49–05	2.51–05	2.50–05	2.49–05	2.47–05	2.45–05	2.43–05
Pu-239	1.03–02	1.03–02	1.03–02	1.03–02	1.03–02	1.03–02	1.03–02
Pu-240	4.87–03	4.87–03	4.87–03	4.87–03	4.87–03	4.87–03	4.87–03
Pu-241	9.43–04	9.00–04	8.58–04	8.18–04	7.80–04	7.44–04	7.09–04
Pu-242	2.62–04	2.62–04	2.62–04	2.62–04	2.62–04	2.62–04	2.62–04
Am-241	5.55–05	9.91–05	1.41–04	1.80–04	2.18–04	2.54–04	2.88–04
Am-242M	1.93–07	1.92–07	1.91–07	1.90–07	1.89–07	1.89–07	1.88–07
Am-242	2.32–12	2.31–12	2.30–12	2.28–12	2.27–12	2.26–12	2.25–12
Am-243	1.80–05	1.80–05	1.80–05	1.80–05	1.80–05	1.80–05	1.80–05
Cm-242	5.06–07	1.07–07	2.30–08	5.22–09	1.46–09	6.67–10	4.97–10
Cm-243	1.62–08	1.59–08	1.55–08	1.52–08	1.49–08	1.45–08	1.42–08
Cm-244	1.55–06	1.49–06	1.43–06	1.38–06	1.33–06	1.28–06	1.23–06
Cm-245	2.70–08	2.70–08	2.70–08	2.70–08	2.70–08	2.70–08	2.70–08
Cm-246	2.02–09	2.02–09	2.02–09	2.01–09	2.01–09	2.01–09	2.01–09

After 7 years in the reactor pool, actinide concentrations are drastically reduced with one notable exception, the Pu-239 isotop. The criticality calculation for a fuel bundle burned up to 7500 MWD/tU and cooled 7 years in the reactor pool, give $k_\infty = 0.349$.

The results obtained for the spent fuel intermediate storage (in shipping and storage cask, storage enclosure and concrete storage module) under normal conditions are presented in Table 2.

Table 2 k-effective values for the spent fuel intermediate storage

	shipping and storage cask	storage enclosure	concrete storage module
k-eff	0.3423	0.3421	0.3420
Rel. dif. (%)	-	0.058	0.087

The relative difference values show clearly the subcriticality preservation.

The second task for this paper was to calculate k-effective for 2 accidental scenarios during the intermediate storage of the spent nuclear fuel.

❒ ***scenario no.1***

▪ The spent fuel bundles burned up to 7500 MWd/tU and cooled in the reactor pool for 7 years are stored in the shipping and storage casks.

▪ After colling air admision exits blocking, the fuel heating determines a temperature rising up to 300 °C (160° in normal conditions for spent fuel storage).

After running CSAS26 for criticality calculations we obtain $k_{ef} = 0.339$. When temperature drastically rise up inside the shipping and storage cask following an accidental condition (cooling air admission exists are blocked) occurs a resonance „broadening", so the absorptions weight in neutronic balance increases and the assembly, subcritical in normal conditions ($k_{ef} = 0.3423$), remains faraway from the critical state.

❒ ***scenario no. 2***

▪ A shipping and storage cask that contains 60 spent fuel bundles suspended in the crane hook falls down freely (from aprox. 9m) in the storage enclosure.

▪ Following the impact, the shipping and storage cask brake up and the fuel bundles scatter inside the storage enclosure.

The tests done at ICN Pitesti, to 1 : 1 scale, show that the cask integrity is preserved even if the cask falls down from a height of 10 m. Although these results, we consider the scenarios as real and we calculate the criticality for the worst case, namely when the fuel bundles falls one near the other. After CSAS26 execution an average value of $k_{ef} = 0.3642$ is obtained, so the assembly subcriticality is preserved.

6. Conclusions

No decisions related on the spent nuclear fuel storage mode, place and storage period were taken in countries with nuclear energetic development, until the spent nuclear fuel storage became a limiting factor for the next fuel cycle phase. Consequently, is obvious the primary goal of an appropriate spent fuel storage: to avoid and reduce the biologic risks for personnel and environmental.

As it can be seen from the criticality calculation results for CANDU spent fuel intermediate storage under normal and 2 accidental conditions, the assembly subcriticality is preserved.

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