ESTIMATION OF CRITICAL PARAMETERS FOR A LOSS-OF-COOLING ACCIDENT AT THE CANDU NPP SPENT FUEL POOL

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Abstract

A Loss-of-Cooling Accident scenario at the spent fuel pool (SFP) of a CANDU NPP was considered for analysis. Rising of the temperature of the demineralised water in SFP will primarily lead to the boiling of the water, after a time interval $t_{boiling}$. If the cooling is not yet restored, due to water evaporation, after the additional time $t_{dewatering}$ the spent fuel assemblies positioned at the highest level, on the vertical axis, in SFP, will become subject to dewatering. The parameters $t_{boiling}$ and $t_{dewatering}$ characterizing the accident that occurs at the SFP filled with 37-element CANDU spent fuel bundles, were calculated for various types of fuel (either natural uranium or slightly enriched uranium, SEU, with enrichment in U^{235} ranging between 0.90 wt% and 1.50 wt%), for burn-ups between 6500 MWd/tU and 8000 MWd/tU, using an in-house developed Visual basic 6.0 computer code. The residual thermal powers were estimated using the ORIGEN-S module included in the SCALE4.4a software package.

Keywords: *Spent Fuel Pool, Loss of Cooling Accident, CANDU, Natural Uranium, SEU Fuel*

Introduction

The past incidents/accidents involving facilities for the storage of spent fuel (Bruce A- Unit 4 Reactor, Canada, 1983; Paks cleaning tank incident, Hungary, 2003; Fukushima Daiichi Nuclear Power Station, Japan, 2011) were followed by carefully driven investigations, questions, development of models and simulations [1]. Stress tests and various evaluations involving assessments of possible accident scenarios at facilities devoted to the storing of spent fuel were considered for a lot of Nuclear Power Plants (NPPs) which were involved in incidents. Specific safety guides for the storage of spent nuclear fuel were elaborated and updated [2]. The simulation and modeling of accidents in spent fuel pools made the object of various international meetings [3].

The paper addresses an important aspect of nuclear safety related to the spent fuel management after it was discharged from the reactor core and sent for intermediate temporary wet storage inside the spent fuel pool. A Loss-of Cooling Accident (LOCA) scenario at the spent fuel pool (SFP) of a CANDU NPP was considered for analysis.

The cooling of the spent fuel pool is assured, under normal conditions, by using demineralised water that is circulated thorough a dedicated cooling system. In case of a loss-of-cooling accident (which could occur mainly due to a long-term black-out), the temperature of the demineralised water in the pool cannot be kept anymore in a limited interval. The temperature rising will primarily lead to the boiling of the water in spent fuel pool (after a time interval $t_{boiling}$ with respect to the very first moment of the accident considered as t=0).

If the black-out continues, due to water evaporation, after the time $t_{dewatering}$ (considered also with respect to the moment t=0 when the loss-of-cooling accident started), the spent fuel assemblies/ elements positioned at the highest level, on the vertical axis, in SFP, are most likely to be exposed. These spent fuel assemblies will be exposed to the atmosphere above the level of the boiling water.

The aim of this work was to evaluate, using a dedicated Visual Basic 6.0 code, the parameters $t_{boiling}$ and $t_{dewatering}$ characterizing the loss-of-cooling accident that occurs at the SFP filled with 37-element CANDU spent fuel bundles, for various types of fuel (either natural uranium fuel or SEU, slightly enriched uranium, fuel). The considered U^{235} enrichments of the SEU fuel ranged between 0.90 wt% and 1.50 wt%.

The spent fuel residual thermal powers were estimated by using the ORIGEN-S module, included in the software package SCALE4.4a. According to the presentation of the software developer, ″ORIGEN-S is a SCALE system module used to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms. ORIGEN-S computes time-dependent concentrations and source terms of a large number of isotopes, which are simultaneously generated or depleted through neutronic transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates. The calculations may pertain to fuel irradiation within nuclear reactors, or the storage, management, transportation, or subsequent chemical processing of removed fuel elements″ [4].

The considered fuel burn-up before the discharge from reactor ranged between 6500 MWd/tU and 8000 MWd/tU, which is in agreement with the fuel maximum burn-up at discharge around 7000 MWd/tU typical for ″standard″ operation of a CANDU6 reactor (as Units 1 and 2 from Cernavoda NPP).

Considerations on the spent fuel storage pool (SFP)

Based on a detailed description, from literature, of a typical SFP corresponding to a CANDU NPP [5], a rectangular spent fuel pool was considered for the present work, which is characterized by the following dimensions:

- length L = 20 m;
- width w = 12 m;
- depth h = 9.8 m (out of which the water itself goes up to 8.8 m).

The spent fuel discharged from the reactor to the SFP is cooled down by circulating the demineralised water from the SFP through a dedicated cooling system, based on several pumps. The normal operation of this cooling system allows the operators of SFP to maintain the temperature of water below 37 °C. In order to operate the facility, including the cooling system, a 24/7 availability of a power station is needed. Hence, an eventual black out would have severe consequences, first of all by making impossible to keep the temperature of water in SFP under control.

The manner of storing the fuel in SFP is presented in Figs. 1 and 2. The design of the manner of operation of SFP involves a handling area (see Fig. 2) where the spent fuel assemblies could stay only for a limited amount of time, during the transfer operations between the reactor, the SFP and the dry storage facility (where the spent fuel is finally moved, after a long enough residence in SFP).

The practical manner of storing the spent fuel in SFP is subject to specific procedures of the facility. The present work was not strictly based on this kind of procedures from a SFP facility of a specific NPP, due to restricted access to such procedures. However, several simplifying, intuitive hypotheses were considered based on screening of the public information available in the literature and webpage of these facilities. First of all, the fuel recently discharged from reactor was supposed to be placed as close as possible to the floor of the SFP. The spent fuel was periodically fully rearranged in SFP, so that the spent fuel having the longest residence time in SFP would further be placed near the surface of SFP (as a caution procedure, in case of an event that could lead to the exposure of such spent fuel to the air).

As far as the residence time of the spent fuel in SFP is concerned, this varies from one day or even shorter (for the spent fuel introduced in the SFP in a specified day considered for study) up to longer periods, going for most to eight years back from the moment considered for study. This initial moment (t_0 = 0) of interest for the proposed scenario will be, of course, the moment when the loss-of-cooling accident is supposed to happen.

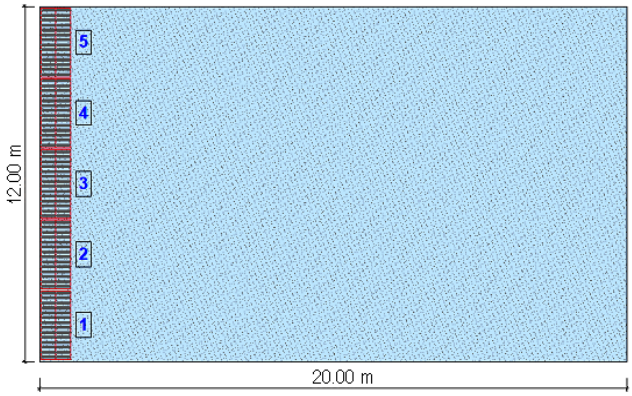


Fig.1. Storing of spent fuel in SFP (along the pool width)

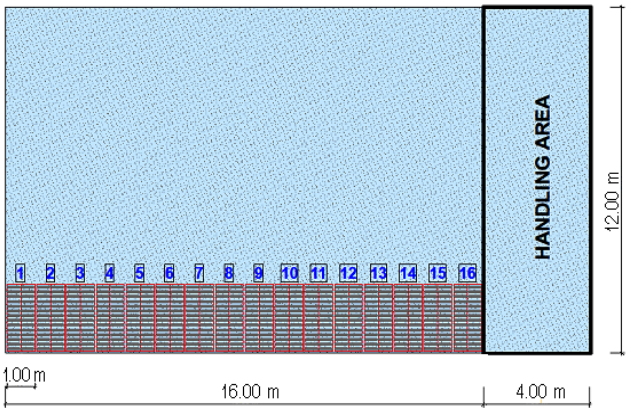


Fig. 2. Storing of spent fuel in SFP (along the pool length)

The loss-of-cooling accident scenario assumptions

The loss-of-cooling accident scenario considered for present study is based on the following assumptions:

• Before the loss-of-cooling accident takes place, 16 spent fuel assemblies (SFAs) (corresponding to the discharge of 8 SFAs out of two fuel channels in the reactor) were already stored in the SFP, on daily bases, shortly after their discharge from the reactor core;

• Before the loss-of-cooling accident takes place, spent fuel assemblies were stored in SFP, according to above presented manner, on daily bases, for 11 months of the year. During a whole month per year, the normal operation of the NPP is stopped for maintenance. This is the so-called ″yearly outage period″.

• The SFAs are kept in SFP for 8 years; once that period elapses, they are removed from SFP, being transferred further to a dry storage on-site facility. Under normal conditions of operation, the rate at which SFAs are removed from SFP is the same with the one at which new SFAs are introduced in SFP (16 SFAs per day in / 16 SFAs per day out). In a specific day, randomly selected, the 16 SFAs that are transferred from SFP to the dry storage facility are supposed to be those SFAs that were introduced in SFP exactly 8 years ago.

• The loss-of-cooling accident is supposed to happen at least 10 years after the NPP was commissioned (which means that the equilibrium operation regime was reached for at least 8 years). In other words, the loss-of-cooling accident is supposed to happen, in our scenario, at a moment when the SFP was fully charged with SFAs, at its maximum capacity of receiving SFAs. This hypothesis is of a conservative type (″the worst case scenario″). During 8 years of operation, the maximum number of SFAs discharged from the CANDU reactor and introduced in SFP can reach 42,784 assemblies.

• All the heat generated in the SFP by SFAs is considered to be transferred solely to the water filling the SFP. This is another conservative element of the considered accident scenario. In other words, the transfer of heat to the metallic structure of the spent fuel, to the metallic structure of the racks in which SFAs are stored, to the walls of SFP and to the atmosphere above the SFP, and in the SFP building, are totally neglected.

Results and discussions

In order to estimate the critical parameters $t_{boiling}$ and $t_{dewatering}$ characterising the loss-of-cooling accident, the residual thermal powers of the spent fuel for a period of 9 years of storage in SFP, starting from the moment of their discharge from the reactor, were evaluated. These kind of calculations were performed both for natural Uranium and SEU (Slightly Enriched Uranium) spent fuel. The considered enrichment of SEU fuel in U^{235} was in the range 0.90 wt% to 1.50 wt%.

For estimation of the spent fuel thermal residual powers, an appropriate time grid was also considered, denser shortly - up to 1 year - after the discharge of the spent fuel out of the reactor than for the rest of the 9 years storage period for which the calculations were performed. Based on the calculated residual thermal powers, it was possible to estimate (again conservatively estimate, by a slightly overestimation) the heat produced by all the SFAs in SFP during a given, user specified time interval, starting from any chosen moment after the onset of the accident. To be highlighted the main (conservative) assumption of the accident scenario, namely that the spent fuel generated heat is fully transferred to the water filling the SFP.

Although the burn-up of SFAs in SFP is not practically the same for all of them, in the calculations (of the residual thermal powers) a fuel burn-up unique value was assumed, an ″equivalent″ burn-up which is the average value of the distribution of the burn-up [6]. The calculations have been performed for 4 different values of (equivalent) burn-up, ranging from 6500 MWd/tU up to 8000 MWd/tU (with a step of 500 MWd/tU).

The residual thermal powers were estimated using the ORIGEN-S code, which is part of the software package SCALE 4.4a, developed by the Oak Ridge National Laboratory, USA.

Following the ORIGEN-S calculations, by analyzing the total thermal (residual) powers – i.e. the sums of powers corresponding, respectively, to light elements, actinides and fission products – a similar behaviour of their time dependence was found for both natural uranium and SEU fuels, regardless the considered enrichment in U^{235} for SEU fuel (see Fig. 3).

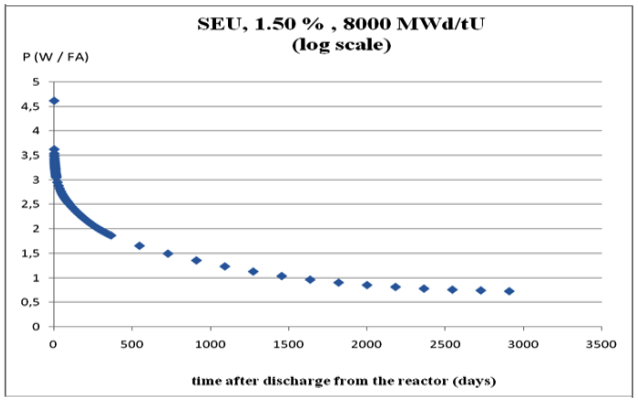


Fig. 3. Variation of total residual thermal power with time (SEU spent fuel, 1.50 wt% enrichment in U^{235}; fuel discharge burn-up 8000 MWd/tU)

The critical parameters characterizing the loss-of-cooling accident, $t_{boiling}$ and $t_{dewatering}$ (both being considered with respect to the onset moment of the LOCA accident) were calculated based on the residual thermal powers previously estimated, using an in-house developed Visual Basic 6.0 computer code. The code was designed in order to allow calculation of the heat produced by all SFAs stored in SFP during any period of time Δ, starting at any moment t_0 following the onset of LOCA accident.

Simultaneously, the amounts of heat $Q_{boiling}$ and $Q_{dewatering}$ leading to the water boiling in SFP, and to dewatering of the most exposed SFAs, respectively, were calculated. By setting the moment $t_0 = 0$ (exactly at the onset of the LOCA accident) and by varying successively the parameter Δ so that the assessed heat Q produced by SFAs in SFP equate as closely as possible $Q_{boiling}$ and $Q_{dewatering}$, the parameters $t_{boiling}$ and $t_{dewatering}$ were evaluated.

For a given fuel burn-up, $t_{boiling}$ and $t_{dewatering}$ are not depending on the type of the fuel (natural uranium or SEU fuel) or the enrichment in U^{235} for the SEU fuel. In fact, this type of result is fully consistent with the fact that, after all, the heat generated is directly related to the number of acts of fissions in the (spent) fuel. Of course, this depends only on the fuel burn-up.

As far as the dependence of these parameters on the fuel burn-up is concerned, they are both monotonously decreasing while the burn-up is increasing. The same trend is obvious for the dependence of the time elapsed between the boiling moment and the dewatering moment ($t_{boiling \rightarrow dewatering} = t_{dewatering} - t_{boiling}$). Table 1 presents these assessed parameters depending on the considered fuel burn-up.

Table 1. $t_{boiling}$, $t_{dewatering}$, $t_{boiling \rightarrow dewatering}$ dependence on the fuel burn-up

Fuel burn-up [MWd/tU]	**$t_{boiling}$ [days]**	**$t_{dewatering}$ [days]**	**$t_{boiling \rightarrow dewatering}$ [days]**
6500	3.3	11.8	8.5
7000	3.1	11.3	8.2
7500	3.0	10.9	7.9
8000	2.9	10.5	7.6

Conclusions

The goal of the present work was to assess, quantitatively and comparatively, the evolution of a loss-of-cooling accident at the spent fuel pool of a CANDU NPP.

For this study, 37-element type CANDU fuel bundles with natural uranium, and slightly enriched uranium (with enrichments in U^{235} in the range 0.90 wt% to 1.50 wt%), respectively, were considered. The fuel burn-up ranged in the interval 6500 MWd/tU ÷ 8000 MWd/tU.

By means of the ORIGEN-S code, the residual thermal powers of the spent fuel were evaluated, considering a period of 9 years storage in the spent fuel pool, starting from the moment of the spent fuel assemblies' discharge from the reactor core. This kind of calculations were performed for both natural UO_2 and SEU spent fuels, considering the above mentioned values for enrichment in U^{235} (in the case of SEU fuel).

Using a Visual Basic 6.0 computer code developed in-house by L.C. Dinu (author), the critical parameters characterizing the loss-of-cooling accident, $t_{boiling}$ and $t_{dewatering}$ (both being considered with respect to the onset moment of the LOCA accident) have been assessed, based on the residual thermal powers previously estimated by ORIGEN-S code.

The values obtained for the parameters of interest, $t_{boiling}$ and $t_{dewatering}$, do not depend on the type of the fuel (natural uranium or SEU fuel) or the enrichment in U^{235} (in the case of SEU fuel) considered.

As the fuel burn-up increase, a shortening of the interest parameters value was noticed. Thus, for a fuel burn-up of 8000 MWd/tU, $t_{boiling}$ was 2.9 days and $t_{dewatering}$ was 10.5 days, whereas for the lowest fuel burn-up considered in this study, 6500 MWd/tU, $t_{boiling}$ was 3.3 days and $t_{dewatering}$ was 11.8 days.

The time elapsed between the boiling moment and the dewatering moment, $t_{boiling \rightarrow dewatering}$, varies, depending on the fuel burn-up, between 7.6 days (8000 MWd/tU) and 8.5 days (6500 MWd/tU).

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