ASSESSMENT OF RADIATION SAFETY OF THE SPENT NUCLEAR FUEL DRY STORAGE CASK

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Abstract

Assessment of radiation safety of the spent nuclear fuel dry storage cask CASTOR RBMK-1500 has been performed by means of the computer modelling (computer codes SAS2H and DORT) and by measurements of the dose rate. The estimated values do not exceed the limited ones. The measured values are several times smaller than the calculated ones due to the conservative approach.

The spent nuclear fuel (SNF) of RBMK type reactor was loaded into a metallic dry storage cask CASTOR RBMK-1500 (produced by company GNB, Germany) for the first time. The first five casks were taken to the open storage site of Ignalina NPP during May – August 1999. Each cask is filled with 102 subassemblies of SNF (total mass of SNF in each cask 6 t).

TABLE 1. THE ACTIVITIES OF THE MAIN ACTINIDES AND FISSION PRODUCTS IN THE SNF CASK

<table>
<thead>
<tr>
<th>Isotope</th>
<th>A, Bq</th>
<th>Isotope</th>
<th>A, Bq</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{234}$U</td>
<td>1.29e+11</td>
<td>$^{85}$Kr</td>
<td>4.47E+14</td>
</tr>
<tr>
<td>$^{235}$U</td>
<td>3.21E+09</td>
<td>$^{90}$Sr</td>
<td>6.40E+15</td>
</tr>
<tr>
<td>$^{236}$U</td>
<td>2.96E+10</td>
<td>$^{90}$Y</td>
<td>6.40E+15</td>
</tr>
<tr>
<td>$^{238}$U</td>
<td>6.60E+10</td>
<td>$^{106}$Ru</td>
<td>9.98E+12</td>
</tr>
<tr>
<td>Np</td>
<td>3.49E+10</td>
<td>$^{106}$Rh</td>
<td>9.98E+12</td>
</tr>
<tr>
<td>$^{238}$Pu</td>
<td>2.32E+14</td>
<td>$^{125}$Sb</td>
<td>3.81E+13</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>8.00E+13</td>
<td>$^{134}$Cs</td>
<td>1.88E+14</td>
</tr>
<tr>
<td>$^{240}$Pu</td>
<td>7.13E+13</td>
<td>$^{137}$Cs</td>
<td>9.60E+15</td>
</tr>
<tr>
<td>$^{242}$Pu</td>
<td>2.04E+11</td>
<td>$^{137m}$Ba</td>
<td>9.06E+15</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>4.04E+14</td>
<td>$^{144}$Ce</td>
<td>1.32E+12</td>
</tr>
<tr>
<td>$^{242m}$Am</td>
<td>1.58E+12</td>
<td>$^{147}$Pm</td>
<td>7.02E+14</td>
</tr>
<tr>
<td>$^{242}$Am</td>
<td>1.58E+12</td>
<td>$^{154}$Eu</td>
<td>4.15E+14</td>
</tr>
<tr>
<td>$^{241}$Cm</td>
<td>1.31E+12</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{243}$Cm</td>
<td>1.13E+12</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{244}$Cm</td>
<td>9.53E+13</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>8.089E+14</td>
<td></td>
<td>3.33E+16</td>
</tr>
</tbody>
</table>
This work deals with characteristics of the first filled cask, which was taken to the open storage site on 11 May 1999. The nuclear composition of SNF was estimated by the computer code SAS2H [1] from a package SCALE 4.3. The following data were taken for estimation: mixture of fuel unit cell UO₂; weight of uranium in assembly 114.5 kg; U isotopic composition \(^{235}\text{U} 0.015\%\), \(^{235}\text{U} 1.8\%\), \(^{238}\text{U} 98.185\%\); fuel pin cell geometry: fuel lattice – triangular, center-to-center spacing between fuel pins 1.605 cm, outside diameter of fuel rod 1.152 cm, Zr clad outside diameter 1.363 cm, clad inside diameter 1.198 cm; number of fuel-rods per assembly 18; outside radius of fuel assembly 3.95 cm; moderator mixture water and graphite; average power of the assembly 2.3 MW; fuel irradiation period of reactor 1000 d; cooling time of SNF 13 years; weight (kg) of light elements in assembly: Zr 20, Nb 0.4, Fe 0.875, Cr 0.2, Ni 0.11, Co 0.01. It should be noted that resultant fuel burn-up was taken 20 MWd/kg. The calculated activities of the main nuclides in storage cask are presented in Table 1. Table 2 presents the calculated SNF neutron emission intensity. The SNF \(\gamma\) and neutron source activity and spectrum are calculated by the computer code SAS2H. The results are presented in Fig. 1 and Fig.2.

**TABLE 2. NEUTRON SOURCE IN STORAGE CASK**

<table>
<thead>
<tr>
<th>Neutron production reaction</th>
<th>n/s</th>
</tr>
</thead>
<tbody>
<tr>
<td>((\alpha, n)) reaction</td>
<td>1.80E+07</td>
</tr>
<tr>
<td>Spontaneous fission</td>
<td>3.39E+08</td>
</tr>
<tr>
<td>Total</td>
<td>3.56E+08</td>
</tr>
</tbody>
</table>

**FIG. 1. The SNF neutron source spectrum**
FIG. 2. The SNF γ source spectrum

These source data were applied for estimation by computer code DORT [2] of dose rates on the surface of the cask, in the storage site and outside of the protective wall. Calculations were fulfilled with assumption of homogeneous contents of cask. The mean density of homogeneous contents is 2.7 g/cm³. The four chemical elements: U, Zr, Fe and O were taken for calculations. The thickness of cask wall is 290 mm and the outer radius is 1036 mm. The cask material density is 7.09 g/cm³. It consists of cast iron with 3.6 % of carbon and other admixtures: 2 % Si, 0.6 % Mn, 0.01 % S, 0.05 % Mg, 1.3 % Ni. The parameters of concrete wall: 8 m from cask, thickness 0.6 m, density 1.96 g/cm³, the composition of concrete: 13 % Ca, 34 % Si, 0.3 % Al, 0.1 % Mg, 51 % O, 0.8 % H.

The dependence of calculated dose rate on cask wall surface upon the height and the dependence of calculated dose rate on 1.5 m height upon the distance are presented in Fig.3. and Fig.4.

FIG. 3. The dependence of calculated dose rate on 1.5 m height upon the distance
FIG 4. The dependence of calculated dose rate on cask wall surface upon the height

The measured dose rate data near the cask are presented in Table 3.

TABLE 3. THE MEASURED DOSE RATE DATA [3] NEAR THE CASK

<table>
<thead>
<tr>
<th></th>
<th>The side surface</th>
<th>1 meter from the side surface</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\gamma$ dose rate, $\mu$Sv/h</td>
<td>85</td>
<td>22</td>
</tr>
<tr>
<td>neutron dose rate, $\mu$Sv/h</td>
<td>200</td>
<td>74</td>
</tr>
</tbody>
</table>

The estimated values do not exceed the limited ones. The measured values are several times smaller than the calculated ones due to the conservative approach.

References