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RADIATION PROTECTION STUDIES REGARDING THE TRANSPORTABILITY OF SECONDARY SOURCE ASSEMBLIES WITH A TN13/2 SHIPPING CASK

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ABSTRACT

As part of the startup of the PALUEL 2 reactor, a transport of active SSAs (secondary source assemblies) was recently carried out with a TN13/2 shipping cask (designed and operated by ORANO TN), commonly used to transport spent fuel assemblies.

Due to the extended shutdown of the PALUEL 2 reactor, its SSAs were expected not to be active enough to ensure appropriate neutron counting during the startup. This led EDF to organize a project to transport active SSAs from other power plants towards PALUEL-2.

As part of this project, radiation-protection studies regarding the transport and operation of a TN13/2 loaded with two SSAs were carried out.

Best-estimate shipping cask models based on those previously developed by EDF for spent-fuel transportation were adapted by describing the SSAs and the additive shielding elements needed for the transport (dummy fuel assemblies with lead pins instead of fuel pins and Al-B_4C shielding bars). Simulations were carried out using the radiation propagation Monte Carlo code TRIPOLI-4®, developed by CEA. Dose equivalent rates (DERs) were computed around the shipping cask using different hypotheses on the additive shielding elements and a wide range of cooling time of the SSAs.

The results show that the transport of 2 SSAs respects the transport DER limitations without any additive shielding element if the SSAs are cooled at least 120 days after core shutdown. We demonstrate that using dummy fuel assemblies with lead pins and Al-B_4C shielding bars allow to transport 20-day cooled SSAs with comfortable margins regarding the DER limitations.

The simulation of an operation configuration also shows that DERs around the shipping cask are lower than those computed for a standard spent fuel evacuation in a close configuration.

Finally, DER measurements carried out during the transport of 2 SSAs towards PALUEL-2 are compatible with the simulation given the conservative assumptions on the source terms. By consolidating the safety demonstration and helping evaluate the operators' dose, this work contributed to successfully plan and carry out the project.

INTRODUCTION

In EDF 1300 MW and 1450 MW PWR cores, secondary source assemblies (SSAs) are inserted in some fuel assemblies. Their purpose is to locally amplify the neutron flux to ensure a high enough count rate on the ex-core neutron detectors, especially during subcritical startup operations. Initially non-radioactive, they are activated during the first cycle of core irradiation and become neutron sources.

Due to the extended shutdown of the PALUEL 2 reactor, its SSAs were expected not to be active enough to ensure appropriate neutron counting during the startup. This led EDF to organize a project to transport four active SSAs from other power plants towards PALUEL 2.

The transport was carried out using a TN13/2 shipping cask, commonly used to transport spent fuel assemblies. As designer and operator of the TN13/2, ORANO TN carried out the regulatory safety studies. As a liable operator, EDF carried out additional radiation-protection studies regarding the transport and operation of a TN13/2 loaded with two SSAs, especially with regard to the operators' dose. This work was carried out using the EDF reference radiation propagation Monte Carlo code TRIPOLI-4® [1], developed by CEA.

CASK AND CONTENT

The shipping cask used to transport the SSAs is a TN13/2B with additional neutron shielding, designed by ORANO TN. This cask is commonly used to transport 12 Uranium oxide spent fuel assemblies from the 1300 MW and 1450 MW EDF reactors to the ORANO La Hague plant. The schematic view of a TN12/2, the equivalent of the TN13/2 for the fuel assemblies of 900 MW reactors, is shown in Figure 1.

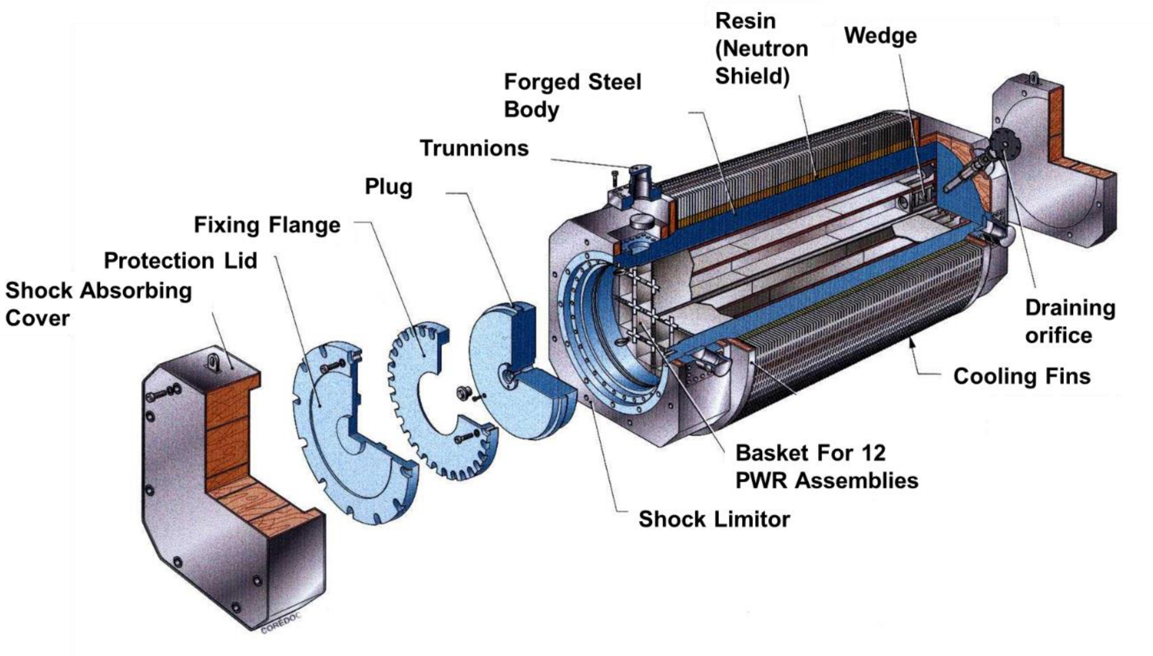

Figure 1 – Main elements of a TN12/2. This cask is very similar to the TN13/2 considered in this document. Credits ORANO TN.

A SSA (Figure 2) is a stationary 24 rods cluster, inserted in a fuel assembly guide-tubes. Four of them are source rods composed of Sb/Be pellets stored in a steel cladding. The total length of a source rod is around 4500 mm but the Sb/Be pellets represent a 2600-mm high column at the bottom part of the rod. The other twenty rods are 130-mm long plug rods, made of steel.

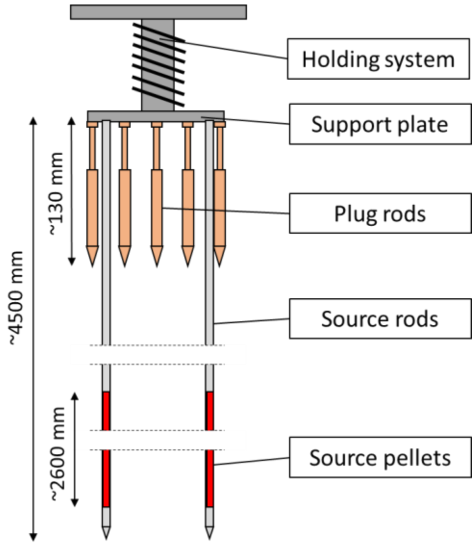

Figure 2 – Schematic View of a SSA.

Regulatory safety studies carried out by ORANO TN showed that two SSAs could be simultaneously transported, provided additional shielding content:

- Two dummy "fuel" assemblies (skeletons of fuel assemblies with lead pins instead of fuel pins), to hold the SSAs;
- Two specific aluminum wedges;
- Ten shielding bars made of aluminum and borated aluminum (Al-B_4C) rods stored in a steel tube.

SSAs, dummy assemblies, and wedges are positioned symmetrically in two central sockets. The ten other sockets are occupied by shielding bars.

The set composed of a SSA, a dummy assembly, and a wedge is represented in Figure 4.

CHARACTERIZATION, EVALUATION, AND MODELLING OF THE RADIATION SOURCES

Characterization

The working principle of a SSA is as follows:

1. The Sb/Be mix constituting the pellets is irradiated by the core neutron flux:

$$n + {}^{123}Sb \rightarrow {}^{124}Sb$$

2. ^{124}Sb decays into ^{124}Te ($T_{1/2}$ = 60.2 d), producing in particular 1.69 MeV (r = 0,488) and 2.09 MeV (r = 0,.056) gamma rays:

$$^{124}Sb \rightarrow {}^{124}Te + e^- + \gamma$$

3. 25 keV photo-neutrons are produced by gamma-Be interaction (threshold: 1.67 MeV).

$$\gamma + {}^{9}Be \rightarrow 2\alpha + n$$

In addition, lower energy gamma-rays are produced by the activation of the SSAs pellets and steel structure (cladding, pellet rod plugs, plug rods). Three types of radiation sources must then be considered:

- 25 keV neutrons emitted by the Sb/Be pellets;
- Gamma-rays emitted by the Sb/Be pellets;

- Gamma-rays emitted by the steel structure of the SSAs (source rod cladding, source rod plugs, plug rods).

Evaluation

The gamma-ray sources are estimated by simulating the activation of the pellets and structure using the evolution code DARWIN [2], [3]. To evaluate the neutron source, the gamma flux induced at the center of the pellets by ^{124}Sb 1.69 MeV and 2.09 MeV gamma rays (other rays are either under the 1.67 MeV (γ, n) ^{9}Be reaction-threshold or have a too weak intensity) is first estimated with TRIPOLI-4®. Then, this multi-group flux is multiplied by the (γ, n) ^{9}Be reaction cross-section.

Conservative hypotheses are made regarding the profile and intensity of the neutron flux irradiating the SSA and the time it spent in core considered in the DARWIN simulations. More realistic simulations would lead to reduce the neutron and gamma sources due to the Sb/Be pellets by a factor 2 to 3. Concerning the gamma sources due to the activation of the steel structure, this would lead to reduce the contribution of the source rod plugs and plug rods by a factor 10 to 100 (as these elements are in reality exposed to a much lower neutron flux).

The material compositions used in theses simulations are shown in Table 1. They are chosen so as to maximize the quantities of impurities leading to radioactive elements.

Table 1 – Composition of the SSAs Sb/Be Pellets and Structure.

Element	Mass Fraction (%)	
	Sb/Be pellets (d = 3.5 g/cm^3)	Structures (d = 7.74 g/cm3)
Sb	78.9	-
Be	20.1	-
Mg	0.02	-
O	0.39	-
Pb	0.04	-
Fe	0.10	59.72
Mn	0.04	1.86
Si	0.25	0.75
C	0.10	0.06
Al	0.05	0.09
Co	-	0.13
Cr	-	19.47
Ni	-	16.01
S	-	0.03
Cu	-	0.90
Mo	-	0.37
Nb	-	0.60
Ta	-	0.01

Total neutron and gamma source activities are summed-up in Table 2 for the different cooling times (time between the reactor shutdown and the beginning of the SSA transport) considered in the studies. The main contributing isotopes to the gamma sources are:

- Pellets: ^{124}Sb (70-90% depending on the cooling time);
- Structures: ^{55}Fe, ^{51}Cr, ^{60}Co, ^{54}Mn, ^{58}Co.

Table 2 – Total Neutron and Gamma Source Activities.

Cooling time (d)	Neutrons (n/s/SSA)	Gammas (γ/s/SSA)			
	Pellets	Pellets	Cladding	Source rod plugs	Plug rods
20	3.97E+10	5.20E+15	7.28E+13	8.00E+12	1.34E+14
45	2.97E+10	3.94E+15	6.16E+13	6.76E+12	1.13E+14
60	2.50E+10	3.35E+15	5.72E+13	6.28E+12	1.05E+14
120	1.25E+10	1.76E+15	4.68E+13	5.16E+12	8.62E+13
166	7.38E+09	1.10E+15	4.32E+13	4.74E+12	7.91E+13
180	6.28E+09	9.56E+14	4.24E+13	4.64E+12	7.78E+13

Modelling

Neutron sources are modelled by 25 keV neutrons distributed on the source rods following a distribution derived from an average spent fuel assembly irradiation profile. Gamma rays originating from the pellets are described by 18 energy groups and are distributed on the active rods following the same distribution as the neutron sources. Gamma rays originating from the structures are described by 10 energy groups and are uniformly distributed on each corresponding volume.

GEOMETRIC MODEL

Cask and basket

The TRIPOLI-4® model of the TN13/2 shipping cask and basket is taken from previous EDF R&D studies regarding spent fuel transport analyses. It is composed of almost 2000 cells in order to describe as well as reasonably possible the cask and basket. Only a few approximations are made. For instance, the set of resin plates constituting the neutron shielding are modelled as a cylinder and homogenized with the copper cooling fins. These approximations are validated and have a limited impact on the results.

Content of the basket

Figure 3 shows a schematic radial view of the basket, with two central sockets occupied with SSAs (red) fixed in dummy assemblies (black) and the other sockets occupied by shielding bars. Figure 4 shows an axial view of a set composed of a SSA, a dummy assembly, and a wedge.

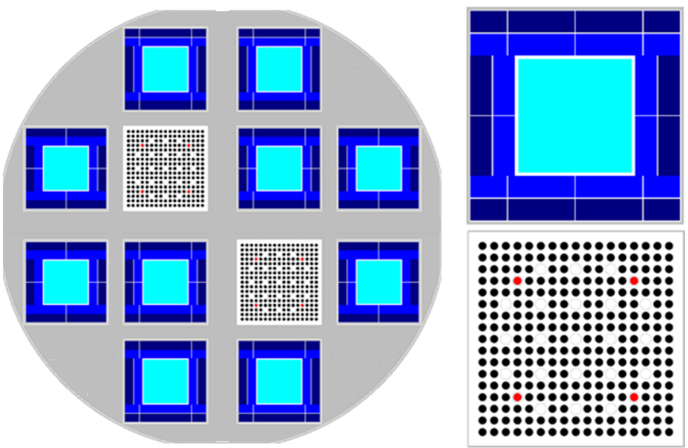

Figure 3 – Left: Schematic View of the Basket Content; Upper Right: Shielding Bar; Lower Right: Dummy Assembly (Black) and SSA Rods (Red).

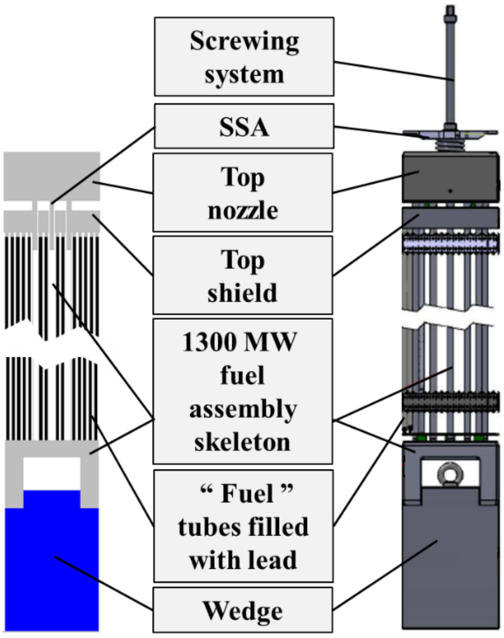

Figure 4 – Axial View of a Set Composed of a SSA, a Dummy Assembly and a Wedge (left: TRIPOLI-4® Visualization).

SIMULATIONS AND RESULTS

Dose equivalent rates (DER) H*(10) are computed in 11 zones surrounding the cask (Figure 5) corresponding to the actual measurement zones. In each zone, DER are computed at 0 m, 1 m, and 2 m from the cask surface. ICRP 74 [4] flux to dose conversion factors are used. The detector responses with respect to particle energy are not taken into account, except for the results presented in Table 7 and Table 8 which are compared to measurements.

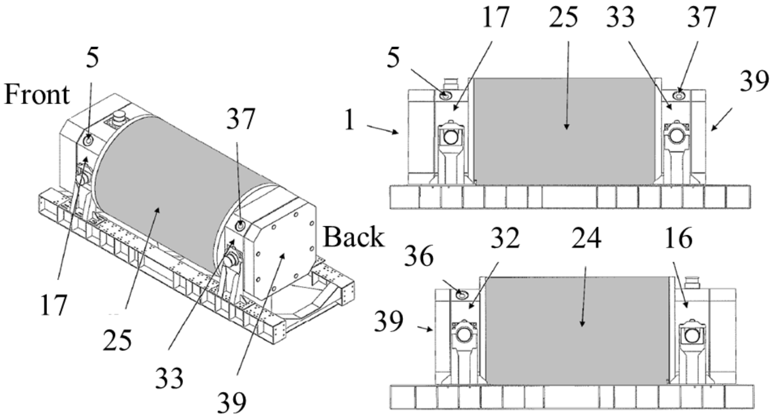

Figure 5 – Positions of the Measurement Zones.

Different shielding configurations are studied. They are named as follows in the next paragraphs and tables:

- "No shielding": each SSA is hold by an assembly skeleton without lead pins and a standard wedge used for spent fuel transport; the other ten sockets are empty.
- "Lead and wedges": each SSA is hold by a dummy assembly with lead pins and a specific aluminum wedge as described above; the other ten sockets are empty.
- "Lead, wedges and bars": each SSA is hold by a dummy assembly with lead pins and a specific aluminum wedge as described above; the other ten sockets are occupied by Al-B_4C shielding bars.

Effect of the shielding elements

Table 3 and Table 4 present respectively neutron and gamma DERs computed in 3 zones for 20-day cooled SSAs, for the 3 shielding configurations described above. The results show that neutron and gamma DERs are respectively reduced by up to 50 % and 90 % thanks to dummy assemblies with lead pins and aluminum wedges. The addition of Al-B_4C shielding bars allows to reduce neutron DER up to 92 % and gamma DER up to 98 %.

Table 3 – Neutron DER for Different Shielding Configurations, for 20-Day Cooled SSAs.

Zone	Distance	Neutron DER (µSv/h)			Difference wrt no shielding	
		No shielding	Lead and wedges	Lead, wedges and bars	Lead and wedges	Lead, wedges and bars
24	Contact	2.1	1.8	0.4	-16%	-83%
	1 m	1.3	0.9	0.1	-28%	-89%
	2 m	1.2	0.6	0.1	-47%	-88%
32	Contact	142.4	95.9	12.4	-33%	-91%
	1 m	14.2	8.6	1.2	-39%	-92%
	2 m	4.6	2.8	0.4	-39%	-91%
39	Contact	132.9	62.8	13.5	-53%	-90%
	1 m	35.2	17.1	3.1	-51%	-91%
	2 m	12.5	6.2	1.0	-50%	-92%

Table 4 – Gamma DER for Different Shielding Configuration, for 20-Day Cooled SSAs.

Zone	Distance	Gamma DER (µSv/h)			Difference wrt no shielding	
		No shielding	Lead and wedges	Lead, wedges and bars	Lead and wedges	Lead, wedges and bars
24	Contact	988.5	337.3	50.9	-66%	-95%
	1 m	459.1	152.8	24.9	-67%	-95%
	2 m	262.0	84.7	14.2	-68%	-95%
32	Contact	653.1	206.3	19.9	-68%	-97%
	1 m	258.7	86.5	11.4	-67%	-96%
	2 m	177.7	57.3	10.2	-68%	-94%
39	Contact	1200.1	177.7	36.2	-85%	-97%
	1 m	423.2	49.7	9.2	-88%	-98%
	2 m	198.6	21.3	4.1	-89%	-98%

Transportability of 2 SSA as a function of their cooling time

Table 5 shows the maximal DER (neutrons + gammas) for the 3 shielding configurations described above, as a function of the SSAs cooling time. Values in red are above the transport DER limitations (2 mSv/h on contact with the cask and 0.1 mSv/h at 2 m).

Results show that the transport DER limitations are respected without any additive shielding element if the SSAs are cooled at least 120 days. This value decreases to 20 days (the minimal cooling time for which simulations were carried-out) if dummy assemblies with lead pins are used, but with a very low margin. The addition of Al-B_4C shielding bars allow to transport 20-day cooled SSAs with comfortable margins.

Table 5 – Maximal DERs and Corresponding Zones for Different Configurations and Cooling Times. Cells are colored in dark-grey when no simulation was carried-out.

Cooling time (d)	Distance	No shielding		Lead and wedges		Lead, wedges and bars	
		DER (µSv/h)	Zone	DER (µSv/h)	Zone	DER (µSv/h)	Zone
20	Contact	1333	39	412	16	53	25
	1 m	460	25	156	25	31	5
	2 m	263	25	92	25	15	25
45	Contact	1003	39			40	25
	1 m	347	39			30	5
	2 m	197	25			12	25
60	Contact	786	39				
	1 m	264	39				
	2 m	138	25				
120	Contact	441	16				
	1 m	146	16				
	2 m	68	25				
180	Contact	430	16			33	16
	1 m	139	16			28	5
	2 m	62	16			10	5

Results in an operation configuration

Simulations were carried out in a representative configuration of an operation situation (loading or unloading). In this configuration, the shock absorbing covers, the protection lid and the fixing flange are removed from the cask. Dummy assemblies with lead pins, aluminum wedges and Al-B_4C shielding bars are present. The results are presented in Table 6 for zones 1 and 39 (axial front and back zones), which are the most impacted, for two cooling times.

The total DER are respectively 30 times and 4 times higher in zones 1 and 39 compared to the same loading in a transport situation for the two cooling times. In a close operation configuration, the same cask loaded with twelve 46 GWj/t spent fuel assemblies cooled for two years show respectively twice and 10 times higher DER in zones 1 and 39 than two 45-day cooled SSAs.

Table 6 – DER in Operation Configuration for Cooling Times of 45 d and 180 d.

Zone	Distance	DER (µSv/h)					
		Cooling time = 45 d			Cooling time = 180 d		
		Neutron	Gamma	Total	Neutron	Gamma	Total
1	Contact	0	183	183	0	170	170
	1 m	0	55	55	0	52	52
	2 m	0	24	24	0	22	22
39	Contact	108	52	160	23	11	34
	1 m	24	16	40	5	3	8
	2 m	9	7	15	2	1	3

Comparison with measurements

Table 7 and Table 8 show measurements done after loading two 166-day cooled SSAs in a TN13/2, compared with simulations, respectively on contact with the cask and 1 m from it. In order to reduce bias in the comparison, simulation results are corrected for detector responses (Canberra CRAMAL 31 for neutrons and Thermo FH 40 GL10 for gammas).

Given the conservative assumptions on the source terms evaluations (see corresponding section above), simulated DER are systematically above measurements. The highest differences are observed on gamma DER, for zones 1, 16, and 17. In these zones, the main contribution is due to the activation of source rod plugs and plug rods, which is highly overestimated (factor 10 to 100). The overestimation of the neutron and gamma sources due to the Sb/Be pellets activation (factor 2 to 3) is also coherent with the observed differences.

Table 7 – Measured and Simulated DER on Contact with a Cask Loaded with Two 166-Day Cooled SSAs.

Zone	Measurement		Simulation	
	Gamma	Neutron	Gamma	Neutron
1	0.2	0.0	6.9	0.0
16	0.8	0.0	34.1	0.0
17	0.6	0.0	25.0	0.0
24	4.3	0.0	10.9	0.2
25	2.5	0.0	11.1	0.2
32	3.0	2.5	5.4	7.1
33	1.5	2.5	4.2	5.8
39	4.5	2.5	11.0	5.5

Table 8 – Measured and Simulated DER at 1 m from a Cask Loaded with Two 166-Day Cooled SSAs.

Zone	Measurement		Simulation	
	Gamma	Neutron	Gamma	Neutron
1	0.25	0.0	2.2	0.0
16	0.2	0.0	18.3	0.0
17	0.2	0.0	13.8	0.0
24	1.5	0.0	5.5	0.1
25	2.0	0.0	5.6	0.1
32	0.9	0.0	2.8	0.6
33	0.7	0.0	2.3	0.6
39	1.0	0.0	2.4	1.3

CONCLUSIONS

These studies show that the transport of 2 SSAs respects the transport DER limitations without any additive shielding element if the SSAs are cooled at least 120 days after core shutdown. We demonstrate that using dummy fuel assemblies with lead pins and Al-B_4C shielding bars allow to transport 20-day cooled SSAs with comfortable margins regarding the DER limitations.

The simulation of an operation configuration also shows that DERs around the shipping cask are lower than those computed for a standard spent fuel evacuation in a close configuration.

Finally, DER measurements carried out during the transport of 2 SSAs towards PALUEL 2 are compatible with the simulation given the conservative assumptions on the source terms. By consolidating the safety demonstration and helping evaluate the operators' dose, this work contributed to successfully plan and carry out the project.

ACKNOWLEDGMENTS
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