Safety Analysis For Prototype MNSR HEU Core Unloading And Storage

Lu Jin[1], Wu Xiao-bo[1], Peng Dan[1], Hong Jing-yan[1], Li Yi-guo[1]

1)*China Institute of Atomic Energy, 275 mailbox,*Bejing ,China,102413, panlujin@sina.com

Corresponding author: panlujin@sina.com

Abstract. Prototype Miniature Neutron Source Reactor (MNSR), a low-power research reactor, was designed and fabricated by China Institute of Atomic Energy. It adopts HEU as fuel, beryllium as reflector, light water as moderator. The heat of reactor core is removed through natural circulation for cooling. There is one fuel cage with dimension of φ241×278mm in the reactor core, U-Al4 alloy a total of 376 rod fuel element. The activity of core is calculated by ORIGEN2 program according to the operation history of the prototype MNSR, and the temporary storage cask of the spent fuel (HEU) is designed according to calculating result of the core activity by MCNP program; The K_{eff} values are calculated at the different positions of spent fuel cage during the unloading of spent fuel cage with MCNP program and the accident critical safety analysis is carried out with the reactivity change during the unloading. Results show that The source term activity of reactor core fuel element is 5.22×10^{12} Bq after 12 months of shutdown, of which γ radioactive activity is 3.74×10^{12} Bq. K_{eff} value is less than 1 during the unloading of spent fuel (HEU) cage from the reactor core, which meets the requirements of radioactive safety. The inner dimension of the temporary cask isφ280×309mm, outer dimension isφ540×691mm, and the thickness of Pb is 90mm; the max value of γ rate on the cask surface is 0.54 mSv/h, which is less than 2 mSv/h.

1.Introduction of Prototype MNSR

Prototype Miniature Neutron Source Reactor (MNSR), a low-power research reactor, is independently designed and fabricated by China Institute of Atomic Energy. It adopts high-enriched uranium(HEU)as fuel and beryllium as reflector, light water a moderator. Moreover, the heat of reactor core is removed through natural circulation for cooling.

It is necessary to conduct low-enriched Uranium (LEU) conversion in order to prevent nuclear proliferation, which means converting HEU core to LEU core. Prototype MNSR LEU conversion only replaces the original reactor core without changing any other components. This paper introduces unloading and storage of prototype MNSR HEU core and related safety analysis.

2. Reactor Core Source Term Calculation

There is one fuel cage with dimension of φ241×278mm in the reactor core, composed of upper and lower grid plate, tie rods and fuel pins. 11 rings of fuel elements which are displayed in concentric circle with 10.48 mm gap between each ring, as shown in Fig.1. There are altogether 417 grid sites (411 fuel element sites, 5 tie rod sites and 1 control rod site).

There are 376 normal fuel pins and 35 depleted uranium pins. The fuel meat is uranium-aluminum alloy, of which ^{235}U enrichment is 90.3% and its density is 3.403g/cm^3 and the weight ratio of uranium is 27.63%.Fuel rod measures φ 5.0×270mm and fuel meat is sized at φ 4.0×250mm with 303-1 aluminum alloy cladding material. The space between upper and lower grid plate is 255mm. At the ends of fuel elements, the thicknesses of upper and lower plates are 3.5 and 5.0mm respectively. The distance from upper plate bottom surface to fuel meat central surface is 129mm on average. Fuel assembly weighs 7.5 kg.

Prototype MNSR reached full power on March 31, 1984 for first time and continued working till shutdown in March 1, 2014, totally in term of 7258 hours in full power. The integrated power is 1.95×105kWh and integrated neutron fluence is 2.61×10^{19}n/cm^2. Until March 2015, the reactor has been shut down for 12 months and ready for unloading. Due to of high photoneutron background from beryllium reflector, 3×10^8n/cm^2/s low neutron fluence rate remains in the reactor after shutdown.

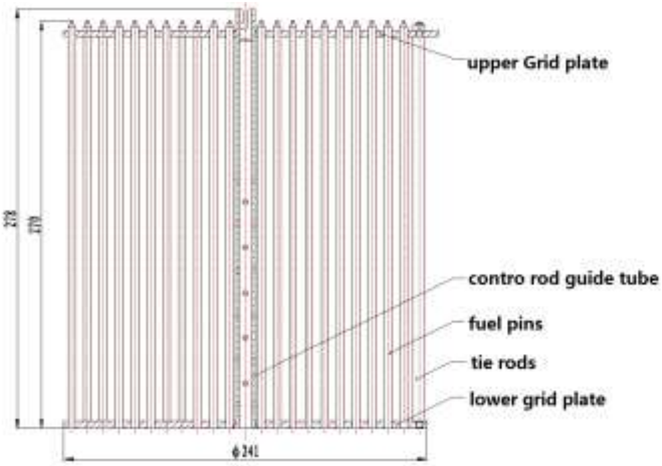

Fig 1. The structure of HEU core

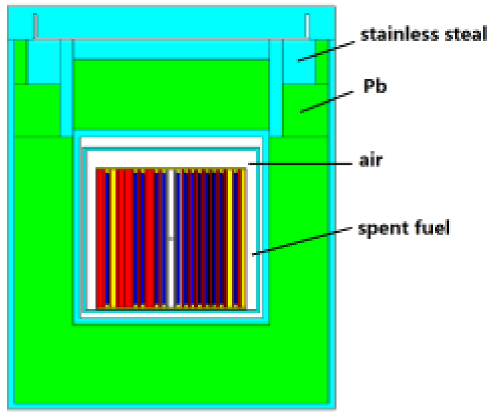

Fig 2. The structure of temporary cask

Source terms of prototype MNSR core fuel element are calculated with ORIGEN2 on a basis of included thermal neutron database. The source term activity of reactor core fuel element is 5.22×10^{12} Bq after 12 months of shutdown, of which γ radioactive activity is 3.74×10^{12} Bq. Radioactive activity of actinide is 1.34×10^{9} Bq and neutron source intensity of spent fuel assembly is 1.69 Bq. The mass of ^{235}U、^{236}U and ^{238}U are 903.6g, 1.912g, 98.16g respectively and decay heat is 8.1 watt.

2.1. Design of temporary cask

The spent fuel is unloaded from the reactor vessel and put into a spent fuel container, and then moved to Swing-pool reactor pool for storage. To meet the transportation requirement at home and abroad, the temporary cask is designed. Assuming that MNSR operates under normal conditions in full power for 2.5 hours per day in 40 years, the calculation results show that, the gross activity for 6 months after shutdown is 4.03×10^{13}Bq，the total photon source intensity is 2.872×10^{13}n/s, neutron releasing rate is 7.15n/s, decay heat releasing rate is 3.6W. Thus, ignoring neutron dose in shielding calculation is reasonable. The calculation model of container is shown in Fig 2 and the results of γ dose rates in the surface of the container and one or two meters away from the surface are shown in TABLE.1 with different lead thickness.

Tab 1. The calculation value of the γ dose rate

Lead thickness /cm	distance to surface /cm	dose rate /mSv/h	Lead thickness /cm	distance to surface /cm	dose rate /mSv/h
9	0	4.07E+00	10	2	2.68E-02
9	1	1.58E-01	11	0	1.16E+00
9	2	4.81E-02	11	1	5.04E-02
10	0	2.09E+00	11	2	1.53E-02

10	1	8.91E-02			

3. Safety Analysis of Unloading Procedure

3.1 Components Relevant to Reactivity

The grid holes of prototype MNSR HEU core in each ring are 6, 12, 19, 25, 32, 38, 44, 45, 57, 63, and 70 respectively. It contains 376 core elements and 35 depleted uranium rods. The dimension of the side Be reflector are 242mm in inner diameter, 440mm in outer diameter and 260mm in height, the bottom Be reflector is 340mm wide in diameter and 50mm deep with 20mm wide cavity in the central; The thicknesses of upper and lower grid plate are 4.5mm and 5.5mm respectively with 241mm diameter; the top beryllium shim dimension is 266mm in inner diameter, 270mm in outer diameter and 2.0mm in thickness; there are 4 inner irradiation channels and 5 outer irradiation channels and 4 regulator tubes in side beryllium reflector. Figure 3 shows radial cross section of reactor core. The total height of upper beryllium is 37.94mm with the reactivity of 10.04mk reactivity, the excess reactivity of the reactor is 3.10mk before the unloading of HEU core.

3.2. Unloading Procedure

Main procedure is as following.

1)Put 5 cadmium strings into 5 inner irradiation channels respectively (one cadmium string consists 3 cadmium capsule); put cadmium tubes into experiment channels;

2)Raise up central control rod;

3)Lift upper beryllium reflector;

4)Elevate HEU fuel assembly.

The total reactivity of 5 strings is -12.36mk(experimental value), reactivity for one cadmium tube is -0.33mk (experimental value) and the reactivity of central control rod is -7.0mk. The cadmium capsule dimension is 1.5cm in outer diameter, 0.1cm in thickness and 5cm in length. Moreover, the cadmium tubes dimension is 28mm in inner diameter, 30mm in outer diameter and 250mm in length. The cladding material of cadmium tube is stainless steel with dimension of 33 mm in outer diameter, 260mm in length. Therefore, the change of reactivity in reactor core is presented in Table 2 according to procedures 1-3.

TABLE.2 The Reactivity in Each Step

Operation	Reactivity/mk	State of reactor
Initial shutdown	-3.10	subcritical
Put into 5 cadmium strings	-15.46	subcritical
Put into Cadmium tubes	-15.79	subcritical
Lift up control rod	-8.79	subcritical
Lift up upper beryllium reflector	-18.83	subcritical

2.4. Accident Critical Safety Analysis of Prototype MNSR Unloading

The most possible accident is that that fuel assembly falls into the pool and forms new geometry arrangement therefrom during the unloading of HEU fuel assembly from core to transport cask. The analysis is done by MNCP program. In order to calculate maximum reactivity of core fuel element under water and in rearrangement of fuel element, a calculation model is built to evaluate the critical safety of equally spaced fuel element in various concentric circles under water. There are 376 fuel elements and 35 depleted uranium rods and 5 aluminum tie rods in calculation model.

Figure 4 shows the relation between K_{eff} and core concentric circles. It can be found from the figure that smaller concentric circles are joined with smaller K_{eff} and vice versa. When the space is 1.3cm, the maximum K_{eff} is 0.9086. After that K_{eff} decreases when the circle distance increase. Prototype MNSR fuel assembly stay away from criticality and remains in sub-critical state without beryllium reflectors.

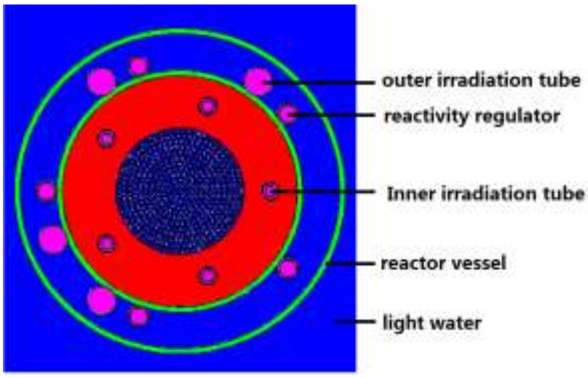

FIG 3. The components related to safety

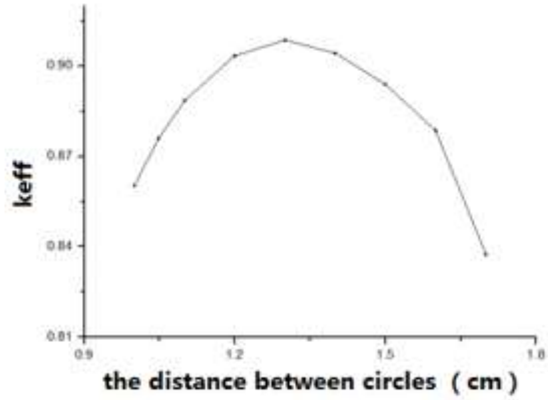

FIG 4. K_{eff} changes with the change of distance

3. Dose Calculation During Unloading Procedure

Spent fuel container is put into stainless steel bucket during unloading and the bucket is filled with deionized water. Then lift spent fuel into container and cover it. The bucket is used for shielding to reduce the dose on operators.

The dimension of the bucket is 130cm in diameter and 150cm in height, the water height is 140cm in bucket. the dose rates are calculated with different height of the bucket using MCNP program, the results are 6.27×10^{-2}mSv/h in the side sueface, 6.56×10^{-3}mSv/h at 1m from the side surface, 3.62×10^{-1} mSv/h on the top water surface of bucket, and 5.30×10^{-2} mSv/h at 1m from the top water surface of bucket.

4. Conclusions

Based on the analysis above , it shows that each procedure of unloading and storage of the HEU core, the reactor remains subcritical and has little influence on environment and work staff. In the accident situation, K_{eff} is far less then 1 even if the fuel assembly has the best geometry structure; the shielding container design of spent fuel meets the demands, thus the dose for operators would below the target value set for unloading. These calculation results and measured data could be provided as important references for other MNSRs unloading and storage.