Using SAFRAN Software to Assess Radiological Hazards from Dismantling of Tammuz-2 Reactor Core at Al-tuwaitha Nuclear Site

Mezher Abed Gatea¹, Anwar A Ahmed, Saad jundee kadhum, Hasan Mohammed Ali and Abbas Hussein Muheisn

¹Iraqi Decommissioning Directorate, Ministry of Science and Technology, Iraq

Abstract. The Safety Assessment Framework (SAFRAN) software has implemented here for radiological safety analysis; to verify that the dose acceptance criteria and safety goals are met with a high degree of confidence for dismantling of Tammuz-2 reactor core at Al-tuwaitha nuclear site. The activities characterizing, dismantling and packaging were practiced to manage the generated radioactive waste. Dose to the worker was considered an endpoint-scenario while dose to the public has neglected due to that Tammuz-2 facility is located in a restricted zone and 30m berm surrounded Al-tuwaitha site. Safety assessment for dismantling worker endpoint-scenario based on maximum external dose at component position level in the reactor pool and internal dose via airborne activity while, for characterizing and packaging worker endpoints scenarios have been done via external dose only because no evidence for airborne radioactivity hazards outside the reactor pool. The in-situ measurements approved that reactor core components are radiologically activated by Co-60 radioisotope. SAFRAN results showed that the maximum received dose for workers are (1.85, 0.64 and 1.3mSv/y) for activities dismantling, characterizing and packaging of reactor core components respectively. Hence, the radiological hazards remain below the low level hazard and within the acceptable annual dose for workers in radiation field.

1. Introduction

For facilities, decommissioning is the final phase in the life cycle after siting, design, construction and operation. It is a complex process involving operations such as detailed surveys, decontamination and dismantling of equipment and facilities, demolition of buildings and structures, and the management of the resulting radioactive and other hazardous waste and materials, while taking into account the need to provide for the health and safety of workers and the general public, and protection of the environment. A general requirement in decommissioning is the development of a decommissioning plan which includes, or has associated with it, an evaluation of the potential radiological consequences to the public and workers during planned decommissioning activities and as a result of any credible accidents that might occur during these activities [1]. Safety assessment can contribute directly to safety by identifying potential hazards and appropriate mitigatory measures that can be put in place to protect workers, the public and the environment. Safety assessments are used to show that facilities will comply or continue to comply with established safety principles, standards and licensing conditions [2]. Hence, it is a process which is required to evaluate the safety of radioactive waste management facilities and activities [3,4,5,6].
There are many sites in Iraq have some degree of radiological contamination and require decommissioning and remediation in order to ensure radiological safety. Many of these sites are located at the nuclear research center at Al-tuwaitha which suffered substantial physical damage during the first Gulf War and several have been subject to looting of materials and equipment as a consequence of the challenging security situation in the country [7]. Tammuz-2 reactor 'figure 1' core 'figure 2' is one of nuclear facilities located within Al-tuwaitha nuclear site which need decommissioning. It is swimming pool type reactor operates in a maximum thermal power of 500kW, used for training, neutron radiography and for research purposes. The reactor was totally destroyed in the 1991 Gulf War and has been de-fuelled [8]. The reactor pool is approximately at the center of the building, between levels (0 and – 7 m). It is parallelepiped in form, with (4m×4m) cross-section area, corresponding to a volume of (112 m³), made of a stainless steel liner, (6mm) thick. This liner is embedded in a reinforced concrete wall, providing the mechanical strength of structure and a biological shield. The pool block is fully separate from the floors at level (0) and (– 4m), to avoid transmission of a shock wave resulting from a borax type explosion in the pool block to the external walls.

A probabilistic safety assessment approach is provided in the (Safety Assessment Plan and report for the Decommissioning of Tammuz-2 Reactor).

On the other hand, a key parameter in any decision making process for selecting the appropriate measures is the distribution of individual doses to the population affected by the radioactive materials in the area. The inhalation of contaminated dust is often a major exposure pathway, and sometimes the associated doses cannot be measured, even though the contamination levels may be rather high. In such cases the doses should be estimated on the basis of model calculations, with input from the radiological monitoring programme and with realistic scenarios [9].

The Safety Assessment Framework (SAFRAN) software tool was implemented here for safety analysis [10]. It developed to apply the methodology developed within the Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) project. The International Atomic Energy Agency (IAEA) organized the International Project on SADRWMS to examine international approaches to safety assessment for predisposal management of all types of radioactive waste, including disused sources, small volumes of waste, legacy and decommissioning waste, operational waste, and large volume naturally occurring radioactive material residues. The initial outcome of the SADRWMS Project was achieved through the development of a series of flowcharts which were intended to improve the mechanisms for application of safety assessment methodologies for predisposal management of radioactive materials [11].

The deterministic safety analysis is used here to verify that the dose acceptance criteria and safety goals are met with a high degree of confidence for all works. A deterministic approach to safety assessment and the identification of safety control measures are recommended as being effective in providing adequate protection for workers and the public during decommissioning activities. However, probabilistic approaches can also be applied in a complementary manner.
It is important to ensure appropriate consideration of radioactive waste management in the development of safety assessment for decommissioning. For this purpose, it is essential to establish clear boundaries and interfaces between waste management and decommissioning activities and the scope of the associated safety assessments.

2. Materials and Methods

- MIP10 digital meter (Canberra Company) with STTC Geiger Muller probe for measuring high dose of Gamma radiation with range (0.3µSv/h-10Sv/h) 'figure (3-a)'.
- Radeye meter with two probes, the first is Scintillation detector probe (NaI crystal) (41S/MHV) model for measuring Gamma dose rate (unit µSv/h), and the second is ZnS Scintillation detector (DP6BD) model with 100cm² active area for measuring Alpha, Beta and Gamma contamination (unit Bq/cm²) 'figure (3-b)'.
- Interceptor (Thermo Company) consists of (2) CZT finder and identification detectors and (1) 3He neutron detector 'figure (3-c)'.
- Ludlum (type 3030) Alpha Beta radiation sample counter. It has radiation detector ZnS(Ag) adhered to plastic scintillation material with 0.4mg/cm² aluminized window 'figure (3-d)'.
- The particulates monitor LB9140 is used for measuring airborne Alpha/Beta particles in the presence of naturally occurring (Radon) activity and fluctuating Gamma backgrounds. Si-CAM detector unit 600mm² for simultaneous separated Alpha/Beta measurement on a flat dust collection area of 25x25mm². Air flow rate of approximately 3.3m³/h is possible 'figure (3-e)'.
- Scale of 1000kg maximum load used for weighing the radioactive dismantled segmented and forklift machine 'figure (3-f)' for lifting and transferring the segments into specified accumulation zone.

For portable radiological detection devices, a daily response check for each instrument is done by using standard radiation sources. Then it compares with the calibration certificates which supplied by the manufactured company for each instrument.

Safety assessment calculations have done by using SAFRAN (Safety Assessment Framework) tool version 2.3.2.7 software that incorporates the methodologies developed in SADRWMS (Safety Assessment Driven Radioactive Waste Management Solutions) project. SAFRAN calculations are based on the maximum external dose to the worker that comes from practicing to fulfill work activity. For dismantling activity, the maximum external dose is determined with accordance to worker position.
for dismantling each component which mentioned in the components description and the maximum doses mentioned in (table 1). For characterization activity, the maximum external dose for characterizing reactor core in step by step manner was the maximum dose at the top of reactor core while, the maximum external dose that taken from characterization of each component after removing outside the reactor core is the same dose that worker of packaging activity undergo (table 2).

SAFRAN calculations for internal doses which came from the inhalation pathway are based on the maximum Co-60 concentration in air during work. The results of particulates monitor LB9140 device showed that the maximum Co-60 concentrations in air at reactor core are (0.001Bq/m^3) but, there is no evidence to present concentrations outside the reactor core. Assessment for affecting of internal doses have been used for worker who charged in dismantling activity while, workers of characterizing and packaging activities are assess due to effect of external dose only with the judgment results of Co-60 concentrations inside and outside the reactor pool.

The assessments covered work took place over 91 day period which spent to fulfill dismantling and removal of Tamuz-2 reactor core. One type of endpoints was considered in exposure assessment scenarios. It refers to worker who contributed in dismantling and removal of reactor core. In this assessment the worker endpoint is defined as a cumulative endpoint in SAFRAN. The worst case is a generic worker who charged with different activities. The annual dose for this worker is then calculated as the sum of all exposures for all the mentioned activities. Dose to the public was neglected and is not numerically assessed in the SAFRAN file due to that Tamuz2 facility is located in a restricted zone far away from the public, relatively low level radioactivity for the affected area, no evidence for airborne radioactivity hazards outside the reactor core and 30m berm surrounded Altuwiatha site. Assessment for accident conditions, were also neglected because no accident occurs in all work activities. The main components in Tamuz-2 reactor core which was dismantled and removed are:

- Dummy are four experimental cells, made of net alloy AG-3, its form simulate the actual fuel cell, each one has cross-section area (8.4x8.4cm), length 80cm, thickness 5mm, contains four iron resistance rods.
- Upper chimney is nonradioactive item, made of stainless steel, 4.25m length, 5mm thickness, bolted with movable upper grid with 22 screws; it is positioned at level -4.25m measured from the top of reactor pool.
- Movable upper grid is nonradioactive material, made of net alloy AG-3, fixed at the top with upper chimney by 22 screws and fixed with AG-3 Aluminum chimney at the bottom by 22 screws. It is position at level -5m and -6m measured from the top of reactor pool.
- AG-3 Aluminum chimney was aligned with and extends the core grid (15cm) above the top surface of the pool water. The chimney assembly consists of two parts; a lower section made of AG-3 net alloy (thickness 2.5cm, height 80cm) and bolted to the core grid. An upper section made of stainless steel (thickness 5mm, height 4.2m). It was at -6m from the top of reactor pool.
- Zircloy sheets are four plates compose of solid AG-3 net alloy. Two of them have (120x62x3.35cm) dimensions and another have (120x71x3.35cm). Zircloy plates are positioned in core grid holes to provide shielding. It was at -6m from the top of reactor pool.
- Control rods are connected to their drive mechanisms by shafts located inside the control rod guide thimbles. The guide thimbles can be inserted into any core channel. Each drive mechanism was bolted to the control rod drive stand, which was perforated with a large number of holes to accommodate these bolts. The drive mechanisms can be placed in the position on the control rod drive stand required to insert the control rod in the core. Control rods are 6rods, 0.5cm thickness, (3.5"x3.5") section area, 1.65m length and 6kg weight for each one. It was at -6m from the top of reactor pool.
• Internal core grid is made of AG-3 net alloy, has (56) square channels measuring (3.5") on a side and the channels height was (1.1m). Thickness of the external wall measured at core level was (3.5mm). Channels were separated by (3mm) thick walls.

• External core grids are made of 150mm thicknesses net alloy AG-3 are designed to support the experiment rigs installed around the outside of the core. Two of them have dimensions (1.4x0.65m) and weighing 274kg while, another two have dimensions (2x0.65m), (0.7x0.6m) and weighing 378.4kg, 151kg respectively. It was at -6m from the top of reactor pool.

• Water box composes of a vertically-positioned, truncated cone made of AG-3 net alloy and weighing 700kg. This cone is formed by mechanically-welded elements and bolted to eight anchor studs adjustable by screws and located on the pool liner, for leveling vertical positioning. The lower part of the cone was closed by a plate, made of AG-3 net alloy (80mm) thick and perforated with (56) recessed holes, used to anchor support shafts for fuel elements or control rod guide thimbles. This plate positions the core assembly on the bottom of the pool and supports the weight of the core. It was at level -7m from the top of reactor pool.

The activities radiological characterization, dismantling and packaging were practiced to manage radioactive waste according to work time for each activity 28h/y, 225h/y and 18h/y respectively. Work time was 3h/day, 5day/week, 4week/month and 12month/year. The activities are:-

• Characterization is an initial step in the safety assessment process to provide a reliable database of information on quantity and type of radionuclides. It requires a logical approach in order to obtain the data necessary for planning a decommissioning program. The characterization program provides radiological information, which enables decisions on dismantling and removal of components and equipment. The characterization procedure has been done according to Multi-agency Radiation Survey and Site Investigation Manual (MARSSIM) [12]; standard approach for implementing the necessary radiological survey which derived by Nuclear Regulatory Commission, the U.S. Department of Energy, the U.S. Environmental Protection Agency and the U.S. Department of Defense and this guidance document used in the most of nuclear facilities under the decommissioning. The characterization program comprises review of historical information, in situ measurements, review and evaluation of the obtained data. The in situ measurement represents the flagship of the characterization program which involves dose rate measuring of reactor core and each segment, radionuclide identifying and taking swipe test for each segment to assure no loose contamination. The radiological characterization worker is positioned at the ground level 'figure 4' to measure the dose rate each 1m along the depth from the bottom to the top of reactor pool (before and after dismantling of each segment). The characterization was done by using Geiger Muller probe with cable detector 40m length. The average readings for each level were taken by measuring one and eight readings at the center and the surrounded area respectively. Then, each dismantled segment was obeyed characterization after removal 'figure 5' from the reactor core; to be adequate for packaging and transferring into specified accumulation zone.

Figure 4. Reactor core characterization
Figure 5. Component characterization
• Dismantling manners have been selected on the basis of previous experience on international decommissioning projects, taking into account availability of the dismantling technique and the low doses that existed inside the reactor core. Working area has been laid out and equipped in accordance with the applicable health and safety, fire protection and radiation protection requirements. Workers have provided with the industrial safety requirements like safety shoes, head covers, helmets, thick gloves and working suits. They also provided with radiation protection requirements like full face mask and personal radiation detection devices 'Figure 6'. The priority was given to dismantle and remove the Dummy from reactor core due to its relatively high dose (63mSv/h) which forms harmful to workers. Dummy removal is so important step and need to remove debris and dust from the reactor pool and core which generated from the destroyed of the reactor building during the second Gulf War in 1991. A simple hand tools were made to remove the rubbles while, compressed air system used to agitate the settled soil in reactor core and suck it up by a vacuum cleaner system to pack it exactly in the connected barrel and minimize the aerosols. The touched dose rate of the Dummy outer face is (63mSv/h) which need to attenuate into no more than (2mSv/h) [13], to be agreed with the waste acceptance criteria for the final waste receptor. Hence, a special container was designated to containerize the Dummy after removal. The manufactured Dummy container 'figure 7' is Carbon steel embedded by a pure lead layer of (12mm) thickness which cast in special ovens in the General Company for Batteries/Ministry of Industry to avoid formation of vocabularies during casting and assure that the touched dose rate of the container within the acceptable dose. Dismantling and containerize the four Dummy in the manufactured container has done by manufacturing a steel basket to hold the Dummy (through annular ring) during lifting it from its place into the container which placed away from the work area, to avoid the unjustifiable dose. Three times experimental lifting had made to train the worker man of crane to avoid accidents might be occurring. After Dummy removal, an encased steel structure by plastic glass was constructed above the reactor pool 'figure 8', to permit the light entering and protect the workers and the reactor pool from the weather variations. The structure also contains (2ton) overhead traveling crane to left and transfers the dismantled segments outside the reactor pool. Workers under control of radiation protection limitations and industrial safety requirements are dismantled components from top to the bottom of reactor pool. The dismantled worker was provided with the required hand tool and positioned at level facing the required segment in reactor core to unbolt the screws that fixed the segment 'figure 9' then, transferring it by the overhead crane outside the reactor pool.
Packaging of reactor core segments have done according to Waste Acceptance Criteria (WAC) forms that established by Radioactive Waste Treatment Directorate (RWTD)/Ministry of Science and Technology (MoST) which is responsible for the final waste status. It is important to note that WAC guidelines are agreed with IAEA-TECDOC-1515 [14]. Hence, each dismantled piece of reactor core was warpped in a thick nylon stratum and was stored in freight container by using suitable equipment like forklift machine to assure a safety distance for packaging worker and transferring each segment safely due to its relatively heavy weights 'figures 10, 11 and 12'.

![Figure 10. Packaging](image1)  ![Figure 11. Transferring](image2)  ![Figure 12. Store](image3)

Regulatory limitations which implemented here are (20mSv/y) maximum dose to the workers from all pathways; (0.4Bq/cm²) clearance levels for surface contamination of radioisotopes have β and γ emitters, (0.1Bq/g) clearance levels for Co-60 radioisotopes in bulk materials [15, 16, 17].

To assure that the received doses within the principal As Low As Reasonably Achievable, the main safety principles for protection from the ionization radiation have been implemented here like calculation and limitation of work time according to in process safety assessment, increasing the distance (as possible) between the worker and the radiation source, using suitable containers to containerize the dismantling components and only the number of the necessary required workers have been permitted to fulfill the target activity.

3. Results and Discussion

With accordance to laboratory analysis of swipes tests which were taken from each component, sides of reactor pool, ground of reactor pool and the scattered materials; there is no evidence for loose contamination. The in situ measurements (by portable detectors) approved that all components are radiologically activated by Co-60 radioisotope. The manner in process survey was followed by measuring dose rate in reactor pool after dismantling and removal of any component as indicated in (table 1). The dose rate for each component after dismantling and removal away from reactor core also measured and tabulated in (table 2) below.

| Height from the top of reactor pool (m) | -7 | -6.5 | -6 | -5 | -4 | -3 | -2 | -1 | 0 |
|--------------------------------------|----|------|----|----|----|----|----|----|----|
| Dose before Dummy removal            | 9900 | 63000 | 1900 | 79 | 60.3 | 47.8 | 21.6 | 17.9 | 13.6 |
| Dose after Dummy removal             | 39  | 40   | 40  | 37 | 4.5  | 3.3 | 2.6 | 0.7 | 0.5 |
| Dose after ampules removal           | 37  | 38   | 38  | 36 | 4.5  | 3.3 | 2.6 | 0.6 | 0.5 |
| Dose after AG-3 chimney removal      | 37  | 38   | 38  | 35 | 4.0  | 3.0 | 2.4 | 0.5 | 0.5 |
| Dose after Zircoloy sheets removal   | 75  | 80   | 78  | 65 | 2.6  | 0.85 | 0.7 | 0.6 | 0.3 |
| Dose after Control rods removal      | 66  | 72   | 68  | 45 | 1.7  | 0.73 | 0.5 | 0.5 | 0.3 |
| Dose after Internal core grid removal| 3.4 | 4    | 2.88 | 0.65 | 0.15 | B.G² | B.G | B.G | B.G |
| Dose after External grids removal    | 3.1 | 0.4  | 0.1 | B.G | B.G | B.G | B.G | B.G | B.G |
| Dose after water box removal         | B.G | B.G  | B.G | B.G | B.G | B.G | B.G | B.G | B.G |

²B.G = radiological Background = 0.09 µSv/h
Table 2. Dose rate for each component away from reactor pool.

| Pool component                                      | Contact Dose (µSv/h) |
|-----------------------------------------------------|----------------------|
| Dummy                                               | 63000                |
| Ampules                                             | 550                  |
| Upper chimney and Movable upper grid                | nonradioactive        |
| AG-3 Aluminum chimney                               | 1.21                 |
| Zircaloy sheets                                     | 79.8                 |
| Control rods                                        | 8.04                 |
| Internal core grid                                  | 63.5                 |
| External core grids                                 | 3.88                 |
| Water box                                           | 3.15                 |

Results of SAFRAN calculations are printed in 'figures 13, 14 and 15' for worker endpoint who charged at dismantling, characterizing and packaging activities respectively.

![Figure 13. Dismantling scenario](image-url)
Figure 14. Characterization scenario

Figure 15. Packaging scenario
Table 1 shows the in situ measurement for dose rate by held detectors which has done with step by step manner. The dose rate was taken each 1m from the bottom to the top of reactor core depth after removing of every component. We can also see that the dose rate decreases after every removing process of components but the dose increases in case of Zircoloy removal and this actual matter because Zircoloy sheets work as a biological shield for reactor core.

Table 2 represents the maximum touched dose for each component which measured after dismantling and removal of a component outside the reactor core.

'Figure 13' indicates the dose rate that effected to the dismantling group of reactor core equipment. The worst case does at dismantling and removal of control rods gloves stage. The received dose rate in this stage (1.2mSv/y) formed 65% from the total dose rate (1.85mSv/y) which was received from the overall dismantling process but, in this context the total received dose rate (1.85mSv/y) remained below the allowable dose rate (20mSv/y) and formed about (9.25%).

'Figure 14' represents the received dose rate to the field measurement worker during characterizing of reactor core and ultimately characterizing of the each segment after dismantling. In this stage, the total received dose rate (0.64mSv/y) formed (3.2%) of the permissible dose rate (20mSv/y).

'Figure 15' shows the received dose rate to the field pretreatment group. This stage comprises packaging and relocating of the dismantling reactor core segments. The total received dose rate was (1.3mSv/y) which formed (6.5%) from the permissible dose rate (20mSv/y).

From figures 13 and 14, the worst case is at characterizing and packaging of ampules which scattered at reactor pool ground. The dose is 0.28mSv/y for characterizing worker and 0.83mSv/y for packaging worker which formed 44% and 64% from the total received doses for each case.

4. Conclusion
A hazard identification and radiological risk assessment study was conducted for the implemented activities to fulfill dismantling and removal works of Tammuz-2 reactor pool and core. Safety assessment results proved that the radiological hazard for all endpoint-scenarios remain below the low level hazard and within the acceptable annul dose for the worker in radiation field. Assessment taking into account specific aspects like contact dose rates, concentration of contaminants in air. The measures which were identified in the safety assessment are elicited through detail characterization and formally laid down in operational procedures and work instructions.

The International Commission on Radiological Protection (ICRP) derives the limit of an average of (20mSv/y) over five years for the occupational dose limit and (1mSv/y) for the public dose limit. The maximum worker dose is (1.85, 0.64 and 1.3mSv/y) for activities dismantling, characterizing and packaging of reactor core components respectively which formed (9.25%, 3.2% and 6.5%) from annul permissible dose (20mSv/y). Dose to the public was neglected and no accident was mentioned during works. Thus, the implemented manners to complete decommissioning of Tammuz-2 reactor core are considered to be adequate for the associated radiological risks As Low As Reasonably Achievable (ALARA).

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