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DOSE ASSESSMENT FOR DECOMMISSIONING PLANNING OF THE GREEK RESEARCH REACTOR PRIMARY COOLING SYSTEM

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Abstract

Greek Research Reactor (GRR-1) is a light-water cooled and moderated heterogonous research reactor with a thermal output of 5 MW. The reactor is in extended shutdown for refurbishment since 2004. The refurbishment plan includes replacement of the Primary Cooling System. The decommissioning strategy consists of dismantling and removal of the bulky elements as whole pieces without preliminary segmentation and cutting and dismantling of the pipe works and PCS support structures. In this work the dose assessment and methodology in the context of planning for decommissioning of the PCS is presented.

Keywords

MTR type nuclear research reactor, decommissioning planning, primary cooling circuit, dismantling, dose assessment

1. Introduction

Greek Research Reactor (GRR-1) is an MTR pool-type, light water moderated and cooled reactor. GRR-1 first criticality was achieved on 1961. Since 1964, the reactor operated at thermal power of 1 MW. In 1971 the reactor fuel was changed from Low Enriched Uranium (LEU) to High Enriched Uranium (HEU) and the reactor power was upgraded to 5 MW. The upgrade works included replacement of the cooling system, replacement of the pool tile liner with stainless steel and installation of new power supply systems. The cooling system consisted of a primary and a secondary circuit equipped with heat exchangers and cooling towers. In 1990 beryllium reflectors were introduced. In the period 1999 to 2004 a gradual replacement of the HEU (93%) fuel elements by LEU (19.75%) elements was performed. The reactor was controlled by five rods composed of Ag-Cd-In alloy with composition 80%, 5% and 15%, respectively. GRR-1 was shut-down for refurbishment and modernization in 2004. A major refurbishment of the reactor building was completed in 2008.

One of the aims of the refurbishment plan is the replacement of the Primary Cooling System. The PCS decommissioning strategy comprise of dismantling and removal of the bulky elements as whole pieces without preliminary segmentation and cutting and dismantling of the pipe works and support structures. Similar decommissioning strategy has been followed by other nuclear research reactor facilities [1, 2]. The PCS parts to be decommissioned include the pumps, heat exchangers, demineralization units, delay tanks, instrument gauges, flanges, valves, pipe works and support structures. The radiological conditions in these systems were evaluated and appropriate decommissioning methods were derived. In the present work the dose assessment and methodology in the context of planning for decommissioning of the PCS is discussed.

2. System description

A detailed description of the PCS system can be found elsewhere [3]. Briefly, the PCS comprises of three sequential delay tanks, two centrifugal pumps, two heat exchangers, a water

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Table 1. Evaluated surface activity at the different parts of the PCS

PCS part	Surface activity (*Bq/cm²*)						
	^{137}Cs	^{60}Co	^{108m}Ag	^{152}Eu	^{110m}Ag	^{55}Fe	^{63}Ni
Pool Outlet	0.004	1.52	0.46	0.13	0.006	2.3	3.8
Outlet to Delay tanks	0.01	0.06	0.04	0.01	0.001	0.1	0.15
Delay tanks to Heat exchangers	0.01	0.06	0.04	0.01	0.001	0.1	0.15
Heat exchangers to pool	0.003	0.03	0.55	0.072	0.02	0.05	0.08
Delay tanks	0.004	1.50	0.40	0.10	0.006	2.2	5

treatment system, safety valves, gauging instruments and aluminum pipes. The diameter of aluminum pipe changes along the circuit from 6 to 10 inches. The coolant water flows through the core and then is led to an outlet pipe. The outlet pipe penetrates the pool leading to the delay tank system. At the outlet of the delay tank system the circuit is divided into two branches, each with its own pump, flow meters and heat exchanger. Coolant water returns to the pool through inlet pipes. Two butterfly safety valves are positioned close to the penetrations. Finally, heat is transferred to the environment through a secondary cooling circuit and two cooling towers. The main water purification system was composed of two "mixed bed" ion exchangers, using resins of regenerative type. Two auxiliary ion exchanger units were also used for purification of the water of the spent fuel storage and for the water used for replacing system losses.

All fuel assemblies, control rods, beryllium reflector blocks and active core supporting components (grid plate, plenum etc.) and irradiation devices have been removed from the pool and transferred to shielded interim storage structures at other locations within the facility [4]. The PCS system coolant water has been transferred to an adjacent storage tank and the reactor pool and piping system have been dried.

3. PCS radiological characterization

A radiological survey was performed aiming to evaluate of the nature, magnitude and extent of radioactive contamination of the different PCS components. The survey was performed in two stages. First a preliminary survey based on in-situ gamma ray spectrometry was performed [5]. The objectives of the preliminary survey were to identify potential gamma emitters, to confirm the impacted radiological status of the PCS and to provide information for planning detailed characterization activities. The preliminary survey was followed by a detailed radiological characterization campaign that was based on analysis of samples using laboratory techniques [6]. The objectives of the detailed radiological characterization campaign were to investigate for the presence of alpha and pure beta emitters; to extend the results of in-situ analysis by measuring gamma emitters that were below the minimum detectable activities of the in-situ technique and to provide further information for planning radiation protection measures taken during the decommissioning activities, as well as for the materials clearance procedures.

The results of the radiological characterization of the different PCS parts are shown in Table 1. The reported surface radioactivity value per radionuclide is the maximum value of total (fixed and removable) surface specific radioactivity evaluated for each part of the system. The activities of Ag-108m, Cs-137, Co-60, Eu-152 and Ag-110m activity were measured by gamma ray spectrometry, while the radioactivity of the pure beta emitters 55Fe and 63Ni were calculated on the basis of their ratios relative to Co-60. Both 55Fe and 63Ni are usually observed together with Co-60 as a result of corrosion of steel parts in reactor plants. The used isotopic vectors were 1.5 and 2.5 for 55Fe and 63Ni, respectively [7]. In summary, the results of the radiological survey showed that the main isotopes of concern were Co-60 and Ag-108m. The Co-60 activity was detected in localized areas of the system. For example, spots of Co-60 activity were found at the bottom of the heat exchangers and delay tanks as well as under the pool inlet. On the other hand, Ag-108m was the dominant and most frequently identified nuclide at all the other parts of the circuit. It has to be stressed that since the reactor was shut down in 2004 all short lived radioisotopes had decayed.

4. Dose estimates

4.1 External dose

In order to assess the external dose by direct radiation gamma dose rate measurements were performed in contact with the pipes along the PCS circuit. The results of the measurements were found to be within the fluctuation of the background measurement, in the range of 40 to 80 nSv/h. However, measurements of PCS components such as heat exchangers and delay tanks showed locally higher dose rates. In particular, the average measured dose rate over the upper part of heat exchanger I was (0.22 ± 0.01) μSv/h while a value of 300 μSv/h was observed at a specific spot below the component. Moreover, the average dose rate measured over the heat exchanger II was (0.27 ± 0.02) μSv/h while a dose rate of 3 μSv/h was measured at a spot below the component. The average external dose rate measured over the surfaces of the delay tank was (0.08 ± 0.02) μSv/h with a ´hot´ spot of 3 μSv/h at its bottom surface. All these ´hot´ spots were attributed to the presence of Co-60 from transferred activated steel corrosion products.

4.2 Internal dose due to airborne contamination

The dose estimation due to airborne contamination was performed for inhalation of re-suspended surface activity during the cutting of the PCS aluminum pipes and delay tanks. No cutting activities of contaminated materials are planned for the dismantling of the heat exchangers and demineralization units, at this stage of the project.

The committed effective dose from inhalation for the dismantling of the k-part was calculated using the following expression:

$$H_{inh,k} = \sum_i D_{inh,i}\, V\, R\, t_k\, A_{k,i} \qquad (1)$$

where, is the committed effective dose from inhalation for the decommissioning of the k part, $D_{inh,i}$ (μSv/Bq) is the conversion factor for radionuclide *i* for workers, conservatively taken for the slow metabolic model and 1 μm particles [8], V (m^3/h) is the breathing rate (1.5 m3/h for an adult performing light activities [9]), R is the re-suspension factor, $t_k(h)$ is the work time for part *k* and $A_{k,i}$(Bq/cm^2) is the surface activity for radionuclide *i* over part *k* of the circuit.

The factor R contains the re-suspended activity fraction as well as the ventilation information and is calculated as following:

$$R = \frac{f_A \cdot S_r}{V_r \cdot E_r} \qquad (2)$$

where, fA ((Bq/m)/Bq/cm2)) is the release per cut length normalized by the surface activity, is the segmenting rate, is the volume of the work hall and is the air exchange rate. The value of parameter R depends on the segmenting technique and the workplace conditions.

The pipe length, diameter, cut length and estimated cutting time for each part of the system is shown in Table 2. Based on the experience and tests performed by our mechanical workshop engineers, the estimated segmenting rate, Sr, of aluminum of 1 cm thickness for circular saw cutting is 10 m/h (~15 cm/min). However, during the actual PCS dismantling activities, the pipe segmenting rate value will be verified and if required revised, appropriately.

The exposure time for the cutting activities of the PCS pipes was derived from the assumption that the PCS piping is going to be segmented in 1 m long pieces. Based on this assumption and the pipes dimensions the corresponding cutting length is 103 m. The pipes cutting time is then derived to be equal to about 10 h based on the assumed segmentation rate of 10 m/h.

The exposure time for the cutting activities of the delay tanks was derived under the assumption that the tanks will be segmented in pieces of 1×1 m^2 pieces. Based on this assumption and the tank dimensions, the total cutting length for the three tanks was estimated to be 210 m. The tank cutting time is then derived to

Table 2. PCS parts, their length and measured surface activity

PCS part (line)	Pipe diam. (*in*)	Pipe Length (*m*)	Total Length (*m*)	Cut length (*m*)	Cut time (*h*)
Pool outlet	8	5	5	3.2	0.32
Outlet to Delay tanks	10/8/6	30.5/10.5/1.5	42.5	32	3.2
Delay tank to Heat exchangers	10/8/6/2/0.5	26/13/15/3/30	87	43.5	4.4
Heat exchangers to pool	10/8/6	7/25/5	37	24	2.4
PCS pipe total			171	103	10.3
Delay tanks				210	21

Table 3. Committed effective dose via inhalation (H_{inh}) during PCS cutting activities

Nuclide	D_{inh} (*Sv/Bq*)	H_{inh} (*μSv*)				
		Pool outlet	**Outlet to Delay tanks**	**Delay tank to Heat ex-changers**	**Heat ex-changers to pool**	**Delay tanks**
Cs-137	6.70E-09	4.5E-05	1.1E-03	1.7E-02	9.3E-03	3.0E-03
Co-60	2.90E-08	7.4E-02	2.9E-02	4.0E-02	2.2E-02	4.9E+00
Ag-108m	3.50E-08	2.7E-02	2.4E-02	2.1E-01	1.1E-01	1.8E+00
Eu-152	3.90E-08	8.5E-03	6.6E-03	0.0E+00	0.0E+00	5.6E-01
Ag-110m	1.20E-08	1.2E-04	2.0E-04	0.0E+00	0.0E+00	7.9E-03
Fe-55	9.20E-10	3.6E-03	1.5E-03	1.9E-03	1.0E-03	2.3E-01
Ni-63	5.20E-10	3.3E-03	1.3E-03	1.8E-03	9.8E-04	2.2E-01
Sum		0.12	0.06	0.27	0.15	7.7
Total						8.3

be equal to 21 h for the assumed segmenting rate of 10 m/h.

The release per cut length, f_A, value was derived by assuming that all activity corresponding to the zone defined by the width of the cut, taken to be equal to 3.5 mm, will be resuspended to the air. This scenario resulted in a value for f_A, of 35 (Bq/m)/ (Bq/cm^2).

The average volume of the working halls or temporary containment, V_r, was taken conservatively to be of 100 m^3. The actual cutting activities will take place in several working halls, along the PCS piping, of a larger room volume.

The air exchange rate in the room depends on the ventilation rate and on the total volume of air. For the PCS decommissioning activities a supplementary system will be employed with a ventilation rate of 1500 m^3/h. Based on the above data an air exchange rate, E_r, of better than 5 h^{-1} is expected. However, for the purposes of the present calculation a value of 1 h^{-1} for E_r was assumed, conservatively. Typical renewal rates for industrial conditions lie in a range 5 to 10 h^{-1} [10].

The calculated committed effective doses due to inhalation are given in Table 3, for the different radionuclides and PCS parts, along with the respective inhalation dose coefficients, D_{inh}. We note that the highest value (conservative approach) of the inhalation dose coefficients per radionuclide was used from reference [8].

From Table 3 it can be observed that the effective dose via inhalation to an individual performing the whole segmenting task, in 31 work-hours without utilization of respiratory equipment, is 8.3 μSv. The dose corresponding to the segmentation of the pipes and delay tanks was 0.6 μSv (10 h) and 7.7 μSv (21 h), respectively. We stress that the calculation of the committed effective dose was based on the maximum surface specific contamination va-

Table 4. Dose estimation by work activity

No	Task	Work Time (*h*)	Number of Exposed Individuals	Estimated External Dose Rate (*μSv/h*)	External dose per individual (*μSv*)	Internal dose per individual (*μSv*)	Collective Dose (*μSv*)
1	Planning	10	2	0.5	5	-	10
2	Radiological Survey						
	Pipes	5	2	0.1	0.5	-	1
	Heat exchanger 1	1.5	2	0.3	0.5	-	1
	Heat exchanger 2	1.5	2	0.3	0.5	-	1
	Delay tanks	5	2	3	15	-	30
	Water treatment	2	2	1	2	-	4
3	Decontamination						
	Pipes	10	2	0.1	1	-	2
	Delay tank	5	2	3	15	-	30
4	Dismantling						
	Pipes cutting, dismantling and transport	30	2	0.1	3	1.8	9.6
	Heat exchangers dismantling and transport	10	2	0.3	3	-	6
	Delay tanks cutting, dismantling and transport	63	2	3	189	23	424
	Water treatment	10	2	1	10	-	20
5	Radiation Protection	30	3	0.5	15	8	69
6	Activity Control	10	2	0.5	5	-	10
7	Waste Management	20	3	1	60	-	60
8	Final Surveys	10	3	0.1	1	-	3
9	Reporting and Verification	-	-	-	-	-	-
	Total	**223**			**325.5**	**32.8**	**680.6**

lues measured per each PCS region and conservatively estimated duration of work.

In order to take into account the presence of the other workers close to the cutting area performing dismantling and transfer tasks, the working time was multiplied by a factor of 3 that is 93 work-hours which results to an internal dose of 25 μSv. The dose corresponding to the segmentation of the pipes and delay tanks is 1.8 μSv in 10 h and 23 μSv in 21 h, respectively.

4.3 Collective dose

Table 4 shows the dose estimates by task. The estimation did not include cutting of heat exchangers or primary coolant water treatment system, since these components will be transferred to the interim storage intact. The collective radiation dose to complete the task of decommissioning of the GRR-1 PCS was estimated to be 680 μSv for a collective working time of 223 h. The work crew involved in the project was estimated to consist of 15 persons. The average estimated effective dose per individual is 45.4 μSv. Since large variation in individual doses is expected this average dose estimate could be tripled and therefore, the estimated maximum dose to an individual could be assumed to be of about 150 μSv. This dose is lower, by a factor of 40, than the administrative limit of GRR-1 which is set at 6 mSv.

Despite the low estimated collective dose, of 680 μSv, corresponding to the whole PCS decommissioning activity it is of interest to note that about 60 % of this dose is due to work related to the delay tanks.

5. Discussion and Conclusions

The basis for the dose assessment before the decommissioning was the thorough radiological characterization of the PCS. In situ measurements of gamma dose rates, laboratory measurements using gamma ray spectrometry of shavings samples, and referenced isotopic vectors were employed to evaluate the internal and external exposure levels. The operational experience was taken into consideration to estimate the man power requirements and duration and working times. With the above information the dose estimation on a realistic basis was enabled.

The results of the dose estimations for the decommissioning planning of the GRR-1 primary cooling system have shown that the works can be performed for a collective dose of less than 1 mSv. This low dose may be attributed to the following project´s features: a) the reactor has been shut down since about 10 years and therefore all short lived radioisotopes that were present in the circuit have been decayed, b) the active components have been removed from the pool to special interim storage facilities and c) the decommissioning strategy consists of dismantling and removal , without preliminary segmentation, of the bulky elements as whole piecesand cutting/ dismantling of only the pipes and PCS support structures.It has to be stressed that the dose estimations presented in this work were performed for planning purposes. During the actual decommissioning work the exposure estimates and control will be subject to continuous revision and updates, in accordance to the requirements of the radiation protection program. Revisions will be based on the measurements of radiation levels and durations of work during performing the different tasks. The activity work control will follow a formal plan including procedures and engineering controls that reduce the exposures as low as reasonably achievable. In this context, procedural as well as engineering controls will be the preferred methods for maintaining exposures to radiation and radioactive materials at minimum possible levels. These controls will include the following, subject to applicability: control of access to the radioactivity areas, remote handling, reduction of exposure times, increase of distance between the individual and the source, use of Personnel Protective Equipment (PPE) including respiratory protection, containment or confinement structures for radioactive materials, controlled ventilation, radiation shielding, training and review/ approval of procedures.

The presented data are specific for the radiological conditions encountered in the GRR-1 facility. Nevertheless, it may also provide valuable information assisting the decommissioning planning of other MTR type research reactors.

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