ALARA IMPLAMANTATION PRINCIPLES AT THE DESIGN AND DECOMMISSIONING STAGES AT ANPP

Aida Avetisyan,
Armenian Nuclear Regulatory Authority
4 Tigran Metz, Yerevan, Armenia
a.avetissyan@anra.am

Vovik Atoyan,
Armenian NPP, Medzamor, Armenia
anpp@anpp.am

The Armenian Nuclear Power Plant (ANPP), the only nuclear power plant in the region, consists of two VVER/440/270 units (that is a modified, seismic design VVER/440/230). Unit 1 started its commercial operation in 1976 and Unit 2 in 1980. Both units were shouted down shortly after the 1988 Spitak earthquake. Re-commissioning works were performed from 1993 to 1995 and in November 1995 Unit 2 restarted operation. At this moment the ANPP Unit N1 is in conservation regime (long-term shut down). The new unit construction of ANPP is approved by Government of Armenia and the siting stage has been started.

The Armenian regulations on nuclear and radiation safety clearly defined the ALARA implementation at the design and decommissioning stages.

RP at The Design Stage of Nuclear Power Plants

The regulations stated that during designing of the protective shields against external exposure to personnel and population the safety factor equal to 2 should be accepted when estimating the annual effective dose. It is necessary also to take into consideration the availability of other radiation sources and further increase of their capacity. The design of protective shields against external exposure should take into account purpose of premises, category of exposed persons and duration of exposure.

For calculation of the authorized discharges and releases from atomic energy utilization installations intended for practices with radiation sources it is accepted that the effective dose from annual releases and discharges during lifetime (70 years) does not exceed the dose constrain value. To ensure radiation protection, NPP design shall identify all real and potential sources of ionizing radiation and shall provide measures for ensuring the necessary technical and administrative control over their use.

The requirements with regard to the classification of zones and compartments, radiation monitoring, the individual protection means and the access control are established by a different regulation.

To keep the exposure of personnel and public as low as reasonably achievable during plant operation, the design of the reactor coolant system shall arrange for:

1. use of structural materials with minimum content of chemical elements with high activation cross-section and producing long-living radioactive corrosion products;
2. coolant purification from fission and corrosion products;
3. water chemistry control;
4. minimum length of the pipelines with a minimum number of isolation valves and connections;
5. leak-tightness testing of operating components;
6. decontamination of SSCs outer and inner surfaces;
7. prevention of uncontrolled radioactive leaks in the NPP premises.

The layout of the plant, its buildings and SSCs shall facilitate the operation,
inspections, maintenance, repair and replacement of systems and components and shall limit the personnel exposure to ionizing radiation.

The buildings, compartments and components, which may be contaminated with radioactive substances, shall be designed in a way that allows easy decontamination by chemical or mechanical means.

The personnel access to compartments of high contamination level shall be controlled by means of locking devices with interlocks and indication for actuation and unavailability.
Biological protection shall be designed in a conservative way, taking into account the build-up of radio nuclides over the plant lifetime, the potential loss of shielding efficiency due to effects of interactions of neutron and gamma rays with the shielding, due to reactions with other materials, decontamination solution, and the expected temperature conditions in design basis accidents.

The choice of materials for the shield shall be made on the basis of the nature of the radiation, the shielding, mechanical and other properties of materials and space limitations.

Ventilation systems shall be installed to:

1. prevent spreading of gaseous radioactive substances in plant compartments;
2. reduce and maintain compartments' airborne concentrations below the established limits and as low as reasonably achievable in all operational states and design basis accidents;
3. cleanup the air in premises containing inert or harmful gases.

In designing a ventilation system, the following factors shall be taken into account:

1. mechanisms of thermal and mechanical mixing;
2. limited effectiveness of dilution in reducing airborne contamination;
3. exhausting of the air from areas of potential contamination at points near the source of contamination;
4. ensuring adequate distance between exhaust air discharge point and the intake point;
5. providing a higher pressure in the less contaminated zones in comparison with the zones of higher contamination level;
6. preventing the spread of fire-released smoke products to neighboring compartments.

Design shall provide for ventilation and air cleaning systems before discharge of gaseous radioactive substances to the environment.

Filters of air cleaning systems shall be sufficiently reliable to perform their function with the necessary decontamination factor in all operational modes. The design shall provide means to test their efficiency.

Provisions shall be made in the design for an automated system for radiation monitoring at the workplace and at the NPP site, and a system for radiation monitoring at the radiation protection and the monitored areas. These systems shall ensure the collection and processing of information on the radiation conditions, on the effectiveness of protective barriers, on the radionuclide activity, and information necessary to predict changes in the radiation conditions in all operational states and accident conditions.

The equipment of the automated system for radiation monitoring shall enable the implementation of:

1. radiation monitoring of technological environment
2. individual monitoring;
3. radiation monitoring at the workplace and at the NPP site;
4. area monitoring for limiting the spread of radioactive contamination.

The system of radiation monitoring shall be able to process and archive the data.

The laboratory methods and technical means of the system for radiation monitoring at the radiation protection and monitored areas shall ensure measurement of the content of human induced radio nuclides in soil, water, deposits, vegetation, water flora and fauna, and agricultural products.

The Regulatory Body shall review the submitted documents based on these statements.

Radiation Protection and Decommissioning

The main approach for this stage is that all Radiation safety requirements for decommissioning of atomic energy utilization installations(including nuclear installations) should be provided in accordance with the requirements of the radiation safety regulations after complex radiation and technical studies of technological systems, equipment, building structures and adjacent territory of installation.

The main regulations for nuclear and radiation safety, the safety requirements, especially radiation protection requirements during preparation of Decommissioning plan of ANPP are the follows:

- The decommissioning plan for shall include safety measures for different phases of decommissioning; shutdown, conservation, dismantling, re-use and disposal.
- Decommissioning program particularly shall include the time period and volume of works for the preliminary complex engineering – radiation investigations.
- Assessment of the total activity of spent fuel and the activity of separate radionuclide
- Assessment of the radioactivity of primary circuits and radioactivity of the main facilities, equipments, pipelines and engineering construction (both the activated and the surface contamination)
- Assessment of radiation contamination in premises of ANPP and the prognoses of changes in timeframe - up to design end time and final shutdown.
- The amount/volume activity of liquid and solid radioactive wastes (including radioactive bulk materials), the available treatment methodology and the characteristics of available storage and/ or disposal facilities.
- Assessment of the amount/volume of non radioactive wastes (chemical, construction materials and others)
- Evaluation of the radiation situation at the beginning of decommissioning and the prognoses during each stages
- Assessment of projected /expected collective doses of personnel and public in each stages based on the radiation characteristics
- The real and expected liquid and airborne discharges
- Assessment of accrued unusual accidental situation during the commissioning of ANPP and their radiological consequences.

Based on this requirements the ANPP has prepared the first stage of radiological characterization of ANPP.

Radiological Characterization in Support of ANPP Decommissioning Scope and Objectives

The objective of radiological characterization is so provide a reliable database of information on quantity and type of radionuclides, their distribution and their physical and chemical states. Characterization involves a survey of existing data, calculations, in situ measurements and/or sampling and analyses. Using this database the decommissioning planner may assess various options and their consequences, considering:

- operating techniques: decontamination processes, dismantling procedures (hands on, semi-remote or fully remote working) and tools required;
- radiological protection of workers, general public and environment;
- waste classification;

- resulting costs.

Comparison and optimization of these factors will lead to the optimization of the process as a whole.

Data obtained during radiological characterization are extremely important for the work planning and corresponding decision making, taking into account that the decommissioning, as a rule, is a stage-by-stage process and each subsequent stage is planned and realized on the basis of the information received at earlier stages.

The measurement and calculation methods used to estimate radiation conditions and radioactivity levels are different depending on the specific objectives. Before and during the NPP decommissioning, various surveys should be carried out aimed at radiological characterization of the facility/site. For example, radiological characterization of the power plant systems and equipment includes:

- Calculation of the of neutron induced activity in the pressure vessel and the concrete shielding structures, confirmed by experimental studies;
- In-situ measurements.

As a rule, the last item includes measurement of gamma dose rate □in premises. The purpose is classification of the premises on the basis of their dose rates and, consequently, their potential contamination.

- Measurement of radioactive contamination of equipment/components both by using in-situ instruments and by sampling and analysis;

During normal operation the measurement of gamma dose rate are aimed, mainly, at:

- minimization and optimization of the personnel dose loads.
- monitoring of the conditions in the systems, components and fuel integrity to prevent the radioactivity release into the environment.

The purpose of dose rate measurements during decommissioning depends on specific stage of the decommissioning. As a rule, the dose rate measurements are carried out at the following stages:

a) Planning and preparation

The objective is a preliminary evaluation of the radiation conditions at NPP and definition of problematic premises / zones. Data received at this initial stage serve for initial classification of the surveyed premises and/or rooms. Preparation of the initial dose rate gamma maps is aimed mainly at estimation of potential radioactive contamination and provides with important information, thus allowing avoiding an unreasonable exposure of the personnel during the subsequent activities.

b) Post-operational stage

More careful measurements of gamma dose rate should be performed immediately after the final shutdown and unloading of the fuel. The information collected at this stage can form a basis for further works and the design decisions concerning:

- Development of personnel protection and safety programs.
- Decision-making on what systems should be decontaminated and/or if the contamination should be fixed.
- Decision-making on what premises need an additional analysis aimed at determining the surface contamination and/or fixing of contamination.
- What components need additional shielding to reduce the personnel dose loads.

c) Safe enclosure stage

At this stage the program of radiation monitoring is carried out aimed mainly at the monitoring of the integrity of barriers to prevent potential deterioration of their condition leading to the radiological consequences.

d) Decontamination and dismantling stage

After completion of the previous stage, dismantling stage follows during which the necessary radiological characterization is to be carried out aimed mainly at specifying the additional shielding to prevent high dose loads on the personnel.

e) After completion of decommissioning

At this stage the survey activities also should be carried out to confirm the fact that the radiation conditions at the site conform to the specified requirements for the site release from the regulatory control.
Taking into account that the process is as a whole an iterative one, it is generally accepted that the corresponding database should be created and maintained during whole the decommissioning process as an important source of information.

Radioactive Contamination of Equipment/Components

Characterization of equipment provides a basis for decision-making - what to do with specific equipment during NPP decommissioning, i.e. either decontaminate up to levels enabling releasing from regulatory control (if possible) or dispose off as the waste.

Data obtained at radiological characterization of equipment/components, are extremely important for scheduling of decommissioning activities and making corresponding decisions taking into account that decommissioning is, as a rule, is a stepwise process, and each subsequent stage is planned and implemented on the basis of the information obtained at earlier stages.

The activities on the measurement of the dose rates from the equipment and piping of the primary circuit of both units were performed, as well as the measurements of activity and the nuclide composition of the radioactive deposits. These activities were aimed at finding correlation between the γ-dose rates and the activity of the deposits on the internal surface of the equipment and piping. The data obtained were analyzed. The verification of the data by using smear samples was performed.. The data on dose rate and surface contamination measurements are given below in the Table 1,2,3,4,5.
Dose rates from the some equipment of Unit 2.

Table 1

Data on dose rate measurements

Equipment	Dose rate(microSv/sec)
Main circulating pump -6	1,0 – 1,6
Control rod	0,05 - 1,6
The block of the protective pipes	0,5 - 0,6
Main circulating pump sealing unit	0,03 - 0,15

The smear samples were taken from the internal surface of the equipment/piping. The data are given in the Table 2.

Table 2

The radionuclide contents on the internal surface of the equipment

	SG-2, cold header	SG-2, hot header,	Control rod	Upper unit

Nuclide	activity Bq/cm^2	activity Bq/cm^2	activity, Bq/cm^2	activity, Bq/cm^2
Mn-54	6,3E3	1,4E4	3,0E4	2,2E4
Co-58	1,2E4	2,4E4	2,3E4	2,8E4
Fe-59	1,5E3	3,0E3	8,0E3	2,9E3
Co-60	3,5E4	5,3E4	1,2E5	1,1E5
Zn-65	2,6E3	3,5E3	7,5E3	5,5E3
Nb-95	-	-	4,8E4	5,6E3
Ag-110m	2,5E4	2,7E4	1,0E4	9,6E2
Cs-137	1,2E3	1,4E3	3,5E3	3,4E3

The measurements of radionuclide composition and activity of surface contamination for the primary circuit equipment and piping of Unit 1 were performed. The measurements were carried out by using "Canberra" portable gamma-spectrometer equipped with a laser guidance device allowing performing the remote measurements (ISOCS system). The major contaminants of the primary circuit equipment and piping of Unit 1 are ^{60}Co and ^{137}Cs. In addition to the identified nuclides ^{54}Mn, ^{110m}Ag and ^{51}Cr were identified in equipment and piping of the Unit 2. The preliminary data on the specific activity of the surface contamination of the equipment and piping had been obtained.

Analyzing the data obtained it was found that to receive the more accurate information it is necessary to bring into compliance the templates used in the computing codes with the real layout of the piping and equipment. In order to verify the calculated data it was planned to perform the sampling and analysis of the deposits from the piping and equipment of the primary circuit.
The measurements results for the surface contamination of the steam generator (SG 5, Unit 1) headers (both "hot" and "cold") are given below in the Table 3.

Table 3
Measurements results for the surface contamination of the steam generator (SG 5, Unit 1)

Nuclide	Surface activity of the steam generator headers, SG5, Unit 1, 5x10^3 Bq/cm2	
	Hot header	Cold header
Mn-54	1,4	0,6
Co-60	18,3	2,1
Ag-110-m	0,4	1,9
Cs-134	0,6	9,6
Cs-137	0,16	29,5

The measurements results for the surface contamination of the Units 1 and 2 are given below in the Tables 4-5.

Table 4
The measurements results for the surface contamination of the some primary circuit equipment and piping of the Unit 1 [kBq/cm^2]

nuclide	Measurement points								
	1K2	2K2	3K2	4K2	5K2	6K2	7K2	8K2	9K2
Cs-137	1,74	1,16x10^{-2}	5,2	3,41	4,48	1,93	2,58	0,23	11,4
Cs-134				4,92		9,25	5,75	0,14	3,06
Ag-110m	38,2		128,8	18,19	8,04	9,3	18,6	0,67	292
Co-60	154	1,2	91,07	77,07	47,1	27,7	40	1,86	205,4
Mn-54	41,5			13,3	9,0	13,3	12,2	0,33	181,2
Zn-65			55,6		11,22	15,6	1,48		

1K2 The cold leg, straight part of the pipe- bend ∅500* to steam generator-2, pipeline without the heat- insulating cover.
2K2 The cold leg, pipe bend ∅500 of the steam generator-2.
3K2 The cold leg, direct part of the pipeline ∅500.
4K2 The cold leg, the part of the pipeline ∅500 between Main Circulating Pump-2 and Main Isolation Valve. Pipeline without the heat- insulating protection
5K2 Cold leg, pipeline ∅500 junction to the Main Isolation Valve
6K2 The hot leg, pipeline ∅500, the part between Main Isolation Valve and reactor outlet.
7K2 Hot leg, inlet of the pipeline ∅500 into the Main Isolation Valve. The pipeline without the heat-insulating protection
8K2 The cold leg, junction of ∅500 pipeline with the Main Circulation Pump-2.
9K2 The part of junction of ∅500 pipeline to Main Circulation Pump -1, main circulating, pipeline with heat-insulating protection.

Table 5

The measurements results for the surface contamination of the some primary circuit equipment and piping of the Unit 2 [kBq/cm^2]

Points of measurement	Nuclide, activity (kBq/cm^2)			
	Cs-137	Cs-134	Ag-110m	Co-60
The center of the pipe bend of the pipeline ∅-500 (cold leg)	0,0152	--	0,31	0,162
The beginning of the bend of the pipeline ∅-500	0,022	0,1	0,27	0,15
The pipeline ∅-500 (cold leg), the outlet of main circulating pump	0,027	--	--	0,069
The bottom part of the pipeline ∅-500 (inlet to Main Isolation Valve).	0,0116	--	--	0,13
The inlet of the pipeline ∅-500 to Main Isolation Valve, (cold leg)	0,0146	--	--	0,125
The inlet of the pipeline ∅-500 in Main Circulating Pump-6 (cold leg)	0,0157	--	--	0,113
The inlet of the pipeline ∅-500 in steam generator-6 (cold leg)	0,0039	--	--	0,27
The outlet of the pipeline ∅-500 from reactor (hot leg)	0,0082	--	--	0,092
Connection of the pipeline ∅-500 with the pipeline ∅-300 (hot leg)	0,014	--	--	0,18
The bend of the pipeline ∅-500 (hot leg)	0,02	--	--	0,17
The pipeline ∅-500 (hot leg) near steam generator -6.	0,017	--	--	0,18
The pipeline ∅-500 (hot leg), the beginning of the pipe bend under steam generator -6.	0,0095	--	--	0,14
The pipeline ∅-500 (hot leg), the middle of the pipe bend under the steam generator -6.	0,016	--	0,12	0,099
The connection of the pipeline ∅-500 to the pipeline ∅-300.	0,0229	--	--	0,118
The wall of reactor protection.	0,0138	--	--	0,031

* ∅-500 – diameter 500 mm

The corresponding database had been compiled

Radioactive Contamination of Construction Materials and Protective Structures

Radiation inspection of construction materials and protective structures was carried out in premises of 1st and 2nd units and special building.

Research includes definition of radionuclide composition of radiation sources on the basis of samples of cores drilled in protective structures, as well as layer-by-layer definition of specific activity of protective materials and measurement of the depth.

Measurement of samples was carried out using a gamma spectrometer facility with pure Ge detector and software GENIE 2000, as well as UMF-1500 low background device. The results of sample analysis have shown that activity of samples varies very substantially both within the boundaries of one premise and in each surveyed unit in whole.

Results for the samples of paint, plastic compounds, cast floor, cement covering and concrete have shown that the basic radionuclide containing in the contaminated materials are ^{137}Cs; ^{134}Cs; ^{60}Co.

The study of contamination penetration to cement covering and concrete was carried out by means of core sampling (diameter 25 mm, depth up to 30 mm) using a special tool. The core samples taken were cut into layers by using a mini concrete cutting machine. Then the measurement of each 2 mm thick core layer was carried out. Activity of the contaminated concrete is basically determined by ^{137}Cs, ^{134}Cs и ^{60}Co radio nuclides.

It should be noted that more than 90 % of activity are concentrated in the first 8мм of concrete structure. In Table 6 below are presents the data on specific activity of the contaminated materials:

Table 6

Activity of various materials in the premises of controlled area.

Paint	plaster	Plastics	Underlayment (floor)	self-leveling floor	Concrete (wall)
5-100 Bq/g	2-100 Bq/g	2-1200 Bq/g	4-1200 Bq/g	1-100 Bq/g	2-200 Bq/g

In many places the cast floor of premises has distressed up to underlayment. Activity of such samples is considerably above the average level specified in the above Table and reaches up to 2500 Bq/g. However, depth of penetration does not exceed 10 mm.

In the Table 7 below the detailed results are presented of the measurements of contamination penetration into the construction materials.

Table 7

Detailed results of the measurements of contamination penetration into the construction materials.

Room #	Isotope	Activity distribution in depth, Bq/g										
		2mm	4mm	6mm	8mm	10mm	12mm	14mm	16mm	18mm	20mm	22mm
A-108/1	^{137}Cs	987	874	725	648	288	112	30	12	12	0	0
concrete floor	^{134}Cs	612	514	366	188	76	38	22	10	8	0	0
	^{60}Co	434	345	212	130	66	23	8	5	5	0	0
A-108/1	^{137}Cs	100	30	6	0	0	0	0	0	0	0	0
concrete wall	^{134}Cs	40	10	4	0	0	0	0	0	0	0	0
	^{60}Co	25	8	4	0	0	0	0	0	0	0	0
A-129/1	^{137}Cs		1180	0	0	0	0	0	0	0	0	0
plastic	^{134}Cs		620	0	0	0	0	0	0	0	0	0
compound	^{60}Co		580	0	0	0	0	0	0	0	0	0
A-129/2	^{137}Cs		850	0	0	0	0	0	0	0	0	0
plastic	^{134}Cs		560	0	0	0	0	0	0	0	0	0
compound	^{60}Co		520	0	0	0	0	0	0	0	0	0
B-111/1	^{137}Cs		1200	0	0	0	0	0	0	0	0	0
plastic	^{134}Cs		670	0	0	0	0	0	0	0	0	0
compound	^{60}Co		340	0	0	0	0	0	0	0	0	0
A-045/1	^{137}Cs	890	430	122	65	18	4	0	0	0	0	0
concrete floor	^{134}Cs	645	276	64	22	6	0	0	0	0	0	0
	^{60}Co	425	124	34	12	5	0	0	0	0	0	0
A-219/1	^{137}Cs	880	418	124	42	14	4	0	0	0	0	0
concrete floor	^{134}Cs	643	286	98	24	8	0	0	0	0	0	0

	^{60}Co	312	144	64	18	6	0	0	0	0	0	0
A-219/1 concrete wall	^{137}Cs	165	60	8	0	0	0	0	0	0	0	0
	^{134}Cs	85	22	6	0	0	0	0	0	0	0	0
	^{60}Co	42	16	5	0	0	0	0	0	0	0	0
B-110/1 concrete floor	^{137}Cs	1145	654	210	64	22	8	2	0	0	0	0
	^{134}Cs	755	298	112	43	6	0	0	0	0	0	0
	^{60}Co	324	166	78	16	4	0	0	0	0	0	0
B-110/1 concrete wall	^{137}Cs	144	62	10	0	0	0	0	0	0	0	0
	^{134}Cs	78	28	6	0	0	0	0	0	0	0	0
	^{60}Co	34	18	5	0	0	0	0	0	0	0	0
B-002/1 concrete floor	^{137}Cs	2130	1643	812	432	216	94	28	10	6	0	0
	^{134}Cs	1725	845	472	248	114	42	12	4	0	0	0
	^{60}Co	896	451	244	132	68	26	8	0	0	0	0
B-002/1 concrete floor	^{137}Cs	214	98	8	0	0	0	0	0	0	0	0
	^{134}Cs	65	14	4	0	0	0	0	0	0	0	0
	^{60}Co	32	10	3	0	0	0	0	0	0	0	0
B-002/1 Walls (paint)	^{137}Cs	94	0	0	0	0	0	0	0	0	0	0
	^{134}Cs	52	0	0	0	0	0	0	0	0	0	0
	^{60}Co	28	0	0	0	0	0	0	0	0	0	0

* Note: for the last row of the Table (paint), depth of penetration is 1 mm.

As a result of above mentioned measurements a number of results have been obtained describing the following characteristics:

- Gamma mapping of all ANPP premises.
- Levels of radioactive contamination of the ANPP premises and equipment.
- Data about radionuclide content and activity of surface contamination of ANPP premises and equipment.
- Activity and radionuclide content of deposits on the equipment internal surfaces.
- Data on penetration of radioactive substances into construction and protective materials of the controlled area premises.
- Amount and activity of all radioactive wastes accumulated at ANPP and their forecasted volumes.

The obtained information forms a base for:

- Optimization of planning activities for further radiological characterization.
- Decision-making on the systems in respect of decontamination..
- Planning of the personnel dose loads taking part in decontamination process.
- Development of a radiation protection program during decommissioning.
- Estimations of amount and activity for all categories of radioactive waste which can be formed during the ANPP decommissioning.
- Development of radioactive waste management process.
- Determination of scope for the environmental radiation monitoring during the ANPP decommissioning.
- Selection of most optimum scenario for the ANPP decommissioning.