Review of Existing and Future Requirements for Decommissioning Nuclear Facilities in the CIS

A report produced for The European Commission, Directorate General XI

January 1999

The picture on the front page is the Beloyarsk NPP Units 1 and 2 in the Russian Federation - forerunners to the RBMK reactor series.

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Executive Summary

The countries that form the Commonwealth of Independent States (CIS) have inherited a significant legacy of nuclear reactors from the former Soviet Union. These were constructed during the period 1950-70 within a programme for the production of both electrical power and nuclear weapons material that paralleled similar programmes in Western Europe and North America. Russia now has some 15 VVER (pressurised water) Nuclear Power Plants (NPPs), 17 RBMK or similar (water-graphite) reactors, 13 further graphite moderated industrial reactors used for the production of plutonium, 1 fast reactor and around 43 research reactors, 52 critical assemblies and 18 subcritical assemblies. Other CIS countries also have significant nuclear facilities.

Many of these facilities have now reached or are close to reaching their design lifetime. In Russia, four NPPs have already been shut down, 2 at Novovoronezh and 2 at Beloyarsk. In Ukraine three NPP units are shut down, all at Chernobyl NPP and the remaining one (unit 2) is expected to shut down in the near future while in Armenia one unit has been shut down. In the short term (within the next 10 years or so) a further 10 NPPs are expected to shut down; Bilibino 1-4, Leningrad 1-2, Novovoronezh 3-4 and Kola 1-2. The plutonium-producing (so-called 'industrial') reactors at Mayak, Krasnoyarsk and Tomsk have been shut down (apart from three). The great majority of the research rectors are either permanently shut down or currently inactive.

This paper reports on the results of a study funded by the Directorate General XI of the European Commission (EC) on a review of existing and future requirements for decommissioning nuclear facilities in the Commonwealth of Independent States. The report provides an overview of the current state in CIS countries concerning decommissioning requirements for a number of nuclear facilities with the power reactors being the largest decommissioning liability. The objective was to establish the technical conditions and requirements for decommissioning of nuclear facilities in the CIS-countries. The study is part of the European Union co-operation with individual CIS Member States.

The shut down of all these reactors and their associated supporting facilities, such as reprocessing and waste treatment facilities, leaves a major nuclear decommissioning legacy. Since the break-up of the Soviet Union and its communist economy, the funds that have been available for this decommissioning have been extremely limited. Such estimations of when a NPP is expected to shut down can not be exact and are influenced by a balance of the anticipated end of design lifetime of 30 years with the pressures to extend such lifetimes to continue to produce power and support the countries' economic situation. Similarly, research reactors are flexible by nature and can be refurbished and continue to operate well beyond the original design lifetime providing safety and funding conditions are met.

Decommissioning of large, multi-unit NPP units is both technically complex and has significant implications for the local social and economic situation. In CIS countries decommissioning of older generation units on a site is linked to the construction of new units to replace the power generated and maintain the social and economic regime that exits to support the NPP site. It is not uncommon for a NPP site to support up to 50,000 (on average) local inhabitants who are linked directly or indirectly to the operation of the site.

Many issues remain to be resolved before successful decommissioning can be achieved and include:

- Clearly defined regulatory requirements for decommissioning of nuclear facilities.
- Decommissioning in the CIS is still relatively new and is constrained by the lack of funding and regulatory framework in which to carry out decommissioning activities. Funds for decommissioning NPPs have only been recently set up and do not have sufficient funds to cover the final decommissioning costs. This is exacerbated by the current financial difficulties the CIS currently face in adopting a market led economy from the original centrally led economies of the former Soviet Union. Funds for decommissioning research reactors are non-existent and the responsibilities have now been passed from the government to the operating organisations to finance the decommissioning. This drives the operating organisations to consider lifetime extension of facilities in order to generate additional funds, some of which may be used for future decommissioning.
- Clearly defined waste management and disposal routes. Although waste management can include the long term storage of wastes above the ground ultimately a final disposal site is required such as a deep geological facility. A national solution for a repository is required, possibly based on regional considerations if more than one is identified to serve the needs of the CIS.
- Although large scale decommissioning of NPPs is not expected to commence in the short term, there will be a significant amount of decommissioning work related to Stage 1 activities and preparing reactor systems for future decommissioning strategies such as storage and surveillance. Consequently there is a requirement for the exchange of experience in decommissioning tools and techniques. In addition an exchange of

experience in developing decommissioning documentation has been identified to maintain best practice.

- Graphite from RBMK units is particularly susceptible to long term deterioration without suitable treatment or disposal. Co-operation with interested parties including those from the CIS should be developed on an international level to spread best practice and experience.
- The management of spent fuel is dependant on the nuclear reactor type. Fuel from RBMK units have no disposal route through reprocessing and are currently stored on the NPP sites. Fuel from VVER units does have a disposal route via reprocessing at Mayak but this is a costly option requiring funds to be made available. Fuel from research reactors varies in configuration and type and requires a number of reprocessing techniques. Some of these are available at sites such as Mayak but others are not and fuel is being stored on the site of the research reactor. Again, reprocessing, if available, is a costly option.
- The management of contaminated sodium is, again, a significant problem with many countries having significant quantities of sodium from fast reactor programmes. The treatment, conditioning and disposal of sodium is a complex requirement. Co-operation with interested parties including those from the CIS should be developed on an international level to spread best practice and experience.
- With the tendency to defer decommissioning later then techniques to remove or fix mobile activity are required. Techniques for decommissioning, particularly decontamination methods for research reactors in the CIS countries are limited.

Contents

List of Abbreviations

AEAT	AEA Technology
AEP	Atomenergoproekt All-Russian Research and Design Institute
EBRD	European Bank for Reconstruction and Development
EC	European Commission
EPG	Channel Graphite Low Capacity Reactor Unit
CERI	Complex engineering and radiation inspection
CIS	Commonwealth of Independent States
CSC	Council of Scientific and Technical Co-ordination
FR	Fast reactor
GAN RF	GOSATOMNADSOR of Russia; Nuclear and Radiation Safety Authority of Russia
GOST	State standard
HL-RW	High level radioactive waste
IAEA	International Atom Energy Agency
ICRP	International Committee for Radiation Protection
IL-RW	Intermediate level radioactive waste
ISO	International Standardisation Organisation
KAB	Kraftwerks- und Anlagenbau GmbH Berlin
LL-RW	Low level radioactive waste
LNPP	Leningrad Nuclear Power Plant
LRW	Liquid radioactive waste
Minatom of Russia	Ministry of Russian Federation of Atomic Energy
Ministry of Health of Russia	Ministry of Russian Federation of Health and Medical Industry
NIS	NIS – Ingenieurgesellschaft mbH
NPP	Nuclear Power Plant
NVNPP	Novovoronezh NPP
OKB "Gidropress"	Gidropress Experimental and Design Bureau(Podolsk)
OST	Industrial standard
RBMK	Channel type large power energy reactors with graphite moderator
REA	State Concern ROSENERGOATOM
RF	Russian Federation
RR	Research reactor
RRC KI	Russian Research Centre “Kurchatov Institute”
RW	Radioactive waste
SNF	Spent Nuclear Fuel
SRW	Solid radioactive waste
TACIS	Technical Assistance Programme for CIS-countries and Mongolia
TOOTD	Typical Operational, Organisational and Technological Documents
USDOE	United States Department of Energy
VGNIPKII Atomenergoproject	All-Russian Scientific-Research Design and Investigating Institute "Atomenergo-project"
VNIIAES	All-Russian Research Institute for Nuclear Power Plant Operation
VNIPIET	All-Russian Science Research and Design Institute of Power Engineering Technology
VVER	Water-Water-Energy-Reactors

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1. Introduction

This paper reports on the results of a study funded by the Directorate General XI of the European Commission (EC) on a review of existing and future requirements for decommissioning nuclear facilities in the Commonwealth of Independent States. The report provides an overview of the current state in CIS countries concerning decommissioning requirements for a number of nuclear facilities with the power reactors being the largest decommissioning liability. The objective was to establish the technical conditions and requirements for decommissioning of nuclear facilities in the CIS-countries. The study is part of the European Union co-operation with individual CIS Member States.

The terms of reference for this project identifies a set of tasks as follows:

1. Establish a database of facilities that are closed or likely to be shut down in the short term (around 2005).
2. For each facility in the database, establish whether a significant decommissioning study has already been undertaken, and whether access to that study can be obtained.

Identify a ‘representative’ facility, for which a reasonable body of data relevant to decommissioning already exists, so that it could be taken as the basis for a ‘generic’ decommissioning plan for the following types of facilities:

- RBMK reactors
- VVER-440 213 and 230 reactors
- VVR research reactors
- BN-type fast reactors
- Research laboratories
- Fuel reprocessing facilities

3. For each ‘representative’ facility prepare the outlines of a generic decommissioning plan.
4. For each category of facility, highlight those facilities which are significantly untypical and give an indication of the decommissioning measures which will be required (i.e. significant departures from the generic plan)

In the course of the work programme a number of changes were made to the scope to take into account the limited information available to the Consortium through contacts with CIS organisations involved in decommissioning.

It was found that information on research facilities was limited and only information relating to those facilities attached to research reactors was available as part of the work on collecting information on the reactors. Similarly information on reprocessing facilities at Mayak, Tomsk and other sites was unavailable to the consortium. These facilities were still considered to be commercial sensitive and had aspects of military security attached to them.

Accordingly the work programme concentrated on:

- RBMK reactors
- VVER-440 213 and 230 reactors
- VVR research reactors
- BN-type fast reactors

This Section (Section 1) introduces the project and scope of work defined under the contract. Section 2 discusses and explores the general state of decommissioning in the CIS including policy and regulations, decommissioning strategies and current decommissioning works being undertaken. Section 3 assesses the decommissioning requirements for nuclear facilities of the CIS by reviewing those nuclear facilities chosen as representative of the types operated in the CIS. Finally, Section 4 identifies a number of conclusions and recommendations resulting from this work programme.

The sources of information are mentioned for every section. The authors of this report wish to thank the Local Partners involved in this project for their help and advice. All other information was used from open literature and presentations of various experts at meetings, for instance at meeting of the Working Group “Decommissioning and radioactive waste management” of the International Economic Association INTERATOMENERGO.

More information is becoming available on the world wide web and several sources have been used for details of many reactor systems in the former Soviet Union including photographs of reactor plant including the following:

- http://www.insc.anl.gov/
- http://nuke.handheld.com/

The authors acknowledge these information sources.

2. General State of Decommissioning of Nuclear Facilities in the CIS

2.1 Overview

2.1.1 The Nuclear Legacy

The CIS and other countries that made up the former Soviet Union have inherited a significant legacy of nuclear facilities. These facilities were constructed during the period 1950-70 within a programme for the production of both electrical power and nuclear weapons material that paralleled similar programmes in Western Europe and North America. Russia, as the country with the majority of nuclear facilities now has some 15 VVER (pressurised water) Nuclear Power Plants (NPPs), 17 RBMK or similar (water-graphite) reactors, 13 further graphite moderated industrial reactors used for the production of plutonium, 1 fast reactor and, according to a 1993 report by Gosatomnadsor, there were at last count, 45 research reactors, 52 critical assemblies and 18 subcritical assemblies.

The shut down of all these reactors and their associated supporting facilities, such as reprocessing and waste treatment facilities, leaves a major nuclear decommissioning legacy. Since the break-up of the Soviet Union and its communist economy, the funds that have been available for this decommissioning have been, at best, limited or non-existent.

Table 2-1 shows the number of civil nuclear facilities in the CIS while Figure 2-1 shows the NPP sites and reprocessing sites for the CIS. The NPP sites are expanded on later in this section.

Table 2-1 Civil nuclear facilities in CIS-countries

Country Facility	Russia	Ukraine	Kazakhstan	Belorussia	Armenia	Others
VVER-units	14	11	0	0	2	0
RBMK-units	11	3	0	0	0	0
Other graphite moderated NPP units	6	0	0	0	0	0
Fast reactors	1	0	1	0	0	0
Research reactors	45	1	4	1	0	2

Figure 2-1 NPP units in the CIS

The split of facilities into countries, types and period of construction is shown in Figures 2-2 and 2-3. In addition, Table 2-2 gives further details of the NPPs while Table 2-3 gives details of the research reactors in CIS countries. These form the majority of nuclear facilities in the CIS countries and represent a substantial decommissioning liability.

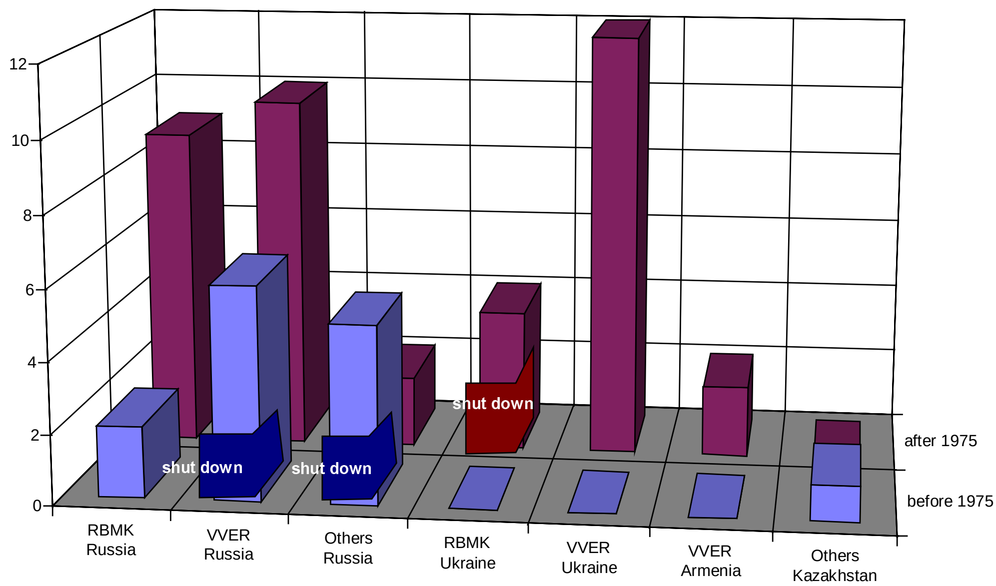

Figure 2-2 Breakdown of NPP units in the CIS

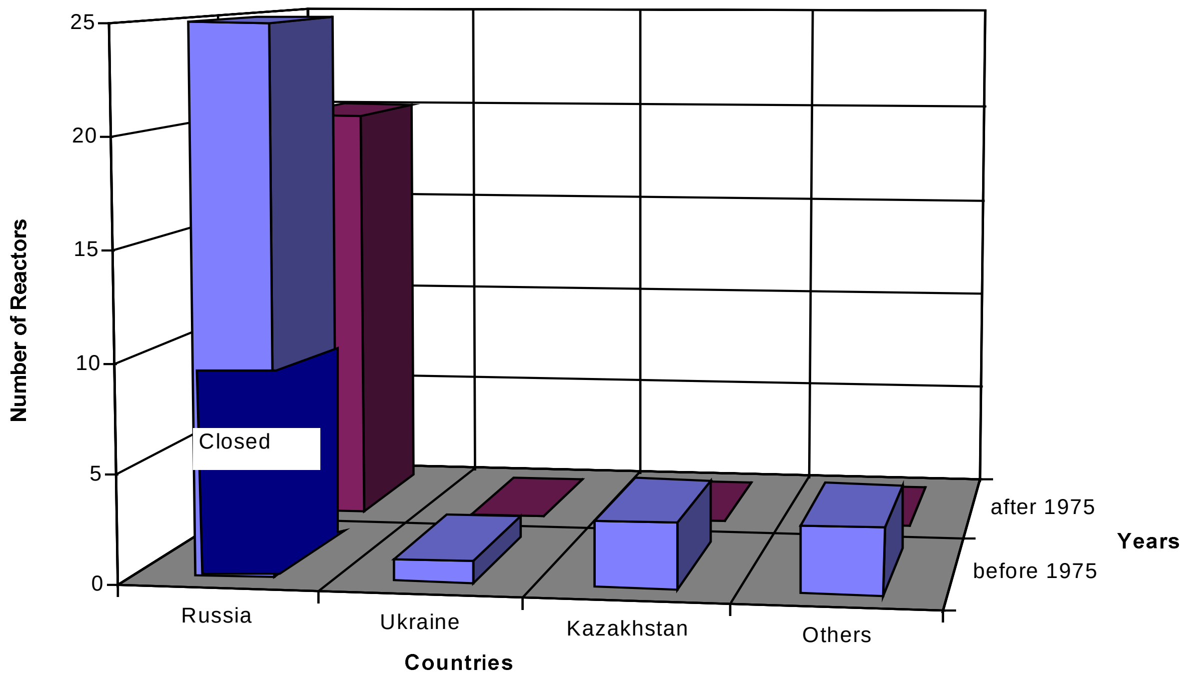

Figure 2-3 Breakdown of research reactors in the CIS

As the figures show a significant number of facilities were built before 1975. As the nominal design life of NPPs in particular is generally within 25 to 35 years then obviously these facilities will be coming to the end of their design life. This timescale is complicated by further factors, not least of which is the accepted practice to extend operating lifetimes of NPPs provided safety and technical requirements are met.

2.1.2 Nuclear Facilities in Russia

As mentioned earlier, the Russian Federation has the majority of nuclear facilities in the CIS by virtue of its size and as the main country behind the former Soviet Union. Figure 2-4 shows the NPP sites that exist in Russia and the reactor types. The majority of sites are based in the western part of Russia reflecting the major population densities. Only one NPP site lies east of the Ural mountain range, Bilibino, lying in the arctic circle.

Power transmission is the responsibility of the Russian Integrated Power System Joint Stock Company which was privatised in 1995 but with 51% of the company still under state ownership.

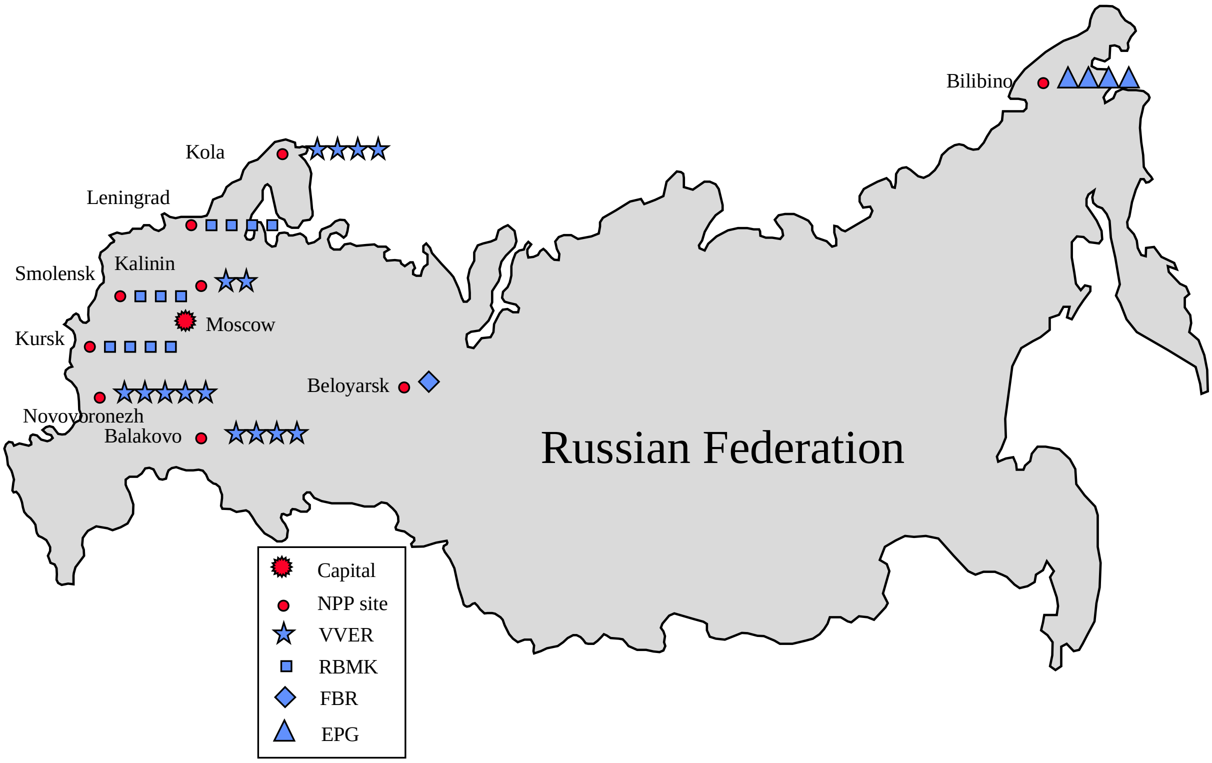

Figure 2-4 NPP sites in Russia

Research reactor sites are not shown here because of the large number that exist. Table 2-3 gives the location of the research reactors in Russia. These facilities also tend to concentrate in western Russia but there are a number in former 'closed cities' such as Tomsk and Mayak (see Figure 2-1) that were built up to support the weapons production facilities in Russia.

Russia contains the majority of the nuclear fuel cycle facilities from the former Soviet Union. It has one operating mine at Tulukai operated by the Priargunsky mining and chemical combine. Four sites convert and enrich uranium at Angarsk, Krasnoyarsk, Tomsk and Ekaterinburg (see Figure 2-1).

2.1.3 Nuclear Facilities in Ukraine

Ukraine has a large number of NPP sites, second only to Russia in number (see Table 2-1) but only one research reactor (see Table 2-3). The NPP sites are shown in Figure 2-5. Chernobyl is perhaps the most famous after the explosion of Unit 4 - a first generation RBMK design.

Ukraine also has two operating underground uranium mines at Ingulskoye and Smolino. Conversion is carried out at Zheltye Vody.

Figure 2-5 NPP sites in Ukraine

2.1.4 Nuclear Facilities in Kazakhstan

Kazakhstan has only one NPP at Aktau (see Figure 2-6) and is the only BN 350 fast reactor system (a second fast reactor unit the BN 600 at Beloyarsk is the next generation of design).

Kazakhstan has an enormous uranium reserve with mining being undertaken at a number of sites with a number of these sites being able to convert the uranium to hexaflouride or yellowcake. The country also has a fuel fabrication facility at Ust-Kamenogorsk.

Figure 2-6 NPP sites in Kazakhstan

2.1.5 Nuclear Facilities in Other CIS Countries

Armenia has the only other NPP in the CIS - Metsamor containing two VVER one of which is currently operating and is shown in Figure 2-7.

Figure 2-7 NPPs in Armenia

Belorussia (IRT), Uzbekistan (VVR-S) and Georgia (IRT) each had a research reactor but which are now either closed or dismantled - see Table 2-3.

Table 2-2 Nuclear power plants in the CIS countries (operating or shut down)

n°	Name of the NPP and n° of the unit	Reactor type	Project Code	Installed power [MW]	Reactor generation	Operating organisation	Start of operation	Planned finish of operation
				RUSSIA				
1	Balakovo -1	VVER-1000	V-320	1000	2	Rosenergoatom	1985	2015
	Balakovo -2	VVER-1000	V-320	1000	2	Rosenergoatom	1987	2017
	Balakovo -3	VVER-1000	V-320	1000	2	Rosenergoatom	1988	2018
	Balakovo -4	VVER-1000	V-320	1000	2	Rosenergoatom	1993	2023
2	Beloyarsk -1	AMB-100	-	100	1	Rosenergoatom	19634	shut down 1983
	Beloyarsk -2	AMB-200	-	160	1	Rosenergoatom	1967	shut down 1989
	Beloyarsk -3	BN-600	-	600	1	Rosenergoatom	1980	2010
3	Bilibino -1	EGP-12	-	12	1	Rosenergoatom	1974	2004
	Bilibino -2	EGP-12	-	12	1	Rosenergoatom	1974	2004
	Bilibino -3	EGP-12	-	12	1	Rosenergoatom	1975	2005
	Bilibino -4	EGP-12	-	12	1	Rosenergoatom	1976	2006
4	Kalinin -1	VVER-1000	V-338	1000	2	Rosenergoatom	1984	2014
	Kalinin -2	VVER-1000	V-338	1000	2	Rosenergoatom	1986	2016
5	Kola -1	VVER-440	V-230	440	1	Rosenergoatom	1973	2003
	Kola -2	VVER-440	V-230	440	1	Rosenergoatom	1974	2004
	Kola -3	VVER-440	V-213	440	2	Rosenergoatom	1981	2011
	Kola -4	VVER-440	V-213	440	2	Rosenergoatom	1984	2014
6	Kursk -1	RBMK-1000	-	1000	1	Rosenergoatom	1976	2006
	Kursk -2	RBMK-1000	-	1000	1	Rosenergoatom	1979	2009
	Kursk -3	RBMK-1000	-	1000	2	Rosenergoatom	1983	2013
	Kursk -4	RBMK-1000	-	1000	2	Rosenergoatom	1985	2015
7	Leningrad -1	RBMK-1000	-	1000	1	Leningrad NPP	1973	2003
	Leningrad -2	RBMK-1000	-	1000	1	Leningrad NPP	1975	2005
	Leningrad -3	RBMK-1000	-	1000	2	Leningrad NPP	1979	2009
	Leningrad -4	RBMK-1000	-	1000	2	Leningrad NPP	1981	2011
8	Novovoronezh -1	VVER-213	-	210	1	Rosenergoatom	1964	shut down 1984
	Novovoronezh -2	VVER-365	-	365	1	Rosenergoatom	1969	shut down 1990
	Novovoronezh -3	VVER-440	V-179	417	1	Rosenergoatom	1971	2001
	Novovoronezh -4	VVER-440	V-179	417	1	Rosenergoatom	1972	2002
	Novovoronezh -5	VVER-1000	V-187	1000	2	Rosenergoatom	1980	2010
9	Smolensk -1	RBMK-1000	-	1000	2	Rosenergoatom	1982	2012
	Smolensk -2	RBMK-1000	-	1000	2	Rosenergoatom	1985	2015
	Smolensk -3	RBMK-1000	-	1000	2	Rosenergoatom	1990	2020
				UKRAINE				
	Chernobyl -1	RBMK-1000	-	1000	1	Goskomatom	1977	2002?
	Chernobyl -2	RBMK-1000	-	1000	1	Goskomatom	1978	1991
	Chernobyl -3	RBMK-1000	-	1000	2	Goskomatom	1981	2002
	Chernobyl -4	RBMK-1000	-	1000	2	Goskomatom	1983	Destroyed 1986
	Khmelnitsky -1	VVER-1000	V-320	1000	2	Goskomatom	1988	2018
	Rovno -1	VVER-440	V-213	402	1	Goskomatom	1981	2011
	Rovno -2	VVER-440	V-213	416	1	Goskomatom	1982	2012

n°	Name of the NPP and n° of the unit	Reactor type	Project Code	Installed power [MW]	Reactor generation	Operating organisation	Start of operation	Planned finish of operation
	Rovno -3	VVER-1000	v-320	1000	2	Goskomatom	1987	2017
	South Ukraine -1	VVER-1000	V-302	1000	2	Goskomatom	1982	2012
	South Ukraine -2	VVER-1000	V-302	1000	2	Goskomatom	1985	2015
	South Ukraine -3	VVER-1000	V-302	1000	2	Goskomatom	1989	2019
	Zaporozhye -1	VVER-1000	V320	1000	2	Goskomatom	1985	2015
	Zaporozhye -2	VVER-1000	V320	1000	2	Goskomatom	1985	2015
	Zaporozhye -3	VVER-1000	V320	1000	2	Goskomatom	1987	2017
	Zaporozhye -4	VVER-1000	V320	1000	2	Goskomatom	1988	2018
	Zaporozhye -5	VVER-1000	V320	1000	2	Goskomatom	1989	2019
					ARMENIA			
	Metsamor -1	VVER-440	V-230	440	2	Armatomenergo	1979	Shutdown 1989
	Metsamor -2	VVER-440	V-230	440	2	Armatomenergo	1996	2026?
					KAZAKHSTAN			
	Aktau	BN-350	-	350	-	MAEC	1973	2003

Table 2-3 Research reactors in the CIS Countries

N°	Owner, management, site	Name	Type	Power designed	MW_{th} after reconstr-uction	Date of Commis-sioning	Condition	Remarks
				RUSSIA				
1	Russian Research Centre "Kurchatov Institute" (RRC KI) 123182, Moscow Kurchatov place 1	F-1	Uranium-graphite	0,024		1946	operating	
2	RRC KI	VVR-2	pool type, water-water	0,3	3	1954	closed 1983	
3	RRC KI	RFT	Channel, graphite	10	20	1952	shutdown 1962	
4	RRC KI	MR	water-beryllium, channel type, pool type	20	40	1963	closed 1993	
5	RRC KI	IRT	pool type	2	8	1957	shutdown 1979	
6	RRC KI	IR-8	pool type	8		1981	operating	
7	RRC KI	"Hydra" (IIN-3M)	Homogeneous, impulse type	0,01 30 MJ in impulse		1972	operating	
8	RRC KI	GAMMA	vessel type, water-water	0,125		1982	operating	
9	RRC KI	"ARGUS"	Homogeneous	0,02		1981	operating	
10	RRC KI	OR	tank type, water-water	0,3		1989	operating	
11	RRC	"ROMASHKA"	High temperature on the basis of medium neutrons with thermoelectric transformers	0,04		1964	closed 1966	

N°	Owner, management, site	Name	Type	Power designed	MW_{th} after reconstr-uction	Date of Commis-sioning	Condition	Remarks
12	State Scientific Centre RF "Institute for Theoretical and Experimental Physics" (ITEF) of Minatom of Russia 117259 Moscow ul. B. Shterjomushkinskaja 25	TVR	Tank type, heavy water	0,5	2,5	1949	closed 1986	
13	Moscow Engineering and Physic Institute (MIFI) Ministry of High Education of Russia 115409 Moscow Kashirskoje Shose 32	IRT	pool type	2	2,5	1967	operating	
14	Scientific Research and Constructor Institute of Electrotechnique (NIKIET) of Minatom 101000, Moscow, P-Box 788	IR-50	pool type	0,05		1961	under reconstruction 1994	
15	State Scientific Centre of RF "Physical-Energetical Institute" (FEI) of Minatom of Russia 249020, Obninsk, Kaluga region	ÀÌ	Channel type, uranium-graphite	30	10	1954	operating	
16	FEI	BR-10	fast reactor, sodium cooled	5	8	1958	operating	
17	FEI	BARS-6	fast reactor, impulse type	0,01; 160 GW impulse		1995	operating	
18	Department of the State Scientific Centre RF "Institute for physical chemistry - Karpov) Ministry for Economy of Russia 249020, Obninsk, Kaluga Region	VVR-Ö	tank type, water-water	15		1964	operating	
19	State Scientific Centre RF "Scientific Research Institute for Nuclear Reactors" (NIIAR) of Minatom of Russia 433510, Dimitrovgrad, Uljanov Region	SM-3	Vessel type	50	100	1961	operating	reconstructed 1974 and 1993
20	NIIAR	MIR-M-1	Channel type in pool	100		1966	operating	reconstructed 1975
21	NIIAR	BOR-60	fast reactor, sodium cooled	60		1969	operating	
22	NIIAR	RBT-6	pool type	6		1975	operating	
23	NIIAR	RBT-10/1	pool type	10		1983	operating	
24	NIIAR	RBT-10/2	pool type	10		1974		
25	NIIAR	BK-50	vessel type, boiler reactor	140	170	1971		
26	NIIAR	ARBUS	vessel type, organic medium cooled	12				
27	Complex Institute for Nuclear Research (OIJaI) 141980 Dubna Moscow Region	IBR-30	fast reactor, impulse type	0,03 (in average) 90 V (in impulse)		1988		

N°	Owner, management, site	Name	Type	Power designed	MW_{th} after reconstr-uction	Date of Commis-sioning	Condition	Remarks
28	OIJaI	IBR-2	fast reactor, impulse type	4 (average 8.3* 10^6 impulse		1984		
29	State Research Institute for Devices (NIIP) of Minatom of Russia 140061, Lytkarino, Moscow Region	IBR-M1	pool type	2		1974	from 1990 under reconstruction	
30	NIIP	BARS-2	Impulse type	2.0 MJ impulse		1971	operating	
31	NIIP	BARS-3M	Impulse type	3.2 MJ impulse		1988	operating	
32	NIIP	BARS-4	Impulse type	4.2 MJ impulse		1984	operating	
33	NIIP	TIBR-1M	Impulse type	4.3 MJ impulse		1976	operating	
34	Petersburg Institute for Nuclear Physic (PIJaF) of the Academy of Science 188350, Gatshina, Leningrad Region	VVR-M	Tank type	10	18	1959	operating	
35	State Research Institute of Nuclear Physic at Tomsk Polytechnic High School of Ministry of High Education of Russia 63061 Tomsk	IRT-T	pool type	2	6	1967	operating	reconstructed 1984
36	Sverdlovsk Branch of NIKIET of Minatom of Russia 624051, Saretshni, Sverdlovsk Region	IBB-2M	pool type	10	15	1966	operating	reconstructed 1983
37	All-Russian Scientific and Research Institute for Experimental Physics - Russian Federal Nuclear Centre (VNIIEF) Sarov, Nishegorod Region	BIGR	uranium-graphite- impulse type	300 MJ impulse		1977	operating	
38	VNIIEF	BR-1	uranium-graphite- impulse type	50 MJ impulse		1978	operating	
39	VNIIEF	BIR-2M	Solution, impulse type	81 MJ impulse		1979	operating	
40	All-Russian Scientific-Research Institute for Theoretical Physics Russian Federal Nuclear Centre (VNIITF) of Minatom of Russia Shneshinsk, Cheljabinsk Region	BARS-5	fast reactor, impulse type	0,01; 160 GW impulse		1986	operating	
41	VNIITF	IGRIK	impulse type, homogeneous	0,03; 25 GW impulse		1975	operating	
42	VNIITF	Jaguar	impulse type, homogeneous	0,01; 40 GW impulse		1990	operating	
43	VNIITF	FBR-L	impulse type, homogeneous	0,005; 800 GW impulse		1981	operating	

N°	Owner, management, site	Name	Type	Power designed	MW_{th} after reconstr-uction	Date of Commis-sioning	Condition	Remarks
44	Norilsk Mining Combinat (NGMK) of the Concern "Russian Nickel" 663300, Norilsk Taimirskij Autonomous Region	RG-1M	Tank type	0.1		1970	operating	
45	Central Scientific Research Institute Krylov of the Ministry for Economy of Russia St. Petersburg	U-3	pool type	0.005		1964		
			BELORUSSIA					
46	Institute of Nuclear Power, Belorussia	IRT-M	pool type	2	4	1962	shut down 1968	
			UKRAINE					
47	Institute for Nuclear Research, Kiev, Ukraine	VVR-M	Tank type	10		1960	operating	
			GEORGIA					
48	Institute of Physics, Tbilisi, Georgia	IRT-M	pool type	2	8	1959	shutdown 1990	
			UZBEKISTAN					
49	Nuclear Physics Institute, Tashkent, Uzbekistan	VVR-CM	Tank type	2	10	1959	unknown	
			KAZAKHSTAN					
50	Nuclear Physics Institute, Almata, Kazakhstan	VVR-K	Tank type	10		1967	shutdown 1988	shutdown for enhancing seismic stability systems

2.2 Decommissioning Policy, Regulations and Considerations

2.2.1 Introduction

The Russian objective for NPP decommissioning can be described as follows [2-1]:

> *"The final aim of NPP decommissioning is bringing the site, on removing the radioactive facilities, to the state, when radioactivity levels do not exceed background values, while the site can be used for any other economic activity without restrictions; i.e. neither spent nuclear fuel nor radioactive wastes are to be left on the site of decommissioning."*

This aim does not necessarily include decommissioning to a green field but rather allows an industrial use if suitable. The Russian decommissioning strategy depends on a number of conditions and issues which have been changing regularly over the last few years as the regulatory and market economy conditions develop. Developments in the field of NPP decommissioning were started in the former Soviet Union in the 1980s after the safety code 'OPB-87' was published.

The basic normative document defining the safety requirements for research reactors in Russia is the "General Provisions on safety of research reactors" (OPB RR-94) [2-2]. According to the requirements of this normative document the operating organisation is responsible for the safety at all stages of the life cycle of the research reactor, including its decommissioning. In this normative document the process of decommissioning a research reactor is defined as a complex of measures after shut down of the research reactor, excluding its further use for the designated operation as a reactor, and guaranteeing the safety of the personnel, public and environment. The operating organisation should ensure the development of the decommissioning documentation for the research reactor.

<u>2.2.1.1 Current conditions for NPP decommissioning</u>

The current conditions influencing decommissioning strategy are described as follows [2-3]:

- The territory of the nuclear facilities and their sanitary-protected zone is handed over to the operating organisations by the decree of the President of the Russian Federation from 07.09.1992.[1]

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[1] The operating organisation of the most Russian NPPs is the state owned concern ROSENERGOATOM. The present exception is the Leningrad NPP. It is forecast that more local operating organisations will be created in future, for instance for the NPPs in Siberia or in the Russian Far East.

- The decommissioning works may be carried out on sites with currently operating units. The impact of decommissioning works adjacent to the operating units must be considered.
- The creation of new NPP sites is difficult in the Russian Federation as it is in western Europe and North America. Therefore the existing NPP sites must be reused for nuclear requirements.
- In most cases the infrastructure of the whole area around the Russian NPPs depends on the operating NPP. New settlements were erected alongside the NPP units to provide supporting services and house NPP staff. The closure of the whole NPP would have significant social and economic consequences for these settlements.
- Federal or regional stores for radwaste do not exist at the present. The operational radwaste is stored at the NPP site where it was created.
- The radwaste treatment facilities at the Russian NPP are underdeveloped. Therefore the processing and conditioning of operational and decommissioning radwaste for long-term storage is a difficult and hazardous task.
- Up until 1990 a fund for decommissioning work did not exist. In the former Soviet Union it was foreseen to finance the decommissioning work on the basis of the unified state budget. Unfortunately such a unified state budget does not now exist under the new market economy conditions. Consequently in 1990 is was decided to establish a decommissioning fund by the operating organisation on the basis of the price of the electricity generated. Currently this fund is very small and is influenced by Russian inflation and current market conditions.
- The existing guidelines in the nuclear field do not cover the decommissioning works to any great detail. The introduction of licensing changes have a significant influence to the decommissioning strategy.
- The costs for the complete dismantling of an NPP unit will be significant. Furthermore the Russian industry does not produce special tools for remote-controlled dismantling of high activated equipment.

The decommissioning of NPP units is not a common occurrence in Russia. It is worth mentioning that no large commercial nuclear power reactor has been decommissioned to stage 3 according to the IAEA definition. Consequently, Russia has to collect its own experience in the field of NPP decommissioning using its own and Western expertise.

2.2.1.2 The general strategy

All Russian sources describe the current general strategy for NPP decommissioning as follows:

"The lack of federal and regional stores, the difficulties with the site infrastructure and the lack of financing make decommissioning Stage 3 (green field) unattainable. It seems appropriate to abandon the aim of 'liquidation' as a final state of decommissioning as impracticable for the Russian strategy of decommissioning, since the idea of a green field under conditions of a multi-unit NPP is far from realistic." [2-3]. The objective is to remove the nuclear fuel from the closed units and to transfer the unit into safe storage conditions ('storage with surveillance' or 'entombment').

It is necessary to undertake the following works for the shut down of NPP units:

- development of the decommissioning concept for the NPP site
- approval of the site decommissioning concept by the Nuclear Regulatory Authority
- development and approval of the decommissioning documentation - this documentation has to include a section justifying the safety of the decommissioning of the NPP unit. This decommissioning documentation must be developed for each unit independently on the multi-unit site structure of the Russian NPPs.
- dismantling of the low level activated or non-activated equipment for disposal and/or re-use
- treatment and condition of radioactive waste from operations and decommissioning
- preparation of a storage for receipt of spent fuel and radwaste, if it was not undertaken during the normal operation of the reactor. After modification it would be possible to use the buildings of the NPP units for radwaste processing and/or interim storage
- development and approval of all necessary instructions on maintenance of the unit in case of its storage with surveillance
- safe storage of the highly activated reactor equipment on-site without any dismantling

The safe storage or entombment stage of the shut down NPP units is foreseen for a period of up to 50 years.

All decisions on decommissioning of NPP units in the CIS are connected to the decisions on the construction of new NPP units on the same site. The reason given is to guarantee the site infrastructure and maintain the social and economic regime that exists to support the

NPPs. Because of the difficult economic situation in all CIS-countries the construction of new NPP units is delayed. Consequently the decommissioning steps are also delayed, and - possibly - the construction of following units will also be delayed. Such delays drive the requirement to extend the life-time of existing NPP units and which is supported by the operating organisations, and the NPP staff and NPP management. The result of this discussion will depend on the financial situation and on the position of the Regulatory Authorities.

2.2.1.3 Decommissioning phases and its duration

The following most probable phases of decommissioning of NPP units in the Russian Federation are expected to be [2-3], [2-4]:

- phase 1a (prior to the final shut down)
  - development of the decommissioning programme
  - complex engineering and radiation inspection of the unit [2], including the development of the breakdown of costs, manpower and other resources
  - evaluation of the real technical status of the structures, facilities and equipment
  - development and design of measures to preserve the structures, facilities and equipment
  - development of suitable software for safety calculations
- phase 1b (after the final shut down):
  - construction of liquid and solid radwaste management facilities
  - construction of an interim storage facility for nuclear spent fuel in case reprocessing of the spent fuel is not possible
  - decontamination of the reactor coolant circuit and the reactor equipment

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[2] The complex engineering and radiation inspection has to include:

- Drawing up of a map of radiation fields in all rooms, and also inside the biological shield of the reactor;
- Investigation of the nuclide structure of radioactive contamination in the basic components of the reactor, of the equipment and pipes;
- Analysis of the status of the systems ensuring the safety of the shut down reactor;
- Additional analysis of the status of building structures;
- Analysis of a the status of the metallic reactor structure, of the storage for spent fuel assemblies (FA) and solid radwaste.

- physical and radiological investigation systems including the civil construction and assessment of the NPP area
- completion of the complex engineering and radiation inspection of the unit

- phase 2:
  - dismantling of equipment and technological facilities (except the reactor vessel)
  - on-site localisation of the reactor systems
  - processing of liquid and solid radioactive waste
  - decontamination and preparation of rooms for radwaste processing and/or interim stores for conditioned radwaste
  - storage of conditioned radwaste

- phase 3:
  - safe operation of the interim storage for nuclear spent fuel and radwaste and the localised equipment, facilities and constructions.

The duration of these phases is as follows [2-5]:

- Phase 1: 5 years. The present safety guidelines require the development of the decommissioning conceptional design up to 5 years before the end of the nominal life-time of the NPP unit - this design has to give an answer to the question whether decommissioning or extension of the life-time is to be the strategy [2-3].
- Phase 2: 5 years
- Phase 3: 30 - 100 years (depends on the stability of the civil construction)

The Russian decommissioning concept is mostly orientated to economic and social factors. The main purpose of the concept is the development of the Russian nuclear energy complex. The concept is developed for the current difficult economic and financial situation in Russia and reflected in other CIS countries. The main direction is the postponed dismantling of the high radioactive components. One of the reasons is the that there are no radioactive waste repositories. It should be pointed out that the responsibility for NPP radwaste management and for final radwaste disposal is divided between different organisations in the Russian Federation and therefore adding to the constraints on decommissioning options.

2.2.2 CIS Legal and Regulatory Framework

The following basic safety criteria must be observed in case of decommissioning of all nuclear facilities for all CIS countries:

- Maintenance of safety of the operational personnel, public and environment during storage with surveillance of the reactor and during dismantling, transportation and disposal of fuel, radioactive equipment and materials;
- Absence of any kind of contamination of the environment by radioactive waste, both during storage with surveillance and dismantling of the reactor, and during long-term storage of the units and systems;
- Maintenance of minimal dose rates during the decommissioning work execution and during disposal of the radioactive equipment;
- Maintenance of concrete structures and localisation of the reactor installation and also maintenance of reliable safety systems containing the radioactive elements of the reactor structure and of the equipment against probable influences by various natural phenomena such as weather condition and seismic events.

In the field of the nuclear industry the former Soviet norms and standards are the basis for all the CIS. Nevertheless the process of changes in this field is ongoing, especially in the Russian Federation and in the Ukraine because of their large nuclear liabilities.

The basis for the concept of a system of nuclear codes and standards in the Russian Federation is the existence of three general levels of documents that have to ensure the optimal scientific-technical and organisational management of radioactive waste (see Figure 2-8 below).

1st level	FEDERAL LAWS OF THE RUSSIAN FEDERATION ⇓ LEGAL ACTS OF THE PRESIDENT OF THE RUSSIAN FEDERATION ORDERS OF THE GOVERNMENT OF THE RUSSIAN FEDERATION LEGAL ACTS OF THE AUTHORITIES OF THE SUBJECTS OF THE RUSSIAN FEDERATION
	⇓
2nd level	FEDERAL CODES AND STANDARDS IN THE FIELD OF ATOMIC ENERGY ⇓ GUIDELINES OF THE GOSATOMNADSOR OF RUSSIA, OF MINISTRY OF HEALTH OF RUSSIA ⇓ STANDARDS, CODES AND STANDARDS FOR CIVIL CONSTRUCTIONS, REGULATIONS OF THE ATOMIC ENERGY MANAGEMENT
	⇓
3rd level	ORGANISATIONAL, OPERATIONAL AND TECHNOLOGICAL PROCEDURES

Figure 2-8 Structure of nuclear codes and standards

The fundamental objectives in this field are the:

- protection of the health of the population at present time
- environmental protection at the present time
- simple and cheap health and environmental protection for future generations

2.2.2.1 First level of documents

At this level the following codes will be included either completely or partially:

- Laws of the Russian Federation
- Decrees and legal acts of the President, Government of the Russian Federation, and Authorities of subjects of the Russian Federation corresponding to the mentioned field.

Table 2-4 shows the most important laws and orders that defines the legal basis for the use of atomic energy and for the nuclear and radiological safety. It should be noted that the legislation in Russia is in a process of significant change. The table gives only a guide to the laws and federal documents existing at this time.

Table 2-4 Russian laws and Federal documents in the field of radwaste management

N°	Title
1	Federal Law "On the Environment Protection"
2	Federal Law "On the Use of Atomic Energy "
3	The draft of the Federal Law “On radwaste management"
4	Federal law "On Radiation Protection of the Population"
5	Federal Law "On sanitary-epidemiological Welfare of the Population "
6	Rules “Licensing of Activities in the Field of Use of Atomic Energy
7	Federal target programme “Management of radioactive waste, nuclear spent fuel, and other nuclear spent materials, their re-use and disposal in the period 1996 - 2005”

2.2.2.2 Second level of documents

The second level includes codes and standards of the Russian Federation, regulations of GOSATOMNADSOR of Russia and of the Ministry of Health of Russia, standards, civil standards, and instructions of the Atomic energy management.

In correspondence to the Federal Law ‘On the use of Atomic energy’ the atomic energy management (MINATOM of Russia, and other departments, responsible for the mentioned field) is entitled to develop and issue federal codes and standards for the utilisation of atomic energy. The radioactive management is also included.

GOSATOMNADSOR of Russia and the Ministry of Health of Russia are entitled to develop, approve, and issue federal standards in the field of nuclear and radiological safety for use of atomic energy.

The state regulation for nuclear safety in the field of utilisation of atomic energy is organised on the federal level in correspondence with the following responsibilities:

- GOSATOMNADSOR of Russia (GAN): nuclear and radiological safety
- Ministry of Health of Russia: radiological safety (sanitary items)
- GOSGORTECHNADZOR of Russia: industrial safety
- Ministry of Interior Affairs: fire safety

The federal codes and standards, the rules of GOSATOMNADSOR and the Ministry of Health of Russia, standards, civil standards, and other rules and instructions of the Atomic energy management are written as principles, criteria and rules (see Table 2-5).

There are no guidelines to describe practical aspects of the undertaking of the defined requirements. They do not describe methods for specific technologies or equipment for particular facilities at defined NPPs.

Recently the Russian legislative authorities have made important efforts to develop Federal laws and decrees concerning the radwaste management which were missing up to now. The regulators, principally GOSATOMNADSOR of Russia and the Ministry of Health of Russia, have developed the structure of the Regulatory guidelines (level 2).

Table 2-5 List of Federal nuclear standards, regulations and key documents

N°	Title	Authorised by
1	General regulations for guarantee of the safety of nuclear plants (OPB - 88)	GOSATOMENERGONADSOR USSR
2	Radiation protection rules of operation of nuclear plants	Ministry of Health USSR
3	Basic sanitary rules for the handling of radioactive materials and other sources of ionising radiation	Ministry of Health USSR, GOSATOMENERGONADSOR USSR
3.1	Draft of the "General sanitary regulations of radiation safety"	
4	Sanitary rules for design and operation of nuclear plants	Ministry of Health USSR
5	The typical content of measurements for the safety of the NPP staff in the case of an accident	MINATOMENERGO USSR, GO USSR, MIA USSR
6	Radiation Protection Norms – 96	GOSSANEPIDNADSOR of Russian Federation
7	Methodological recommendations for the monitoring of radioactive substances in the environment	Ministry of Health USSR

These guidelines include the already issued principal new document NRB-96 (this document is valid for existing plants - all NPPs - only after the year 2000), the newly issued second level document in the field of radwaste management 'Safety Rule for NPP radioactive waste management'. GAN of Russia has issued their own structure of a system of Regulatory documents - see Table 2-6.

Table 2-6 The Structure of the Nuclear Regulatory Codes and Standards

0	Safety of radioactive waste management - General principles
1	Collecting, processing, conditioning and storage of radioactive waste - Safety requirements
1.1	Management of gaseous radioactive waste - Safety requirements
1.2	Collecting, processing, storage and conditioning of liquid radioactive waste - Safety requirements
1.3	Collecting, processing, storage and conditioning of solid radioactive waste - Safety requirements
2	Disposal of radioactive waste - Safety requirements
2.1	Disposal of radioactive waste - Safety principles, criteria and requirements
2.2	Surface disposal of radioactive waste - Safety requirements
2.3	Disposal of solid and solidified radioactive waste in geological formations - Safety requirements
2.4	Disposal of liquid radioactive waste in geological formations - Safety requirements
3	Management of radioactive waste generated by the decommissioning of nuclear facilities, radioactive sources and by the rehabilitation of contaminated areas - Safety requirements
3.1	Management of radioactive waste generated by the decommissioning of nuclear facilities and radioactive sources - Safety requirements
3.2	Management of radioactive waste generated by the rehabilitation of contaminated areas - Safety requirements
4.	Guidelines for safe radioactive waste management at plants
4.1	Safety at management of radioactive waste generated by mining, processing and usage of geological resources – Guideline
4.2	Safety at management of radioactive waste generated by mining, processing of radioactive ores - Guideline
4.3	Safety at management of radioactive waste generated by enrichment of uranium and production of nuclear fuel – Guideline
4.4	Safety at management of radioactive waste generated by reprocessing of nuclear fuel - Guideline
4.5	Safety at management of radioactive waste generated by usage of radioactive materials, radioactive substances (of isotope production) at the public economy, at scientific and medical organisations - Guideline
4.6	Safety at management of NPP radioactive waste – Guideline
4.7	Safety at management of radioactive waste generated at ships and other swimming objects with nuclear energy facilities and radioactive sources – Guideline
4.8	Safety at management of radioactive waste generated at specialised enterprises - Guideline
5	Guidelines for assessment and analysis of the safety, registration, reporting and monitoring in the field of radioactive waste management
5.1	Analysis, assessment and safety demonstration at various stages of radioactive waste management
5.2	Analysis, assessment and safety demonstration for release of nuclear materials, substances and activities out of the state safety monitoring
5.3	Reporting and registration of radioactive waste
5.4	Reporting and registration of release of radioactive materials into the natural environment

Remarks:

- Guideline in item 0: First priority
- Guidelines in items 1-3: Second priority
- Guidelines in items 4-5: Third priority

All the mentioned guidelines, codes and standards cover, generally, all the requirements, mentioned in Figure 2-8 as objectives.

2.2.3 Recycling Policy

Up to now the recycling of surplus equipment, for instance, of equipment dismantled during decommissioning, is difficult, because norms and standards were not established in the field of recycling and re-use of surplus materials in Russia (and other CIS-countries). Furthermore norms and standards for the exemption of materials also do not exist.

The Russian documents require [2-5]:

- The re-use of dismantled equipment if possible, particularly if the end of design lifetime is not finished, and the technical parameters conform to the passport data. For operation under different conditions the area and conditions for a possible re-use of residual materials and dismantled equipment must be defined. In this case it is necessary to certify the materials and equipment for the re-use.

- The dismantled equipment must be controlled and treated as suspect for possible radioactive contamination of surfaces. The limits of radioactive contamination are defined which must be achieved by the decontamination. These definitions are based on the probable replacement of the equipment in case of re-use. The decontamination must be undertaken by a method which does not destroy the integrity of the equipment, and does not change the physical and chemical qualities of materials and will keep, for instance, the protective zones on welded seams.
- If all the technical and radiation control requirements to the equipment are met, the area and conditions of its operation will be defined. This decision must be approved by the designer, by the manufacturer and by the regulatory authorities.
- In case of re-use of the equipment at another enterprise, the transfer will be accompanied by exit documents. These exit documents must contain the basic characteristics data and the results of the technical and radiation control.
- Solid NPP waste is considered as exempt if the specific activity of radionuclides is less than the limit according to OSP ORB-96 (A_{sp}= 0.5 Bq/g).

2.2.4 Waste Management

The most important document that describes and defines the strategy in the area of NPP radwaste management is the "*Working Programme for NPP radwaste management*" of ROSENERGOATOM. It was authorised by the First Deputy of the Minister of Atomic energy of Russia in 1993 and is directed to the solutions of problems of providing NPP with modern technologies and installations for conditioning of radioactive waste.

The *Working Programme* was established to meet the demands of the authorities, first of all GOSATOMNADSOR of Russia, and to increase the speed and the quality of the equipment development for treatment of radioactive waste so as to force its introduction at NPP site. It also allows for the co-ordination of activities of the various organisations engaged in the NPP radioactive waste management processes.

The following activities have the highest importance:

- Development of Typical Operational, Organisational and Technological Documents (TOOTD) regulating the NPP radwaste management;
- Creation of cementation plants for liquid radioactive waste (LRW);

- Creation of press compaction plants for solid radioactive waste (SLW);
- Creation of incineration plants for combustible radwaste;
- Creation of vitrification plants for fused salt and ashes;
- Development of technologies and equipment for clearing tanks of salt deposition layers;
- Development of technologies and equipment for the retrieval of used filter materials and sludge from tanks and their processing;
- Development of technologies for retrieval of radwaste from storage cells;
- Organisation of sorting of SRW and transfer to the treatment facilities;
- Creation of a plant complex for treatment of metallic radwaste;
- Manufacture of containers for solidified and press-compacted radwaste;
- Design of a storage facility for the regular storage of conditioned solid and solidified waste.

Tasks which were planned to be implemented within the framework of the *Working Programme* were divided into common branch-related tasks that ROSENERGOATOM was assigned to and individual tasks on perfection of the systems of radwaste management of the NPP sites.

The financing of these activities had to be performed on the non-budgeted programme of scientific and design work, which is included in the cost-price of power production, and using funds connected to transportation, processing and storage of radioactive waste. It was planned to create a special fund to finance the NPP radwaste and spent fuel management, and to include these costs in the electricity prices.

It was anticipated to finish the activities with regard to branch-related and individual tasks on the development of technology and equipment by 1995-1996, and their introduction at NPP sites by 1998. It was also planned to equip all NPPs with the necessary equipment for radwaste management NPP within this period, and to implement the new developed technologies for radwaste processing and storage at NPP.

Unfortunately, since the *Working Programme* had passed, no essential progress has been made in providing Russian NPP with facilities for the conditioning of radioactive waste, mainly caused by lack of available finance. Nevertheless, this programme has played a positive role: the reports of the NPPs about the progress of the implementation of the

Working Programme and the periodic information by ROSENERGOATOM on the problems of radwaste management demonstrate the correct understanding of the importance of the task and the necessity of safe management of radioactive wastes.

At the NPPs attention is paid to minimising the generation of radioactive waste. This is demonstrated by proposals of NPPs directed to the solution of this problem, including the revision of operating procedures of the primary and secondary coolant circuits.

The basic purpose of the *Working Programme* established in 1993 was to increase safety and reliability of radwaste management by the introduction of equipment for the radwaste treatment - first of all for conditioning - and by organisation of the storage of conditioned waste in a way to enable later transfer to final storage. However, during the time that has passed since the *Working Programme* came into effect, significant progress was made in the development of segregation procedures and of radwaste treatment with the aim to extract inactive components from the radwaste, and to re-use the former waste after the cleaning from radionuclides in NPP and in the industry, or for disposal on common industrial waste disposal sites. Moreover, a technology (polishing filters) was developed to polish sewage from typical radionuclides. This procedure allows the prevention of contamination of sewage fields and natural objects in areas where technological sewage is released. The technology has passed tests in some Russian NPPs and recommended for implementation. The introduction of this technology enables a radical reduction of the quantity of liquid radwaste requiring storage and disposal.

The Council of Scientific and Technical Co-ordination (CSC) of ROSENERGOATOM has required the revision of the *Working Programme* with regard to the newly developed technologies and methods to solve these problems in the field of radwaste management. This is one of the reasons why it is necessary to develop a new system of TOOTD.

Supporting the recommendations of CSC, the managers of ROSENERGOATOM decided to modify the *Working Programme*. The activities to revise the Working Programme were started in the second half of 1996.

The new edition of the *Working Programme* is based on organisational and technical measures and scientific and technical decisions to reduce the production of liquid radioactive waste that must be treated, and to allow the extraction of non-radioactive components from the radwaste.

Essential additions to "Working Programme 93" are the technology and equipment for selective extraction of radionuclides from radwaste evaporation remains allowing the classification of the salt remains to the category of non-radioactive waste, as well as the equipment and methodology for the control of the residual activity of the radioactive waste.

The re-use of some components in NPP also should be aided by TOOTD of plant-operating organisations.

The activities stipulated by the *Working Programme* will be financed by their own sources of the NPPs and by the centralised fund of ROSENERGOATOM.

The implementation of the *Working Programme* demands the introduction of high qualified technologies which requires a higher qualification of the working staff and also of the management of these activities. The implementation is planned for 2002.

2.2.5 Responsible Organisations

Figure 2-9 shows the organisational chart of the Russian nuclear industry. Special enterprises for NPP decommissioning do not exist inside the MINATOM structure, because the NPP decommissioning is not included in the first priority tasks.

The INTERATOMENERGO Working Group "Decommissioning of NPP and radwaste management" is playing a specific co-ordinating role in this field. This Working Group unites the interested Russian (and foreign) companies, for instance:

- DECOM Engineering Moscow, a private Russian company
- NIKIET Moscow, involved in special decommissioning jobs for RBMK reactors
- VNIPIET St. Petersburg, working some special fields, for example wet spent fuel storage
- NIKIMT Moscow, working for RBMK-reactors
- VNIIAES, the responsible scientific institute in the MINATOM structure
- ATOMMASCH Wolgodonsk with its Sverdlovsk Branch, working in the field of special remote handling tools
- SVERDNIICHIMMASCH Ekaterinburg, experienced in the field of large decontamination and melting facilities
- AEP Moscow, one of the designers of the Russian NPPs
- Russian Research Centre Kurchatov Institute, experienced in the field of decommissioning of research reactors

and others.

The RADON company must be also be noted, because this enterprise is responsible for the radwaste storage of civil organisations outside the nuclear power industry.

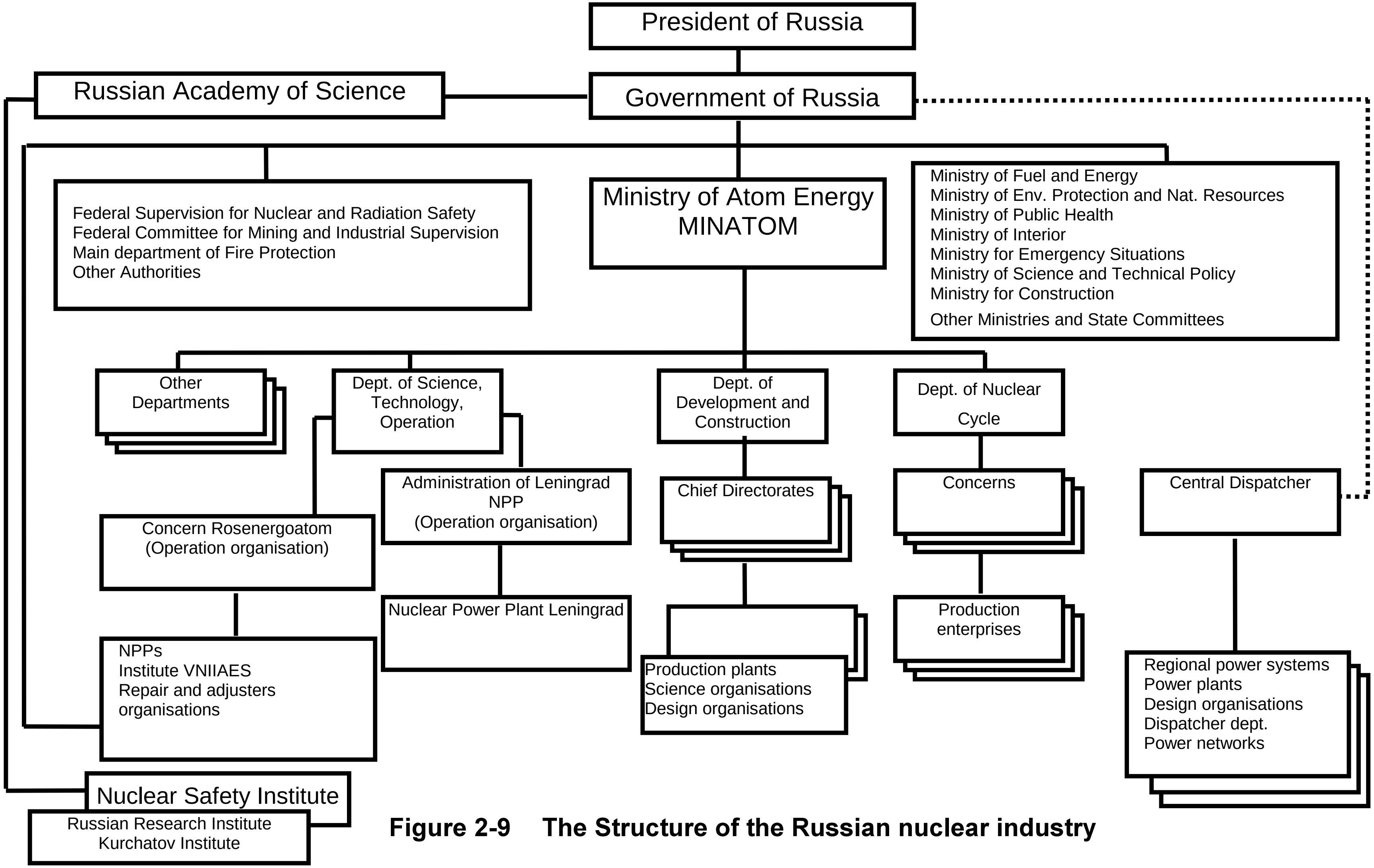

Figure 2-9 The Structure of the Russian nuclear industry

2.2.6 Financial Assurance

2.2.6.1 NPPs

With the purpose to ensure a steady centralised financing of decommissioning of NPP an all-branch uniform decommissioning fund was established at January 1, 1991 inside the Ministry of nuclear engineering and industry of the Soviet Union (EFS) on the basis of submissions from the cost price of production at working NPPs (about 2% of the electricity cost). All sales from dismantled clean equipment, metal, devices and other material assets of the shut down units would be directed also to the EFS. The presence of a uniform fund for NPP decommissioning NPP allowed them to plan the necessary tasks. The EFS would finance all works connected with the NPP decommissioning including research, tests, design, constructional and other works connected to manufacturing of the equipment, to adaptations, to manufacturing of special tool for dismantling and other technological purposes. The EFS was not subject to use for other needs.

The specification of financing at EFS depended on the calculated average annual costs for decommissioning of NPP, including costs for performance of tasks identified earlier, and on the design life time of the units, referred to the established capacity of the unit. The submissions in EFS were made from the operating NPP monthly during the life time of the units, beginning one month after the achievement of the nominal power of the NPP unit.

These rules are still existing in Russia and in the other CIS-countries with working NPPs. It must be added however, that the EFS is very small up to now, because of the financial crisis in the CIS-countries. For instance, in the Russian Federation the Leningrad NPP could get only 7% of the price of the produced energy in actual currency. About 25% of the total price were paid by barter in the form of goods and services, and the some of the large consumers have not paid.

On a final note it should be noted that this financing basis was only established in the early 1990s. No funds were established for the NPP operations before this period. Currently, the current Russian and Ukrainian government do not accept their financial responsibility from the establishment of decommissioning funds for NPP operation in the former times of the Soviet Union. Obviously, this leaves a significant gap in the funds available for decommissioning and provides an overwhelming consideration for the lifetime extension of nuclear facilities or their deferral of major decommissioning tasks such as core removal.

2.2.6.2 Research Reactors

The funding for decommissioning of research reactors is difficult to establish. All institutes operating research reactors were originally state financed institutions in the former Soviet era. They received financing for the undertaking of research works, capital construction and other works, but these institutes did not accumulate any finance for others (off-schedule or unforeseen) work. In the conditions of transition to a market economy these institutes have become independent and have received the status of operating organisations, but have not yet achieved the necessary financial security.

According to the Federal Law, recently accepted in Russia "On the use of atomic energy" the State has transferred on to the operating organisations of research reactors the duty to carry out decommissioning activities using their own forces or with other organisations.

Now that these organisations have economic independence by carrying out research and other applied works for external customers they could form the specialised funds for decommissioning of research reactors, stipulated by the Federal law. Realistically, the work of the institutes does not allow them to accumulate the required financing in these funds up to now. Moreover, as many of these research reactors are old there has not been enough time to accumulate significant funds.

Obviously, it is very difficult to carry out the decommissioning of research reactors without the financial help of the state.

2.3 National Decommissioning Strategies

2.3.1 Russian Federation

<u>2.3.1.1 Nuclear Power Plants</u>

The regulatory requirements for decommissioning in Russia are still being formulated, having regard to international guidelines and recommendations, and are rather incomplete. However the current regulations require that the owner of a reactor should generate a Decommissioning Plan some five years before final shutdown to demonstrate that the plant can be managed in a safe manner during decommissioning. The basic normative document of Russia defining the safety requirements for all stages of the life cycle of NPP units, including the decommissioning stage, is ‘General Safety Rules of nuclear plants’ (OPB AS-88/97).

The basic approach to the decommissioning of nuclear installations was originally developed by the Ministry of atomic energy and industry (Minatom) of the former Soviet Union. In 1991 [2-6] a document defining a conceptual approach to the decommissioning of NPP units was prepared and approved, which sought to define a general strategic approach to the planning and decision process for the decommissioning of NPP units.

In this conceptual scheme, the decommissioning of NPP units is defined as a complex task covering a wide range of activities, beginning with the run-down of operation of the unit as a producer of electrical energy, and ending with the complete dismantling of each unit and returning the site to the status where it is suitable for use for (un)restricted industrial-economic aims. It is required that the ecological consequences for the NPP site should be minimised during and after the decommissioning.

The Decommissioning Concept of NPP units in Russia currently still applies but is being reviewed in light of current conditions in the CIS countries and is based on the basic principles below [2-6]:

1. The decommissioning of the NPP unit (or of the whole nuclear plant), is generally planned after the end of the nominal (designed or specified) life time or in case of the technical impossibility to ensure its further safe operation.

The specified lifetime of the NPP unit is understood as an estimation of an opportunity to extend the lifetime of its operation over the nominal period. A standard nominal lifetime of a NPP unit is 30 years. It is foreseen to extend the lifetime for 5-10 years, taking into account the following circumstances;

Firstly, the decommissioning of the NPP unit is a highly expensive process. Secondly, the construction of a new unit with the replacement of generation capacities is extremely difficult now in Russia. Besides, existing estimations say that the life time extension of a 1000 MW unit for 10 years can give a pure economic benefit from 1,000 up to 2,800 million US dollars.

Certainly, this favourable measure of lifetime extension must be based on the appropriate safety guarantee programme and must be technically substantiated. This programme should provide an improvement of the control and quality of operational service as a compensating measure to ensure the existing or improved safety level. In addition, systems must be replaced which are important for the safety of the NPP unit.

Practically, the financing of the life time extension of the unit should be carried out on the basis of self-financing. Some additional expenses for the operation could be compensated by the improvement of operating efficiencies and by the decrease of expenditure for the overhaul and replacement of equipment.

In the opinion of Russian experts, the major element in the problem of lifetime extension of the NPP unit is the problem of the codes and standards regulating this process. This development will require serious scientific studies for a reliable forecasting of the safe operation of systems and equipment of the unit, that is

foreseen for the extended life time, especially for units with reactors of the first generation.

2. The decommissioning planning must be based on the replacement principle (complete restoration) or replacement of the removed power production by new, advanced and safer units. This is a typical requirement of the Russian nuclear industry. This would avoid a reduction of the total installed power production of the NPP, and transfer the Russian nuclear energy to a higher level of operational reliability and safety.

The reason of such approach is; the expediency to utilise the existing NPP infrastructure; the preservation of qualified operational staff; and the requirements of the achieved energy production of the country, all of which are dictated by the difficulties with selection and substantiation of new NPP sites, etc.

3. Maximum use of NPP sites which are decommissioned. Therefore it is necessary to accept all measures to fix available sites of operating NPPs and firstly the NPP sites at which units are foreseen for decommissioning. This should conform to the needs of nuclear energy of the country providing the characteristics of these sites and meet the modern normative requirements and criteria from the point of view of radiological and ecological safety of the public and environment.

Maximum use of buildings, components and equipment of decommissioned units, (or NPP as a whole). It is planned to use buildings and components of decommissioned NPP units (mainly of the first units of multi-unit plants) with the intention to modify them to long-term or interim storage facilities for radioactive waste produced during operation and decommissioning.

4. Conversion (re-use) of decommissioned units.

   At the stage of decommissioning planning of separate units (or the whole NPP) it is planned to take into account the opportunity of modification of the specified NPP units for other practical purposes, and first of all, for reuse as nuclear energy.

   It is planned to examine variants of possible modifications of the NPP units to nuclear heating plants or modification of a NPP to a power plant using organic fuel with use of separate buildings, components both part traditional power and auxiliaries at the NPP site, identical for thermal power plants.

2.3.1.2 Research Reactors

Currently in Russia a state concept for the decommissioning of research reactors is not yet developed.

At the same time, the normative documentation, which is available now in Russia, regulating the decommissioning of objects in the nuclear energy and industry generally, recommends two variants regarding the decommissioning of nuclear objects:

- immediate dismantling
- deferred dismantling.

A complex engineering and radiation inspection (CERI) has to proceed before decommissioning of the research reactor, which is carried out with the following objectives:

- Drawing up of a map of radiation fields in all rooms, and also inside the biological protection of the reactor;
- Investigation of the nuclide structure of radioactive contamination in the basic components of the reactor, of the equipment and pipelines;
- Analysis of the status of the systems ensuring the safety of the shut down reactor;
- Additional analysis of the status of building structures;
- Analysis of the status of the metallic reactor structure, of the storage for spent fuel assemblies (FA) and solid radwaste.

Alongside it the social questions of the employment of the operating staff should be resolved together with the question of the use of structures and systems after decommissioning (reuse). The results of the CERI should be used as initial data for the development of the decommissioning documentation for the research reactor.

It is necessary to complete the following tasks for shut down and decommissioning of research reactors:

- To develop and to approve the decommissioning programme of the research reactor (this concept must also be approved by the regulatory authority);
- To develop and to approve the decommissioning documentation of the research reactor (the documentation has to include a section "Justification of the safety of the reactor decommissioning");
- To prepare a storage for reception of spent fuel and radwaste, if it was not undertaken during normal operation of the reactor;
- To make and to co-ordinate the appropriate instructions on service of the research reactor in case of its storage with surveillance.

2.3.2 Ukraine

A decommissioning concept is only available for the NPP units in Ukraine. Information about the plans for research reactors are unavailable, information being difficult to obtain because of the difficulties and changes inside the administration of the Ukrainian scientific research activities.

The decommissioning concept in Ukraine is very similar to that in the Russian Federation - the concept of decommissioning Ukrainian NPPs being based on the following principles:

1. Decommissioning an NPP unit is implemented in order to exclude the possibilities of its further usage for the purposes for which it was primarily constructed. The ultimate objectives of the decommissioning process can be to put the site to the state that allows its limited or unlimited application.
2. The status of the site to be achieved is identified by the operating organisation after shutdown of the NPP when performing feasibility studies for the decommissioning strategy (the preferred option) within the decommissioning programme. In order to identify the final status of the site it is necessary to have information about the final radiological state of the unit, the potential of the further site application for industrial needs, the latest information about specific technical means and technologies as well as the most reliable forecast with regard to the financial requirements at the moment of expected dismantling of a reactor unit.
3. In case the site is put into the state allowing its (un)limited application, all buildings, structures and equipment from the decommissioned units can be used for the industry needs including:
    - converting the buildings into the temporarily radwaste storage facilities of different radiation levels;
    - construction of interim storage facilities for spent fuel and radwaste
    - construction of new units with the improved level of safety;
    - foundation of a research base for performing design and scientific studies to solve nuclear industry problems in the framework of branch-oriented scientific technical programs.
4. In executing the activities for decommissioning NPP units, safety assurance (radiation safety, general engineering and fire safety) is regarded as one of the main objectives.
5. In executing a decommissioning process it is necessary to effectively use the NPP infrastructure, skilled and trained personnel qualified in specific fields and knowledge as well as the peculiarities of multi-unit structure of NPPs in Ukraine including:

- an overall system for NPP site monitoring;
- some general technological schemes;
- the premises, equipment and systems of the unit to be decommissioned for the NPP needs.

6. Activities to provide an engineering support function and to plan a decommissioning process are to be executed in the framework of branch-oriented programmes.
7. All arrangements, engineering measures and activities related to decommissioning during all stages and phases of the process are to be adequately financed in a timely manner.

The concept suggests the preliminary priority of all stages of a decommissioning process (compliant with the new normative requirements on ensuring safety when decommissioning). This priority is to be considered as a basis for development of decommissioning alternatives as well as identifying the anticipated terms for performing separate works and measures. For example, feasibility studies are to be performed in terms allowing submission of inspection results and proposals related to termination or extension of operation at least 5 years before the design unit lifetime is finished.

It is required either for executing a complex of measures to extend the duration of operation (equipment replacement, modernisation and upgrading measures, performance of required analyses, etc.) or to prepare a facility for decommissioning (development of the "Programme of removal a facility from service", other required documentation etc.).

The concept under development can be applied not only in case of planned decommissioning but also in case of unplanned decommissioning of an NPP unit provided the 'beyond the design accident' did not occur. The solution with regard to the closing down or the extension of the unit operation should be based on the principle of cost-effectiveness. According to this principle the solution about the unit decommissioning should be accepted in case the costs required to ensure a safe unit operation for a certain period of time exceeds the anticipated profit to be obtained for energy sold during the mentioned period.

The problem of radwaste management and disposal is closely associated with the decommissioning of nuclear facilities and their commissioning as well. That is why the work to develop concepts to solve this problem started at the beginning of 1990s which resulted in establishing the document "The concept of NPP radwaste management in Ukraine" by GOSKOMATOM of Ukraine.

2.3.3 Kazakhstan

A decommissioning concept is only available for the NPP unit BN-350, the only existing NPP unit at Kazakhstan. Information about the plans for research reactors is unavailable.

According to the general safety rule OPB-88, the operating organisation should submit the decommissioning documentation to the State Regulatory Authorities no later than 5 years before the finish of the normal lifetime of the NPP. The decommissioning should be preceded by a CERI by a commission nominated by the operating organisation. The results of the CERI will be the basis of the decision on shut down of the NPP.

The alternatives of decommissioning will be considered based on the following criteria:

- Maintenance of nuclear and radiological safety of the public and the personnel
- No environmental pollution by radwaste, long storage of radioactive parts of the reactor facility
- Minimisation of dose rates during realisation of decommissioning activities and disposal of radioactive equipment, not exceeding sanitary radiation norms for repair jobs.
- Minimisation of economic expenses on decommissioning and preservation of the reactor facility, taking into account the probable economic efficiency from re-use of exempted equipment
- Maintenance of the radiological safety under all circumstances, taking into account probable natural hazards (earthquakes, floods, hurricanes, etc.)

At the present time the decommissioning alternative of storage under surveillance is preliminary accepted. The storage is planned for 30-50 years. The decision about the decommissioning variants depends on the following problems:

- Further assignment of the buildings and structures of the reactor facility
- The method of preservation of the reactor facility and further conversion of the equipment and coolant.

Despite the fact that the last 5 years of the BN-350 normal operation has begun, it seems that the activities to prepare the decommissioning have started only slowly.

2.3.4 Others

Information about the decommissioning concept for research reactors in other CIS-countries was not available.

2.4 Current State of Decommissioning Works

2.4.1 Overview

NPP units, shut down at present time, are:

- in Russia
  - units 1 and 2 at Beloyarsk NPP
  - units 1 and 2 at Novovoronezh NPP
- in Ukraine
  - unit 4 at Chernobyl NPP after the accident 1986
  - unit 1 and 2 at Chernobyl NPP[3]
- in Armenia: the unit 1 at Metsamor NPP after the earthquake in 1987 (the unit 2 was re-commissioned in 1996)

NPP units, foreseen for shut down up to 2005, are [2-2, 2-3]

- in Russia
  - units 1 - 4 at Bilibino NPP
  - units 1 and 2 at Leningrad NPP
  - units 3 - 4 at Novovoronezh NPP
  - units 1 and 2 at Kola NPP
- in Ukraine: the Chernobyl NPP
- in Kazakhstan: the BN-350 NPP at Aktau

It is important to note that the intentions and plans are changing with time and from organisation to organisation so this is just a current snapshot of the situation.

Table 2-7 shows the reasons of the closure of NPP units and the present state of decommissioning work at these units [2-7]:

Table 2-7 Reasons for shut down of NPP units

NPP Unit	Reason for closure	State of decommissioning work
Beloyarsk NPP unit 1	problems with the reactor channels	Preparatory stage, building and equipment are partially isolated
Beloyarsk NPP unit 2	problems with the reactor channels (meltdown of fuel assemblies)	Preparatory stage, building and equipment are partially isolated
Novovoronezh NPP unit 1	decision of the Minatom	Preparatory stage; engineering and radiological survey and feasibility studies completed
Novovoronezh NPP unit 2	no operating permission after 1990 decision of the Minatom	Preparatory stage; engineering and radiological survey and feasibility studies completed
Armenia, unit 1	earthquake in 1987 caused shutdown	Preparatory stage; engineering and radiological

[3]The Chernobyl NPP is not included in the content of this project. Therefore it is not described in the other chapters.

The situation in the field of research reactors is more difficult to define. Only 8 reactors are closed up to now, but 32 are older than 25 years. This reflects the more flexible nature of these facilities and the ability to re-configure the reactors. The situation in Ukraine, Belorussia, Georgia and Uzbekistan is not clear. Some reactors are closed or are in a "repair" status. From discussions with Local Partner experts, it can be said that approximately 10 - 20 research reactors would be decommissioned in the next ten years.

2.4.2 Nuclear Power Plants

The situation in the field of decommissioning is described below for the decommissioning candidates.

2.4.2.1 NPP Beloyarsk (Russia)

The present state of the NPP units 1 and 2 is that they are finally shut down [2-8]. The situation is as follows:

- Both units are operating on the basis of the instruction 'Shut down for repair conditions'
- At unit 1 the equipment and the pipelines are preserved against corrosion by liquid inhibitors
- At unit 2 the main equipment is dried.
- The reactor cores are defuelled.
- The spent fuel is located in the spent fuel cooling pond at unit 2. The reason is that reprocessing facilities for AMB spent fuel do not exist in the Russian Federation because of missing corresponding technologies and equipment. This problem is a general one for all graphite moderated channel pressurised water reactors in Russia and in the former Soviet republics. Moreover, some of the spent fuel assemblies are damaged
- The systems and facilities are working for:
  - supply of energy, water, and heating
  - radiation protection
  - spent fuel cooling ponds
  - engineered barriers against release of radioactivity
  - infrastructure of both units

- The main control room of unit 2 is working for both units.

The main working steps are described for the following period. These working steps also characterise the main existing problems:

1. Management of nuclear spent fuel and of damaged nuclear spent fuel
2. Preservation of units 1 and 2 on the basis of the existing conception developed by NIKIET
3. Processing of operational and decommissioning radioactive waste
4. Construction of:
    - on-site storage of solidified liquid radwaste
    - on-site storage of solid radwaste
    - interim storage of metal bullion smelted in the existing experimental smelter

The most urgent problem at the present time is the problem of the nuclear spent fuel. The fuel channels are corroded and some channels are damaged. The spent fuel cooling pond is corroded and the cooling water is contaminated. The danger of radioactive release from the cooling ponds is real. Therefore a project entitled: ‘Feasibility study and technology transfer for Beloyarsk NPP on long-term management of spent fuel (R4.01/95)’ is included in the Tacis programme of the EC.

2.4.2.2 NPP Novovoronezh (Russia)

The present state of the NV NPP units 1-2 (the units 3-4 are in operation, the decommissioning preparation has not started) is that they are finally shut down [2-9]. The situation is as follows:

- The systems and facilities, necessary for the service of the units, are defined and in operation. The remaining systems are disconnected. The design of this work step was approved by the Head Designer of the NPP AEP Moscow.
- The new Procedure for operating of the units 1 and 2, taking into account the current stage, is developed, approved by the Authority and implemented.
- The development of technologies and equipment for underwater cutting of high-activated equipment is finished. Cutting is ongoing for conditioning of absorber and control rod drives. The development of technologies and equipment of conditioning of in-core-equipment is ongoing.

- Two facilities of high concentration evaporators have been commissioned. This makes the site able to treat the liquid radwaste generated during operation and during decommissioning.
- The complex engineering and radiation investigation of the units 1 and 2 is finished.
- The transfer of the spent nuclear fuel is not completed.
- The development of the documentation of the first stage for preparation of units for storage under surveillance is ongoing.
- Individual works for decontamination of the equipment, for dismantling (generally, equipment of the turbine hall) and modifications of rooms for the storage of radioactive waste are ongoing at the units. These tasks are carried out under individual documentation. The reconstruction of the buildings and rooms of the units 1 and 2 is ongoing with the aim to prepare an interim storage for drums of waste, generating by the high concentration evaporator UGU-500. The dismantling of the turbines n° 1-3 of unit n°1 was started in February 1996 after the approval of the Authority.

In 1996-97, technical and economic investigations were carried out for variants of modification of the 2nd unit with the aim to place into the confinement a modular reactor for fast neutrons with a lead-bismuth coolant and the use of a part of the existing equipment of the secondary circuit for electric power production with an established capacity ~ 300 MW. These investigations were carried out in connection with a delay in the commissioning of replacement electric power generation at the NV NPP site.

Therefore, the dismantling works on the 2nd unit are suspended up to the completion of the technical documentation and up to the final decision about the opportunity and expediency of the modification.

The nominal lifetime of the units n° 3,4 at the Novovoronezh NPP are:

- unit n° 3 - December 27, 2001,
- unit n° 4 - December 28, 2002.

In discussions with the operators and taking into account the reliable operation of the units, the planned amounts of works for safety improvements, the preliminary forecast of the residual life time of basic elements of the units, and also the delay of commissioning of replacing electric power generation in the region it is forecast that the technical opportunity and economic feasibility of the life time extension of the units n° 3, 4 at the Novovoronezh NPP is likely.

Now, according to the order of the Minister of Russian Federation on nuclear energy the Programme is developed to prepare the units n° 3,4 Novovoronezh NPP for the life time extension. The program includes two directions of works:

- safety improvement;
- definition of the residual lifetime and reassignment of the resource characteristics of the basic elements of the units or their replacement, if their resource is limited.

It is planned to develop the decommissioning documentation of the units n° 3, 4 during their operation at the extended life time. In case of positive experience from the modification of the 2nd unit a similar decision is most probable for the following stage of the life cycle of the units n° 3, 4 at the Novovoronezh NPP. No decisions are made for unit 5.

2.4.2.3 NPP Leningrad (Russia)

All units of the Leningrad NPP are in operation. A modernisation programme is partly finished and partly ongoing. The normal lifetime of the two oldest units, 1 and 2, the forerunners of the whole RBMK-serial, will end in 2003/2005.

In 1992 Leningrad NPP was given the status of an independent operating organisation, therefore the NPP has received the right of independence in the field of decisions of various organisational, technical, economic and financial questions. From this time the Leningrad NPP is now completely responsible for safety in both operation, and decommissioning of the units regulated by GAN.

Leningrad NPP is a state-owned enterprise, but at the same time the NPP is completely self-financed, not receiving state financial support even to realise the current needs of necessary reconstruction of units to ensure their sufficient safety level. In this connection the strategy originally was accepted to ensure the financing of the decommissioning works of units at the Leningrad NPP, to continue the balance between commissioned and decommissioned capacities.

The choice of a similar strategy was justified by management taking the position of preservation of the achieved level of energy production in the region, and the solution of social problems, which will inevitability arise during decommissioning of the NPP units. The Leningrad NPP management together with other designers and constructors has developed this aim by identifying various possible alternatives of replacement of decommissioned units.

However, considering the facts that the nominal life-time of the first unit at the Leningrad NPP will end in 2003, and the construction of the new replacing unit requires from 8 to 10 years both significant material and financial assets are not available at Leningrad NPP now or in the near future. Taking into account this situation and the economic conditions, which currently exist in Russia, the strategy described above was subjected to auditing and was revised. Now the planned decommissioning strategy of the first and second units at the Leningrad NPP is directed to extend the life-time of their operation for a period, which would allow to accumulation of the required material and financial assets for their decommissioning.

Some decommissioning preparation work is ongoing (including international projects).

2.4.2.4 NPP Aktau (Kazakhstan)

The NPP Aktau is based on the fast reactor facility BN-350 with sodium coolant. The normal lifetime suggests that the facility will be shut down in 2003. The reactor is currently in operation.

Real decommissioning preparation works are not ongoing. The development of initial research studies was started in 1990/92. It must be expressed that no experience is available for large sodium cooled fast breeder reactors. The current R&D works have the character of a feasibility study.

In the period from 1992 to 1996, the work on the development of the decommissioning concept was interrupted. In October 1996 at the IAEA headquarters, an international meeting for the development of a effective programme for decommissioning BN-350 was held. It was confirmed that the prospective time for final shut down of the BN-350 would be 2003.

After this meeting the development of the concepts was continued, and now the following tasks are realised:

- A special group of experts for the development of the decommissioning concept was created at the NPP.
- First consultation meetings with experts of IPPE (the scientific leader of the NPP) and VNIIAES were arranged. VNIIAES has developed the draft of the technical specification of decommissioning activities.
- The Government of Kazakhstan finances works on the feasibility study of replacement of capacities of the BN-350 unit.

2.4.3 Research Reactors

Further information is also given in Reference [2-10].

Alongside the end of the nominal lifetime, there can exist other factors, which should be taken into account for decisions of the final shut down and decommissioning of research reactor. Such factors could be the following:

- *Low efficiency of further research reactor operation*: there can be a number of economic reasons, on which could depend the decision of further operation of the research reactor. In this case, the operating organisation is obliged in the normal way to submit the application to the National Nuclear Regulatory Authority about the shut down and decommissioning of the research reactor.
- *Technical imperfection of the research reactor that do not guarantee the safety of its operation*: in this case, as a result of the inspection carried out at the research reactor and subsequent decision of National Nuclear Regulatory Authority the operating organisation should decide to upgrade the research reactor to such a condition at which the level of safety of its operation would be accepted by the National Nuclear Regulatory Authority. Alternatively they could decide to shut down the reactor finally and to begin preparation for its decommissioning.

It is necessary to take into account the following circumstances in the field of decommissioning of research reactors in Russia:

- research reactors do not normally exist separately, and they are operated in structures of large research centres having different research programmes, and also other nuclear and radiologically installations (hot metallographic chambers, critical stands, accelerators etc.);
- research reactors are frequently located near the centre of inhabited areas, that determines difficulties with transport of the nuclear spent fuel and contaminated equipment;

Currently in Russia a state concept for decommissioning of research reactors has not yet been developed. The same situation exists in the other CIS-countries, but with a lower number of research reactors.

The average age of nuclear research reactors in the CIS is significant, therefore the decommissioning process of Soviet Union research reactors started a long time ago. The first experience on a research reactor decommissioning in Russia was connected, as a rule, with the reconstruction of old reactors and their transformation to others research reactors. In some cases the reconstruction included a complete dismantling of all in-core devices.

However the decommissioning experience of research reactor is not extensive in Russia. It is necessary to note, that in general no research reactor has been decommissioned to a Stage 3 according to the IAEA definition in Russia now. One exception could be the demonstration reactor type WDNKH at Moscow.

At the present time the technical policy for research reactor decommissioning in Russia covers, basically, such aspects, as analysis of possible variants of research reactor decommissioning, choice of the final stage of the decommissioned research reactor, definition of principles to guarantee radiation and ecological safety during the research reactor decommissioning period and others.

Complicating the circumstances for planning and realisation of research reactor decommissioning works is their localisation in or near to residential areas. Because of these conditions the long-term localisation of research reactor (some tens or more years) before their dismantling could be unacceptable from the point of view of the reaction of the public living near to it, independently of the fact that it could be maintained in a safe radiological condition for such long-term storage.

The state of the shut down Russian research reactors is described below:

- Reactor VVR-2, RRCKI

This was shut down 1983 after an operating lifetime of 29 years. The reason for its shut down was partly for economic reasons caused by operating problems and partly by public protest regarding the location of the reactor inside the Moscow city

The reactor is now defuelled and the majority of equipment dismantled. Parts of the reactor were used for further facilities.

- Reactor RFT, RRCKI

The reactor was shut down in 1962 after operating for 10 years and a new reactor was built to replace it within the same building. Fuel has been removed and equipment dismantled. The steel reactor vessel together with graphite was filled with concrete

- Reactor MR, RRCKI

The reactor was shut down in 1993 because of economic problems and it was not possible to reconstruct the reactor so that it would meet the current safety requirements

The first phase of decommissioning is finished: the reactor is defuelled, after decay in the cooling pool it was transferred to the dry storage

- Reactor IRT, RRCKI

The reactor was shut down in 1979 as construction of a new reactor IR-8 began in the building of the reactor IR. Fuel has been removed and in-tank equipment is dismantled.

The building and equipment are used for the construction of the new reactor IR-8.

- Reactor “Romashka, RRCKI

The reactor was shut down in 1966 as the research programme was finished. Currently the reactor is being planned for decommissioning.

- Reactor TVR, ITEF

The reactor was shut down in 1986 because it was not possible to carry out the research of some constructions under the conditions of high neutron fluence. The public opinion after the Chernobyl accident was directed against the operation of low safe old research reactors located inside cities.

Currently spent fuel has been removed and transported to reprocessing facilities. The heavy water moderator and cooling medium: contaminated by tritium is currently located inside the reactor building

- Reactor “ARBUS”, RIAR (NIIAR)

The reactor was shut down in 1989 after an operating lifetime of 26 years because the research programme finished. The reactor has been defuelled.

2.4.4 Mayak

2.4.4.1 Brief history of Mayak

The arrangements at Mayak have changed little during the 50 years of its existence (it celebrates its 50th anniversary shortly). It is located at Ozyorsk, a ‘closed city’ with about 100,000 inhabitants, all of whom are directly or indirectly involved in the Mayak plant. In addition there is an ‘offshore community’ living outside the security fence, but enjoying some of the same tax privileges. Its original mission was to produce metallic Plutonium for the weapons facility at nearby Snezhinsk: during the late 1970s it expanded its mission to undertake reprocessing of spent fuel from VVER, Fast Reactor and submarine reactors, and during the past 35 years it has also operated a radio-isotope production facility. Starting in 1987, it has progressively closed down its 5 graphite-moderated reactors (the last, AV3, being closed in November 1990). This was primarily in accordance with US-Russian agreements on the closure of Pu-producing facilities: however the condition of the reactors had been deteriorating rapidly in the last few years of operation, and it would have been increasingly difficult to have kept them going. In the early days, weapon production was given over-riding priority, and a number of environmental mistakes were made, notably the discharge of liquid wastes from the reprocessing plant into the Techa River and into a nearby swamp, which expanded into a 51 hectare lake, Lake Karachai. In spite of substantial national and international efforts, the resulting contamination problems remain unsolved problems. In April 1967, Lake Karachai partially dried out, and a tornado spread dried out sediments containing some 600 Ci up to 70 km. This followed another serious incident in 1957, commonly known in the West as the Kyshtym Disaster, though it in fact happened within the Mayak plant and not at the neighbouring town of Kyshtym, in which a tank holding liquid active wastes with a high nitrate content overheated and exploded, spreading contamination along a 30 km plume, and requiring the evacuation of 10,000 people.

The plant is currently spread out over several square kilometres, and has one operational reprocessing facility (RT-1), two large operational isotope-production reactors (Ruslan and Liudmila), with associated sealed source manufacturing facilities, and a major mechanical workshop, as well as numerous redundant facilities from an earlier era. RT-1 is currently operational, although it has been closed down for about a year on orders from GAN, because the associated vitrification plant for the liquid waste had failed. However they managed to persuade GAN that they had two years of spare tank capacity left, so were allowed to restart recently. Reconstruction of the vitrification plant is going ahead.

2.4.4.2 Graphite reactors

The story of Mayak's graphite reactors begins on 19 June 1948, when the first reactor A went critical. Shortly after that the first radiochemical plant for plutonium separation, and the first metallurgical plant for plutonium metal production came into operation. During the following 4 years, a further 4 reactors were brought into operation - AV1, AV2, AI and AV3 (the last two in 1952). All this work was carried out under the direct supervision of Kurchatov, and subsequently Alexandrov. These reactors all had the same channel size (7600 mm length) but variable numbers of channels (A 1000, AI 150, AV1-3 3000). The stacks were built up out of graphite blocks of length 600 mm. They continued operation until 1987(AI & A), 1989 (AV1), 1990 (AV2&AV3). By the date of closure, all were suffering from severe swelling of the core - in the case of reactor A this had reached 140 mm increase in diameter by its date of closure, and had burst the restraints, caused cracking of 80-90% of the blocks, and was making channel withdrawal very difficult. In each case, defuelling was complete within 6 months of shutdown. During the past 10 years there have been studies, and some limited action, on the decommissioning of these reactors, which are reported in a 1990 paper by Strakhov et al, with participation of VNIPIET, NIKIET and IAE. These studies included neutronic calculations of activation (validated to within 30% by local measurements), sampling of core material, which showed that the dominant activity is C-14 (about 3 MBq/gm - it is specially high because they used N2 to prevent radiation-enhanced oxidation of the stack as the temperature of operation was raised to control swelling of the stack), stack monitoring, which showed that emissions are below 0.25 and 0.2 MBq/month respectively for Cs-137 and Sr-90, and groundwater monitoring, which has shown negligible contamination compared with the (somewhat abnormal) background.

The areas of potential concern are: progressive deterioration of the support structure , build-up of radiolytic hydrogen, leading to an explosion, and sudden release of the trapped Wigner energy. To address the first concern, they are considering injecting concrete into the voids below the base of the core, though this is controversial (Tomsk has done this on its reactors). As regards the second, they have done some studies which suggest that there is no problem. The third concern has been the subject of some limited theoretical & experimental investigations, leading to the conclusion that there is probably no problem.

A feature of all these reactors is that the core is wholly below ground level (the foundations are at about -55m), and there is a generous biological shield, which has been shielded from activation by a water jacket. Ground water leaks into the cavity, and is pumped out. There is a tall above-ground structure, with an overhead crane to permit replacement of the 2000+ vertical water-cooled Al channels within which the fuel is loaded. Currently the charge face is as it was during operation, but there is a plan to cover it with a shield plate (so far only done on reactor A).

<u>2.4.4.3 Environmental problems</u>

The overall scale of releases from Mayak to the environment of long-lived radionuclides (Cs-137 and Sr-90) has been very high (8900 PBq - ie 680 MCi) - as compared with atmospheric testing (1550 PBq), Chernobyl (70 PBq) or Sellafield (47 PBq). Particular components of their environmental problem are:

- Kyshtym accident

  This is Zhores Medvedev’s name for an accident which happened within the Mayak plant in September 1957. There were some 16 stainless steel underground vessels holding liquid wastes containing high concentrations of nitrate & acetate in a concrete canyon with 60 cm thick concrete and a cover plate. Due to a loss of coolant accident, one dried out, and then heated up to 350 C, leading to an explosion. Some 2 MCi were released, leading to dose levels locally of 360 R/hour and a plume of 1-8 km width extending about 30 km NNE, within which all residents were resettled (some 10,000). The isotopic mix had only about 2% of Sr-90 but that has set the lifetime of the contamination, which is now about 150 microCi/m^2. They have in places removed a top layer of 5 cm, and have ploughed about 180 km^2 to a depth of 50 cm, and now monitor regularly and control agriculture.

- Lake Karachai

  This is about 3 km from the reprocessing plant, and expanded over the years to cover 51 hectares with fairly high level liquid waste, with dose levels at the surface of up to 100 R/hour. Following the 1967 typhoon incident, they began a programme to cover it over, thereby pinning down the sediments on the bottom, and preventing wind erosion from the surface, using close-packed arrays of cylindrical bell-like concrete structures which rest on the bottom, and covering the top with soil to provide further shielding and provide access for the lead-shielded vehicles used to deploy them. They have now covered all but the last 13 hectares. This remedial measure does nothing to reduce seepage from the base of the Lake into the ground water and hence into aquifers leading to the nearest river (the Mishelyak). This problem has been studied by BNFL in a DGXI contract, and will eventually become serious if no further measures are taken.

- The River Techa

  This flows through a series of local lakes and then into the Iset and eventually into the Ob and the Kara Sea. In the early days, liquid waste (and cooling water from the rectors) with a total radionuclide content of some 2.7 MCi was discharged directly into the lakes and the Techa, leading to serious pollution of the banks & bottom, and exposure of some 124,000 people. From 1951, this was reduced (when Karachai was used as the main dump), and from 1953 remedial measures were undertaken (including evacuation of 7500 inhabitants from its neighbourhood, and the construction of a series of dams, to localise the pollution and prevent further contamination downstream. The flow of the Techa was diverted into a bypass canal.

- Liquid radioactive waste

  During its 50 years of operation, Mayak has accumulated some 19,000 m^3 of high level liquid radioactive waste (containing some 135 MCi of activity) from its military plutonium production activities, plus some 8000 m^3 (containing some 200 MCi)from its reactor fuel reprocessing plant RT-1. This latter Figure is increasing at about 2-3000 m^3 (100 MCi)pa. Russian publications on the status of the 19,000 m^3 of military wastes are not very informative: however details are given in some Western publications, especially in DJ Bradley's book 'Behind the Curtain' which describes some 90 tanks in five tank farms (see p 383). These wastes are mostly alkaline, and contain sludges with ferrocyanide and aluminium hydroxide, and suffer from the same problems of burping and leakage that have been encountered at Hanford. There are few details on their chemical composition, beyond the vague indications given by Bradley (supernate contains 42-86% sodium and 5-8% aluminium, and possibly they do not have accurate data). Since the 1957 explosion they have been aware of the need for a careful tank management programme. However most of their current R&D is directed towards the acidic wastes which now come from RT-1, and which are the main input to their vitrification programme (though they can admix some of the alkaline supernate).

2.5 Decommissioning Constraints

Finally in this section a discussion is undertaken of the problems or handicaps affecting decommissioning in the CIS. Issues identified earlier are brought together in the section before Section 3 discusses the requirements for decommissioning.

2.5.1 General Overview

The problems in the field of decommissioning of nuclear facilities were discussed with the experts from CIS countries and are identified in the reports procured for the project. They identified a number of constraints that handicap the decommissioning process and include:

- the insufficient financial framework and funds for decommissioning activities
- the lack of a national strategy, for instance, for the decommissioning of research reactors
- the lack of regulatory documents for decommissioning of nuclear facilities
- the lack of radioactive waste processing and disposal facilities
- the lack of national full-scale experience in the decommissioning area
- the lack of or insufficient instruments and tools for dismantling work of high radioactive components

These problems are discussed in the following sections.

2.5.2 Financial Framework and Funds for Decommissioning

Financing is the most difficult and the most important question for all kinds of decommissioning activities in the CIS countries. The transfer from the former central administrated economy with central funds to free market structures has destroyed the financial basis for the accumulation of decommissioning funds. The measures, established for NPPs, are only a start. Up to now it is not clear how the financing basis of billions of

EURO will be created. For a NPP, perhaps, the life time extension of the units on the basis of results of a detailed engineering inspection can be a source to accumulate the necessary funds, if the profit of the energy production will be accumulated for the following decommissioning. However, the current situation, particularly in Russia, exacerbates the funding crisis facing the operating organisations. Payment for electricity sold by the NPPs is usually delayed or a bartering system exists to allow the NPP staff to be paid. Funds for decommissioning are likely to be neglected in the current situation.

2.5.3 Radioactive Waste Management

Final disposal facilities and/or repositories do not exist in the CIS countries [7]. The radioactive waste is stored on-site, mostly at local interim storage facilities. These interim storage facilities include storage for liquid radioactive waste and for solid radioactive waste, mostly bulk stored. The nuclear spent fuel is stored on-site, too. For RBMK reactors spent fuel-reprocessing facilities do not exist and is a significant expense even if it can be used.

The local interim storage facilities and the local radioactive waste processing facilities are insufficient. Therefore, a large volume of untreated radioactive waste exists. An exemption level does exist but is not utilised because there are no rules for the exemption procedures (eg for low active operational waste).

This situation involves a high hazard for the environmental situation at the NPP and research institutions, especially in regions near by or inside settlements and cities. The stores were constructed based on old nuclear and radiological standards. Damages of the liquid radwaste storage tanks can not been excluded.

In summary, the insufficient radioactive waste processing and storage facilities is one of the most important technical handicaps for the decommissioning and dismantling of shut down and closed nuclear facilities now. The real reason for this situation is not the lack of technical capabilities, but the missing financial basis and the missing general strategy for decommissioning.

2.5.4 Experience, Tools, Instruments for Dismantling

Especially in the Russian Federation, a high experience exists in the area of reactor repair and reconstruction work. In addition, large experiences were accumulated during accident liquidation under high radioactive dose rates and contamination. Tools and instruments were developed and fabricated by the national industry.

This could be a sufficient basis for future own decommissioning and dismantling design and realisation.

In the field of dismantling experience, tools and instruments, the international assistance could be helpful. The co-operation between the manufacturers could be arranged on a commercial basis. The necessary training in the field of market orientated project management of the decommissioning managers could be made using the Project management units for internationally financed modernisation measures.

2.5.5 Legislative and Standard Environment

It is interesting to note that the efforts have survived to have a detailed system of instructions for all kind of works in the nuclear industry and research. These efforts have survived the former central administrated socialist economy. In discussing international assistance requirements both the central administration and the local executors expressed the request for detailed instruction for decommissioning and dismantling of nuclear facilities.

The necessary legislative basis exists in the CIS countries in present time. Laws on the use of nuclear energy were created and approved by the national bodies. The development of the second step - the transfer of the central rules to special rules in the area of decommissioning is ongoing. For instance, in the Russian Federation the development of the rules of the Nuclear Regulatory Authority is ongoing, but mostly in the area of radwaste management today. On this basis plant or facility related instructions can be carried out.

In summary, the development of the necessary normative documents for decommissioning of nuclear facilities is an ongoing process in the CIS countries. It could be useful to train the facility's staff to use current methods and standards (for instance: radiological protection standards) for decommissioning and dismantling activities. This would also require international support.

2.5.6 Decommissioning Experience

No civil nuclear reactor in the CIS countries has been decommissioned to the Stage 3 level corresponding to the IAEA classification [2-1, 2-4, 2-7]. Therefore, a complete range of decommissioning and dismantling experience is unavailable in CIS countries. Notwithstanding this, a wide range of experience of reconstruction and upgrading of nuclear reactors and components does exist, mostly in Russia, but this experience was collected under conditions of other safety and radiation protection rules and are therefore somewhat different from international and current Russian standards for decommissioning.

In the CIS-countries, possibly with exception of Ukraine, a decommissioning strategy, approved by the corresponding Bodies and Authorities, does not exist.

2.6 Section 2 References

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3. Decommissioning Requirements

3.1 Introduction

This section considers the decommissioning requirements for the four main reactor types being considered. The assessment utilises the development of outline generic decommissioning plans for the reactor systems based on an example nuclear facility as follows:

- For RBMK systems, the Leningrad NPP is adopted
- For VVER systems the Novovoronezh NPP is adopted
- For Fast Reactors the BN-350 systems at Aktau is adopted
- For Research Reactors the VVR2 and MR pool type research reactors in Moscow are adopted

The main issues arising from the outline generic decommissioning plans are brought into this paper as decommissioning considerations and requirements and include:

- an overview and description of the nuclear facilities identified as representative of the systems
- assessment of the radioactive inventory and radioactive waste management information where available
- assessment of the costs for decommissioning and funding considerations

Where possible the requirements to carry out the decommissioning proposed are identified and listed as examples of the work anticipated for that reactor type. Particularly in the cases of VVER and RBMK reactor decommissioning the considerations can be applied to other NPPs from the representative discussed as many of the issues are common and result from the operation of the reactors. Once the strategy “Later dismantling after a deferral period” is chosen the outline of the decommissioning project includes the following stages:

- Stage 0 "Preparation for closure"
- Stage 1.1 "Preparation for Care and Maintenance"
- Stage 1.2 "Care and Maintenance of the sealed reactor"
- Stage 2.1 "Preparation for Restricted Site Use"
- Stage 2.2 "Residual Care and Maintenance"

3.2 Considerations for RBMK Decommissioning

3.2.1 Overview

NPP sites with RBMK reactors are located at St. Petersburg (Leningrad NPP), Kursk and Smolensk (see Figure 2-1). A small NPP with a uranium-graphite reactor is located in Siberia at Bilibino. The first and second units at the Beloyarsk NPP are also equipped with uranium-graphite power reactors, on which basis the RBMK reactors were developed. Today only the Beloyarsk units are shut down, but it would be also necessary to prepare the decommissioning documentation for the first constructed NPP units at the Leningrad NPP, the forerunners of the whole series of RBMK-1000 nuclear units. Their nominal end of life time will be in around 2005-2007 assuming a 30 year design life. Outside the Russian Federation NPPs with RBMK reactors are located at Ignalina and Chernobyl. The Ignalina NPP has the newest and largest Soviet designed reactors type RBMK-1500, and is not involved in the current report. The decommissioning of the Chernobyl NPP is a special international assistance project, and therefore it is not included in the present report.

3.2.2 Description of the Leningrad Nuclear Power Plant

The Leningrad Nuclear Power Plant is used as the representative NPP for the series of NPP with RBMK systems. The reason is that the first units of the LNPP are the forerunner of all other RBMK-NPPs, and that these units are the oldest ones, and therefore are likely to be the first to be decommissioned. Generally speaking, these units and their location have no significant differences in comparison with the other RBMK-locations.

The Leningrad nuclear power plant (LNPP) is located near by the city of Sosnovy Bor in the Leningrad region (Leningradskaja oblast). It has two first and two second generation units. Construction began in 1970 on the first units and the LNPP is now a multi-unit operating plant typical of most NPP sites in CIS countries. Both units 1 and 2, as well as units 3 and 4 form a double-unit with some common systems and facilities.

Currently two new units 5 and 6 are planned. However the construction of these new plants is now dependant on financing being available.

The Leningrad NPP is the only one which is not operated by ROSENERGOATOM (see Figure 2-8). The Leningrad NPP has the status of a operator-organisation since November 20, 1992 and reports directly to Minatom. From this moment the management and the staff of the Leningrad NPP are completely responsible for safe operation of units (rather than Rosenergoatom who operate the remaining NPPs in Russia). Plant management is responsible for everything from choosing a site for new reactors to decommissioning old once and selecting licensed contractors to undertake the work required. [3-1]

The design end of life of unit 1 would end around 2003 based on a 30 year lifetime. Therefore currently some activities are being undertaken in the field of decommissioning planning. For the last two years the Russian Research Centre Kurchatov Institute in co-operation with US-enterprises have worked on some concept documents in the decommissioning field. This work was financed by a grant from the US Department of Energy (USDOE).

The current plan of the plant administration is to start the decommissioning work of the first units only after the commissioning of new two units, but the first priority is to undertake activities with the aim to extend the lifetime of the oldest units.

3.2.2.1 Historical Overview

Table 3-1 shows the main milestone dates for the operation of the LNPP site.

Table 3-1 History of the Leningrad NPP

1970	Start of the construction work at unit 1
1973/74	Commissioning of unit 1
1975/76	Commissioning of unit 2
1979/80	Commissioning of unit 3
1981/82	Commissioning of unit 4
1991	Completion of a major upgrading programme at unit 1
1992	'Privatisation' of the plant
1995	Start of the EBRD upgrading programme

Water-graphite type reactors - RBMK - were developed in the former USSR. The development began in the 1940s, when the development started on the first reactors of similar design.

The research nuclear reactor in Obninsk was developed and commissioned as the initial research facilities. The following stage of development of water-graphite reactors started in 1958 with the water-graphite channel reactor of the Siberian NPP (Swersk), used for military R&D. After that two channel water-graphite reactor type AMB were installed and commissioned at the Beloyarsk NPP with a electrical capacity 100 and 200 MW.

Leningrad NPP is the prototype of an NPP series with reactors of this type. About 10 units with RBMK-1000 reactors are in operation at other NPP sites currently, and at the Ignalina NPP (Lithuania) two units with RBMK-1500 reactors are in operation with a individual electrical capacity 1500 MW as the most current design of this reactor type.

The construction of Leningrad NPP begun in 1967. The constructional work for the Leningrad NPP was carried out by two phases of units, accordingly first and second (first generation) units then the third and fourth (second generation) units.

Power start-up of the first unit of the first phase of the Leningrad NPP was in December, 1973. About one year of commissioning works were performed and on November 1, 1974 the first unit of the Leningrad NPP achieved its nominal electrical capacity 1000 MW. In 1975 the second unit of the first design generation of the Leningrad NPP was commissioned. Currently at Leningrad NPP four units are in operation each with an electrical capacity of 1000 MW.

As the Leningrad NPP was the first among a series of installed NPP with RBMK. The design was carried out according to the standards of the 1960s and 1970s. Therefore, current safety requirements, which are defined for nuclear stations by modern normative documents, are not completely met by the design of the NPP.

The work at the Leningrad NPP follows three stages:

- 1973 to 1981: commissioning and increasing the power to the nominal capacity of the units (the stage is characterised by lower parameters of capacity and more shutdowns than later operations);
- 1982 to 1988: establishing the normal mode of operation of the units, elimination of design omissions and equipment faults (the stage is characterised by high technical and economic parameters and a small amount of shutdowns. The operating ratio of the established capacity of nuclear station was above 80 %, and for separate units exceeded 90 %);
- 1989 up to present time: a period of large-scale reconstruction works of the units of the first generation (first and second units). These works have required long shutdowns, and for this reason there was a decrease of production of electric power and operating ratio of the established capacity of the nuclear station.

An IAEA asset mission visited the LNPP in 1993 to advise on safety and a follow-up visit in mid-1996 reported that above 90% of the recommended improvements plan had been implemented successfully. However, it noted that consideration should be given to interim measures to prevent a recurrence of operational failures. [3-1]

The safety of LNPP is the focus of considerable international attention. An upgrading programme, financed by a 35 M€ grant of the EBRD, is ongoing.

<u>3.2.2.2 Location</u>

The Leningrad nuclear power plant is located inside a large nuclear industry complex 70 km in the west of St. Petersburg - see Figure 3-1

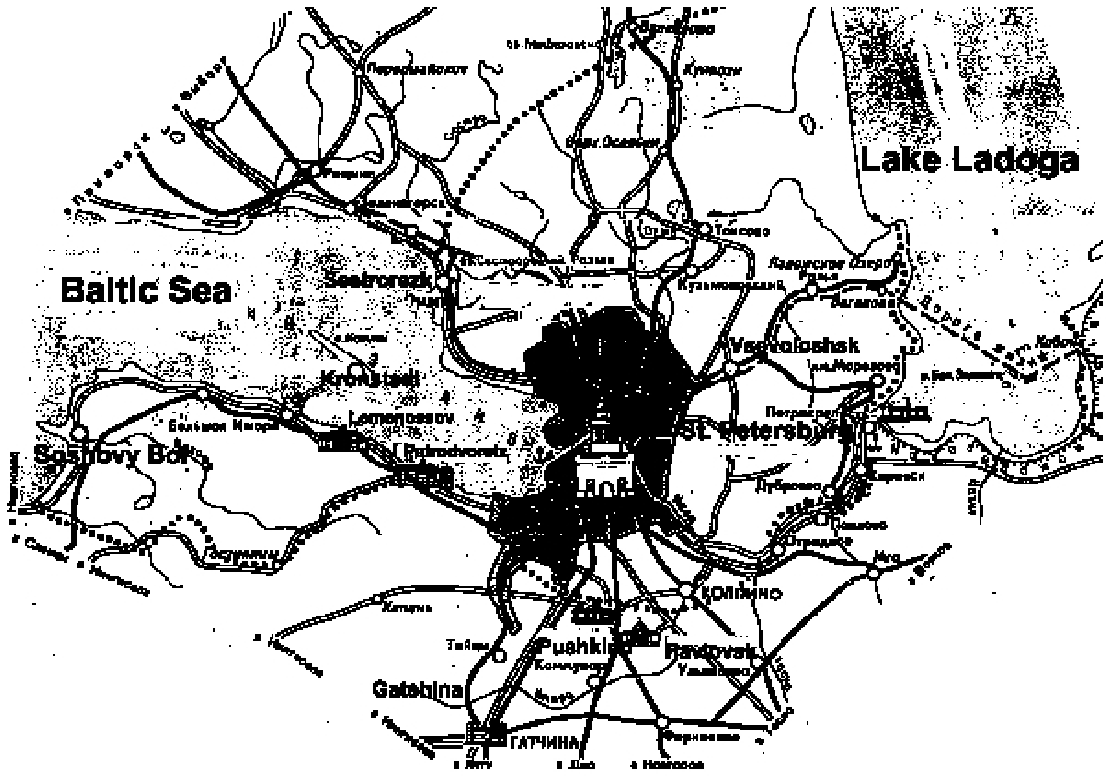

Figure 3-1 Map of the Leningrad region - in the west: Sosnovy Bor, the location of the Leningrad NPP

The NPP is located at the banks of the Baltic Sea. In the neighbourhood are located:

- the Institute NITI (North-West scientific-industrial power institute)
- the Radon nuclear repository

NITI plans to construct one (two reactors) new VVER-type NPP (AP-640/407) some 15 km from the existing one.

Sosnovy Bor, the nearest settlement to the Leningrad NPP, is located at a distance from NPP of approximately 6 kilometres. The population of Sosnovy Bor city is approximately 62000.

About 700 large and small enterprises are located in the region near by the Leningrad NPP. About 650 of these are small enterprises.

The NPP site is located 100 km from the state border line of the Russian Federation to Estonia at the sea border line to Finland. Therefore the location is integrated in the state border regime with controlled access. Some military objects are also located in the neighbourhood, for instance the base of the Russian Baltic Fleet at Kronstadt.

3.2.2.3 Description of the facility Site

The layout of the LNPP site is given in Figure 3-2.

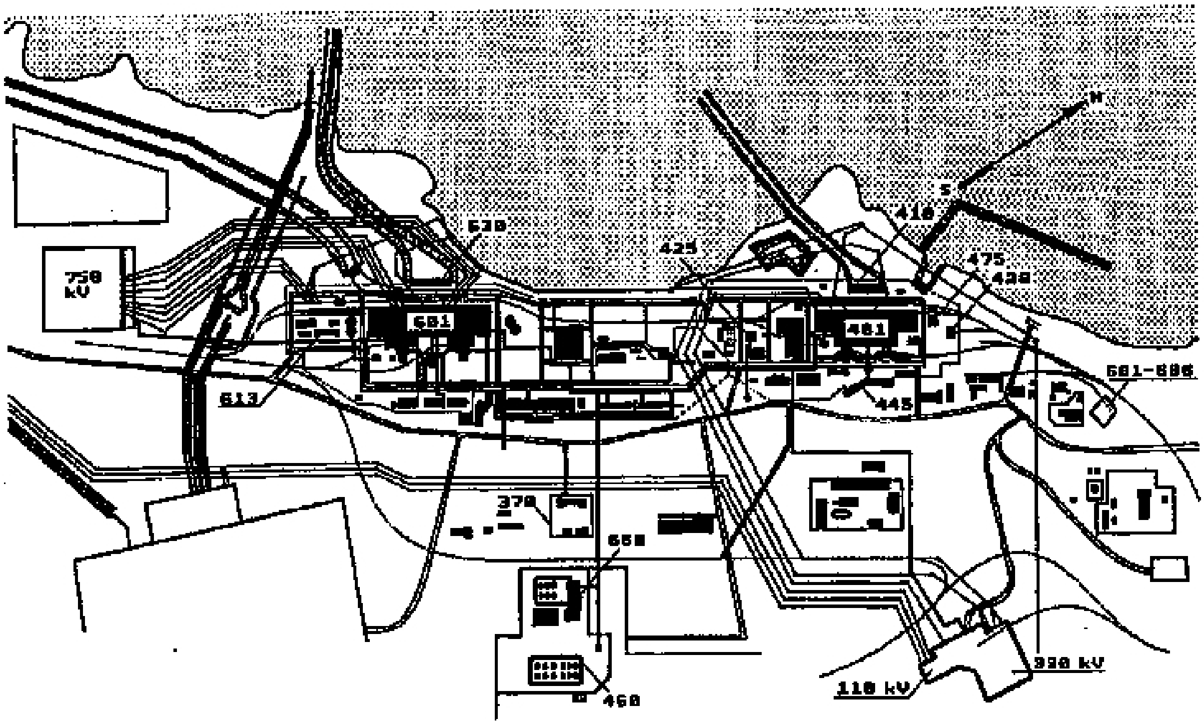

Figure 3-2 Layout of the Leningrad power plant site

370	Fire department	445	Admin building	630	Pump station of the second order
401	Units 1 and 2	460	Liquid radwaste storage	660	Radwaste treatment complex
410	Pump station of the first order	475	Diesel station of unit 2	681/686	Oil warehouse
425	Diesel station of unit 1	601	Units 3 and 4		
428	Spent fuel storage	613	Diesel station of units 3 and 4		

3.2.2.4 Physical interfaces

Leningrad NPP enters into the energy-network of the joint-stock company "Lenenergo", which carries out centralised electricity and heating power supply to consumers of St.-Petersburg and the Leningrad area.

The territory of the Lenenergo network, which is included in the electric power pool of Northwest network of Russia, includes a area of 85.9 thousand sq. km with a population of about 6.7 million people.

The surplus electric power is transferred to the networks of the Northwest and Centre. Export of electricity is also carried out by the Leningrad network to Finland.

The designed power production of Leningrad NPP is 28 GW hours per year. The NPP consumes 8-8.5 % from the produced electric power for its own needs.

In spite of the fact that since 1989 the electric power production of the Leningrad NPP has decreased, the nuclear station continues to have an important part in the supply of electricity to both the Leningrad region and the Northwest network.

The Leningrad NPP covers about 50 % of loading in Lenenergo network, and together with the Kola NPP more than 40 % of Northwest network. Leningrad NPP is also the basic heating energy supplier of Sosnovy Bor and the industrial zone, located approximately 6 km from the NPP.

The water supply to the nuclear power plant is managed directly from the Baltic Sea via a special pumping station.

The radioactive waste generated is stored at the territory of the NPP, partially in bulk storage for solid waste and also in large tanks for liquid radioactive waste. An interim storage facility for spent fuel was constructed in the eighties because reprocessing facilities for spent RBMK fuel does not exist in the Russian Federation (and the rest of the CIS).

The staff of the Leningrad NPP live mostly in Sosnovy Bor, the special NPP settlement. This settlement has increased to a medium-large industrial centre with research institutes, manufacturers and service companies. Nevertheless, the NPP is currently the largest employer in the territory.

<u>3.2.2.5 Description of the unit 1 at final closure</u>

Closure of the whole plant is not foreseen. It is foreseen, possibly after a lifetime extension, to close the units step by step following behind the commission of new units. In this case attention must be paid to the links between closing and operating units at the double-unit structure.

The unit structure includes (see Figure 3-3) [3-2]:

- the reactor building (radiation controlled area with controlled access)
- the turbine hall (radiation controlled area with controlled access)
- the unit's radwaste treatment facility (radiation controlled area with controlled access)[4]
- the electrical switchboards and control rooms (radiation controlled area with controlled access)
- the sanitary and access building

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[4]Special radwaste treatment facilities do not exist inside the units. They are centralised for the complete NPP. Only small special technological systems are existing at the units.

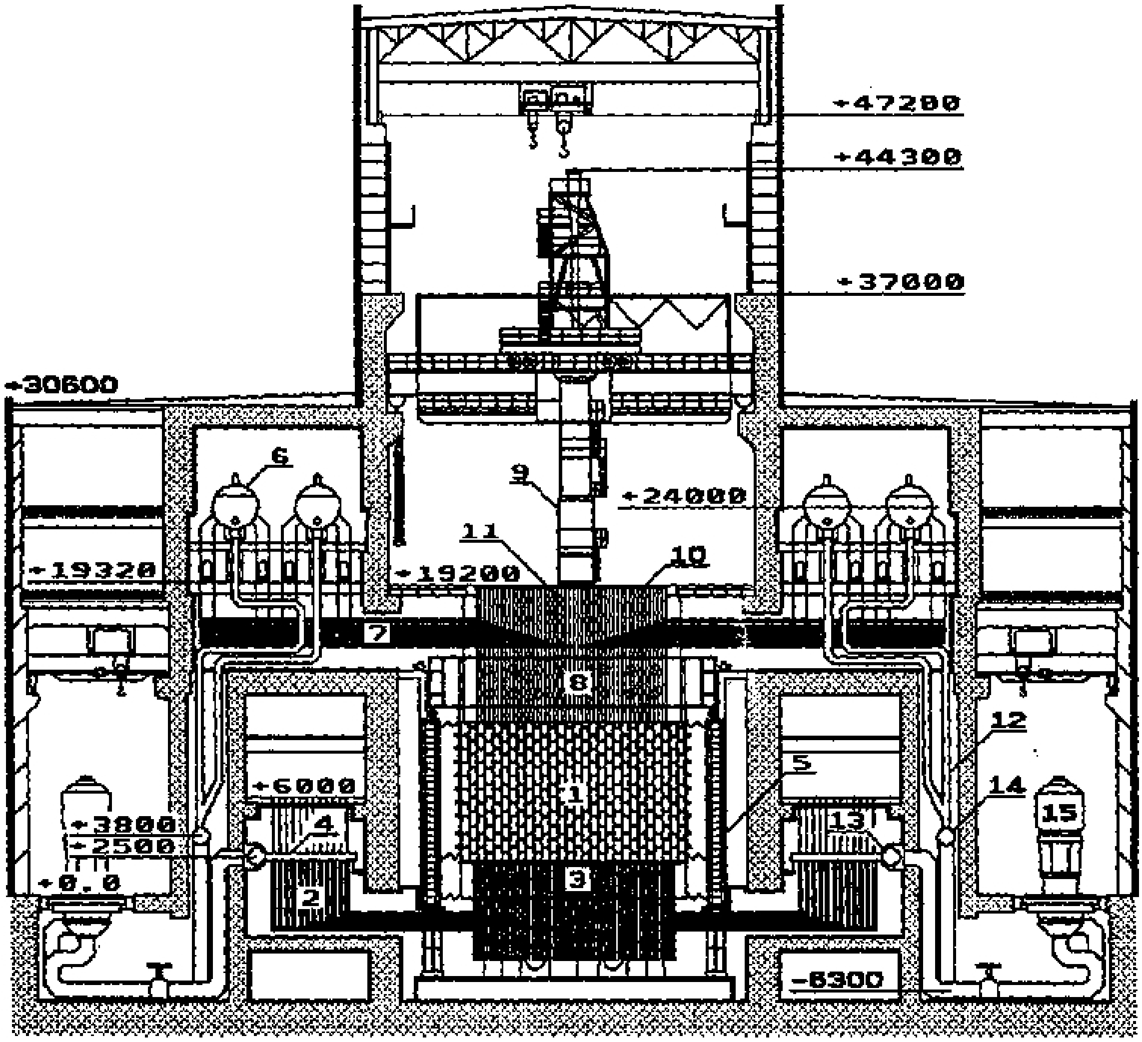


Figure 3-3 Cutaway of the nuclear island

The main structural components of the RBMK reactor systems are given in Table 3-2.

Table 3-2 The basic structural parameters of a main building of one unit with reactor type RBMK-1000

Parameter	Value
Volume of civil construction, m^3/kW	1.14
Volume of concrete and iron concrete, m^3/kW	0.25
Weight of metallic structure, kg/m^3	38.0
Weight of technological equipment, kg/kW	36.6
Weight of stainless steel, kg/kW	10.24

The content and the weight of the main equipment for the unit of the Leningrad NPP are summarised in Table 3-3:

Table 3-3 Masses of the equipment of the NPP unit

Name		Mass [ton]
1.	Reactor (together with the refuelling machine and the support)	8200
2.	Equipment of the reactor part	4940
3.	Turboset	11320
4.	Equipment of the turbine hall and the deaerator building	3420
5.	Low pressure pipelines at the main building	4250
6.	High pressure pipelines at the main building	3460
7.	Equipment of special water treatment	1820
8.	Equipment of specialised ventilation systems	1630
9.	Equipment of communications of external facilities	5575
	Total	44615

3.2.2.6 Decommissioning Requirements for the Facilities

The main parts of the NPP will have the following requirements for decommissioning to take place:

- Reactor building - the physical structure must be investigated. The design lifetime of civil constructions at the NPP is approximately 100 years.
- Reactivity and coolant control system - the physical structure of the equipment and pipelines must be investigated immediately after the final shut down of the reactor.
- Reactor fuel ponds - the physical structure of the equipment and pipelines must be investigated immediately after the final shut down of the reactor.
- Spent fuel interim storage - the common usage of the existing and new installed equipment of the operating units must be planned.
- Liquid and solid waste treatment facilities and interim storage - the common usage of the existing and new installed equipment of the operating units must be planned.
- Service facilities - the common usage of the existing and new installed equipment of the operating units must be planned.

In addition the following systems are identified as being required for decommissioning to take place:

- Heating system - if heating can not be provided from the operating units
- Power generation - if power supply can not be provided from the operating units

- Liquid waste retrieval and solidification facilities - if liquid waste treatment and solidification facilities are not available in operating units
- Solid waste retrieval and processing facilities - if solid waste processing facilities are not available in operating units

3.2.2.7 Operating history of LNPP UNIT 1 relevant to decommissioning

Unit 1 of the Leningrad nuclear power plant was designed in the sixties on the basis of the common industrial rules and standards existing in the Soviet Union at that time and special norms and standards for nuclear power plants were not developed. For instance, it was foreseen to collect the radioactive waste at special facilities and stores of the NPP territory and to transport the nuclear fuel for reprocessing to special plants. No special design for decommissioning was foreseen.

The most important incidents and accidents impacting to the decommissioning activities are given in Table 3-4 [3-1].

Table 3-4 Accidents and incidents impacting on decommissioning works

January 1974	Unit's 1 radioactive gas reinforced concrete gas holder burst.
February 1974	Intermediate circuit ruptured when the water boiled.
November 30, 1975	There was a major incident when a fuel element melted down in one of the 1,673 fuel channels of unit 1 resulting in partial destruction of the core.
July 1976 and September 1979	Two fires at the plant's solid radioactive waste store.
Between 1992 and mid-1993	Over 150 safety related events at the plant, mostly affecting unit 1
February 22, 1994	Accident shutdown of unit 1 as a result of a breach in a pipeline

From 1989 to 1991 the first upgrading programme was executed. The second upgrading programme is currently ongoing. This programme is financed by a grant of the European Bank for Reconstruction and Development (EBRD). According to this project a large amount of reconstruction work is executed both on the first and second units Leningrad NPP as these units are the first generation of NPP with RBMK reactors. The work is expected to be completed by 2001.

Annually Leningrad NPP, as operating organisation, presents a report on the current condition of operational safety on each NPP unit to GAN. According to the latest conclusion the current condition of operational safety of all units of the Leningrad NPP is recognised as satisfactory.

Unit 1 is due to be decommissioned in 2005 and unit 2 in 2007 when their service life will end. However, the modernisation and upgrading measures could extend their operation life by up to ten years.

3.2.3 Radioactive Inventory/Hazardous Material

Decommissioning starts normally with a shut-down of the reactor, similar to the shut-down for re-fuelling, but with no intention to restart again and accompanied by activities which make a restart impossible. At this time there will remain on the facility site significant amounts of materials that are potentially harmful (radiologically and conventionally) to people and to the environment. There are hazardous materials stored in waste vaults as well as within the main and auxiliary systems and components of the NPP which have to be detached from these systems directly after shutdown. Also there are hazardous materials which will arise during the short-term and long-term decommissioning activities.

Based on the information supplied by Local Partners this section gives information about the amounts of radioactive waste existing at the Leningrad NPP site, especially at unit 1. Besides overall information about the main radwaste categories, information will be given with data on waste category, waste amount, waste origin, radiological waste characteristics with the total nuclide content and the most important nuclides. Information on conventionally hazardous materials are not available up to now and thus not given here.

For decommissioning purposes of unit 1 of Leningrad NPP more information than currently available is necessary. Information is needed about amounts of radioactive waste existing at the NPP and/or the unit site at final shutdown and about the amounts expected at the end of the various decommissioning stages (Stage 1, Stage 2). Furthermore, the existing overall information about the main radwaste arisings must be supplemented by detailed information for the unit in a system by system way with data on waste category, waste amounts, waste origin, radiological waste characteristics with the total nuclide content and the nuclide content of the most important nuclides at various time periods after final shutdown (eg 0, 5, 10, 20, 50, 100 years, and at the assumed times for the end of Stage 1 and 2).

Corresponding information for conventional hazardous materials is required. Because of different amounts of different conventionally hazardous materials in different forms, it is necessary to establish a register with records describing the materials, the protection measures, waste treatment processes, input, arisings due to decommissioning activities, and outputs over the whole decommissioning period. Records are also needed on the personnel in contact with the materials and on corresponding health data.

3.2.3.1 Inventory at final shutdown

At the final shut down there are radwastes from the operation of the NPP as well as from maintenance and backfitting or upgrading activities carried out in the past which are stored at the NPP site in interim storage facilities. Furthermore, there are radwastes which are in the waste treatment system under processing. Finally, there are radionuclides within the media of various technological systems of the unit and within components the construction material of which are carrying radionuclides either from activation or from contamination.

There are six main kinds of radioactive material at shutdown to deal with:

- fuel within the reactor
- spent fuel either in ponds near the reactor or in storage facilities at the NPP site
- solid and liquid operational radioactive wastes in the storage facilities at the NPP site
- radionuclides within the metal reactor materials
- radioactive graphite
- radionuclides within the components of the primary cooling circuit
- contaminated construction materials such as concrete and metal

The present status regarding radioactive waste at the Leningrad NPP and the unit 1, respectively, is described on the basis of investigations carried out by Local Partners. Data on conventionally hazardous material is not available.

3.2.3.2 Spent fuel

At present there are about 26,000 spent fuel assemblies in total at the Leningrad NPP site, either within the spent fuel storage facility at the NPP site (KHOYAT) or in reactor ponds. The mass of the spent fuel assemblies amounts to more than 3000 tons. More than 8200 fuel assemblies of them have arisen from unit 1. The total activity of the spent fuel at the Leningrad NPP is about 4.2×10^{19} Bq. Table 3-5 estimates the total amount of spent fuel at final shutdown.

Table 3-5 Spent fuel in unit 1 (expected at final shutdown)

	in the reactor	in unit spent fuel pond(s)
number of fuel elements	1600	2900
total activity / Bq[(*)]	about 2.6×10^{18}	about 4.7×10^{18}

[(*)] Estimation based on the number of fuel assemblies expected and the total number of fuel assemblies with their total activity present at the NPP site.

3.2.3.3 Liquid waste

The following classification of liquid radioactive wastes is applied at the Leningrad NPP:

- Low-level waste (LLW): less than 3.7×10^5 Bq/l
- Intermediate-level waste (ILW) from 3.7×10^5 to 3.7×10^{10} Bq/l
- High-level waste: more than 3.7×10^{10} Bq/l

At 31 December 1995 there was approximately 11923 m^3 of liquid radioactive wastes coming from the operation of the Leningrad NPP. That means that the available storage facilities are filled up to about 70% of capacity. The total capacity of the storage facilities is 17020 m^3. The total activity of the liquid radwaste is 1.55×10^{14} Bq.

From the unit 1 of the Leningrad NPP the following arisings of operational liquid radioactive wastes can be expected:

1. Residues from evaporator facilities:

- arising, m^3/yr 300 to 500
- mean salt content g/l 300 to 350
- specific activity Bq/l 1.85×10^6

2. Low-level sorbents

- arising, m^3/yr 16
- specific activity Bq/l 1.11×10^8

3. High-level sorbents

- arising, m^3/a 22
- specific activity Bq/l 1.85×10^9

4. Perlit

- arising, m^3/a 25
- specific activity Bq/l 7.40×10^7

5. Waters

- arising, m^3/a 10^5 to 1.3×10^5
- specific activity Bq/l 7.4×10^4

6. Ion exchange resins

- arising, m^3/a 84

- specific activity Bq/l 1.11 x 10^8

Solid waste

The following classification of solid radioactive wastes is applied at the Leningrad NPP:

a) Dose rate limits 10 cm distant from surface:

Low-level waste (LLW):	3.0 to 3.0 x 10^2 µSv
Intermediate-level waste (ILW):	3.0 x 10^2 to 1.0 x 10^4 µSv
High-level waste:	more than 1.0 x 10^4 µSv

b) Specific activity for beta nuclide bearing radwaste

Low-level waste (LLW):	7.4 x 10^1 to 3.7 x 10^3 Bq/g
Intermediate-level waste (ILW):	3.7 x 10^3 to 3.7 x 10^6 Bq/g
High-level waste:	more than 3.7 x 10^6 Bq/g

c) Specific activity for alpha nuclide bearing radwaste

Low-level waste (LLW):	7.4 to 3.7 x 10^2 Bq/g
Intermediate-level waste (ILW):	3.7 x 10^2 to 3.7 x 10^5 Bq/g
High-level waste:	more than 3.7 x 10^5 Bq/g

(Based on the dose limit of 10 µSv /a for the public in Russia, an unrestricted use of alpha and beta nuclide bearing materials is possible for activities between 0.1 and 7.4 Bq/g.)

At 31 December 1995 there are about 15724 m^3 solid radioactive wastes coming from the operation of the Leningrad NPP. That means that the available storage facilities are filled up to about 66%. The total capacity of the storage facilities is 24000 m^3. The total activity of the liquid radwaste is 2.96 x 10^{13} Bq.

From the unit 1 of the Leningrad NPP the following arisings of operational solid radioactive wastes can be expected:

- arising, m^3/a 400
- 1 / 2 / 3 group, % up to 65 / up to 30 / up to 5

60 % of the solid wastes at the Leningrad NPP site are concrete and other construction materials, 30 % are metal, and 10 % are combustible.

The total activity of the reactor construction materials is about 3.11 x 10^{18} Bq, whereas 82 % comes from the pipes of the technological channels, 0.2 % from the radioactive graphite, and the rest from the other materials of the reactor. The radioactivity of the components of the primary loop amounts to 1.70 x 10^{14} Bq.

The most important nuclides of the metals are Cr-51, Mn-54, Fe-55, Fe-59, Co-58, Co-60, Ni-59, Ni-63, Nb-94, Zr-93, and Zr-95.

The most important nuclides of the graphite are H-3, C-14, Cl-36, Co-60, Ni-59, Ni-63, Fe-55, and Fe-59.

3.2.3.4 Inventory at end of Stage 1

Preparation and execution of decommissioning activities resulting in radwaste generation are initiated immediately after final shutdown. During the Stage 1 phase the processing of the radwaste present at the final shut down must be continued with the objective to store them for a time period at on-site storage facilities, if possible in a conditioned form, or exempt them from further control, reuse them, or discharge them to off-site disposal. At the same time radwaste will arise from decommissioning activities. They have to be processed as the radwaste present at final shut down. At the end of Stage 1 there are radwaste from the Stage 1 decommissioning activities as well as from the operation phase of the NPP which have to be processed during Stage 2 phase and must be discharged from the NPP site latest at the end of Stage 3. During Stage 1 the fuel will be removed from the unit so that the main source of radiological risk will be eliminated.

3.2.3.5 Inventory at end of Stage 2

During the Stage 2 phase the processing of the remainder of the radwaste present at the final shut down and from Stage 1 activities must be continued with the objective to store them for a time period at on-site storage facilities, if possible in a conditioned form, or to exempt them from further control, or to give them to a further utilisation, or to discharge them to off-site disposal. At the same time radwastes generated during Stage 2 have to be processed in the same way as the radwaste present at final shut down and produced during Stage 1 phase. At the end of Stage 2 there are radwastes from the Stage 2 decommissioning activities as well as from the operation phase and from Stage 1 of the NPP which have to be processed during Stage 3 phase and must be completely discharged from the NPP site latest at the end of Stage 3.

3.2.3.6 Graphite Waste Management

The radioactive graphite coming from nuclear installations has different characteristics than other radioactive waste due to its physical and chemical properties and also because of the presence of tritium and carbon-14. Even after many years of irradiation, graphite retains most of the good mechanical properties and is relatively insoluble and not otherwise particularly chemically reactive. It appears therefore to fulfil most of general requirements for a solid radioactive waste suitable for disposal. However the evaluation of the radioactivity inventory of graphite moderators and other graphite details applied in nuclear reactors show that this graphite cannot be accepted by existing disposal sites without particular conditioning.

Different options have been studied for the management of the radioactive graphite, but the final and generally accepted solution for its conditioning and/or disposal has not been decided yet. In practice the main option is for a period of long term storage before final disposal. Three basic solutions are often proposed for disposal of reactor core graphite:

- direct disposal after suitable packaging ;
- disposal after incineration with consequent ash conditioning and with efficient filtration system of the off gas (except for carbon 14 which is difficult to trap);
- disposal after chemical treatment (liquid and/or gaseous extraction), conditioning (impregnation) and proper packaging.

Both disposal options: near surface repositories and also deep geological formations have been evaluated and disposal on the seabed was also analysed.

The Russian AMB power production reactors at Beloyarsk and the combined heat and power reactors at Bilibino contain a unique fuel design where the fuel elements are located in system of graphite blocks. These graphite blocks are part of the fuel assembly and are removed with the fuel.

Most of the RBMK reactors will have the fuel channel tubes replaced during operation. This involves not only removal of the zirconium tubes but also removal of the graphite rings. This will create a significant amount of graphite waste. In addition to this there is a small amount of graphite used as displacer elements associated with the control rods. Although this displacer graphite is small in quantity, it is possible it may contain a significant amount of stored energy owing to the low temperature at which it was irradiated.

In conclusion, the management and disposal of graphite from nuclear facilities is a significant problem for a number of countries. Graphite from RBMK units is particularly susceptible to long term deterioration without suitable treatment or disposal. Solutions to the problem of graphite waste management should be applicable to CIS countries with graphite waste streams.

3.2.3.7 Conventional hazardous materials

Besides radioactive wastes there are also conventionally hazardous materials at the NPP site at the final shutdown. These materials are either stored in conditioned or unconditioned forms or are within the NPP main or auxiliary systems. During the following decommissioning activities these materials have to be discharged from the NPP site in a controlled manner. Protective measures for the personnel as well as for the public and the environment must be applied. These conventionally hazardous materials can exist together with radioactive materials. In this case radiation protection measures are needed, too. In addition to conventionally hazardous materials in the unit from the operation phase, during the decommissioning process at every time there will be input and output streams of conventionally hazardous materials caused by the decommissioning activities themselves.

Considerations of conventionally hazardous materials have to include:

- explosive materials
- fire promoting materials
- fire dangerous materials
- toxic materials
- etching materials
- allergy causing materials
- cancer causing materials
- reproduction toxic materials
- cell damaging materials
- environment damaging materials.

Among conventionally hazardous materials the following are of great importance because of their damaging potential and/or the amounts one is faced with:

- asbestos
- oil
- organic solvents
- acids
- bases.

At present, for the Leningrad NPP no data is available on existing amounts of conventionally hazardous materials and expected arisings from operation and future decommissioning activities There is no information on conventionally hazardous waste routes including storage, treatment, disposal and on applied technologies.

Conventionally hazardous materials of certain amounts are present within the unit at final shut down in mobile and fixed forms. During decommissioning, e.g. during dismantling, fixed conventionally hazardous materials can be mobilised with an increase of their hazard potential (as is the case with asbestos). Furthermore, conventionally hazardous materials will be brought additionally into the unit during decommissioning and due to applied technologies. In contrast to radioactive materials where there will be a continuing decrease of waste amounts and radioactivity levels with the ongoing decommissioning process, there are varying amounts of conventionally hazardous materials with varying associated hazard levels and requiring suitable protection measures. Thus it is important to establish an adequate register of conventionally hazardous materials which has to be updated over the decommissioning process so that records are available for every time to say what kind of materials are within the unit, what are the corresponding amounts, what it is the origin of these materials, how they will be treated and disposed of, and what protection measures are necessary for involved staff, public and the environment. This register should also include the results of medical surveillance for the persons who have contact with the conventionally hazardous materials including data of safety related occurrences.

3.2.4 Cost Estimates and Funding Requirements

This section gives an estimation of the decommissioning costs. It must be explained, that it is very difficult to define the real decommissioning efforts and costs for the CIS-countries. The economic situation is currently moving from a central controlled to a market orientated economy. Therefore the efforts and costs are described in this report as they would be at Western Europe. The Euro is used for currency (€). The costs given are developed using Western European labour costs scaled down to reflect the cheaper rates experienced in the CIS countries.

The special multi-unit structure of RBMK type nuclear power plants is the reason that most of the new facilities that are necessary for decommissioning and radioactive waste management of the decommissioning of all units must be constructed for the decommissioning of the first unit. Because of this situation the decommissioning costs for the first unit appear very high.

It was not possible to calculate the decommissioning costs for decommissioning stages after the deferral period. For this period literature sources were used. In this section the decommissioning costs are estimated up to ten years after the final shut down of the reactor, e.g. up to the safe enclosure of the reactor building and the dismantling of the interiors of the turbine hall.

The estimation of the decommissioning costs could be a basis for the detailed economic and commercial assessment of the financing for decommissioning. This financial planning must also include the detailed identification of sources, investigations of all foreseeable risks and the assignment of corresponding assurances as well as the detailed identification of activities which will be done by the NPP staff and works for which contractors will be invited, to define a detailed financial plan.

In this section the following main cost elements are described:

- design and construction of new facilities (excluding the construction of repositories)
- development of the necessary operational, design and licensing documentation
- the man-power work for the stages 0, 1 and 2 for
  - efforts for operation and monitoring of the shut down unit
  - dismantling works
  - operation of waste treatment facilities

The necessary R&D efforts are not included in the cost estimation.

The planning of the decommissioning costs is made beginning with the year of the decommissioning decision. Table 3-6 gives the values estimated for the decommissioning costs per year (no discounting is used).

Table 3-6 Total decommissioning cost estimation

Facility	Costs	Year													
	[million €]	-3	-2	-1	0	1	2	3	4	5	6	7	8	9	10
New facilities	255	8	13	19	40	40	30	15	10	15	15	15	15	10	10
Documentation	27	1	2	3	4	4	3	3	2	2	2	1	0	0	0
Stage 0 Operating & Dismantling	52	13	13	13	13										
Stage 1 Operating & Dismantling	102					16	16	16	14	16	12	12			
Stage 2 Operating & Dismantling	28												9	10	9
Total [million EURO]	464	22	28	35	57	60	49	34	26	33	29	28	24	20	19
Total from the start [million EURO]		22	50	85	142	202	251	285	311	344	373	401	425	445	464

Figure 3-4 shows the development of decommissioning costs, Figure 3-5 the sharing between the individual parts.

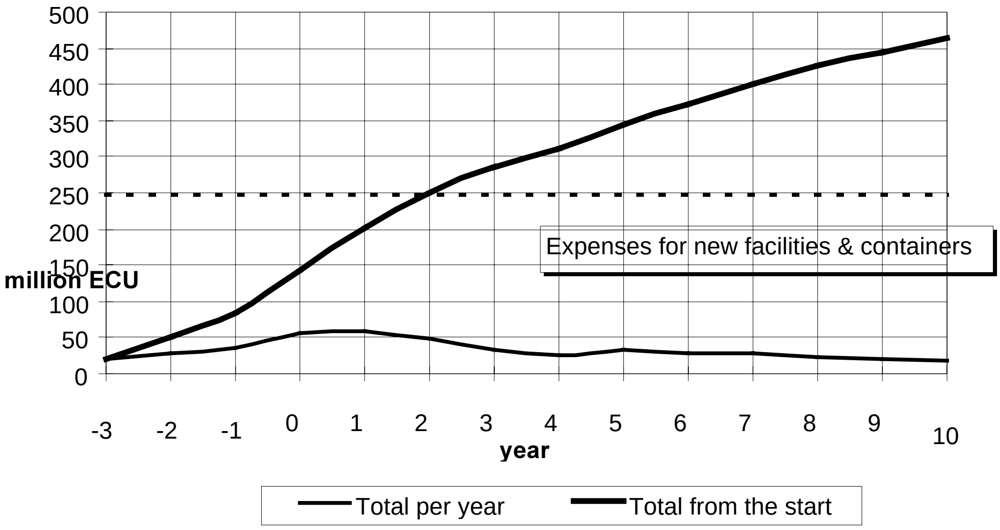


Figure 3-4 Total Decommissioning costs

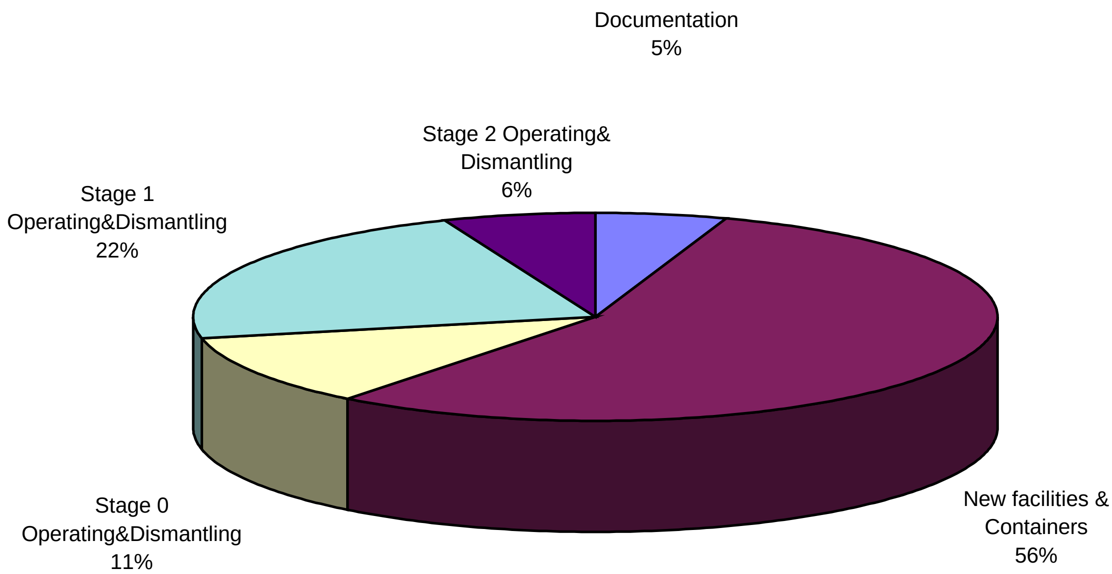


Figure 3-5 Breakdown of decommissioning costs

The costs for stage 3 were estimated and shown in Table 3-7 [3-3]. This source gives the following estimation for RBMK-NPPs:

Table 3-7 Total decommissioning cost estimation

Stage 0	Stage 1	Stage 2	Stage 3
117 million €	156 million €	15.5 million €	40.5 million €

For the stages 0 - 2 the current estimation gives an amount of 404 million €, the information in [3-3] gives 288.7 million €.

3.2.4.1 Costs for new and modified facilities

It is planned to construct a liquid radwaste treatment facility with the aim of solidification of all existing and generated by decommissioning (decontamination) liquid radioactive waste. This could be a cementation facility. Currently it is planned to construct such a cementation facility using the financing support of the EBRD. Nevertheless the costs for such a facility are included, but it should be explained that this facility must be constructed to meet the operational and decommissioning requirements of all four units at the LNPP site. Solid radwaste treatment facility with the aim of sorting, conditioning and packing of all existing solid radioactive waste, bulk stored in existing interim storage facilities, and generated by decommissioning and dismantling of the units.

3.2.4.2 Costs for new facilities and modifications of existing facilities

The following facilities are required:

- Radwaste interim storage facility with the aim of an interim storage for a period of approximately 50 years of solid and solidified radioactive waste generated at the LNPP site. The reason for such an interim storage facility is the lack of a national or regional repository.
- Spent fuel storage facility with the aim of an interim storage of all LNPP spent fuel including the operation of all four units up to the end of their life time. The reprocessing and/or the final disposal of RBMK-spent fuel is not decided up to now.
- Special decommissioning (decontamination) and dismantling tools and instruments. Significant remote handling instruments are not required for these decommissioning stages because the reactor dismantling and the dismantling of high and intermediate radioactive level equipment is not planned.

Because the costs for the new facilities are the highest of all the specified costs Table 3-8 shows the breakdown for new facility costs.

Table 3-8 Costs for new facilities

N°	Facility	Financing period [years]	Costs [million EURO]
1	Liquid radwaste treatment	-3 ... +1	20
2	Solid radwaste treatment	-3 ... +1	20
3	Radwaste container	0 ... +10	90
4	Radwaste interim storage facility	-3 ... +2	20

5	Spent fuel interim storage facility	-4 ... +3	85
6	Tools and instruments	+5 ...+8	20
	Total		255

The long-term repository for radioactive waste is not included.
The facilities 1, 2, 4, and 5 are planned for the decommissioning of all four units at the LNPP.

The estimated costs include:

- the project management
- the procurement
- the design
- the construction
- the commissioning

in the form of turn-key contracts (for example). The estimated costs do not include the operation of the facilities.

3.2.4.3 Costs for decommissioning documentation

The decommissioning requires the following:

- the documentation for the complex engineering and radiation inspection
- the decommissioning design and licensing documentation
- the decontamination and dismantling documentation
- some new or modified operational documentation
- the detail design documentation for the safe enclosure of the reactor building for the storage under surveillance period

This documentation must be carried out for each unit. It is suggested that for the second unit the efforts will be somewhat smaller by making use of the experience on the first unit. The project manager of the NPP has to decide who or which institution must be involved in the development of the documentation. It would be necessary to contract some development with external contractors, for instance, the detail design for safe enclosure or the special dismantling design. The licensing documentation has to involve a safety documentation, which could be prepared by a scientific centre. The costs for basic research & development (R&D) are not included.

The basis for the cost estimation carried out is that one man-year costs 50000 €.

3.2.4.4 Costs for decommissioning stages

The decommissioning costs for the stages include the manpower-costs, involving the

- the costs for operators (existing units and surveillance facilities)
- the costs for dismantling work
- the costs for operation of the radwaste treatment facilities and of the storage facilities

3.2.4.5 Costs for operation of existing and surveillance facilities

The operational costs include:

- the management cost
- the costs for the shift operators and repair staff - the six shift mode is estimated. At the moment of reactor shut down the shift team consist of approximately 25 operators and 10 repair mechanics. The number of shift operators and repair mechanics is subsequently decreasing. The repair staff will also undertake the disconnection and sealing inside the reactor building. The operators and the repair staff will undertake the decontamination of the systems using existing equipment.
- the costs for security staff - after the safe enclosure of the reactor building the security staff will be increased
- the costs for the radiological monitoring staff - after the safe enclosure of the reactor building and during the complex dismantling of the equipment inside the turbine hall this staff will be increased.

Decommissioning costs are not foreseen in the period before the reactor shut down. The costs of the plant services (fire department, social and medical services, etc.) are included in the price of the man-years.

3.2.4.6 Costs for operation of radwaste processing facilities

The costs for the operation of the radwaste processing facilities are estimated by the same method as for the operating costs. The facilities will be also working for units 2 - 4 of LNPP. A one-shift mode is planned.

3.2.4.7 Costs for dismantling activities

The dismantling will start with the final reactor shut-down. Most of the efforts will occur after receiving the decommissioning license (after the reactor defuelling and the removal of the fuel of the unit) at the turbine hall. The method for cost estimation is the same as described above.

3.3 Considerations for VVER Decommissioning

3.3.1 Overview

In the CIS countries a large number of VVER reactors were built and commissioned as well as in Finland, Germany, Hungary, Bulgaria, Czech and Slovak Republics. In the CIS countries VVER-440 reactors are in operation in Russia, Ukraine and Armenia. The VVER-440 units and the two forerunners are listed in the Table 2-2.

In ***Russia*** VVER-440 reactor NPP units are located at Novovoronezh and Kola. The two units at Novovoronezh are the prototypes of the VVER-440, equipped with 73 control rods. At the Kola site two units of the older V-230 project and two units of the advanced V-213 VVER-440 reactor type are in operation.

In ***Ukraine*** two V-213 VVER-440 units at Rovno are in operation.

In ***Armenia*** two VVER-440 units are located at Metsamor. Both units are the modified variant of the V-230 project with a seismic design and were switched off after an earthquake in 1989. Unit 2 was recommissioned after the earthquake to produce much needed electricity. The unit 1 was shut down and "cannibalised" to provide parts for repairs of the unit 2, and therefore making recommissioning of the older unit increasingly unlikely.

In the field of the decommissioning of VVER-440 reactors there are some international experiences. These originate mainly from the decommissioning of the five VVER-440 reactors of the Greifswald NPP in Germany. Four of these units - the units 1-4 - are similar to the Novovoronezh units 3 and 4. These units belong to the first generation of the VVER-440 reactor type. All VVER type reactors operated in the former German Democratic Republic (GDR - East Germany) were shut down in 1990 and their decommissioning was decided in 1990/91.

In the former Soviet Union in 1990 a first outline of possible variants of further procedure after the termination of operation was given in the "Conception of NPP shut down in the USSR and other countries" [3-4]. In general two paths are described, the life time extension and the decommissioning of the reactor units (see Figure 3-6). The given outline is currently the basis for the options considered for decommissioning or further operation of the NPP units.

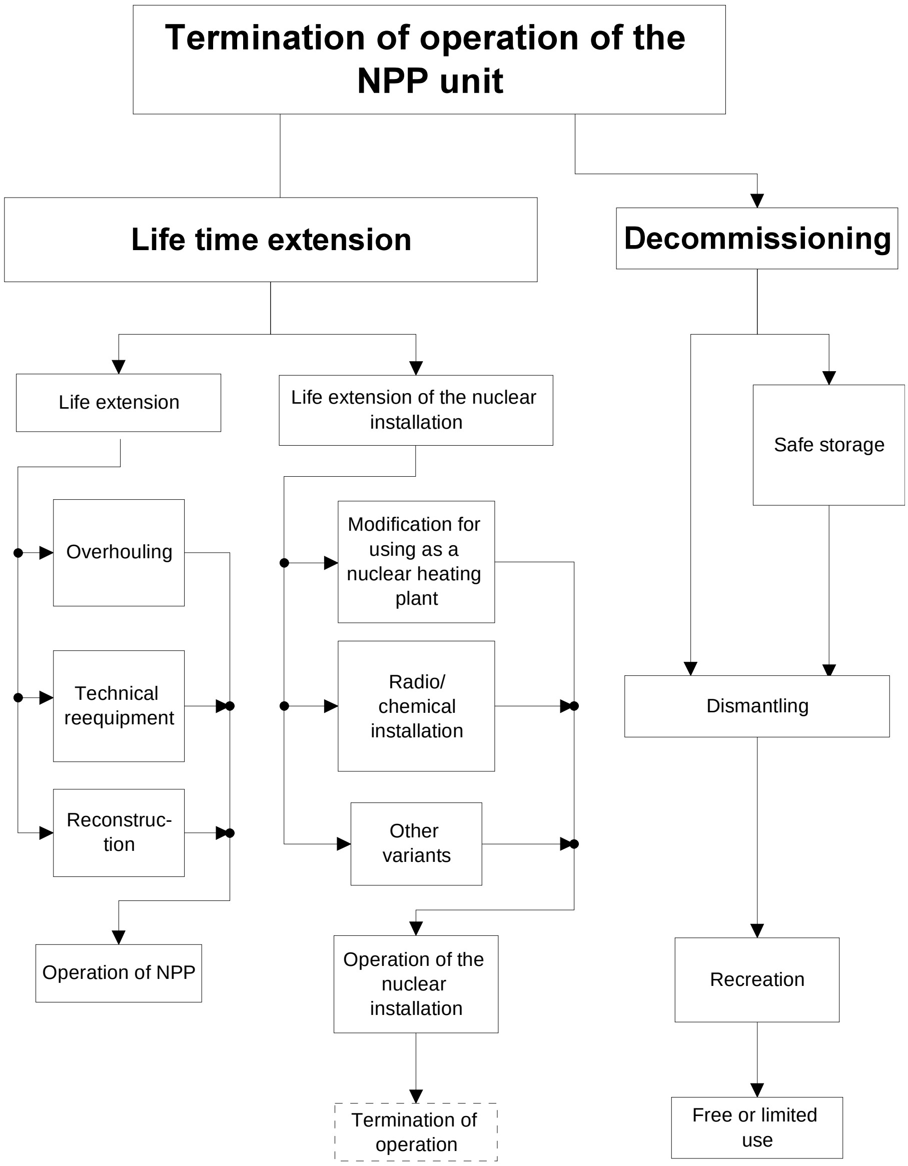


Figure 3-6   Possible options for life extension and decommissioning of NPPs

In the early 1990s plans for the reconstruction of older NPP units and also conceptions for the dismantling of VVER reactors were developed [3-5, 3-6].

The existing national experiences in the field of decommissioning in the CIS-countries are gathered from the ongoing decommissioning works, mainly from the unit 1 and 2 of the

Novovoronezh NPP (NVNPP). The decommissioning activities of unit 2 are temporarily halted until a final decision concerning the future use of the unit is made. This has been affected by the ongoing discussions concerning the option of replacing the reactor by a new fast reactor with a lead cooling system. Information about the ongoing decommissioning activities are available from information supplied by the Local Partners and from the reports of the decommissioning group INTERATOMENERGO. With regard to the destiny of the Novovoronezh NPP units 1 and 2 the plan is to transfer the units to the stage “Safe storage with surveillance”.

The basic documents for the decommissioning activities of the VVER 210 and 365 units of the NVNPP are given in Table 3-9:

Table 3-9 Basic planning documents for decommissioning of the NVNPP units 1 and 2

Year of issue	Title of the planning document
1991	First draft of the decommissioning conception
1993	ROSENERGOATOM programme “Working programme on management of radioactive waste generating during NPP-operation and decommissioning”
1995	Operational Procedure for systems and facilities remaining systems and facilities that are in operation for safe cooling of the spent fuel cooling pond and of systems that are relevant for radiation protection

The first draft of the decommissioning concept was carried out by experts of VNIIPIET, VNIIAES, OKB "GIDROPRESS", VGNIPKII, "AEP" in 1989-90. The concept was carried out as a feasibility report for the long-term storage of 1 and 2 units at the NVNPP on the basis of a contract with the NVNPP.

In Russia experience was gained in this field of repair work from the dismantling and conservation activities at the two final shut down units in Novovoronezh and other industrial and military units [3-7, 3-8]. Investigations were carried out for the radiological inventory complex at the shut down Novovoronezh units and also at the Metsamor NPP unit 1 [3-9].

In the field of the remote controlled dismantling activities at VVER-reactors experiences from the dismantling of the Greifswald VVER-440 NPP units will be available in the near future.

3.3.2 Description of the Novovoronezh NPP

The Novovoronezh Nuclear Power Plant is used as the representative NPP for the series of NPPs with VVER-440 reactor systems. The reason is that the third and fourth units of the NVNPP are the forerunner of all other VVER-440 (project V-179) reactors. Units 3 and 4 are the oldest ones of the VVER-440 reactor type units, which are foreseen for decommissioning first. Therefore these units will be not only the forerunner for the operation of the VVER-440 NPPs, but also the forerunner of their decommissioning. The differences between the modifications of the VVER-440 reactor types V-270 (the earthquake resistant modification of the standard V-230 model) and V-213 (second generation of VVER-440) are not essential with regards to their decommissioning. The Novovoronezh VVER-440 and the standard type V-230 are different mainly due to the number of control rods.

The design lifetime of the unit 3 and 4 ends in 2001 and 2002, respectively. At present all work in the field of decommissioning are aimed at the decommissioning of the units 1 and 2, which are already shut down a few years ago. The decision about the future of the units 3 and 4 depends on a lot of boundary conditions. The most important condition being the progress in the construction of the new units in a similar manner to the situation at Leningrad. Consequently, for the operating organisation the first priority concerning the units 3 and 4 have activities with the aim to extend the lifetime of these units.

At present the following units are located on the same site of the Novovoronezh NPP. The units 1 and 2 (VVER-210 and VVER-365) and 5 (VVER-1000) were erected first at the Novovoronezh site as prototypes. The units 3 and 4 are the forerunners of the first standardised VVER-440 models V-230. The VVER-number gives the approximate gross electrical power of the nuclear installation.

3.3.2.1 Historical Overview

Soviet-designed reactors are essentially variations on two basic designs. The VVER- or pressurised light water - type, and the RBMK - the graphite moderated, channel reactor. Water-graphite type of reactors were used as “industrial” reactors for the production of plutonium. The first unit at Novovoronezh was the prototype of a large VVER reactor. Up to the third generation (VVER-1000) of the VVER the reactor units were erected without a containment structure. The first two units were trial reactors for development of large water cooled and water moderated reactors. In 1964, the first unit started up at the Novovoronezh site. The second unit was commissioned in 1969. The increased capacity of the second unit, whose size is nearly the same as that of the first one, was attained due to updating the reactor core, modifying the operating conditions and introducing other improvements (the first NPP in the former GDR in Rheinsberg also belongs to the first generation of VVER reactors).

The development and commercial trial of the reactors of this type have made it possible to build and set in operation nuclear power plants with production line similarities the water-cooled and water-moderated reactors of the VVER-440 type. The third unit of 440 MW capacity and the fourth one of the same capacity were commissioned in December 1971 and 1972 respectively. These reactors belong to the second generation of VVER reactors. The reactors and the turbines of the units 3 and 4 of the NVNPP are located in one building. Tables 3-10 and 3-11 give some details of the designs of the VVER units at Novovoronezh [3-10, 3-11].

The experience gained during the operation of the VVER-440 reactors has, in its turn, been used in the development of a reactor design and its basic equipment for a higher unit capacity, type VVER-1000. The first unit with the VVER-1000 was commissioned in 1980 also at the Novovoronezh site. This unit belongs to the third VVER generation. The VVER-1000 design was developed between 1975 and 1985 based on the requirements of a new Soviet nuclear standard that incorporated some international practices, particularly in the area of plant safety. The VVER-1000 design was intended to be used for many plants, and 18 units now operate in Russia and Ukraine.

Table 3-10 VVER reactor development at the Novovoronezh site

Unit no.	unit 1	unit 2	unit 3 and 4	unit 5
Version	-	-	V-179	V-195
Type	VVER-210	VVER-365	VVER-440	VVER-1000
Mass of the reactor [Mg]	470	523	572	730

Table 3-11 Main data of the Novovoronezh NPP units 1 - 5

Data	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5
Reactor type	VVER-210	VVER-365	VVER-440	VVER-440	VVER-1000
Electrical power [MWe]	210	365	417	417	1000
Construction period	1957-1964	sixties	sixties	early seventies	seventies
Commissioning	1964	1969	1971	1972	1980
Final shut down	1984	1990	operating	operating	operating
Technical data					
Number of primary loops and steam generators	6	8	6	6	4
Number of isolation valves per loop	4	2	2	2	2
primary pressure [MPa]	10.0	10.5	12.5	12.5	16.0
Turbines	3	5	2	2	2

The basic design of the units 1 - 4 is similar. Therefore the generic decommissioning requirements for VVER-440 covers in general also the VVER prototypes 210 and 365. The units 1 and 2 were shut down in 1984 and 1990. The reason for the final shut down of the unit 1 was that the unit achieved the end of the designed life time and limited compliance with present safety requirements. The reason for final shut down of the unit 2 were limited compliance with present safety requirements and also the wrong reactor vessel material being used limiting its operating regime.

3.3.2.2 Location

The Novovoronezh NPP is located 30 km in the South-East of the city Voronezh (Woronesch in Figure 3-7) in South Russia at the left riverside of the river Don. The plant settlement is named Novovoronezh. The town has between 50,000 - 70,000 citizens and is located about 3 to 4 km from the NPP.

In the town of Voronezh a nuclear heating plant AST-500 is under construction, which was foreseen for the heating supply of the town. The construction works have been suspended until a final decision about the destiny of this nuclear facility.

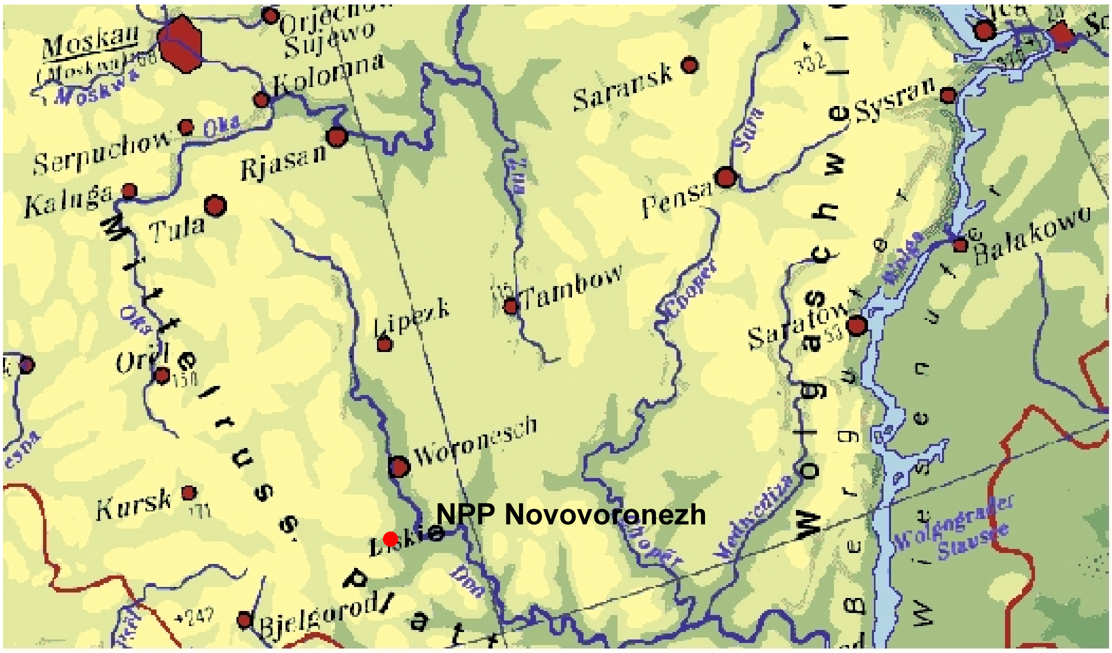


Figure 3-7 Location of the Novovoronezh NPP

3.3.2.3 Description of the facility site

The NVNPP is a multi-unit NPP with the following sections:

- Section 1: Units 1 and 2 - see Figure 3-8
- Section 2: Units 3 and 4
- Section 3: Unit 5
- Section 4 (planned): units 6 and 7 (VVER-1000)

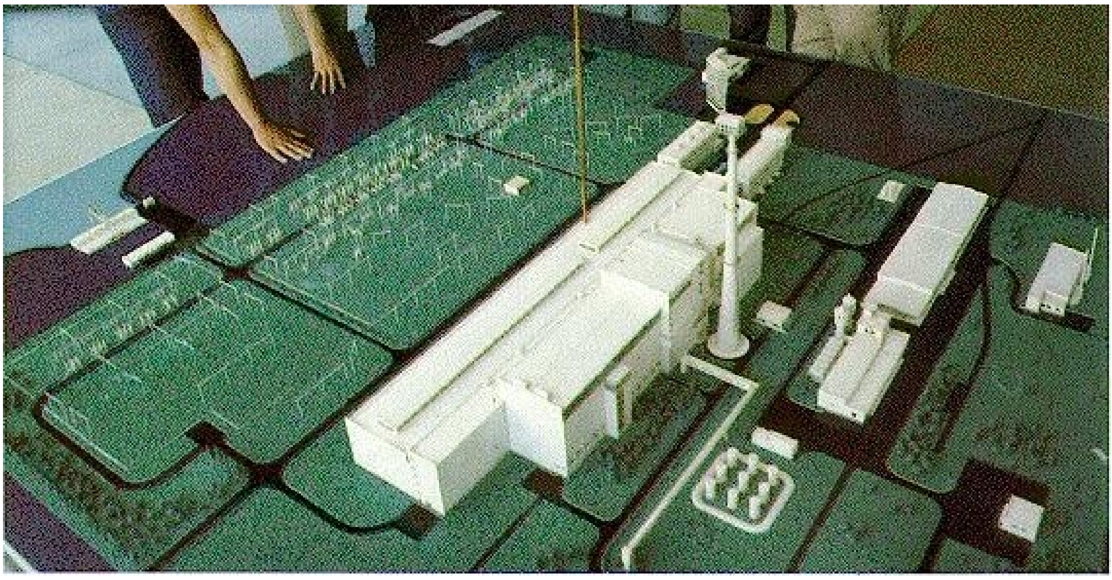


Figure 3-8 Picture of the Model of the Novovoronezh NPP units 1 and 2

Figure 3-9 shows the second generation VVER reactors of the NVNPP. In the picture are shown from left to right the auxiliary building, the reactor building, the ventilation shaft, the sanitary-laboratory building and the administrative building. In the background the turbine hall and one of the cooling towers are located.

Figure 3-9 Picture of the Novovoronezh NPP units 3 and 4

Figure 3-10 shows unit five. This unit is the VVER-1000 prototype.

Figure 3-10 Picture of the Novovoronezh NPP unit 5

At present one or two new units (6 and 7) based of newly designed VVER-1000 are planned. Some preparatory works for the erection of the new units have been carried out but no significant construction work is planned until issues such as funding are resolved. Without this, the site management are reluctant to consider the decommissioning requirements for units 3 and 4.

The Novovoronezh NPP is operated by ROSENERGOATOM.

The layout of the whole NVNPP site is given in Figure 3-11.

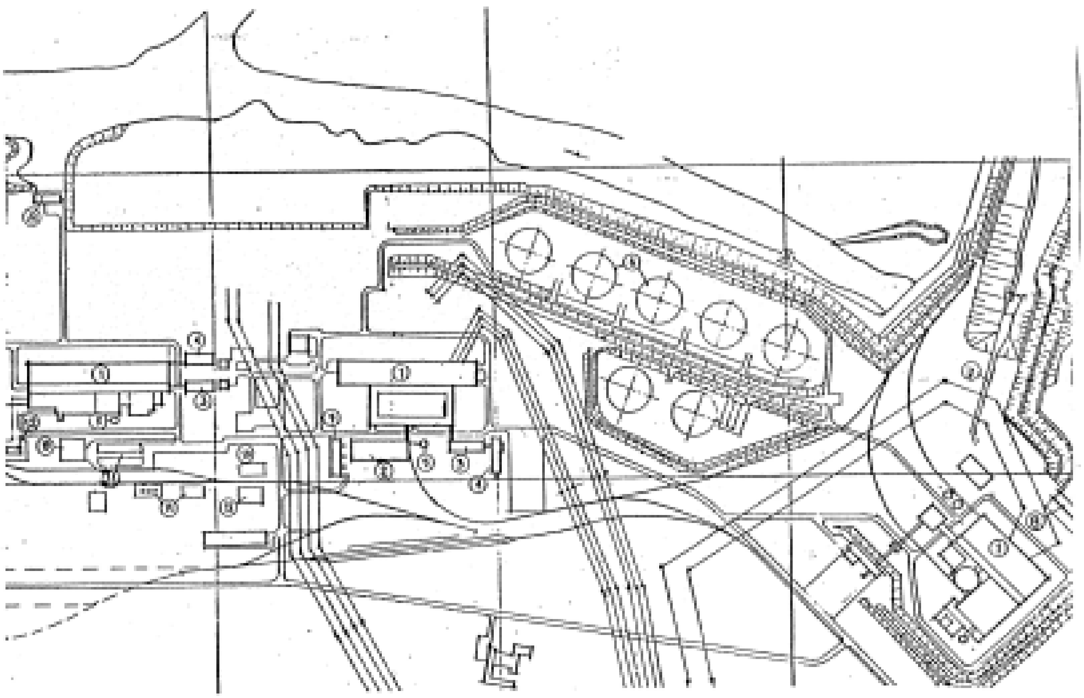


Figure 3-11 Layout of the Novovoronezh NPP site

1	Reactor building
2	United (auxiliary) special building
3	Sanitary-laboratory building
4	Administrative building
5	Ventilation shaft
6	Cooling towers
7	Open cooling water channel
8	Circulating pump station
9	Diesel generator
10	Incorporated auxiliary building
11	Chemical water treatment station
12	Nitrogen-oxygen station
13	Acetylene-generating station
14	Heating building
15	Warehouse

3.3.2.4 Physical Interfaces

The Novovoronezh NPP is connected to the Central energy network in Russia.

The water supply to the nuclear power plant is managed from the Don river via a special pump station (see Figure 3-11).

The generated radioactive waste is stored at on the NPP site, partially in bulk storage for solid waste and in large tanks for liquid radioactive waste. An interim storage facility for solid waste and for special storage drum with solidified waste is under construction.

Spent fuel was regularly shipped away for reprocessing at the MAYAK plant. At present the spent fuel is stored in the cooling ponds of the operating units and at the units 1 and 2.

A wide range of industrial enterprises are located in the region. The staff of the Novovoronezh NPP lives mostly at Novovoronezh. This settlement has increased to a medium-large industrial centre with manufacturers and service companies. Nevertheless, the NPP is the largest employer up to now in the territory.

3.3.2.5 Description of the Section 1 and 2 at Final Closure

The first section of the Novovoronezh NPP consists of two units. The first unit with an electrical capacity of 210 MW was started in September, 1964, the second unit with an electrical capacity of 365 MW in December, 1969.

The NPP has two technological circuits. The first circuit consists of the water-water reactor and circulation loops. Each loop includes the main circulation pump, steam-generator, electrical driven isolation valves (at unit one four valves on each loop, at the second unit - two), pipelines with an internal diameter 500 mm.

The second circuit is non-radioactive, consists of steam-generators, turbines and auxiliaries in the turbine hall. The technical water supply is constructed by direct flow type, with water supply from the Don river.

At present units 3 and 4 are in operation, the decommissioning preparation has not yet started. For comparison the basic characteristics of units 1, 2 and 3 are given in Table 3-12. The unit 4 has parameters similar to unit 3.

Table 3-12 Technical details of the Novovoronezh NPP units 1 - 4

Technical Data	Unit 1 VVER-210	Unit 2 VVER-365	Unit 3 VVER-440
Heating power of the reactor, [MW]	760	1320	1375
Electrical power of the unit, [MW]	3×70	5×73	2x220
Average fuel enrichment, [%]	2,0	3,0	3,6
Average burn-up, [MW d/ton]	13	28	28,6
First core loading, [ton]	38	40	42
Primary circuit pressure, [MPa]	10.0	10.5	12.5
Average temperature of the water at the reactor in-put, [°C]	252	252	268
Average heating-up of the water over the reactor core, [°C]	19.1	25.8	301
Number of primary loops	6	8	6
Diameter of primary lines, [mm]	550 x 25	560 x 30	560
Number of steam-generators in one loop	1	1	1
Number of isolation valves in one loop	4	2	2
Turbines	3	5	2
Power of the steam-generators, [t/h]	230	325	452
Steam pressure in front of the turbine, [MPa]	2.9	2.9	4.4
Specific volume of the building, [m^3/kW]	1.43	1.10	-
Specific expenses of concrete, [kg/kW]	0.29	0.17	-
Specific weight of the equipment, [kg/kW]	74	50	-
working hours of the reactor vessel [h]	123000	129000	-

Figure 3-12 shows the cross-section of the reactor building and the turbine hall [3-11].

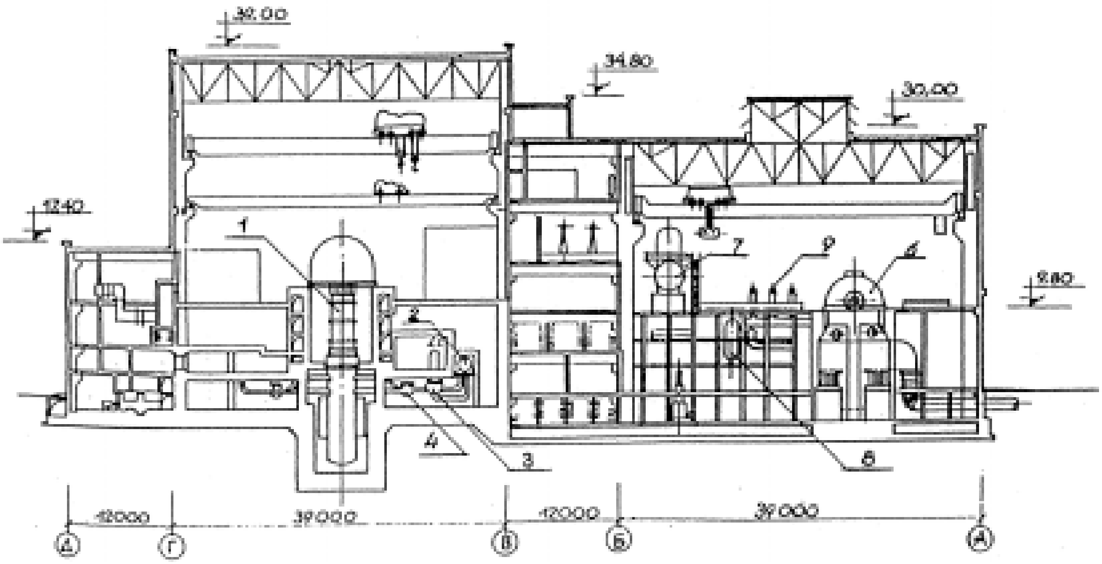


Figure 3-12 Cross-section of the reactor building and the turbine hall

One unit of the double unit section includes the main systems shown in Table 3-13.

Table 3-13 VVER-440 Main systems

No.	Main part	Example
1	Reactor core system	Reactor pressure vessel
2	Primary circuit	Main circulation pumps
3	Auxiliary systems of the reactor	Intermediate cooling circuit
4	Life steam/feed system	Steam generator
5	Emergency systems	Emergency core cooling system
6	Ancillary systems	Ventilation system
7	Fuel handling equipment	Refuelling machine
8	Electrical systems	Normal power supply
9	Monitoring and control systems	Main control room
10	Buildings	Reactor building

Desalinated light water under pressure of 12.5 MPa is used as a moderator and heat-carrier for the reactor (water - water reactor).

The reactor works with enriched uranium oxide fuel with an U-235 enrichment of 2.4-5.6 %. The reload of fuel is carried out under a layer of water after a campaign of 11 months. Spent fuel after unloading from the reactor is stored in the cooling pond. The cooling pond is located in neighbourhood of the reactor in the reactor building . The capacity and sizes of the cooling pond are determined by the opportunity to store spent fuel no less than two years. In the pond 24 fuel element baskets with the spent fuel elements can be placed. One basket contains up to 30 fuel elements. The fuel is stored in the pond under a layer of water, carrying out function of biological protection. The cooling pond is equipped with a special cooling circuit. The reload of spent fuel is carried out using the fuel handling equipment. The fresh fuel is stored in the reactor building.

The technological scheme of the unit consists of two circuits [3-12]. The primary circuit is the so called radioactive circuit. The primary circuit consists of six circulating loops. Each loop includes the main circulating pump, one steam generator, two isolation valves with an electric drive and the circulating pipelines. The pipelines are made from stainless steel with a diameter of 560 mm and a thickness of 52 mm.

The primary circuit consists of systems for heat-up and cool-down of the loops, the primary make-up system with water treatment systems and the emergency input of boric water. The arising liquid waste from the special water treatment systems are stored in the liquid waste store.

The secondary circuit, the so called non radioactive circuit, includes a steam generating part of six steams generator, two turbosets and auxiliary systems.

A possibility of by-pass the steam from the main steam collector in the condensers of the turbines is foreseen (BRU-K). The secondary circuit is equipped with a steam generator blowdown system for cleaning water of the steams generators. One such system serves for both units. The cleaning is made by a filter system.

One unit is equipped with two turbines K-220-44, working on saturated steam with a pressure of 4.4 MPa. Individual capacity of one turboset amounts 220 MW. For the purposes of regeneration heat-up of the feedwater the turbine has 8 unregulated selections.

The feedwater system of each unit consists of two deaerators with a capacity of 800 Mg/h and five feedwater pumps with a capacity of 850 m^3/h (four main feedwater pumps and one emergency pump). In addition, for the case of a voltage drop, each unit is equipped with two emergency feedwater pumps.

The cooling of the systems of both circuits is designed by using closed systems of technical and circulating water supply and by transfer of the heat to the environment through cooling towers. The equipment of the systems of technical and circulating water supply is located outside of the reactor building and serves as a protective barrier for prevention of the release of activity to the environment.

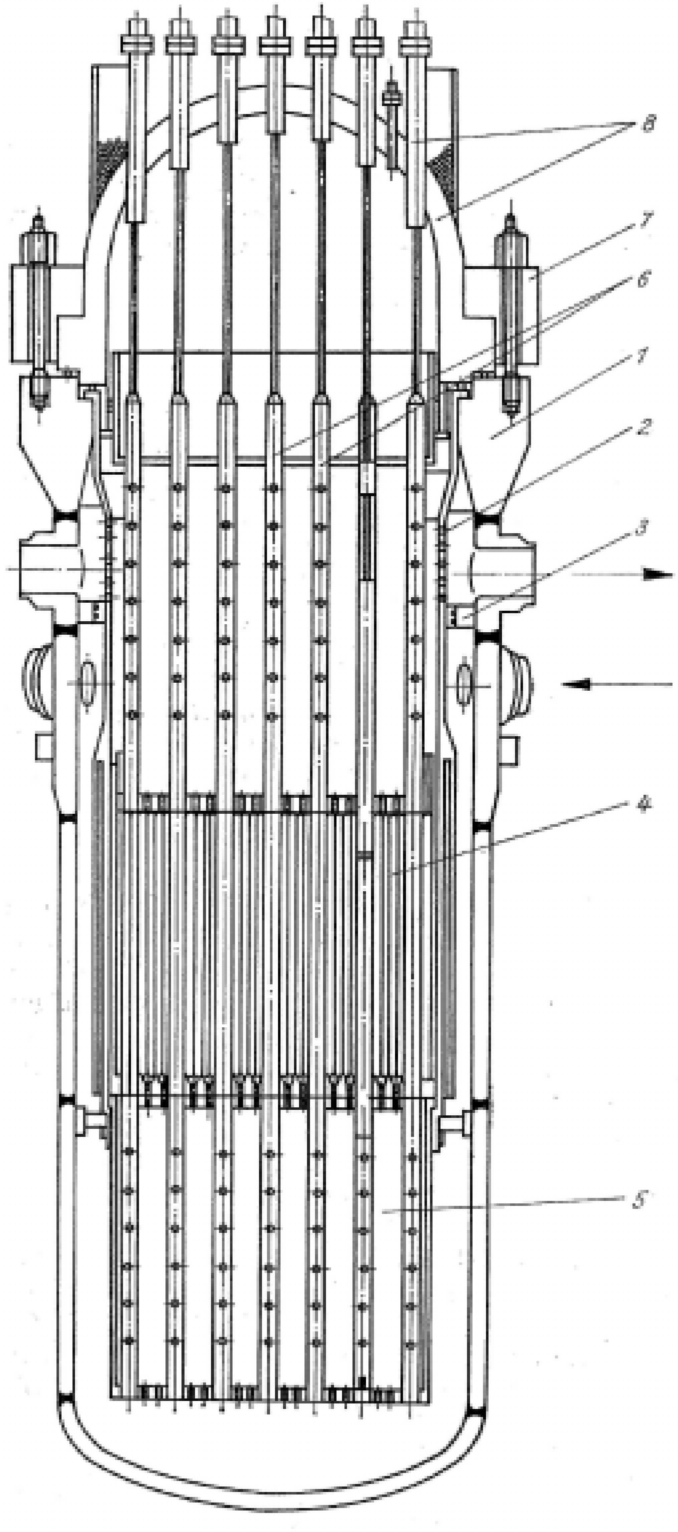


Figure 3-13 VVER-440 Reactor with the main components

1	reactor pressure vessel	5	reactor shaft bottom
2	reactor shaft	6	tube block protector
3	reactor shaft basket	7	pressure ring
4	fuel element basket	8	reactor lid

The reactor and the pressurizer are made from mild steel. The main circulation pipes, the pipelines of auxiliary systems of the first circuit, the closing valves, the pumps and the tube bundle of the steams generators are made from stainless steel.

The structure of NPP management provides:

- a centralised remote control of the basic technological processes
- an automatic control by independent regulators
- a local control and management of the auxiliary systems.

The reactor building is designed in a modular manner. The civil construction is made from reinforced concrete. The foundation plate and the bases of the reactor building are finished with waterproofing materials.

The pressure chamber system of the reactor building is designed in view of withstanding an overpressure of 0.1 MPa. The rooms of the pressure chamber system are designed for a difference of pressure of 0.05 MPa. The rooms of maintenance are designed for a pressure difference of 0.01 MPa. The rooms of the controlled area, depending on their purpose, the temperatures and the possible radioactive contamination, have a special liner made from stainless or mild steel or colouring with chemical proof epoxide colour.

The source of emergency power supply, ensuring the delivery of power for the own needs of the NPP in case of a voltage drop, is served by an emergency diesel power station. The capacity of the diesel power station is chosen depending on the total capacity of the consumers of the network of reliable power supply with a 100 % reserve. The diesel power station consists of three cells, completely isolated among themselves. Each of them consists oft two diesel generators with a capacity of 2 x 1600 kW, on a voltage of 6 kV. All operations on start-up of diesel generators are automated.

The content and the weight of the main equipment for one VVER-440 unit of the Novovoronezh NPP are summarised in Table 3-14.

Table 3-14 Selected mass parameters of the equipment of one VVER-440 unit

no.	Name	Mass (Mg)
Main building		
1	reactor (with inner equipment)	500
2	primary loop with steam-generators, main circulation pumps and isolation valves	1500
3	Pressurizer	130
4	emergency- and auxiliary systems	700
5	cleaning make-up water system	20
6	intermediate main circulation pumps coolant circuit	50
7	intermediate emergency and control system coolant circuit	10
8	emergency boric water system	65
9	spray system (sprinklers)	100
10	cooling pond coolant circuit	30
11	special water treatment system	40
12	Equipment of the specialised ventilation system	60
13	leakage water treatment system	100
14	fuel handling equipment	450
15	electrotechnical equipment	600
turbine hall		
1	Turbine	4500
2	Condenser	400
3	feedwater system	1000
4	emergency feedwater system	30
5	live steam system	350
6	technical water treatment	200
7	water coolant system	250
8	electrotechnical equipment	3000

Tables 3-15 and 3-16 identify the decommissioning requirements for the NPP.

Table 3-15 Facilities needed for decommissioning

1.	Reactor building and the turbine hall	The physical structure must be investigated. The designed lifetime of civil constructions at the NPP is approximately 100 years.
2.	Reactivity and coolant control system	The physical structure of the equipment and pipelines must be investigated immediately after the final shut down of the reactor.
3.	Reactor fuel ponds	The physical structure of the equipment and pipelines must be investigated immediately after the final shut down of the reactor.
4.	Spent fuel interim storage	The common usage of the existing and new installed equipment of the operating units must be planned.
5.	Liquid and solid waste treatment facilities and interim stores	The common usage of the existing and new installed equipment of the operating units must be planned.
6.	Service facilities	The common usage of the existing and new installed equipment of the operating units must be planned.

Table 3-16 New systems needed for decommissioning

Heating system	if heating can not be provided from the operating units
Power generation	if power supply can not be provided from the operating units
Liquid waste retrieval and solidification facilities	if liquid waste treatment and solidification facilities are not available in operating units
Solid waste retrieval and processing facilities	if solid waste processing facilities are not available in operating units

The requirements given in Tables 3-15 and 3-16 are generic in nature and can be applied to all VVER systems in CIS countries.

3.3.2.6 Operating History of Both Generations Relevant to Decommissioning

3.3.2.6.1 First generation

During the inspection of the reactor facility various cracks were found out in metal of transitive cartridges and welded seams of reactor and pipelines ∅ 500 mm. All 12 pipes were pierced to a depth of 11-13 mm, and in them were pressed and are scalded corrosion-proof cartridges.

Practically over all perimeter, in the place of transition of the cylindrical part of the reactor vessel to the bottom part, a deleted zone of width 100-150 mm was formed by the fallen screen inside the corrosion-proof plating. The reason for this case was the pulsing water flow. The metal in this place was smoothly polished off, the metal thickness was decreased by 11-13 mm.

Since 1971 the unit was operated with nominal parameters without a thermal shield, which fragments were stored in a storage.

In 1974-1977 the main circulation pumps type GZN-138 were replaced to type GZN-309A caused by constructive lacks of the first type. The knees and separate sites of the main circulation circuit were dismantled and replaced.

During the planned maintenance in 1980 were found out cracks of intercrystal character in the field of corrosion-proof plating with length of 10-15 mm. The cracks were polished to a depth of 22 mm and 20 mm; thus it was necessary to open the plating over a area 60 x 50 mm and 40 x 40 mm. At these areas stainless plates are welded on with a diameter of 125 mm and 100 mm accordingly.

During 1980 the state of the main isolation valves of the primary circuit was checked, samples of the basic pipelines of the primary and secondary circuits were cut out for research of metal and welded seams after 100 thousand hours of operation. The researches of valves and pipelines have shown their satisfactory state.

In August, 1984 the unit was stopped, the operation was continued on the basis of the power effect up to a level of 30 % nominal power.

The second unit of the NV NPP was started in December, 1969, and since April 1, 1970 has worked with nominal parameters.

In 1986, during the planned preventive maintenance, through cracks at the support of the reactor shaft were found out, and also vibrating deterioration of the pins at the reactor shaft up to 1.35 mm and 2.0 mm accordingly. Research has shown that the damages of the basic support of the shaft and of the fixing devices of the shaft are caused by material fatigue.

The following works are executed to eliminate the damage:

- The basic support of the shaft was repaired.
- The pin slots at the reactor shaft were maintained (sample, layer welding and polishing).
- Inserts were established in the pin slots of the reactor shaft.
- A new arm no. 5 with pin no. 5 was established and welded.
- The working surfaces of the pins no. 1, 2, 3, 4, 6, 7, 8 were polished.
- The device for fixing of the top unit of fastening of the shaft was replaced.
- A layer was welded on the basic surface of the support of the shaft and this surface was polished.

The thermal reactor shield was dismantled and was disposed of in 1986 with the aim to ensure the monitoring of the state of the metal of the cylindrical part of the reactor vessel.

Four defects as cuts were found caused by dismantling of the thermal screen during the planned preventive maintenance 1990. The defects are located between welded seams no. 4 and no. 5 in the core area. The total length of four defects was 900 mm, the maximal depth 35 mm. The defects were eliminated by polishing.

In August, 1990 the decision was made to shut down the second unit at the NVNPP. In December, 1990 the reactor together with the in-core equipment was preserved.

The present state of the NVNPP units 1-2 at final closure can be described as “pre-decommissioning stage” [3-4]. The situation is the following:

- The systems and facilities, necessary for operation of the units, are selected. they are disconnected from the systems out of work. The design of this work step was approved by the Head Designer of the NPP AEP Moscow.

- The new “Procedure for operating of the units 1 and 2”, taking into account the pre-decommissioning stage, is developed, approved by the Authority and implemented.
- The reconstruction of the buildings and rooms of the units 1 and 2 is ongoing with the aim to prepare an interim storage for drums of waste, generating by the high concentration evaporator UGU-500.
- The dismantling of the turbosets no. 1-3 of unit no. 1 was started in February 1996 after the approval of the Authority.
- The development of technologies and equipment for underwater cutting of high-activated equipment is finished. Cutting is ongoing for conditioning of absorber and control rod drives. The development of technologies and equipment of conditioning of in-core-equipment is ongoing.
- Two facilities of high concentration evaporators are commissioned. These evaporators will be used for the treatment of liquid radwaste generated during operation and decommissioning.
- The complex engineering and radiation investigation of the units 1 and 2 is finished.

3.3.2.6.2 Second generation

The soviet designed nuclear power plants differ from the Western NPP in many respects, including plant instrumentation and controls, safety systems and fire protection systems. The earliest pressurised water nuclear power plants were developed between 1956 and 1970. These plants include the Novovoronezh prototypes and the first standardised model V-230. The former planning of reactor units do not reflect the requirements of the decommissioning of the reactors after expiration the life cycle. Furthermore the projects do not include a sufficient store volume for waste and installations for waste processing and conditioning. Up to now no national wide solutions of a final radwaste repository exist.

Information about incidents or accidents at the units 3 and 4 with implications to their decommissioning are not available.

3.3.3 Radioactive Inventory/Hazardous Material

After final closure, there will remain on the facility site significant amounts of materials that are potentially harmful (radiologically and conventionally) to people and the environment.

This section describes the locations, identities, physical forms and quantities of hazardous materials on the facility site at the final shut down, at the end of Stage 1, and at the end of Stage 2 [3-13].

Based on the information from [3-14], the present section gives information on amounts of radioactive waste existing at a typical VVER-440 unit. Information about the Novovoronezh NPP units 3 and 4 are not available. As well as overall information about the main radwaste categories, information will be given on waste category, waste amount, waste origin, radiological waste characteristics with the total nuclide content and the most important nuclides. Information on conventionally hazardous materials and their amounts are also given for a typical VVER-440 unit.

For a further qualification of a Decommissioning Plan more information than that available at present is necessary. Detailed information is needed about amounts of radioactive waste existing at the NPP and/or the unit site at final shutdown and about the amounts expected at the end of the various decommissioning stages (Stage 1, Stage 2). Furthermore, the existing overall information about the main radwaste arising must be supplemented by detailed information for the unit in a system by system way with data on waste category, waste amounts, waste origin, radiological waste characteristics with the total nuclide content and the nuclide content of the most important nuclides at various time periods after final shutdown.

In addition information is needed on the present waste conditioning state, future waste management activities for re-use, recycling, exemption from further control, waste treatment technologies, interim storage, disposal, and timetable for radwaste management activities. Some parts of the radioactive waste can also have conventionally dangerous properties so that in addition to radiation protection measures other protection measures are needed.

For the radiological as well as for the conventional hazardous materials, a regular inventory and the development of a corresponding material balance should be carried out with the beginning in Stage 0 and the end at the completion of the decommissioning to the planned stage. A hazardous material book keeping system should be established. This system can provide the basis for decommissioning and waste management planning including equipment demand. It shows the origin of hazardous materials and the treatment route from the origin to the recycling, re-use or disposal with amount, date, and accompanying documentation.

3.3.3.1 Inventory at final closure

The radioactive inventory is classified by:

- the radiological characteristics,
- the manner of activity (contamination and/or activation),
- the system containing the inventories,
- the buildings that contain the inventories.

The radioactive inventory of the of the unit is divided in the following groups:

1. contaminated components
2. activated (and contaminated) main reactor components
3. spent fuel
4. liquid radioactive waste
5. solid radioactive waste

This grouping method is used for planning of the decommissioning activities and for planning in detail of the dismantling steps.

Contaminated components

- Contaminated components of the secondary circuit

  Contamination of the secondary circuit should be considered. Possible contamination is caused by leakages from the primary to the secondary circuit:

  – in the steam generator (tube damage) and

  – in other auxiliary system heat exchangers.

  Possible contaminated systems (more than 0.5 Bq/cm²) of the secondary circuit are:

  – steam feed pipe

  – feedwater tank with deaerator

  – condenser

  – condensate settling tank

  – pumps of the condensate settling tank

  – cleanup blowdown water treatment system of the secondary circuit

- Contaminated components of the primary circuit

  Contaminated systems of the primary circuit include all primary systems and connections to the primary circuit auxiliary systems, such as the special water treatment systems, e.g.

for cleanup the primary circuit water and the cleanup of leakage water. For some selected contaminated main components of the primary circuit their mass and the dose rate at a distance of one meter from the surface of these components are given later. Also the off-gas system and parts of the ventilation system are contaminated. The inner contamination of the primary circuit components amounts up to some 10^4 Bq/cm² Co-60. Table 3-17 gives dose rates of primary circuit components.

Table 3-17 Dose rate of the contaminated main components of the primary circuit

Component	Mass [Mg]	Dose rate at 1 m [mSv/h]
Steam generator	155	0.1
Main circulation pump	50	0.5
Isolation valves	10	0.3
Pressurizer	110	0.1

There are two common groups of nuclide contamination which are present depending on the location and history of the contaminated components:

– the fission product Cs-137 (30 - 50 %)

– activated corrosion products Co-60 (70 %) and Fe-55, and other nuclides like Ag-110 and Mn-54

The contamination of surfaces of the rooms and floors within the reactor buildings are much less than by some orders of magnitude to the inner contamination of the primary circuit components. However the contamination of the room walls and floors must still be considered.

Main (high active) reactor components

An overview about the activated reactor components is given in Table 3-18 [3-15].

Table 3-18 Radiological characteristics of reactor components after shut down

Reactor component	Material	Mass [Mg]	Maximal Mass-specific Co-60 Activity* [Bq/g]	Co-60 Activity inventory [Bq]	Total Activity inventory [Bq]	Dose rate 1 m [mSv/h]
Tube block protector	Stainless steel	30	2×10^8	5×10^{14}	1.5×10^{15}	200

Reactor component	Material	Mass [Mg]	Maximal Mass-specific Co-60 Activity* [Bq/g]	Co-60 Activity inventory [Bq]	Total Activity inventory [Bq]	Dose rate 1 m [mSv/h]
Fuel element basket	Stainless steel	25	8×10^{8}	5×10^{15}	1.5×10^{16}	7000
Reactor shaft	Stainless steel	60	2×10^{8}	1×10^{15}	3×10^{15}	2000
Reactor pressure vessel (lower part)	mild-steel	200	3×10^{6}	3×10^{13}	3×10^{14}	200
Thermal insulation	mild-steel	30	4×10^{4}	2×10^{10}	2×10^{11}	10
Water ring jacket	mild-steel	60	2×10^{5}	4×10^{12}	3×10^{13}	30
Biological shield	heavy concrete	100	1×10^{0}	-	1×10^{8}	-
Sum					2×10^{16}	

* - for the Biological shield Fe-55

The activity of components is mostly concentrated in the fuel element basket and the reactor shaft. The activity of the biological shield is very low. This is caused by the presence of the water ring jacket. About 99 % of the total activity of the primary circuit is concentrated in the activated components of the reactor pressure vessel.

The nuclide share of the components activity depending on the used material is presented in Table 3-19.

Table 3-19 Nuclide share of activated materials

Nuclide	Stainless steel %	mild-steel %
Mn-54	0.1	0.6
Fe-55	43.7	87.2
Co-60	32.5	11.5
Ni-59	0.2	0.1
Ni-63	23.5	0.8

Spent fuel

The most active waste is the spent fuel. With the removal of the spent fuel from the site the radioactive inventory decreases significantly. Damaged spent fuel elements require special handling (packaging into special cans). Up to 720 spent fuel elements can be stored in the spent fuel pond of one unit. The number of the storage casks and/or the size of the interim storage for spent fuel depends on the remaining number of spent fuel elements after final shut down, which are not shipped away from the unit. Because this number of spent fuel elements at final closure is not available the total number of spent fuel elements for the whole life time of one unit is presented in Table 3-20.

Table 3-20 Spent fuel of one VVER-440 unit

	per operating year and per unit	for a operating period of 30 years

Number of fuel elements	120	3600

Liquid waste

A typical amount of arising liquid waste per year for a VVER-440 unit is given in Table 3-21 [3-16].

Table 3-21 Annual arising liquid waste at one VVER-440 unit

No.	Kind of waste	Amount [m³/a]
1	concentrates	120 - 300
2	resins (spent ion exchange)	20
3	oil	1

Solid waste

A typical amount of arising solid waste per year for a VVER-440 unit is given in Table 3-22. The amount does not split between incinerable, non-compactable and compactable solid wastes.

Table 3-22 Solid waste

No.	Type of waste	Amount [m³/a]
	LLSW	100
	ILSW	30
	HLW	0.5
	after deep evaporation (in case of use a deep overconcentrator)	(40)

The solidified waste arising after treatment of concentrates also belongs to this solid waste generation.

<u>3.3.3.2 Inventory at End of Stage 1</u>

Preparation and execution of decommissioning activities resulting in radwaste generation are initiated immediately after final shutdown. During the Stage 1.1 and 2.1 (see 3.3.4) the processing of the radwaste present at the final shut down must be continued with the objective to:

- store them for a time period at on-site engineered storage facilities, preferably in a conditioned form, or
- exempt them from further control, or

- reuse them, or
- to discharge them to off-site disposal.

At the same time radwaste will arise from decommissioning activities. They have to be processed in the same way as the radwaste present at final shut down. At the end of Stage 1 there should be no waste from the operation phase of the NPP. There remains the radwaste from the Stage 1 decommissioning activities as well as from the processing of waste and waste from the residual operations.

These amounts of waste have to be processed during Stage 2 phase and must be discharged from the NPP site at the end of Stage 3 at the latest.

During Stage 1 the spent fuel will be removed from the unit. Thereby the main source of radiological risk and the risk of a criticality fault will be eliminated. If the radwaste is also processed the activity inventory decreases further.

The remaining activity inventory is mainly enclosed in the systems and components of the primary circuit. After the period of preparation for care and maintenance the decrease in this activity inventory is not significant.

3.3.3.3 Inventory at End of Stage 2

During the Stage 2 phase the processing of the remainder of the radwaste present at the final shut down and from Stage 1 activities must be continued with the objective to store them for a time period at on-site storage facilities.

At the same time radwaste generated during Stage 2 has to be processed in the same way as the radwaste present at final shut down and produced during Stage 1 phase. At the end of Stage 2 there are radwastes from the Stage 2 decommissioning activities as well as from the care and maintenance operation which have to be processed.

The main aim of the deferral periods (stage 1.2 and 2.2) is to reach a significant lower activity inventory. In the case that the radwaste is mainly removed from the unit, the activity inventory is located in the systems and components of the primary circuit within the reactor building.

The decrease of the mass specific activity of reactor components and of the dose rate at 1 meter from the surface of the fuel element basket is shown in Figure 3-14. Considering the nuclide share of the activated components after a long decay period only the activity of the Nickel nuclides Ni-63 and Ni-59 remain (see Figure 3-15).

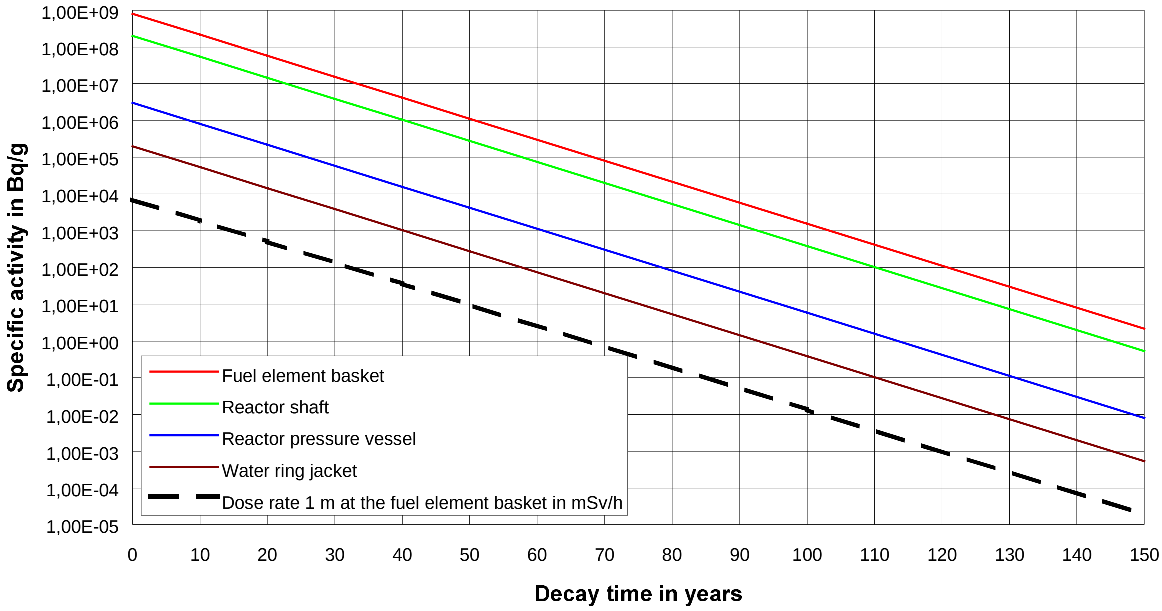


Figure 3-14 Decrease of the mass specific activity of high activated reactor components

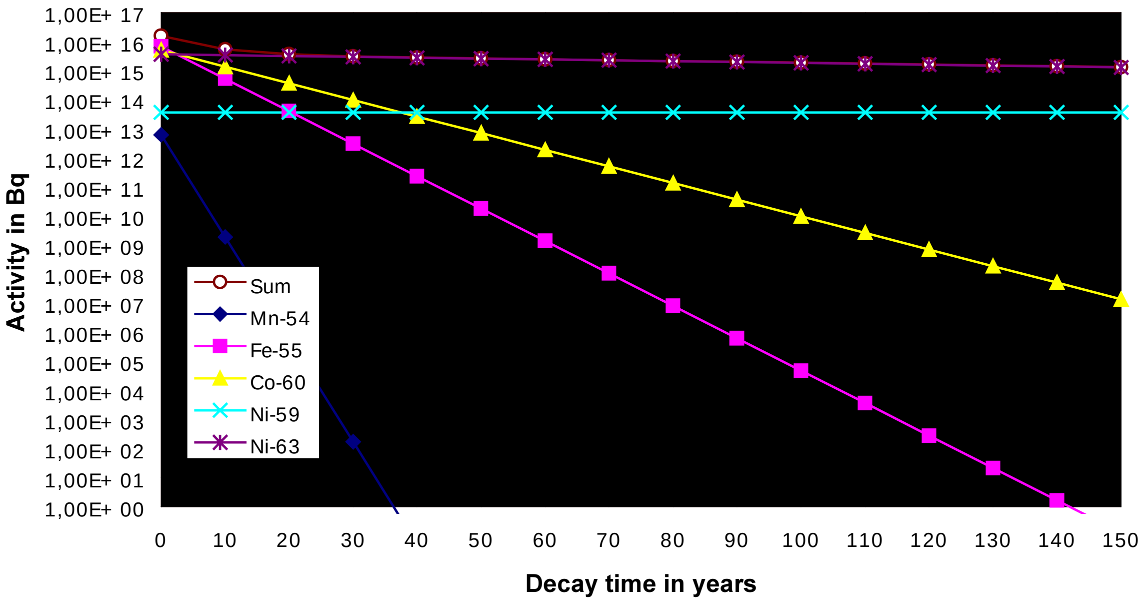


Figure 3-15 Nuclide share of the activity of the fuel element basket

After a decay time of 100 years a remote controlled dismantling of the rector and the built-in components is not necessary, because the dose rate of the highest activated fuel element basket reduces to 10 µSv/h. This is caused by the decrease in Co-60 activity by 6 times.

Depending on the decommissioning and dismantling activities in the special and auxiliary buildings the stored radioactive waste and the contaminated systems, tools and building structures must be considered.

3.3.3.4 Conventionally Hazardous Materials

Beside radioactive waste there are also conventionally hazardous materials at the NPP site at the final shutdown and their description has been covered earlier in this report when considering the RBMK systems.

At present, for the Novovoronezh NPP no data are available on existing amounts of conventionally hazardous materials and expected arising from operation and future decommissioning activities. There is no information on conventionally hazardous waste routes including storage, treatment, disposal and on applied technologies.

The following example gives the typical amount of conventionally hazardous materials at final closure for the Greifswald NPP (see Table 3-23).

Table 3-23 Typical amount of conventionally hazardous materials at VVER-440 units

Group	Material	Volume [m³]	Mass [kg]
Operating agents	hydrazine	5	-
	ammonia	10	-
	potassium dichromate	300	-
Cleaning, decontamination and other chemical agents	Solvents	20	-
	laboratory chemicals	-	50
	lacquers and colours	-	200
Oil		50	-

3.3.4 Cost Estimates and Funding Requirements

This section gives a presentation of a simple decommissioning cost estimate. It must be explained, that the decommissioning efforts depend on a lot of influencing factors. The results of cost estimates strongly depend on the assumptions made for the boundary conditions of the decommissioning project. Cost estimates for Western Europe units are based on defined boundary conditions. These boundary conditions for the CIS-countries are not sufficiently defined, because the economic situation is changing between a central controlled and a market orientated economy.

Considering the multi-unit structure of nuclear power plants, some costs e.g. the overhead costs for site operation cannot be assigned clearly to operating units or units under decommissioning. Furthermore the new installed facilities, necessary for decommissioning and the safe radioactive waste management, are used for all units independent from their operating state.

In this section the decommissioning costs are estimated for the stage 0 and 1.1 up to eight years after the final shut down of the reactor for one unit. The cost for the period of care and maintenance of the reactor building, for the dismantling and the operation of the turbine hall as a waste storage and the operation of other remaining buildings are not included.

As an indication of the whole decommissioning costs literature sources were used [3-3, 3-17, 3-18]. Based on these dates the costs are determined for the complete dismantling of one VVER-440 unit after a deferral period.

The estimate of the decommissioning costs could be a basis for the detailed economic and commercial assessment of the decommissioning financing. This financial planning must also include the detailed information about the primary masses to be decommissioned, the detailed information about the boundary conditions for decommissioning and dismantling. Based on such a detailed decommissioning project it is possible to identify the decommissioning activities and the necessary equipment, facilities and other means.

The efforts and costs are described in this report based on Western Europe experience. The used currency is EURO. The ***decommissioning stages*** are divided into ***working packages***. For further detailed cost estimate the working packages usually are divided into working steps. (For additionally collective dose calculations working areas with typical dose rates are used). The working packages are determined according to functional and cost aspects. This division also serves to provide a clearer structure and reproducibility of the individual decommissioning activities.

The working package "Site operation" (e.g. project management, supervision) is an integral part of all parallel stages. The part of the site operation, which can be assigned only to the decommissioned units must be added to the decommissioning cost.

Three cost categories are usually determined for each working step:

- labour costs which are calculated on the basis of the workload required by a working package and the assumed labour cost unit rate
- investment costs of the equipment and machinery used for a particular working step

- costs of consumables (protective clothing worn in controlled areas, decontamination fluids etc.)

The described cost estimate of a working packages includes only the labour costs, which are the biggest percentage of the total costs of these packages.

The decommissioning project is organised in a task structure according to the decommissioning stages.

In stage 0 the most important working packages are the planning and licensing (P&L) for the decommissioning strategy, the P&L for the decommissioning stage 1.1 and 2.1 and the P&L for the new facilities. A brief characterisation of all working packages of this stage are listed below.

– P&L for the decommissioning strategy

– P&L for the decommissioning stage 1.1 and 2.1

– P&L for the new facilities

The cost estimate of the preparation stage for care and maintenance (stage 1.1) includes the following working packages:

– Planning and licensing of the decommissioning task 1.2 and 2.2 (C&M)

– Erecting of the new facilities - see Table 3-24

Table 3-24 New facilities required

Waste form	Facility	Capacity	Equipment cost [M€]	Number of operators per shift
	Treatment and conditioning			
Liquid waste	Cementation	0.4 m³/h	~ 5	2
Solid waste	Supercompaction	15 (200 l drums/h)	~ 3	3
	Incinerator	50 kg/h	~ 5	3
	Cementation	1 m³/h	~ 1	2
	Decontamination	1 m³/h	~ 1	2
	Melting facility	2 t/h	~ 20	6
	Release measurement facility	5 (200 l drums/h)	~ 0.5	2
	Storage			
Spent fuel storage	Dry spent fuel storage		~ 20	2
Storage for solid and solidified waste	Interim storage for waste drums	3000 drums	~ 10	3

– Shut down reactor, removal of the spent fuel from the reactor core

– Shut down of not used systems

– Processing of operational liquid and solid waste

– Removal of core internals

– Removal of spent fuel

– Modification/Installation of systems and preparation of building

- Site Operation and Supervision

  For the cost estimate the site operation and supervision must also be considered. The overhead cost for the site operation, which should also be assigned to the decommissioning project, amounts to a significant percentage of the whole decommissioning project costs.

  Because the estimate of these costs is very difficult, for the generic decommissioning requirement the tasks of the overhead costs are only listed below:

  – General plant management

  – Decommissioning project management:

  – Miscellaneous (chemicals, conventional waste)

  – Energy, heating

  – Water, sanitary effluents

  – Operation of sanitary area and radioactive laundry

  – Social costs (canteen, travel cost, others)

  – Service (telephone, drivers, office helpers, copies, EDV)

  – Guards and gatekeeper

  – Supervision decommissioning work including:

  – Radiological protection

The results of cost estimate of part of the decommissioning are shown in Table 3-25.

Table 3-25 Labour cost estimates

Stage	Working-package	costs [Million €]
Preparation for closure	Planning and licensing of the decommissioning Strategy	0.3
	Planning and licensing of the decommissioning stage 1.1 (PCM)	1.2
	Planning and licensing of the decommissioning stage 2.1 (PRSU)	1.0
	Planning and licensing of the new facilities	2.0
	Complex investigations	1.0

	Documentation	1.0
	Total (phase)	7
Preparation for Care and Maintenance	Planning and licensing of the decommissioning stage 1.2 and 2.2 (C&M)	1.1
	Erecting necessary waste treatment facilities	-
	Erecting of interim stores for waste	-
	Erecting of interim stores for spent fuel	-
	Shut down reactor and Removal spent fuel from the reactor core	-
	Removal and processing of core internals	0.3
	Shut down of not used systems	1.3
	Processing of operational and decommissioning waste	1.0
	Transport of spent fuel to the interim storage	2.5
	Modification/Installation of systems and preparation of building	5
	Total (phase)	11

For the cost estimate a project management tool was used. To each working package was assigned a working staff with different qualifications. By this way the total labour cost are estimated and the duration of the packages can be planned. The proposed approach is one possibility for a rough cost estimate.

Cost estimate for VVER-440 units were also carried out by the VNIIAES [3-3] and the summary cost for one VVER-440 unit is shown in Table 3-26.

Table 3-26 Summary cost for one VVER-440 unit

Stage	Name	Cost [Million €]
0	Pre-decommissioning stage	50
1	Unit preparation for safe storage under surveillance	53
2	Unit safe storage under surveillance	15
3	Unit complete dismantling*	40
Total		158

*) cost only for dismantling of reactor and all residual radioactive parts of the unit

The cost for the decommissioning and dismantling of the Greifswald NPP was also calculated. For this the programme code STILLKO 2 [3-19] was used. The results of this cost estimate are shown in Table 3-27. By using the STILLKO code for the NPP Greifswald the total costs for all stages up to the green field are estimated by 3 Billion € for all eight reactors and buildings [3-18]. This decommissioning variant for the whole Greifswald NPP includes the following projects (see Table 3-27):

1. Decommissioning project A (variant later dismantling after a deferral period of 30 years, the preparation period and the dismantling period last about 19 years)

   The main objectives of the later dismantling decommissioning variant of the Greifswald NPP is the decommissioning and dismantling of all buildings except the reactor building and the erecting of a large interim store. The Russian variant keeps most of the buildings ready for operation. This is the main difference between these two variants. Dismantling within the controlled area before care and maintenance (working package A 2) includes the decommissioning and demolition of the buildings except the reactor building to decrease the surveyed area within the deferral period (see Figure 3-16). After a correction with regard to lower labour cost rates and the lower dismantling activities (working package A 2) these costs are comparable with the expected costs for a Russian VVER-440 unit.

The working packages ‘Documentation’ and ‘Planning and licensing’ (working packages F2 and F7) (about 9 Million €) are comparable with the planning and licensing and the documentation in the current cost estimate. The working packages ‘Dismantling within the surveyed area’, the ‘Dismantling within the controlled area before care and maintenance’ and the ‘Preparation for care and maintenance’ (about 25 Million €) includes dismantling and preparation work ‘for Preparation for care and maintenance’. To the current cost estimate in the way described above, which includes only the labour cost of one working NPP, must be added the investment cost, cost for license and external cost. All other cost must also be considered, e.g. for new facilities, for waste management and the overhead costs.

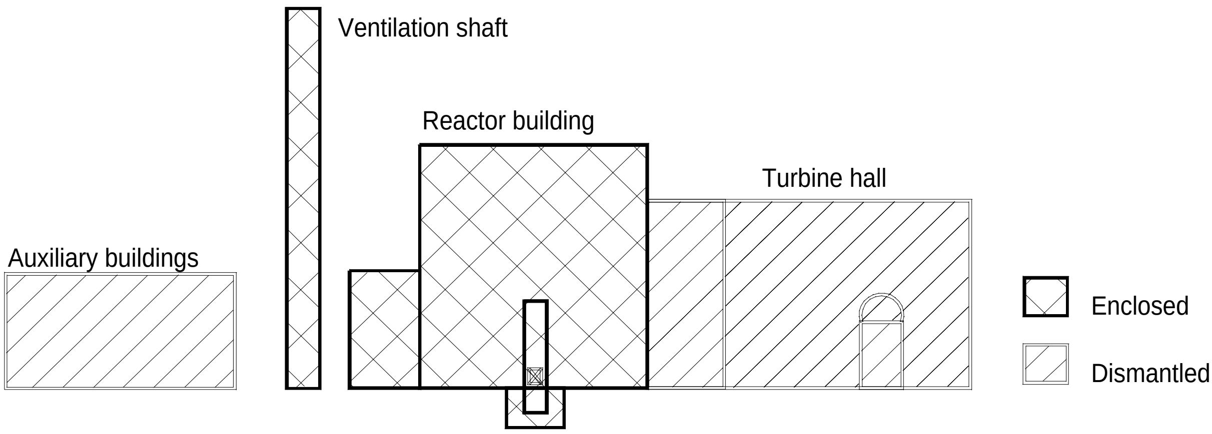


Figure 3-16 Later dismantling option for the Greifswald NPP

2. Modification / New infrastructure for waste handling B

   The project B includes the modification of the central active workshop, the installation of necessary waste treatment and conditioning facilities, additional waste interim stores, the installation of activity measurement devices and the modification or new installation of ventilation systems. After a correction with regard to lower labour cost rates these costs are also comparable with the expected costs for a Russian VVER-440 unit.

3. Interim storage C

The interim storage stores for a long term period (at least 40 years):

- the CASTOR casks with the spent fuel elements
- the waste, for which is foreseen storage for radioactive decay
- large dismantled components (e.g. steam generators)
- other operational and decommissioning waste, which are not disposed, e.g. in the final repository at MORSLEBEN
- waste, for which is foreseen a reuse under controlled conditions
- other waste (medical, industrial and scientific radioactive waste)

The operation of the interim storage is independent from the NPP site. The interim storage includes also a social requirement with offices, workshops, laboratory, material storage, laundry and supply mains. The erecting of such a large interim storage in the CIS countries is not planned. Nevertheless the cost for the interim storage gives a good idea for the expected cost for a interim storage facility.

4. Waste Management D

   The waste management includes mainly expenses for fees of the final disposal of waste (which is very high in Germany). Also the investment costs for containers and other means for packaging and the external (transport) costs are very high. These costs are actually not comparable with the expected cost in the CIS countries. Nevertheless the establishment of an national final repository is necessary.

5. New and reconstruction of infrastructure and buildings E

   This working package includes the improvement of the infrastructure like social buildings, warehouses, streets and other required infrastructure for further operation.

6. Overhead costs F

   The overhead costs includes project management, expenditure for the planning and licensing, documentation, quality assurance, common expenses for the site operation, overhead costs and some expenses for the post operation period. The comparison of these costs is more difficult, because they depend on a lot of factors. It must be noted, that the Greifswald decommissioning project includes only the decommissioning without any operating units. Unlike the single decommissioning project, the overhead cost for an operating site could be assigned to the operating units.

For the cost estimate for one VVER-440 unit in Russia on the basis of the cost estimate for the decommissioning of the Greifswald NPP the following items are assumed:

- Labour cost rates for the CIS country are assumed to be a quarter of the German ones.
- Consumable costs of waste management are assumed to be one third, while the very high German costs for packaging of waste to be one tenth for the CIS countries (except for spent fuel management cost). Fees and external costs are not considered.
- The cost are estimated for the whole Greifswald NPP site. Therefore for comparison the prorated costs of the decommissioning of one VVER-440 unit of the Novovoronezh NPP are determined. Depending on the number of units being considered of the Greifswald NPP a factor is created for the single cost estimate. The Greifswald decommissioning working package costs are divided by this factor for estimate of the single decommissioning cost of one unit. It must be noted that the decommissioning of only one unit under the conditions of the multi-unit structure of the Russian NPP is not reasonable. Therefore for all new installations and activities the planning and licensing should be carried out preferably for all the units on the site.

The results of the cost estimate are presented in the Table 3-27. The total cost for the whole decommissioning variant amounts to 360 Million € (not discounted). It must be considered that this amount is for a period of about 50 years including the total dismantling and demolition of all buildings. The share of the working package cost is shown in the Figure 3-17.

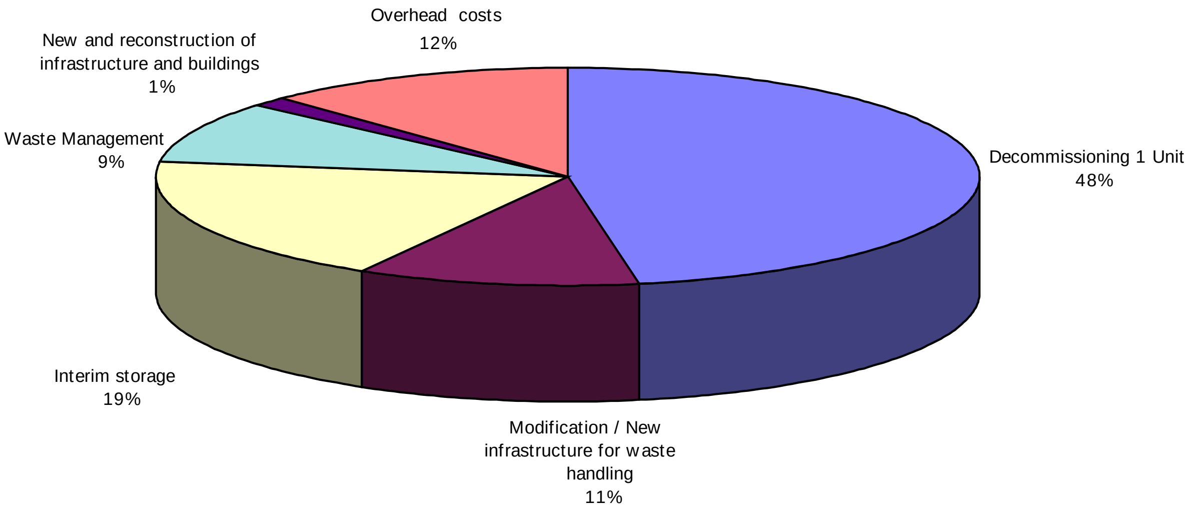


Figure 3-17 Decommissioning cost breakdown

The cost estimate for the Greifswald NPP is carried out by using the NIS programme STILLKO 2. The basis of all cost estimates with the STILLKO code is the analysis of the "primary mass" to be decommissioned and its activity inventory. The decommissioning strategy and all boundary conditions for waste handling, decommissioning techniques, staff costs must be assumed prior to the calculation. For the optimum decommissioning strategy it is necessary to estimate the cost of the decommissioning strategy up to the green field, also considering the re-commissioning of the equipment after a long term deferral period.

The different approaches for cost estimate give a wide spread of results. It depends not only on the different factors like the labour cost rates. Besides the cost for dismantling activities within a facility, the most important factors influencing the cost of a decommissioning project are the duration of the decommissioning project and the possibilities for disposal of spent fuel and radioactive waste.

Table 3-27 Cost estimate for the later dismantling strategy for a VVER-440 unit

Project	Working package	Cost [Million €]		Personnel	Consu mable	Invest	Fees	Extern	Factor	1 Unit	Sum [Million DM]	Sum [Million €]
A 1	Dismantling within the surveyed area	71.8		84%	6%	0%	0%	10%	4	6.6		
A 2	Dismantling within the controlled area before care and maintenance	289.5		82%	4%	3%	0%	11%	5	22.3		
A 3	Preparation for care and maintenance	315.3		82%	4%	3%	0%	11%	5	24.3		
A 4	Operation Care and maintenance (30 years)	593.6		82%	4%	3%	0%	11%	5	45.7		
A 5	Preparation for dismantling	302.5		82%	4%	3%	0%	11%	5	23.3		
A 6	Dismantling within the controlled area	79.9		82%	4%	3%	0%	11%	5	6.2		
A 7	Remote controlled dismantling	266.5		23%	2%	75%	0%	0%	5	44.0		
A 8	Demolition of buildings	502.3		62%	6%	9%	1%	22%	8	33.5		
A 9	Post Operation period	950		46%	19%	35%	0%	0%	5	124.7		
A	Decommissioning Greifswald (KGR)		3.371								331	168
B	Modification / New infrastructure for waste handling		157						2		79	40
C	Erecting and Commissioning of the Interim storage		525						4		131	67
D 1	Disposal of spent fuel (Dry Interim storage within CASTOR casks)	184.3		11%	6%	83%	0%	0%	4	42.4		
D 2	Operational waste KGR	136.2		14%	43%	5%	24%	14%	4	6.2		
D 3	Decommissioning waste within the preparation period of care and maintenance	335.7		14%	14%	24%	29%	19%	5	7.0		
D 4	Operational waste within the period of care and maintenance	72.2		14%	43%	5%	24%	14%	4	3.3		
D 5	Decommissioning waste after the care and maintenance period	305.6		14%	14%	24%	29%	19%	5	6.4		
D	Waste Management		1.034								65	33
E	New and reconstruction of infrastructure and buildings		17						2		9	4
F 1	Project management	65		100%	0%	0%	0%	0%	5	3.3		
F 2	Documentation	11.8		100%	0%	0%	0%	0%	5	0.6		
F 3	Quality assurance	18.9		100%	0%	0%	0%	0%	5	0.9		
F 4	Site operation	11.3		100%	0%	0%	0%	0%	5	0.6		
F 5	Overhead	628.9		94%	0%	0%	6%	0%	5	37.0		
F 6	Post operation	181.6		0%	28%	72%	0%	0%	5	36.3		
F 7	Planning and licensing	55		33%	0%	0%	67%	0%	5	8.3		
F	Superior costs		972								87	44
Total			6.077								701	356

3.4 Considerations for Fast Reactor Decommissioning

3.4.1 Overview

The former Soviet Union developed a fast reactor programme similar in nature to the US and western Europe. Russia's first fast reactor BR-1 was built in 1955. This was a zero energy reactor fuelled with plutonium metal. Following this, the BR-2 reactor - a 100 kW (thermal) plutonium fuelled, mercury cooled reactor was built in 1956. Both reactors were used to investigate fast reactor parameters such as irradiation effects on materials and reactor temperature coefficients.

After dismantling BR-2, some of the components were used to build the BR-5 reactor at Obninsk, near Moscow, which was completed in 1958, achieving criticality in the summer of 1958. The reactor was operated to collect data on fuel burnup and as an irradiation facility. BR-5 was designed for 5 MW (thermal) operation with liquid sodium cooling and was the first to use plutonium oxide fuel. The core contained 88 hexagonal fuel sub-assemblies each containing 19 stainless steel clad fuel pins. In November 1964 the reactor was shut down and reloaded with carbide fuel. In 1972 further modifications increased the rating to 10 MW (thermal) using a plutonium oxide core. The reactor became known as BR-10.

The next design was the BOR-60 (60 MW thermal) fast reactor at Dimitrovgrad which started construction in May 1965 and achieved criticality in 1969. A steam generator was installed in 1970 and a second one of different design was installed in 1973 to generate 12 MW of electricity. BOR-60 was used to provide data on reactor materials. The fuel was enriched uranium oxide pellets clad in stainless steel, grouped in hexagonal sub-assemblies and cooled by a liquid sodium primary circuit.

The world's first prototype fast reactor power plant was the BN-350 at Aktau in Kazakhstan (see Figure 2-1). The 350 MWe uranium oxide fuelled reactor was built on the Caspian shore. Construction began on BN-600, a 600 MW (electrical) plant with completion in 1976 at Beloyarsk in Russia even before completion of the construction of the BN-350. Unlike the previous reactors BN-600 was a pool type design. Criticality was achieved in 1980. The reactor uses a mixed uranium-plutonium oxide fuel in annular pellets. Further reactors were planned in the CIS. BN-800 and 1600 plants reached the design stage but have now been suspended because of the lack of funding and collapse of the former Soviet Union.

The sole operating unit at Beloyarsk, the BN-600, is a sodium-cooled breeder reactor that generates new fuel as it operates. BN-600 is a three-loop "pool" design and is the second-largest breeder reactor in the world, behind the French Super Phoenix. The plant features a modular steam generator design that allows the steam generators to be repaired while the plant is on-line. Beloyarsk has no overhead containment structure; a standard industrial building and a protective shroud cover the reactor.

Table 3-28 Nuclear power plants with fast reactors in the CIS

n°	Name of the NPP and n° of the unit	Reactor type	Installed power [MW]	Reactor generation	Operating organisation	Start of operation	Planned finish of operation
1	Beloyarsk -3, Russian Federation	BN-600	600	1	Rosenergoatom	1980	2010
2	Aktau, Republic of Kazakhstan	BN-350	350	1	Mangishlak Atomic Energy Complex	1972	2003

3.4.2 Description of BN-350 at Aktau

The world's first prototype fast reactor power plant was the BN-350 at Aktau in Kazakhstan (see Figure 2-1 and Figure 3-18). The 350 MWe uranium oxide fuelled reactor was built on the Caspian shore. Criticality was achieved in 1972 with full power operation in 1973. The reactor produces both electricity (150 MW) and desalinated sea water (120000 m³ of fresh water per day). After five of its six steam generators were repaired in 1976 the reactor has run at a rating of only 650 MW (thermal) [3-1].

3.4.2.1 Historical Overview

The BN-350 at Aktau was seen as a prototype for a generation of fast reactors in a similar manner to the Prototype Fast Reactor, operated by the United Kingdom Atomic Energy Authority at Dounreay in the UK and the Superphoenix in France. Aktau was the worlds first commercial sized fast breeder reactor and the forerunner to the BN-600 at Beloyarsk.

The plant is operated by the Mangishlak Atomic Energy Complex (MAEC) which is wholly owned by the Kazakhstan government and is part of the Kazakh State Atomic Power Engineering and Industry Corporation (KATEP) which comprises the 15 biggest enterprises in nuclear power and industry. KATEP is a national joint stock company in Kazakhstan since 1993.

Figure 3-18   View of BN-350 at Aktau

3.4.2.2 Location

The BN-350 reactor is situated on the Mangishlak peninsula of the Eastern Coast of the Caspian Sea approximately 10 km south of Aktau town and on the MAEC operated site - see Figure 2-6.

3.4.2.3 Physical Interfaces

Aktau provides only 10% of Kazakhstan's electrical power supplies with the reminder from coal and oil and gas fired plants as well as hydroelectric plants. About 16% of the electricity is imported from neighbouring countries such as Russia. Aktau operates as a single plant with associated supporting facilities such as fuel and waste stores.

3.4.2.4 Description of the Unit 1 at Final Closure

The following detail the main parameters of the BN-350 plant:

- Country: Kazakhstan
- Location: Aktau,
- Name: Mangishlak Atomic Energy Complex
- Acronym: MAEC
- Project type: BN (FBR)
- Operating company: MAEC
- Model: BN-350
- Operation start: 1972
- Power and unit: Nominal 1000MW (therm.); 150 MW electrical+120000 m^3/day
- Total operation to date (1998): 26 years
- Effective operation to date: ~19.4 years (4087048 MW×cal.day)

Reactor BN-350 is loop type fast breeder reactor and has three circuits. The coolant of the primary and secondary circuits is sodium, the third circuit coolant is water-steam (a flow diagram with nominal parameters is shown Figure 3-19). In normal operation 5 loops are in use and one loop is in standby. Each primary loop includes an intermediate heat exchanger and main circulation pump, the secondary loop includes a main circulation pump and steam generator (with evaporator and superheater). Superheated steam is transferred to turbines of conventional plant or to Desalination Water Plant which is on the site of MAEC.

The core is approximately 1 m in height and 1.6 m in diameter [3-20]. The reactor uses enriched uranium oxide fuel in 226 stainless steel fuel assemblies. Refuelling is normally carried out every 60 days with a maximum fuel burn up of 50 MWd/kg. The reactor vessel is 12 m high and 6 m in diameter to contain both the core and the breeding blanket of uranium oxide. Argon is used as the cover gas.

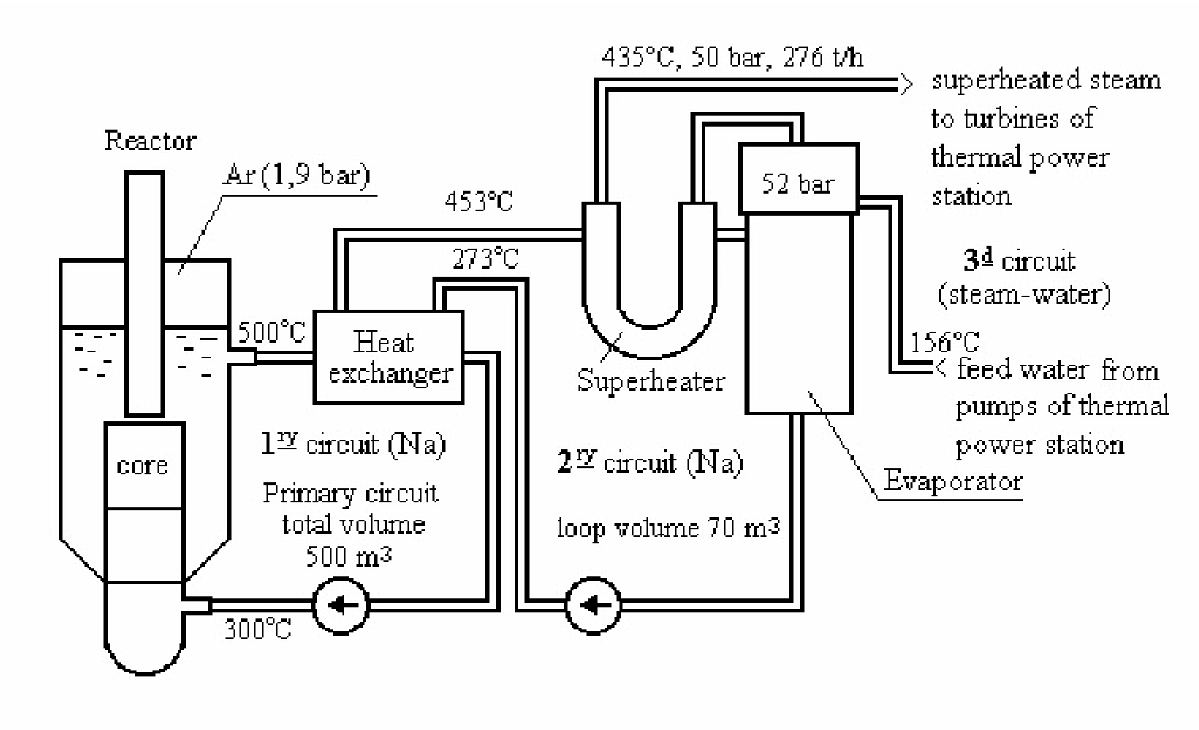


Figure 3-19 Flow diagram of primary, secondary and water-steam circuits of BN-350 (with the nominal parameters).

3.4.2.5 Operating History of the BN-350 relevant to Decommissioning

The physical reactor start-up was in 1972 with the commercial start-up in July 1973. The maximum power level of 750 MW was operated up to 1989. After that the reactor capacity was decreased because of steam generator problems. Since 1995 reactor capacity has been decreased to nominally 420 MW because of the absence of an earthquake engineered core standby cooling system.

The planned date for decommissioning is approximately 2003 based on the end of design lifetime.

According to Perera [3-1] there have been several accidents at the site. The most important occurred in 1974 when the steam generator pipes depressurised and allowed water to enter the liquid sodium metal secondary circuit causing a water-sodium explosion and cracking the heat exchanger.

3.4.3 Radioactive Inventory/Hazardous Material

3.4.3.1 Waste Classification

Solid waste, which activity exceeds 7.4×10^4 Bq/kg (for β-activity waste), 1×10^{-7} g eq.Ra/kg (for γ-activity waste) and 7.4×10^3 Bq/kg (for α-activity waste), is defined as Solid Radioactive Waste (SRW).

Similarly liquid waste with activity exceeding 5.55×10^5 Bq/m^3 (1.5×10^{-8} Ci/l), is defined as Liquid Radioactive Waste (LRW).

The detailed radiological classifications of solid radwaste (SRW) for the BN-350 reactor are given in Table 3-28.

Table 3-28 Radiological classifications of solid radwaste

Type of radwaste according to BN-350 terms.					Type of radwaste according to IAEA terms.
Number of group	Surface dose rate for γ-activity (in the near 10 cm).		Surface β-activity	Surface α-activity	
	mSv/h	MkR/s	1/cm^2 min.	1/cm^2 min.	
1	$(0.03\text{-}30)\times10^{-2}$	Below 8	50-2000	5-200	LLW
2	$(30\text{-}1000)\times10^{-2}$	Below 280	2000-8000	200-800	ILW
3	≥10	≥280	≥8000	≥800	HLW

The detailed radiological classifications of liquid radwaste (LRW) for the BN-350 reactor are given in Table 3-29.

Table 3-29 Radiological classifications of liquid radwaste

Type of liquid radwaste according to BN-350 terms.	Total activity (Bq/l)	Type of radwaste according to IAEA terms.
Low level waste (LLW)	below 3.7×10^5	LLW
Intermediate level waste (ILW)	3.7×10^5-3.7×10^{10}	ILW
High level waste (HLW)	$\geq3.7\times10^{10}$	HLW

3.4.3.2 High Level Solid Operational Wastes

Solid radwaste (SRW) storage (for high-level wastes) is situated on the site of MAEC and consists of concrete walled area (500 m^3) separated into 12 insulated tanks with leakproof hatch covers.

At the present time SRW storage is 85 % full, and the remaining storage is not sufficient for BN-350 decommissioning wastes. Consequently, it is necessary to design additional storage for high-level waste.

There is no SRW treatment plant currently on site and therefore a new facility is required.

Currently the management strategy is to store this waste on site but the long term management has not yet been developed.

The SRW HLW currently amounts to 126 tonnes with an activity of 20 TBq (538 Ci). The different activity streams (alpha, beta/gamma) are not separated

3.4.3.3 Intermediate, Low and Background Level Solid Operational Wastes

Intermediate, low and background level solid radwastes are buried in a trench on the site of MAEC.

At the present time storage for intermediate, low and background level solid radwaste is 80% full and approximately 2000 m^3 remains available in one trench. This is not expected to be enough for BN-350 decommissioning, and therefore it is necessary to design additional storage for intermediate, low and background level wastes.

There are 8 trenches with a total volume 8000 m^3. The total amount of intermediate level waste is 551 tonnes and the total activity of the intermediate level waste is 40 TBq (1085Ci). Low level and background (BW) level waste is also stored in the trench. The total mass is 5147 tonnes with an activity of 2.2 TBq. The wastes are not separated.

3.4.3.4 Liquid Radwaste from Operations

Liquid radwaste (LRW) storage is situated on the site of MAEC. It consists of 10 heavy-walled concrete and steel covering tanks of three types:

- 2 tanks (volume 160 m^3, not used now)
- 6 tanks (volume 1000 m^3, one of these is used for oil and water mixed waste)
- 2 tanks (volume 840 m^3, one of these is not used).

At the present time LRW storage has only one empty tank (volume 1000 m^3).

It is necessary to design further expansion of existing LWR storage because the current storage will not be able to store waste from BN-350 decommissioning. In addition a new design of water treatment plant for final evaporation (to concentration ~500g/l), the cementation and finally packing in steel drums (volume 200 l) is required. The resulting waste will need to be stored on site. This will replace the existing liquid waste treatment plant shown in Figure 3-20.

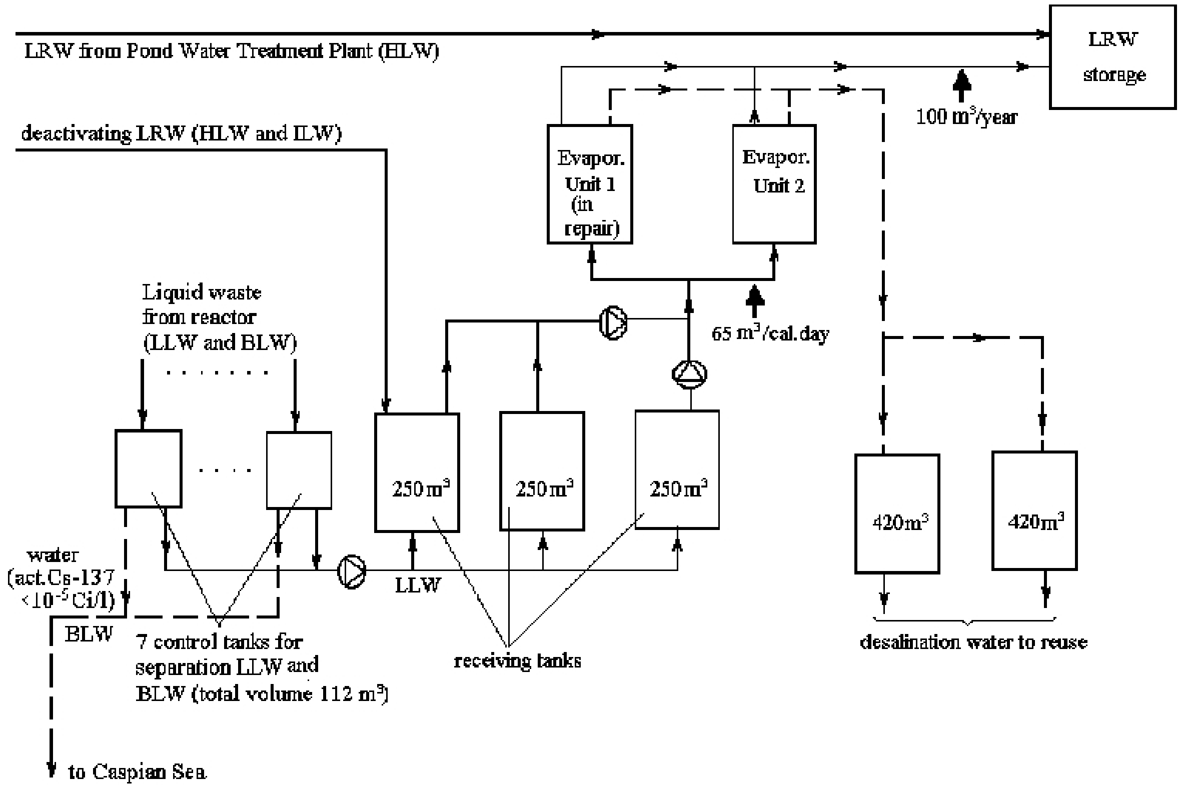


Figure 3-20 Flow chart of existing Special Water Treatment Plant (SWTP).

The total activity of the LRW is 111 TBq (3018 Ci) in a volume of 5050 m^3.

Only the total activity of SRW and LRW is available because:

- β and γ activity estimations are produced only for the radwaste separation stage on the reactor site;
- α activity is not produced at the present time.
- Solid radwaste of types 1, 2 and 3 are buried in trenches together but are accounted for separately.
- At the present time only intermediate level liquid radwaste is stored in LRW storage (3.7×10^7-1.2×10^7 Bq/l). High level liquid radwastes are mixed into intermediate level.

3.4.3.5 Planned Treatment of the Sodium Coolant

Fast reactor systems are unique among nuclear power plants by virtue of the presence of liquid sodium metal as the main coolant for the reactor. Sodium is a very chemically active element which leads to unusual waste treatment considerations.

Therefore when BN-350 is finally shutdown for decommissioning, the sodium coolant disposal is a significant problem. This problem consists of:

- Removal of the sodium coolant from the circuits.
- Washing of the equipment and circuits from the sodium residues and deactivation.
- Sodium treatment.

The total volume of the sodium coolant of the secondary circuit is 450 m^3. The sodium coolant activity of the secondary circuits is approximately 5.9×10^5 Bq/kg (1.6×10^{-5} Ci/kg) (caused by the presence of tritium). The allowable level of activity of the sodium coolant is 7.4×10^4 Bq/kg (2.0×10^{-6} Ci/kg (OSP-72/87)). Consequently the sodium coolant of the secondary circuit must be treated or decontaminated before repeated use.

At the beginning of decommissioning the BN-350 the sodium coolant must be removed from the secondary circuit and the equipment decontaminated.

The storage system of the secondary circuit of the BN-350 is designed only for one-third of the volume of sodium coolant of the secondary circuit. Therefore the sodium coolant of the secondary circuit must be drained in three cycles. The sodium storage system of the primary circuit is designed for the total volume of sodium coolant contained in the primary circuit.

The total volume of the primary circuit is approximately 500 m^3. The activity of the sodium does not allow its reuse without additional decontamination - see Table 3-30.

At the start of decommissioning BN-350 the radioactive equipment will be dismantled and the sodium coolant removed. The equipment should be washed to remove sodium and separate components for dismantling. Part of the equipment and pipelines can be dismantled in this early phase.

Table 3-30 The shut down reactor sodium coolant activity

Type of coolant	Volume m^3	Radioisotopes present	Specific radioactivity, Bq/l
1 Sodium coolant of the primary circuit	500	Na-24	3.7×10^{11}
		Na-22	2.6×10^7
		Cs-137	1.1×10^8
		Cs-134	2.2×10^{10}

		Mn-54	3.7×10^6
		Zn-65	7.4×10^6
		Tritium	1.9×10^6
		Pu-239	1.1×10^2
2 Sodium coolant of the secondary circuit	450	Na-24	3.7×10^2
		Tritium	5.9×10^5

The methods of treatment of sodium radwaste can be listed as follows:

- Repeated use as the coolant on NPPs with sodium cooling system.
- Regeneration of sodium radioactive waste (SORW) for reuse.
- Treatment of SORW for burial.

The properties of SORW differ from traditional radwaste and the development of new processes and methods are required. Such wastes have high chemical activity, they are flammable and explosive on contact to water and some organic environments. Therefore the measures for treatment of SORW are one of most difficult problems for deciding at the decommissioning stage.

The basic tendency of modern technology development and treatment with waste is the processing of sodium for reuse (regeneration of sodium) provided that there is a market for the sodium. This minimises the disposal of sodium containing waste while providing an environmental benefit.

Regeneration of the sodium waste includes two basic processes:

- Removing Cs-137 activity from the sodium.
- Filtration of liquid sodium using hot and cold traps.

To improve the regeneration of SORW allowing ease of transportation and reloading it is expedient to carry out the regeneration 5 - 10 years after shut down of the reactor so that the Na-22 activity is reduced by two or three times.

Clearing Cs-137 from sodium will be carried out with the help of a Cs trap containing nitrogen materials. This absorbs caesium from liquid sodium.

The regeneration of SORW is accompanied by the formation of a new radwaste - spent caesium traps, which can contain significant activity due to the amount of Cs-137 that has been collected. For treatment these traps require special technology.

The final regeneration stage for SORW is filtration of liquid sodium at a temperature above 150 °C for removal of the oxide particles which can absorb nuclides, and particles of carbon containing nuclides of Cs-137.

To reuse SORW from the secondary circuit on other reactors does not require its preliminary clearing of tritium. In the absence of such a route the utilisation of these wastes in other areas of industry requires that the tritium is reduced to a level of specific activity less than 7.4×10^4 Bq/kg. The resulting sodium can be metallic or carbonated - the last is widely used in the chemical and glass industry.

For the option where the sodium is to be buried the choice of technology for processing and solidification of SORW is guided by the following principles:

- Minimising waste during processing.
- To be carried under controlled safety conditions.
- Relative simplicity.
- Low costs.
- Minimum volume of the final products from the processing and solidification methods used.

The last requirement becomes especially important for processing a large volume of SORW following its removal from the reactor facilities and when there is no significant reuse route. In this case and being guided by the interest of protection of the environment, the whole of the spent sodium is subject to processing and burial. For this purpose SORW is to treated into a chemically inactive condition, suitable for long-term (not less than 300 years) storage.

For processing SORW it is expedient to use methods of 'dry chemistry', as they minimise the risk from explosion and fire as well as derive smaller quantities secondary waste. The application of processes of 'dry chemistry' is based on solid-phase or gas-phase oxidation of sodium. A final product in solid-phase oxidation are cement compounds, the results of gas-phase oxidation are cement compounds and glass-ceramics compounds.

The costs of gas-phase and solid-phase oxidation technology are comparable. However, gas-phase oxidation technology is easier to undertake as it is a continuously working process and convenient to use for filling buildings and structures for storage. From considering the variants of processing of spent sodium the method of solid-phase oxidation is the most preferable being considered by the Kazakhstan authorities.

It should be noted that existing and other methods of processing sodium with the purpose burial - in particular the 'NOAH' technology, developed by FRAMATOME, consists of processing radioactive sodium in hydroxide by dissolving in alkali water solutions. However dissolving hundred of tons of radioactive sodium in water solutions is possibly dangerous and a large volume of hydrogen is accumulated and is accompanied by the formation of large quantities of LRW with activity 1.9 x10^7-1.9 x10^8 Bq/l and high contents of salts in a solution. This radwaste must also be treated in addition to the sodium.

Notwithstanding these issues it should be noted that following progress has been made:

- Successful application of this technology for processing of 37 tons of sodium from 'RAPSODIE' during two months (installation 'DESORA').
- Development of the technology and installation for processing 1560 tons of sodium from PFR (PFR Sodium Disposal Plant).
- Presence at FRAMATOME of experience of the development and industrial operation of installations of a similar type as well as experience in licensing the technology,

For these reasons the requirements for disposal of sodium at BN-350 should continue to pay attention to such technology as 'NOAH' and experience by FRAMATOME to develop the choice of possible technologies and firms for its development on the BN-350 NPP.

After removal of sodium from circuits and walls of the equipment and pipelines there is the quantity of sodium (100 g/m^2), which it is necessary to remove. Such experience in washing sodium from such large systems, as the reactor circuits of BN-350, is not present in Russia or Kazakhstan. The nearest experience to this presently is the washing of the sodium primary circuit of the reactor BR-10 (in Russia) before fulfilment of its reconstruction.

However, because of the differences in scale between research reactors and the commercial sized fast reactors further development work is required.

Using current knowledge the following sequence of sodium washing operations is identified:

1. Sodium draining from circuits to drain tanks. The separation of the tanks from the circuits.
2. The removal of non-draining cold traps from circuits.
3. Vacuum sodium topping from circuit.
4. Steam and gaseous circuits washing by submission of the mix to the highest points of the circuit and draining from the bottom points to a special system of steam and gaseous mix reception containers with the sodium residues. An inert gas nitrogen is used.
5. The hot distillate washing of the circuits.

It is recommended to carry out primary circuit deactivation by acid solutions to minimise personnel dose during disassembly operations.

3.4.3.6 Contaminated and Activated Equipment

The primary circuit equipment, pipelines, pumps with contamination up to 2.6 GBq/m^2 are high level radwastes (groups 2 and 3 of SRW). The presence of Cs-137 deters the removal of this contamination in the short term.

The reactor vessel has an induced activity of 26 GBq/kg. During 50 years of surveillance this activity is reduced to 0.59 GBq/kg.

Most of the concrete is not radioactive (probably only its surface is contaminated). The concrete in the region of the sodium pipelines has an activity of 37 MBq/kg.

Reactor vessel internal equipment has high specific induced activities (up to 14 TBq/kg). It is therefore necessary to use robots or special protective cabins for work within it. Reactor vessel internal equipment should be directed in SRW store for long period of storage.

3.4.3.7 Fuel Management

The spent fuel BN-350 is translated to nuclear and radiation safety conditions under a separate programme. The programme for the treatment of spent fuel is developed in common by the Ministry of Science - Academy of Sciences of Republic of Kazakhstan and Ministry of Power USA.

The programme includes:

- Stabilisation and packing spent fuel assemblies.
- Safe transportation to a co-ordinated place of store.
- Storing these assemblies during 50 years in reliable storehouse which have the licence at a level of the requirements required by IAEA safeguards.
- Prospective place long-term storing the spent fuel is ‘Baikal-1’.

Financing of the project is carried out by the USA.

3.4.3.8 Conventional Hazardous Wastes

Unfortunately no information is readily available for conventionally hazardous wastes at Aktau and therefore this information must be gathered at the appropriate time to support the decommissioning requirements.

Only desalination water from SWTP is reused on reactor for fulfilling own reactor needs.

The limit for reuse of desalination water from SWTP is below 10^{-8} Ci/l of Cs-137 activity.

3.4.4 Cost and Funding Requirements

The described cost estimate of a working packages includes only the labour and new equipment costs, which are the biggest percentage of the total costs of these packages. The costs given are developed using Western European labour costs scaled down to reflect the cheaper rates experienced in the CIS countries.

The planning of the decommissioning costs is made beginning with the year of the decommissioning decision (year “-3” - see section 6) and up to the year “+10”. The costs for the storage under surveillance and for maintenance of the turbine hall building, etc., at the deferral period and the costs for the final dismantling are not estimated, because these stages can not be described in detail.

Table 3-31 gives a survey about the decommissioning costs per year.

Table 3-31 Total decommissioning cost estimation

Facility	Costs	Year													
	[million ECU]	-3	-2	-1	0	1	2	3	4	5	6	7	8	9	10
New facilities	84	3	7.6	7.6	11	13	6.7	4.2	5	5.9	5.9	4.2	4.2	2.5	2.5
Documentation	18	1	1.5	1.8	2.7	2.3	2	2.2	1.3	1.3	1.3	0.7			
Stage 0 Operating & Dismantling	9	2	2.3	2.3	2.3										
Stage 1 Operating & Dismantling	59					9.2	9.2	9.2	8.7	9.2	6.7	6.7			
Stage 2 Operating & Dismantling	16												5.4	5.4	5
Total [million ECU]	185	6	11	12	16	25	18	16	15	16	14	12	9.6	7.9	7.5
Total from the start [million ECU]		6	17	29	45	69	87	103	118	134	148	160	169	177	185

Figure 3-21 shows the development of decommissioning costs while Figure 3-22 shows the proportioned costs between the individual parts.

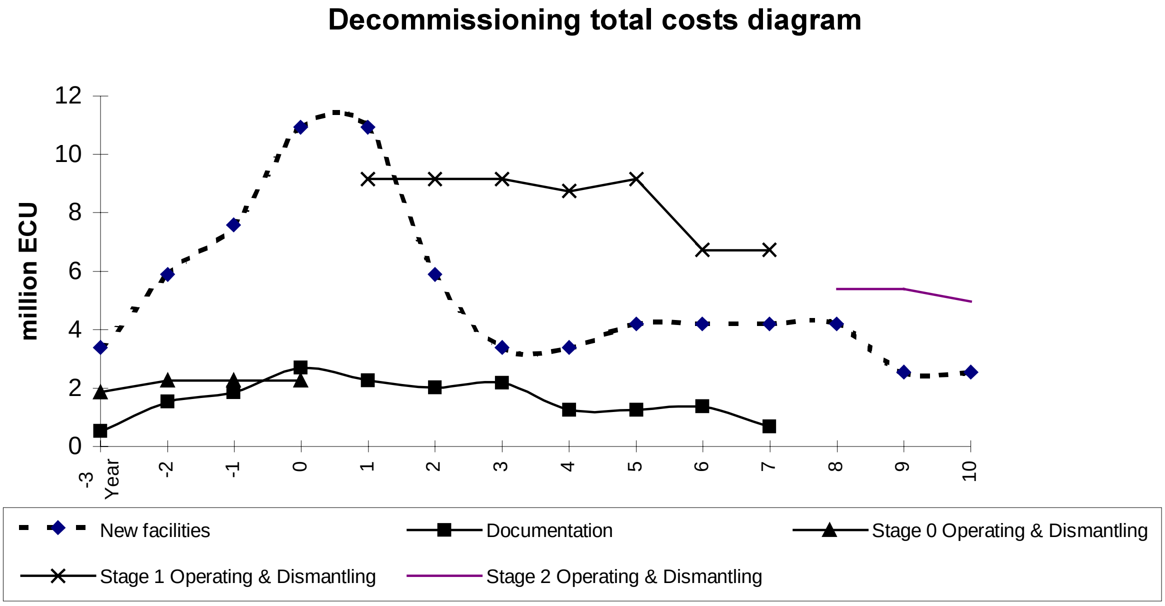


Figure 3-21 Total Decommissioning costs

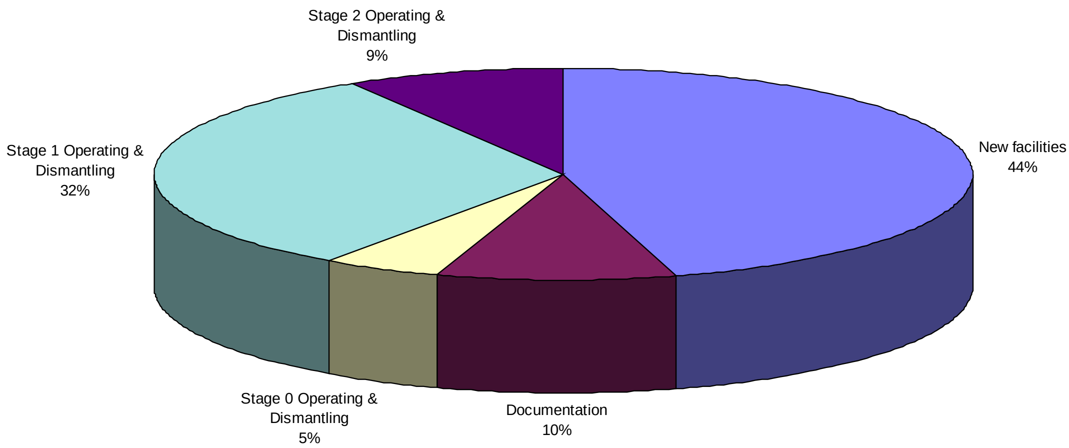


Figure 3-22 Decommissioning costs breakdown

Stage 3 costs were not generated but assessed from known activities associated with this stage. At this point in the decommissioning programme the BN-350 reactor is much like any other reactor and therefore dismantling costs are assumed to be similar to other reactor types of similar size. A figure of 30 M€ is used.

3.4.4.1 Costs for new and modified facilities

It is planned to construct the following new facilities:

- Liquid radwaste treatment facility with the aim of solidification of all existing and generated by decommissioning (decontamination) liquid radioactive waste. This could be a cementation facility.
- Solid radwaste treatment facility with the aim of sorting, conditioning and packing of all existing solid radioactive waste, bulk stored in existing interim storage facilities, and generated by decommissioning and dismantling of the unit(s).
- Sodium treatment plant for the treatment and conditioning of activated and contaminated sodium from the primary and secondary circuits.

3.4.4.2 Costs for new facilities and modifications of existing facilities

The following new facilities are anticipated to be required:

- Radwaste interim storage facility with the aim of an interim storage for a period of approximately 50 years of solid and solidified radioactive waste, generated at the site. The reason for such an interim storage facility is the missing national or regional repository.
- Spent fuel storage facility with the aim of an interim storage of all spent fuel including the operation of the units up to the end of its life time. The reprocessing and/or the final disposal of FBR-spent fuel is not decided up to now.
- Special decommissioning (decontamination) and dismantling tools and instruments. Remote handling instruments are not planned for these decommissioning stages, because the reactor dismantling and the dismantling of high and intermediate radioactive level equipment is not planned.
- Radwaste treatment facilities for solid and liquid wastes as well as a sodium treatment and conditioning plant to process the metal coolant from the reactor

Because the costs for the new facilities are the highest amount of all specific costs, Table 3-31 shows the assumptions.

Table 3-32 Costs for new facilities

N°	Facility	Financing period	Costs [million ECU]	-3	-2	-1	0	1	2	3	4	5	6	7	8	9	10
1	Liquid radwaste treatment	-3 ... +1	13	1	3	3	4	4									
2	Solid radwaste treatment	-3 ... +1	13	1	3	3	4	4									
3	Radwaste container	0 ... +10	13					1	1	1	1	2	2	2	2	2	2
4	Radwaste interim storage facility	-3 ... +2	7	1	1	1	2	2	1								
5	Spent fuel interim storage facility	-4 ... +3	13	1	2	2	3	3	3	1							
6	Tools and instruments	+5 ...+8	10								2	3	3	2	2		
7	Sodium treatment plant	+1 ...+10	16					2	3	3	3	2	2	1	1	1	1
	Total		84	4	9	9	13	16	8	5	6	7	7	5	5	3	3

The long-term repository for radioactive waste is not included.

The estimated costs include:

- project management
- procurement
- design
- construction
- commissioning

It is anticipated that the facilities will be built under turn-key contracts. The estimated costs do not include the operation of the facilities.

3.5 Considerations for Research Reactor Decommissioning

3.5.1 General

An initial review of research reactors in the Russian Federation by the Local Partners identified a number of facilities owned and operated by several institutes associated with atomic energy. These civil facilities are based either in Moscow (and environs), Dimitrovgrad, Obninsk and St Petersburg. Out of the 45 facilities cited they breakdown as:

- 23 pool/tank
- 5 FR
- 5 graphite
- 10 impulse/FR
- 2 unique plants such as ’Romashka’ -thermoelectric, ’Argus - homogeneous’

The list is given in Table 2-3 for research reactors in the CIS countries. Approximately 50% are pool/tank type reactors and of those identified to be shutdown up to 2005, 60 % are pool/tank types. Consequently it has been agreed to develop the generic decommissioning plan for pool type research reactors as the most prevalent in the CIS countries (and indeed other former Soviet Union countries).

The following circumstances should be noted and may need to be taken into account for the decommissioning of the majority of Russian research reactors:

- Research reactors are very old installations; they were operated when the restrictions of military secrecy and high working rates limited the ability to pay due attention to conducting the exact, detailed documentation about many design aspects and which will be significant for decommissioning;
- The power, campaign duration and dimensions of the equipment of research reactors, as a rule, are significant lower than similar parameters of power reactors at nuclear power stations;
- The neutron density in the more powerful research reactors exceeds the density in power reactors, determines a higher level of specific activity in components located near the core;
- The design features, materials used, duration of operation up to the shut down of RR differ strongly from one reactor to another, and therefore complicates the development of common requirements;
- Research reactors have a number of different experimental devices, some of which could be difficult and complex to dismantle (for example, horizontal experimental channels, loop channels and others.

According to the basic Russian safety standard of research reactors (OPB RR — 94) [3-21], the operating organisation is responsible for the safety at all stages of the life cycle of the research reactor, including the decommissioning stage, which is defined as: *a complex of measures for shut down of the RR, excluding its further use and ensuring safety of the personnel, public and environment.*

The details are given below regarding the decommissioning of the two research reactors of RRC Kurchatov Institute, that are chosen as representative RR.

3.5.2 Description of VVR-2 and MR Research Reactors

Both reactors operated within the Russian Research Centre Kurchatov Institute located in the Northwest part of Moscow. The border of the RRCKI territory limits the sanitary-protective zone around the reactors. Originally located on the outskirts of Moscow in the 1950s and 1960s the site now lies within the city limits as the city has expanded around the site. The nearest public buildings are at a distance of 600 m from the reactor buildings. A public street is located at a distance of 200 m from the MR reactor buildings within the RRCKI.

The characteristics of the VVR-2 are:

- **Owner**: Russian Research Centre Kurchatov Institute, Institute of nuclear reactors
- **Designer of the reactor construction**: OKBM (Nizhni Novgorod)
- **Designer of the construction works**: VNIPIET (St. Petersburg)
- **Name of the reactor**: VVR-2
- **Type of the reactor**: Water cooled, water moderated, pool type
- **Thermal power**: 3.0 MW
- **Current status**: dismantled

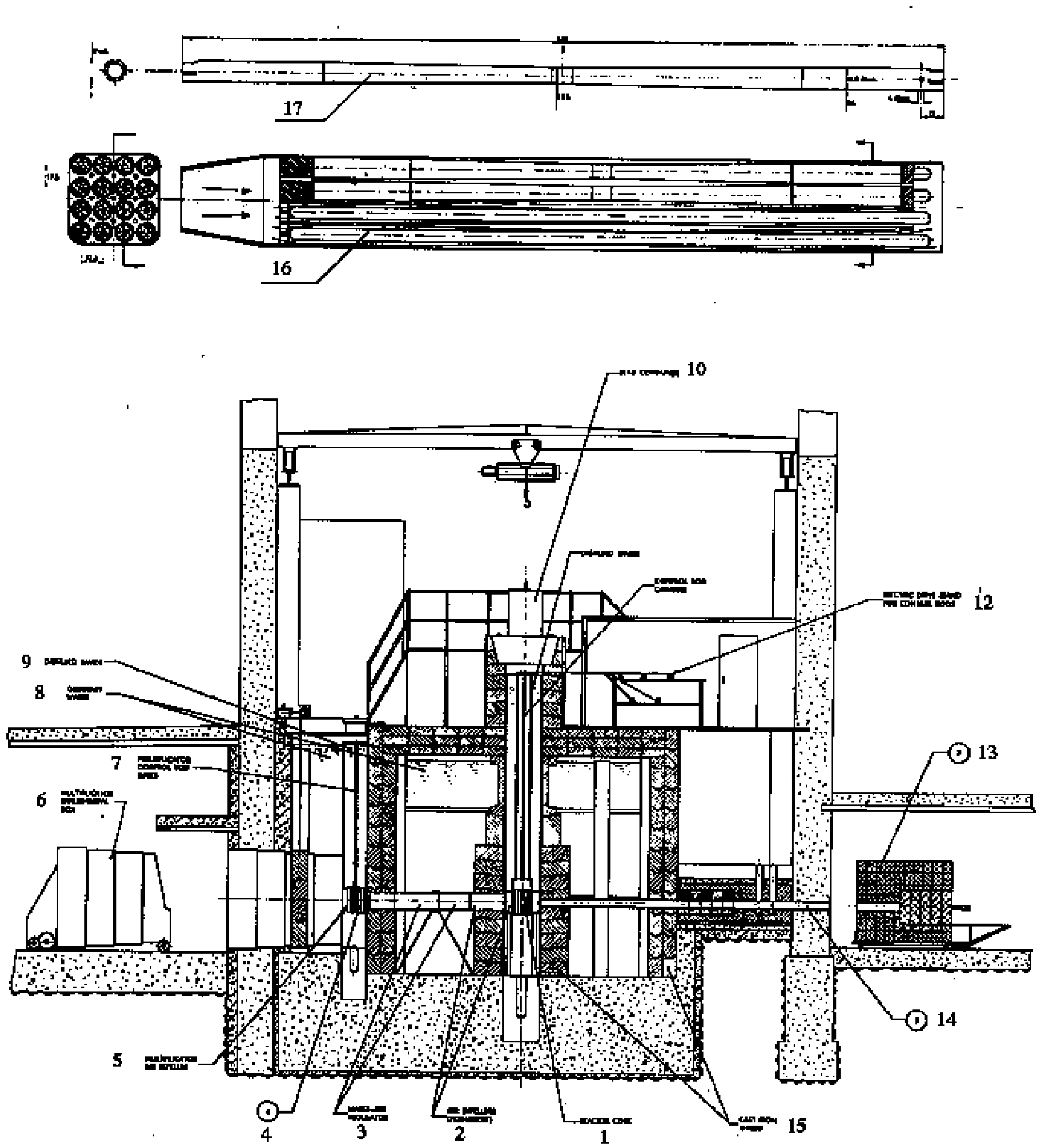
Research reactor VVR-2, was the first of the water cooled and moderated reactors without channels inside the core in Russia and was the prototype of a series of reactors types such nominated VVR-S of pool type with nominal power of 2 MW, and also of reactors type VVR-M, VVR-Z, VVR-1S with nominal power on 10 MW.

VVR-2 is a heterogeneous reactor using thermal neutrons (a vertical plan of the reactor and its fuel assemblies are shown schematically in Figure 3-23). Distilled water is used as coolant and moderator of the reactor, and also for top reactor shielding. It uses uranium fuel (4.5 kg). Maximum density of thermal neutrons at the centre of the core reached $4*10^{13}$ neutrons/cm^2.sec.

The reactor core (Ø 400 mm and height 500 mm) was formed from assemblies of fuel rods of 10 mm diameter of two types (UO_2 with 10 % enrichment of U-235 with an aluminium clad and alloy of uranium metal with 36 % of enrichment with aluminium in an aluminium clad).

The main structures of the reactor (tank, fuel assemblies, elements of the core, lattices, pipelines etc.) are made from aluminium alloys.

Research in the field of thermalisation of neutrons was carried out with various moderators and reflectors. The effects of radiation on insulators, organic and semi-conductor materials was also studied in the reactor. The reactor was used also for research of the efficiency of various radiation protection techniques and for the production of isotopes.



Key:

1 Reactor core

9 Distilled water

2	System for constant de-airing	10	Lead container
3	water-air regulator	11	Channels of regulating rods
4	Multiplicator	12	Floor of drives of regulating cores
5	De-airing of the multiplicator	13	Thermal column
6	Experimental box of the multiplicator	14	Horizontal experimental channel
7	Directing pipes of regulating rods of the multiplicator	15	Elements of pig-iron protection
8	Normal water	16	Fuel assembly
		17	Coolant flow pipe

Figure 3-23 Vertical section of VVR-2 and fuel elements

The initial reactor power, commissioned in 1954, was 300 kW. After the reconstruction in 1957 its power was increased to 3 MW. After the reconstruction, research in loop channels placed at the reflector was started. Figure 3-24 show the reactor hall.

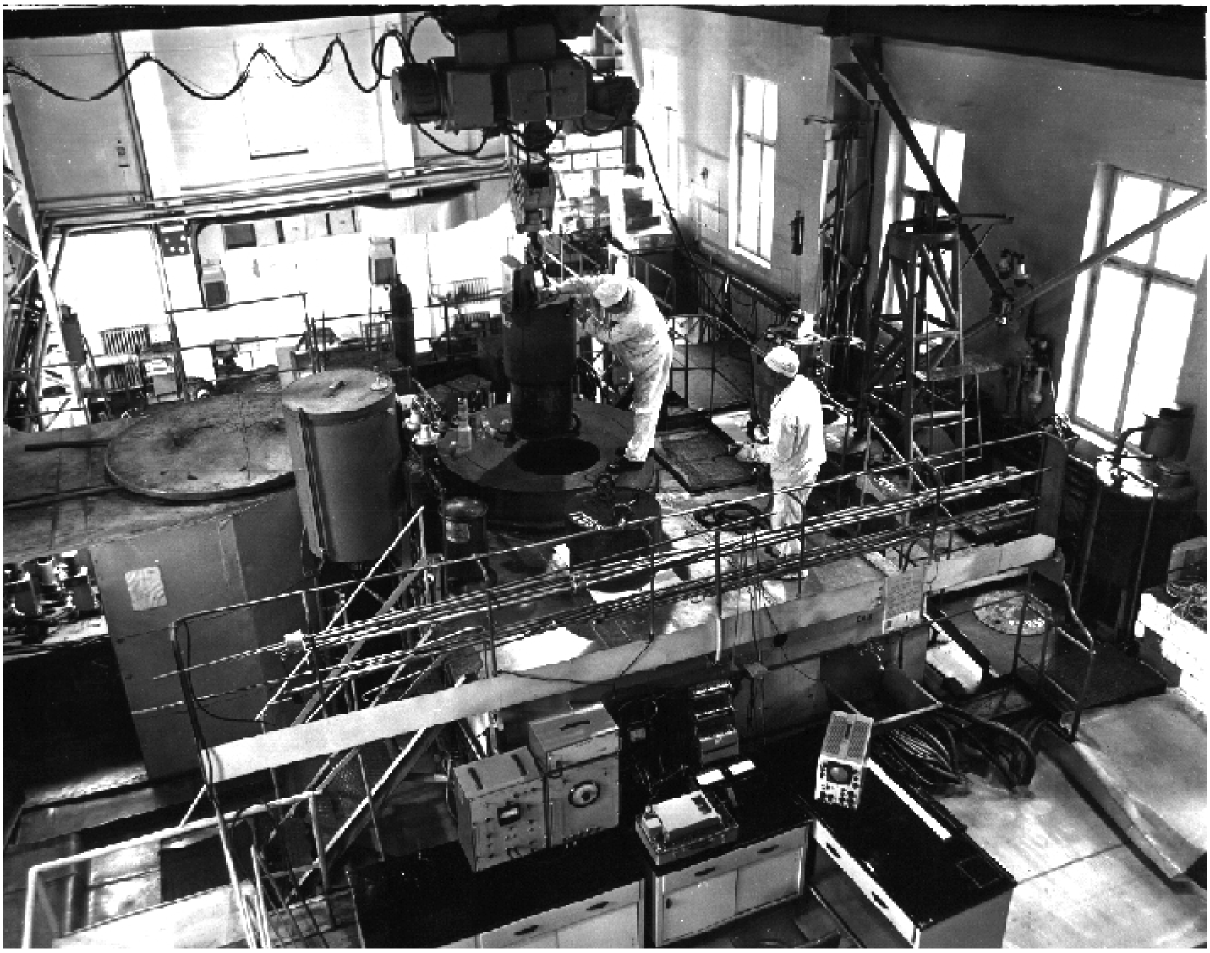


Figure 3-24 View of reactor hall

The MR reactor has the following characteristics:

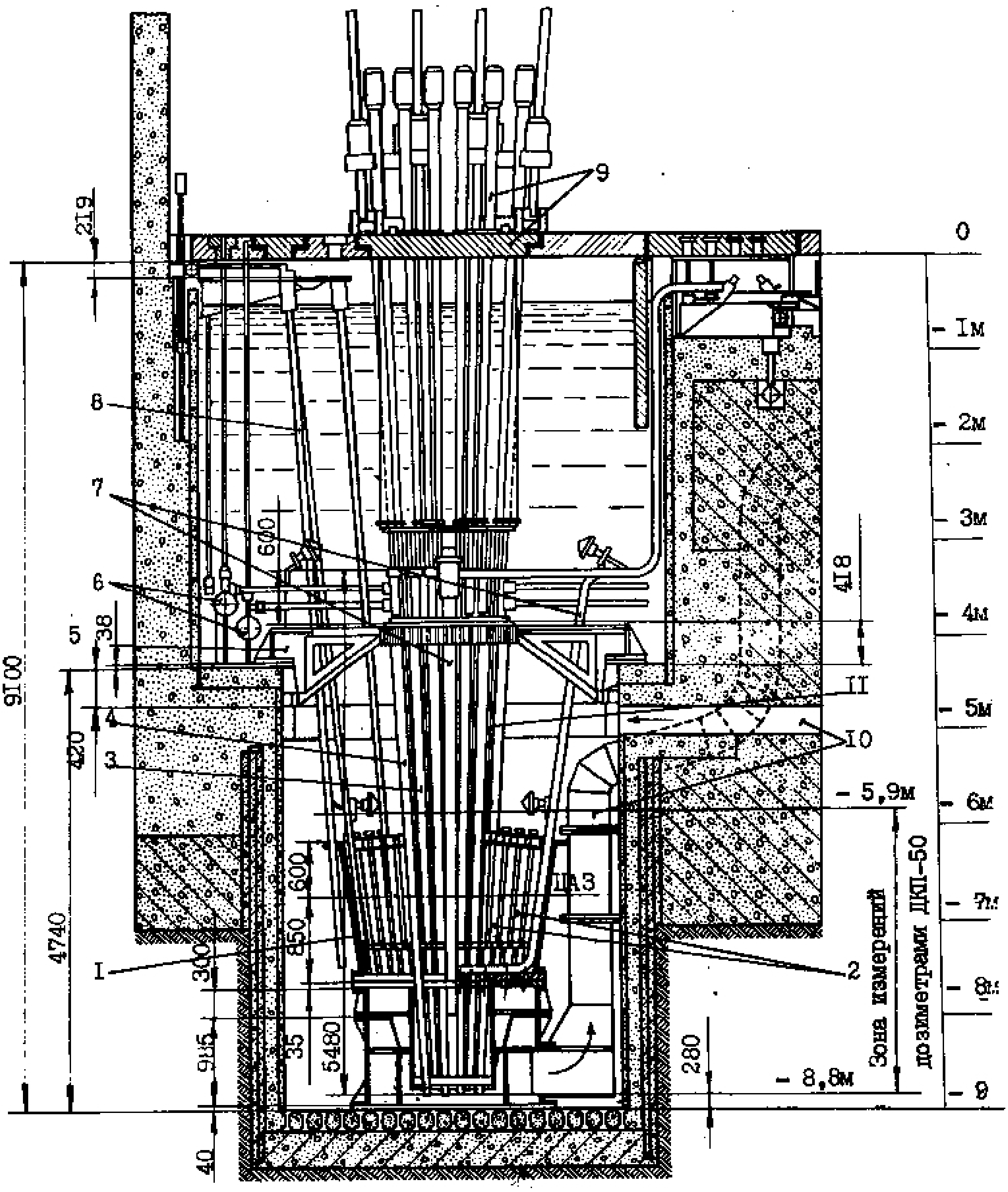
- **Owner**: Russian Research Centre Kurchatov Institute, Institute of reactor technologies and material research
- **Designer of the reactor**: Russian Research Centre Kurchatov Institute

- **Designer of the reactor construction**: Institute "Akademprojekt" of the Academy of Sciences of USSR
- **Name of the reactor**: MR
- **Type of the reactor**: channel, pool type with water-beryllium moderator
- **Maximal thermal power**:
  - of the reactor without loop channels, 40 MW
  - of the reactor with loop channels, 50 MW
  - of one operational channel with fuel assembly, 2.5 MW
- **Greatest possible amount of operational and loop channels in the core and the reflector**: 61
- **Amount of operational fuel assemblies in the core**: 37
- **Present status**: shut down

The multi-loop specialised reactor MR, started in December 1963, was the first representative of a completely new type of research reactors at that time. The MR reactor was the first representative of a new type of loop reactor. This is a reactor of a channel type placed in a pool, and has become the prototype of research reactors MIR (Dimitrovgrad, Russia), "Maria" (Warsaw, Poland). It is thus a channel pool type reactor (the vertical plan of the reactor MR is shown at Figure 3-25).

The reactor core height is 1 m. The operational and experimental channels, coolant pipes and distribution, their collector and a number of metallic constructions of the reactor are located in a pool, filled with distilled water. The pool of the reactor is surrounded by a concrete bioshield for protection. The top part of the reactor tank has a rectangular form of 5000x5600 mm and a height of 4300 mm. The bottom part of the reactor tank has a cylindrical form by a diameter of 3600 mm and a height of 4800 mm. The depth of the reactor pool is 9100 mm.

The MR research reactor advantages are combined from the reactors of channel type cooled by water under pressure and from the ease of use in the field of undertaking experiments which characterise the pool type research reactors.



Key:

1 Vessel of the reactor moderator installation;
2 Beryllium and graphite bricks :
3 Operational channel with motionless fuel assembly;
4 Operational channel with mobile fuel assembly;
5 Support plate with the basic of the operational and loop channels;
6 Collectors of the cooling circuit for the fuel assemblies;
7 Direct flow s-similar loop channel;
8 Channels with detectors SUS;
9 Carriage with drives of SUS-rods and of mobile fuel assemblies;
10 Pipelines of the stuck cooling circuit;
11 Channel with SUS-rods

Figure 3-25 Vertical section through MR reactor

The maximum density of thermal neutrons was:

- $8*10^{14}$ neutrons/cm^2.sec in the spherical trap,
- up to $5*10^{14}$ neutrons/cm^2.sec, inside the beryllium blocks
- $2.75*10^{14}$ neutrons/cm^2.sec inside the operational channels (on uranium)

The maximum fast neutron density with energy more than 0.5 MeV was up to $3.0*10^{14}$ neutrons/cm^2.sec.

The nominal reactor power was 20 MW. In 1967 the reactor was modernised. Its thermal power was increased up to 50 MW (with loop installations) and experimental opportunities of the reactor were considerably extended.

During the modernisation, all the stainless channels water loops of the reactor MR were replaced to zirconium channels with the aim to increase the reactivity reserve of the reactor. This also gave the opportunity of the installation of a large number of loop channels. The design of the reactor provides the application of direct flow loop channels with feed and supply communication equipment located away from the core. The reactor has nine experimental loops for tests of fuel assemblies for power reactors of a various type and assignment. The loops allow carrying out tests of fuel assemblies on a power level up to 3000 kW, at temperature up to 360 ^{o}C with a pressure of up to 200 bar.

Tubular fuel assemblies were used in the reactor from the dispersed type (U-Al alloy or dioxide of uranium in aluminium) with uranium 90 % enriched with U- 235, working at high thermal density and specific energy production with the achievement of a large burnup of fuel.

The following test types were executed at the MR reactor:

- reactor material tests for fuel assemblies of power and research reactors
- works were carried out for neutron radiography
- manufacture of radioisotopes

The complex of the research reactor MR includes some buildings and structures, including: the reactor building; the special ventilation and special sewage building; a building of the reactor cooling system; the spent fuel storage; a ventilation chimney.

3.5.2.1 Operational Characteristics of the VVR2 Reactor

The reactor operated as follows:

- **Date of start-up**: 1954
- **Date of reconstruction**: 1957
- **Date of a final shut down**: 1983.
- **General time of operation**: 29 years
- **Effective time of operation**: about 150 000 hours.

The reactor was operated in a continuous power mode, around-the-clock, with periodic stops for refuelling and maintenance. The average duration of operation of the reactor was 5000-6000 hours per one year.

In 1983 the reactor was shut down for the next reconstruction. The purpose of this reconstruction of reactor and buildings was to meet the requirements of new research tasks for the new scientific programmes and to meet the newly issued safety norms.

The plans of reconstruction of the reactor, however, were not finalised because funds from the Institute were not available. In addition, because of increasing local public pressure to shut down nuclear installations located in Moscow, the decision was made to final stop the reactor and begin decommissioning.

3.5.2.2 Operational characteristics of the MR reactor

The MR reactor had the following operational characteristics:

- **Date of start-up**: 1963
- **Date of reconstruction**: 1967
- **Date of a final shut down**: 1993.
- **General time of operation**: 30 years
- **Effective time of operation**: about 100 000 hours

The physical start-up of the reactor was carried out in December 1963. The nominal power was achieved in October 1964.

The reactor was operated by a cyclic diagram of work. The duration of one cycle was from 25 to 35 days. Duration of stops - from 4 to 7 days. 9 cycles were realised within one year by such mode of operations

The temporary mode of operations of the reactor was determined by requirements to undertake tests with experimental fuel assemblies. The work of the reactor on maximum power amounted to 67 % of calendar time on average, and operating ratio of time of the reactor amounted to 0.77.

During operation of one of the test loops there was a case of infringement of loss of seal of an experimental fuel assembly. The result was a significant contamination of the loop circuit by fission products both as solid and gaseous. In view of absence of outflow from the circuit the failure of the experimental fuel assembly has not given significant increases of gaseous and aerosol emissions to the external environment. The necessity of cleaning of the circuit from fission products required dumping the coolant from the circuit. The coolant was dumped into hermetic tanks taking place under a vacuum to avoid the emission of radioactive gases and aerosols to the environment

The subsequent decontamination of the circuit by distilled water in a hot mode with cleaning by ion-exchangers has resulted in a reduction of the activity below the level which existed before the event.

The average dose rate of the personnel was 12 — 24 mSv/year during the operation of the reactor. The maximum dose rate did not exceed 40 mSv/year. Refuelling works caused 80% of the exposure of the personnel. The total activity contained in air samples at the RRCKI territory and around of it did not exceed 4 to $7*10^{-7}$ B/l, that is the usual background. Total activity in water samples from the nearest to RRCKI part of the Moscow river did not exceed (4 to 7) Bq/l, that is also the usual background.

1990 the State Committee of USSR on Nuclear Regulation, the Academy of Sciences of the USSR and the Ministry of Atomic Energy of the USSR have formed a commission of experts to investigate the safety, reliability and efficiency of research nuclear reactors in Moscow. The Commission has involved independent experts of high qualification.

After a careful analysis, the commission has made the conclusion, in which, in particular, is noted:

"Generally the safe operation of surveyed reactors at Moscow is carried out on the basis of the requirements of norms and rules. The influence on the public of the nearest areas at their operation have not exceeded the background irradiation, and for the personnel - the accepted limits".

At the same time the commission connected to introduction or preparation of new rules or changes for them made a large number of remarks. The commission has required protective measures for the case of improbable and hypothetical events (for example of strong external influences or a crash of an aircraft on a roof of the reactor).

By analysing the remarks of the commission, the management of RRCKI has made the conclusion that the elimination of the noted shortcomings will require a lot of time and financing. It was also necessary to take into account the negative public attitude of inhabitants and urban authorities to the continued operation of nuclear reactors in Moscow.

Because of these reasons the operation of the reactor MR was stopped in 1993.

3.5.2.3 Work on Decommissioning of the Reactors

3.5.2.3.1 VVR2

Decommissioning of VVR-2 began in 1983 and has now been fully dismantled. Parts of the reactor systems were utilised for a new installation on the site of the reactor. Spent fuel was removed and is stored in a separate area at the Kurchatov Institute.

The total activity of all components of the reactor (excluding spent fuel) is $2.6 * 10^{13}$ Bq while the activity of the fuel removed was $2 * 10^{15}$ Bq.

The average dose of personnel as result of dismantling works was less than 50 mSv while labour efforts were about 12000 man-hours.

About 600 tonnes of waste was generated and removed from the decommissioning area, the majority of which was sent to the waste management concern ‘Radon’ which operates a waste disposal facility for Northwest Russia.

Before the start of dismantling activities of the reactor and equipment a storehouse was erected for interim storage of spent fuel, and of high activity reactor elements. The storehouse is shown in Figure 3-26.

Figure 3-26 Spent fuel store at the Kurchatov Institute

The building was also reconstructed to provide for changing clothes and sanitary rooms for the workers participating in the dismantling. The radiological staff monitoring point was equipped with additional devices, and also the points for monitoring of radiologically hazardous tasks were completed for dismantling of the equipment and places for its temporary storage.

The design documentation for undertaking the dismantling works was developed by the Institute of nuclear reactors at the Kurchatov Institute and by the Research design institute of assembly technologies (NIKIMT). The design was completed, the necessary equipment for dismantling was prepared and included lifting mechanisms and transportation devices. Space for a temporary storage of radioactive waste generated as result of dismantling, for conditioning, packing of radwaste and their preparation for sending on a repository was organised at the reactor site.

The basic principles, on which the dismantling works were based, were formulated according to the valid norms of radiation safety [3-22]:

- Observance of limits for accepted annual dose for the personnel (50 mSv)
- Exception of any unreasonable exposure
- Decrease of exposure to as low as possible
- Use of experts only for undertaking special operations
- Use of protective means, shielding and other various devices for remote operations
- Undertaking preliminary training of the personnel for performance of radiological works.

The radiation control of operations and the account of received personnel exposure was undertaken to guarantee the accepted basic principles during the performance of dismantling works.

The dismantling works were carried out in the following sequence:

- Measurements of the actual radiation conditions for performance of operations with the subsequent estimation of exposure dose rate
- Removal of fuel from of the reactor and its transportation into a special storage at the site (Figure 3-26)
- Dismantling of the equipment, pipelines, metallic constructions in the pump and reactor hall, dismantling of the concrete protection and removal of ground with sorting and transfer of waste to interim storage and on the specially prepared spaces at the site (the dismantling operations are shown in Figures 3-27 and 3-28)
- Preparation and packing of radioactive waste in special containers for transport to a repository
- Preparation for removal from the reactor of non-radioactive waste and constant radiation control of all removed and transported waste.

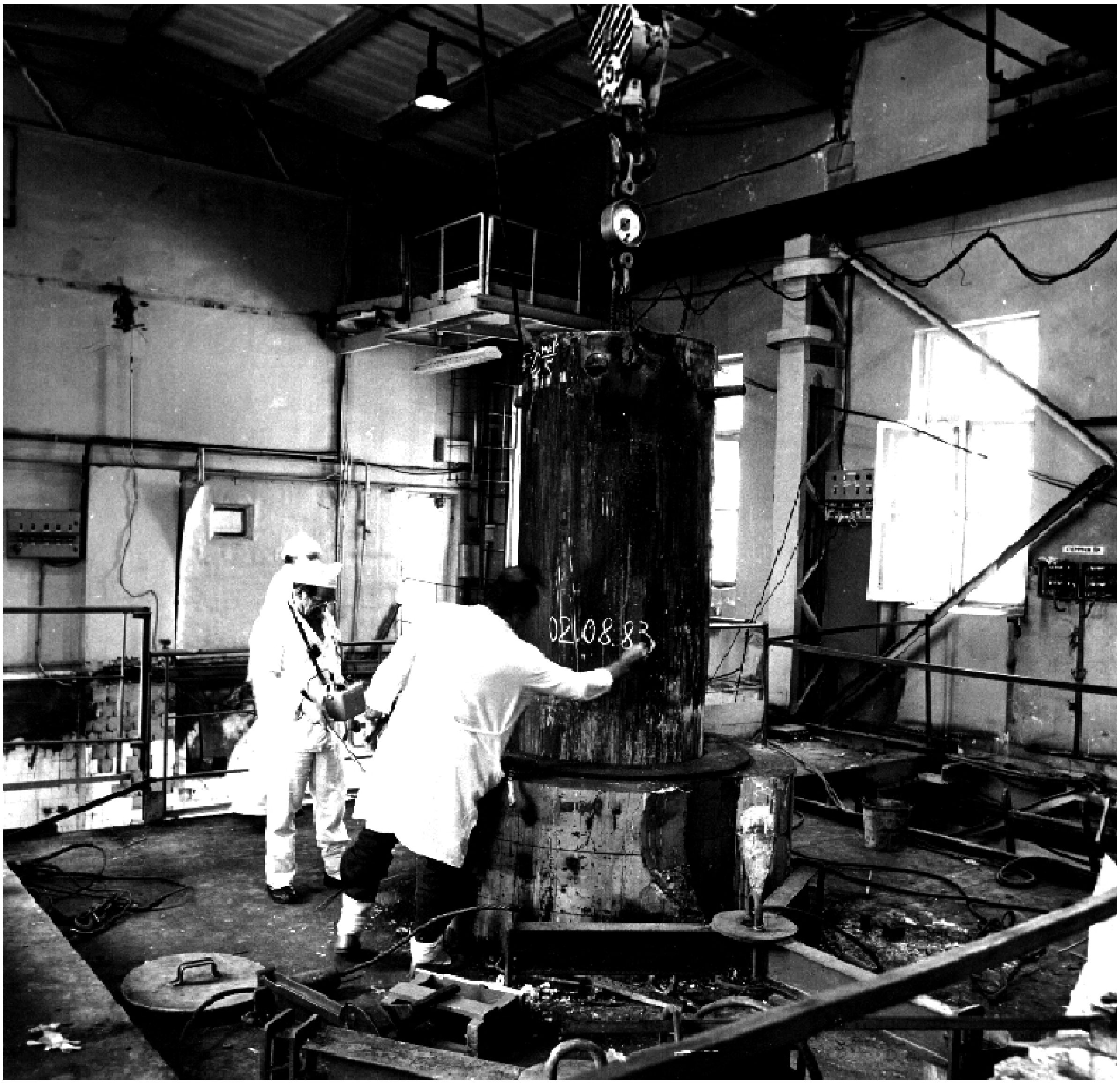


Figure 3-27 VVR-2 dismantling

Figure 3-28 VVR-2 dismantling of components

3.5.2.3.2 MR

Decommissioning works on the MR reactor can be summarised as follows:

- **Start of decommissioning works**: 1993
- **Stage of the realised decommissioning**: reactor is finally stopped
- **Use of installation after shut down:** it is not determined at the given stage
- **Total activity of all components of the reactor (excluding spent fuel) to the moment of its final stop**,: $8*10^{15}$ Bq
- **Activity of spent fuel**,: $1.1*10^{17}$ Bq

- **Average dose for the personnel as a result of decommissioning works**,: < 40 mSv
- **Total quantity of waste (estimation),**: 650 tonne
- **Destination of radioactive waste**: to the special combine 'Radon'

The fuel assembles were loaded into the cooling ponds near by the reactor after the shut down of the reactor MR and the necessary endurance. The loading into the ponds of the reactor was made using the regular technology, which allows the underwater transfer of the fuel assemblies from the reactor to the pond. The reactor was completely cleared of fuel on the basis of a specially developed programme that satisfies its nuclear safety.

An important task to improve the radiation conditions was the evacuation of the reactor pool from operational fuel assemblies and loop channels with experimental fuel assembly, placed in them.

The loop channels were cut and from them the fuel assembly are taken after the realisation of preparatory works (repair of the equipment in the cutting chamber [5], manufacturing of containers for packing of fuel assemblies etc.). Then the fuel assemblies were placed in hermetic stain-less containers. After that the containers were established in a dry spent fuel store.

The preparation of the documentation for development of the first stage of the decommissioning project is finished on the basis of the initial data.

First a complex inspection of radiological inspection of the status at the reactor was realised, also at its systems and experimental installations. The information about the radiation conditions will define the basic parameters of the project in the future.

The analysis of the received results shows that the level of radiation in various systems have a wide range. In some cases the wide range of radiation levels is within the limits of one system and therefore complicates the subsequent decontamination works. These difficulties take place mainly in the experimental loop installations.

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[5] The protective cutting chamber is intended for cutting of fuel assemblies of operational channels from the pendant system. The chamber is connected with the reactor hall by two apertures by a diameter of 250 mm for transfer of fuel assemblies and by diameter of 1000 mm for the container. The chamber is equipped with the cutting machine and manipulators. It is assumed that the chamber will be used for dismantling works to cut the high activated components of operational and loop channels.

It is established by the definition of the radionuclide components that the basic contribution to the radiation is from Cs-137, Cs-134, and Co-60. Therefore an obligatory stage in all kind of decommissioning strategies must be the decontamination of the equipment and reactor systems. The decision of the decontamination task is complicated, if regular systems of processing and solidification of liquid radioactive waste are missing at the reactor MR. The construction of such systems for MR would require long time and large financial expenses. The experts of RRCKI are realising works on research of ways of selective extraction of radionuclides from the decontamination solutions to define the optimum decisions in the field of decontamination. Positive results are received under laboratory conditions.

3.5.3 Radioactive Inventory/Hazardous Material

3.5.3.1 For VVR-2

Total weight of dismantled equipment, pipelines and metallic constructions (including elements of protection) was approximately 630 tons, including about 600 tons that contaminated or activated. The weight of individual elements was up to 5 tons.

The maximum measured level of gamma irradiation of the mild steel elements of biological protection of the reactor, located near to the core, was during the period of dismantling works, approximately 3 Sv/h.

3.5.3.2 For MR

On the basis of radiological inspections carried out at the reactor, the preliminary estimation on quantity of solid radwaste indicates the following:

- Components of the reactor with all elements in place inside its vessel — about 50 tons.
- Core circulation circuit — 60 tons.
- The contour includes the main pumps, the auxiliary and emergency pumps, the heat exchangers, the pressuriser, pipelines with a diameter from 20 up to 366 mm, 225 valves. The maximum weight of one component — 5 tons.
- Auxiliary systems of the primary circuit, for instance: de-aerator systems, gas generators and others — 40 tons.
- Cooling pond cooling circuit — 60 tons.
- Equipment for radioactive rebalance water system — 55 tons.

- Equipment of seven experimental water loops — 175 tons.
- Equipment of helium loop — 160 tons.
- Equipment of a metal loop — 10 tons.
- Equipment of special ventilation systems, storage, cutting chambers and others — approximately 40 tons.

In addition, there are also systems and equipment at the reactor MR, which are not contaminated (including secondary circuit and some auxiliary systems). The weight of metal of "clean" systems is approximately 300 tons.

3.5.3.3 Management Of Radioactive Waste And Spent Fuel

3.5.3.3.1 For VVR-2

The liquid radwaste of low volumetric activity generated by the reactor operations was regularly sent for processing to the special combine "Radon". The remaining (total volume: 14 m^3, total activity: $2.6*10^{10}$Bq) is stored at the reactor site in metal tanks.

The solid radwaste of high specific activity (elements of the dismantled reactor (total weight: 70 tons, total activity: $3.1*10^{12}$Bq) is stored at the reactor site in two underground steel containers with a total volume of 87 m^3 in a specially constructed building.

Others wastes are sent for processing to the special combine “Radon” (Sergiev Posad).

Spent nuclear fuel removed from the reactor (total contents of U- 235: about 31.5 kg and total activity: $2*10^{15}$Bq) is located in a cooling pond at the reactor site. Spent fuel is anticipated to be sent for reprocessing to the “Mayak” enterprise.

3.5.3.3.2 For MR

Liquid waste of low specific activity with a total volume of 20 m^3, and a total activity of $7.4*10^8$ Bq, is stored in three metal tanks at the reactor.

The solid radwaste of high specific activity with a total activity of 10^{15} Bq, is also stored at the reactor.

Storage exists for interim storage of solid radioactive waste near to the MR building. This building is constructed from monolithic concrete structures, in which are placed 127 containers with a diameter of 200-400 mm. Up to now the storage is filled to 78 % capacity. The major part of the solid radwaste of high activity was generated during metallographic investigations.

The main weight of solid radwaste from reactor operations is generated from the experimental channels. During operation about 200 pieces were collected. The maximum activity of the bottom part of the channel is about 0.4 Sv/h. The activity is decreasing with channel height and at the top part it is very low.

Highly activated parts of the channel with a height up to 1500 mm and width up to 280 mm will be loaded in containers for single use, established in a protective container and transported to a repository.

The radiation inspection, which was carried out at the MR reactor, has shown, that available means and technologies will be sufficient for processing and localisation of all kinds of radioactive waste, except some in-core units of the reactor. Weight of such units is about 5 tons, and the radiation levels from them is 16 mSv/h. The problems with these units are expected to be solved at the subsequent stages of decommissioning.

It is required to organise places of interim storage of the solid conditioned radwaste for their preparation for transportation to the enterprise "Radon" in rooms of the reactor MR or near to it.

It will be possible to execute a part of works for localisation of radioactive waste, if necessary, using mobile installations, which the enterprise "Radon" has.

Existing free capacities for storage of solid radwaste at the enterprise "Radon" are not able to accept all radioactive waste caused by dismantling of the reactor MR. Construction of a new solid low and intermediate level radwaste storage will be necessary at the territory of "Radon”. The volume of the storage will be determined by development of the decommissioning documentation.

The nuclear spent fuel (187 fuel assemblies) is placed in a dry storage near by the MR reactor building. Fuel assemblies are stored in aluminium containers, in 2 storeys. A protective steel plate with loading apertures, sealed by fuses covers the storage sections above. The total activity of fission products in the spent fuel is $4.4*10^{16}$ Bq, the total activity of uranium is $4.4*10^{9}$ Bq.

In the dry storage there are also 200 spent loop channels (total activity — $5*10^{16}$ Bq).

The nuclear spent fuel is foreseen for reprocessing at the combine "Mayak".

3.5.3.4 Management of Beryllium

Of some concern is the presence of beryllium inside the MR reactor vessel. Beryllium presents both a problem of conventional toxicity and, following irradiation, a radiation hazard due to the activation of trace impurities e.g. Co-60. Neutron irradiation of beryllium generates H-3 by (n, α) and Be-10 by neutron capture - both beta emitters. Further, at high accumulated neutron fluxes, helium bubble induced accelerated swelling occurs in the beryllium matrix weakening the structure and rendering the material liable to cracking. Experience at the BR2 reactor at Mol in Belgium [3-23] indicates that the initial beryllium matrix, which had accumulated a maximum fast fluence (>1 MeV) of nearly $8x10^{22}$ n/cm^2 had to be replaced since the mechanical interaction between the beryllium channels due to swelling had led to some cracking. For BR2, a maximum allowed flux of $6.4x10^{22}$ n/cm^2 has been defined based on the experimental limits and irradiation temperatures.

These observations pose the question of the mechanical integrity of the beryllium reflectors in MR and suggest that some initial survey work should be envisaged to assess the integrity these materials for handling purposes. Initially, the total neutron fluence of the blocks could be assessed for comparison with the BR2 case. Since the main fixed reflector is likely to be removed contained within it's aluminium shroud this should not prove to be an insurmountable remote handling problem.

For waste management purposes, the beryllium reflectors once removed can be readily stored under 'technological' water - beryllium like aluminium has a protective oxide layer and has a very low solubility in water. In the respect of conventional toxicity hazards (which in this case are greatly outweighed by the radiological problems), the only concern is the potential for future levitation of any beryllium bearing dusts which are a carcinogen when inhaled into the lungs (exposure limit is 2µg/m^3). In contrast, due to the low solubility, ingested beryllium is much less of a hazard.

For the purposes of grouting beryllium into concrete, large sections of material (small surface to volume ratio) are more acceptable in limiting hydrogen generation during the corrosion of beryllium in the alkaline grout. The chemical nature of beryllium indicates that it is probably at least as resistant to alkali attack as aluminium, so that a similar cement formulation could equally used for both materials Smaller beryllium sections and residual dusts (higher surface to volume ratios) will cause more corrosion problems and hence the preference for larger items during waste conditioning. Vented packages are suggested to accommodate these wastes.

3.5.4 Cost Estimates and Funding Requirements

It should be explained that it is difficult to define the real decommissioning resources and costs for the CIS-countries because the economic situation is currently changing from a centrally controlled economy to a market orientated economy with subsequent large gaps in areas such as funding, manufacturing costs, local market forces etc. Therefore the resources and costs are described in this report as they would be in Western Europe but acknowledging the differences where possible. The costs given are developed using Western European labour costs scaled down to reflect the cheaper rates experienced in the CIS countries.

Estimating the decommissioning costs for research reactors is also difficult because of the varied nature of the reactor systems and their supporting services. This section is only able to give a guide to the likely costs. Research reactors are by nature small, but associated with sites involved in research and development in the field of nuclear energy. This usually means that the reactors share a number of facilities with other reactors or research facilities. For example, the Kurchatov Institute in Moscow is made up of 11 institutes with 8 research reactors and a number of critical assemblies, all within the environs of the city of Moscow. NIIAR at Dimitrovgrad has 9 and FEI at Obninsk 2 operating reactors. Experience in decommissioning research reactors in Western Europe and North America suggest that each facility must be considered on an individual basis unless common systems can be identified between reactors.

The previous sections refer where possible to the examples of the VVR2 and MR reactors in Moscow. However, this section will make more general indications for the pool and tank type systems by referring to similar systems within western Europe or earlier studies for Russian designed research reactors. The section also outlines the structure for preparing the cost estimates that are needed in the detail decommissioning planning and documentation phases. The decommissioning must be planned with the supporting infrastructure and adjacent facilities around the RR borne in mind. Facilities identified to assist the decommissioning such as waste treatment and storage should also be utilised for the other facilities if possible bringing economies of scale. This approach develops a site based assessment of the planning and funding requirements rather than each individual RR or nuclear facility on the site. As a consequence the costs of new and supporting facilities will vary depending on the site requirements. Again, as an example, the Kurchatov Institute will want to consider the requirements for waste facilities for all its reactors leading to central facilities where possible to minimise costs and avoid duplication. In fact this requirement may involve the RADON concern (Sergiev Posad) so that they can actually build and operate the waste facilities away from the Moscow environs. This is a more logical requirement to allow the problems of research reactor decommissioning in the Moscow region to have a common waste treatment and disposal facility.

Notwithstanding the discussion above, by reactor decommissioning standards, the decommissioning of pool and tank-type research reactors is a relatively straightforward task - there are no safety, technical and logistical issues beyond the scope of normal nuclear practices.

Once the strategy “Later dismantling after a deferral period” is chosen the outline of the decommissioning project includes the following stages:

- Stage 0 "Preparation for closure"
- Stage 1.1 "Preparation for Care and Maintenance"
- Stage 1.2 "Care and Maintenance of the sealed reactor"
- Stage 2.1 "Preparation for Restricted Site Use"
- Stage 2.2 "Residual Care and Maintenance"

The decommissioning options available for research reactors depend to a large extent on their location. Consequently for RR close to environs like Moscow the options where the removal of the major part of the radioactive inventory is a more likely choice. For this option the costs are somewhat different as the care and maintenance period of the RR may be much less and Stage 3 decommissioning undertaken early.

Three cost categories are usually determined for each working step:

- labour costs which are calculated on the basis of the workload required by a work package and the assumed labour cost unit rate
- investment costs of the equipment and machinery used for a particular working step
- costs of consumables (protective clothing worn in controlled areas, decontamination fluids etc.)

The cost estimate of a work packages includes only labour and new equipment costs which are the biggest percentage of the total costs of these packages.

The planning of the decommissioning costs begins with the year of the decision to decommission and up to the completion of decommissioning or the state of storage with surveillance. The costs for the storage under surveillance and for maintenance at the deferral period because these stages can not be described in detail.

A brief, high level cost assessment was undertaken for the MR reactor in Moscow to develop the levels of costs to be expected for decommissioning. These are then compared below with known decommissioning costs from other assessment on research reactors where information is available to the authors.

Table 3-33 gives a survey about the decommissioning costs per year.

Table 3-33 Estimated decommissioning costs for research reactor

	[million ECU]	-3	-2	-1	0	1	2	3	4	5	6	7	8	9	10
New facilities	24	0.0	3.8	4.6	5.9	6	2.3	1	0	0	0	0	0	0.0	0
Documentation	2	0.0	0.3	0.3	0.5	0.3	0.3	0.3	0.1	0.1	0	0			
Stage 0 Operating & Dismantling	1	0.0	0.3	0.3	0.3										
Stage 1 Operating & Dismantling	7					1.1	1	1.1	0.6	0.6	0.5	0.5	0.5	0.5	0.5
Total [million ECU]	33	0.0	4.4	5.3	6.7	7.4	3.5	2.4	0.7	0.7	0.5	0.5	0.5	0.5	0.5
Total from the start [million ECU]		0.0	4.4	9.7	16	24	27	30	30	31	32	32	32	32.9	33

Figure 3-29 shows the proportioned costs between the individual parts.

Stage 3 costs were not generated but assessed from known activities associated with this stage. Stage 3 costs are estimate to be in the order of 2 M€.

As Table 3-32 and Figure 3-29 show the decommissioning costs are overwhelmed by the costs for new facilities, most of which should be developed as a central facility in line with the IAEA reference design [3-24] and therefore unlikely to be fully attributed to one reactor decommissioning task.

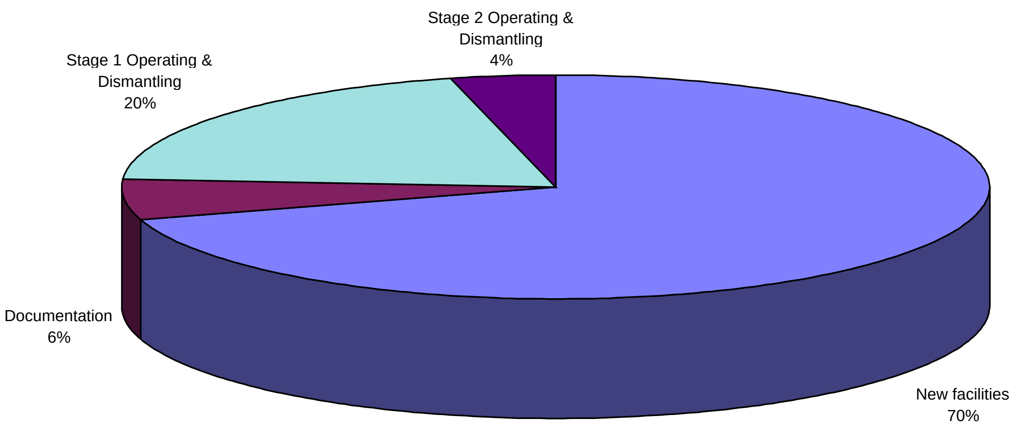

Figure 3-29 Proportion of decommissioning costs

Although decommissioning costs between different research reactors are difficult to compare directly it is useful to highlight the range. For instance, preliminary costs for decommissioning the VVR-S type Magurele reactor near Bucharest in Romania have been carried out between IAEA experts and the Head of the Decommissioning Project of IFIN-HH. This work has identified decommissioning costs which have been developed using two independent costing models and described in [3-25]. The decommissioning programme was similar to that considered here - later dismantling.

The costs use a mixture of project overhead costs for management functions as well as direct costs for staff carrying out the work and equipment and waste management costs during decommissioning. The exercise gave a decommissioning cost range of 1.9 M€ to 2.9 M€ depending of the model used. It is likely that that the actual costs will be somewhere in the middle of this range as the detailed costs estimated by spreadsheet calculations will be reduced by the economies in parallel decommissioning tasks and the effect of staff experience through the decommissioning programme. This is borne out by experience in Western countries on reactor decommissioning.

Detailed costs for the decommissioning of the JASON research reactor in the UK are not available but early assessments suggested that the costs are in the region of 3.5 M€. This includes the Stage 3 costs but excludes fuel storage and reprocessing or the need for new waste management facilities.

For the purposes of assessing the decommissioning costs the basis for the cost estimation is that one man-year costs 50000 €. This is perhaps higher than current labour costs in CIS countries and is part of the reason that the estimate carried out above is higher than similar ones for the Magurele RR or the JASON RR. These costs are given as examples and should not be compared directly with the estimates for the RRCKI reactors.

3.5.4.1 Costs for new and modified facilities

It is planned to construct the following new facilities:

- Liquid radwaste treatment facility with the aim of solidification of all existing and generated by decommissioning (decontamination) liquid radioactive waste. This could be a cementation facility. This is most likely to be built close to the RR being decommissioned.
- Solid radwaste treatment facility with the aim of sorting, conditioning and packing of all existing solid radioactive waste, bulk stored in existing interim storage facilities, and generated by decommissioning and dismantling of the unit(s).

3.5.4.2 Costs for new facilities and modifications of existing facilities

The following new facilities are anticipated to be required:

- Radwaste interim storage facility with the aim of an interim storage for a period of approximately 50 years of solid and solidified radioactive waste, generated at the site. The reason for such an interim storage facility is the missing national or regional repository.
- Spent fuel storage facility with the aim of an interim storage of all spent fuel including the operation of the RR up to the end of its life time (if still operating). The reprocessing and/or the final disposal of RR fuel depends to some extent on the type of fuel used. It is anticipated that the Mayak reprocessing facility is able to take the spent fuel but this has not been finalised and requires funding arrangements to be in place.
- Special decommissioning (decontamination) and dismantling tools and instruments. Little remote handling is planned for these decommissioning stages because the reactor dismantling and the dismantling of high and intermediate radioactive level equipment is to be deferred.
- Radwaste treatment facilities for solid and liquid wastes

3.5.4.3 Costs for decommissioning documentation

The decommissioning process requires:

- the documentation for the complex engineering and radiation inspection
- the decommissioning design and licensing documentation
- the decontamination and dismantling documentation
- some new or modified operational documentation
- the detail design documentation for the safe enclosure of the reactor building for the storage under surveillance

The project manager of the RR has to decide, who or which institution must be involved in the development of the documentation. It would be necessary to contract some development with external contractors, for instance, the detail design for safe enclosure or the special dismantling design. The costs for basic research & develop (R&D) are not included.

3.5.4.4 Costs for decommissioning stages

The decommissioning costs for the stages include the manpower-costs, involving the

- the costs for operators
- the costs for dismantling work
- the costs for operation of the radwaste treatment facilities and of the storage facilities

3.5.4.5 Costs for operation of existing and surveillance facilities

The operational costs include:

- the management cost
- the costs for the operators and repair staff. At the moment of reactor shut down the team consist of approximately 4/5 operators and 2 repair mechanics. The repair staff will also undertake the disconnection and sealing of the reactor building. The operators and the repair staff will undertake the decontamination of the systems using existing equipment.
- the costs for the radiological monitoring staff.

Decommissioning costs are not foreseen for in the period before the reactor shut down. The costs of the plant services (fire department, social and medical services, etc.) are included in the manpower costs as an overhead charge.

3.6 Principal Project Risks

3.6.1 Types of Risk

The cost estimations discussed above are only general in nature as each reactor system must be assessed individually to develop better understanding of the cost elements involved in decommissioning .

The most important risks which could significantly change these costs are:

- Task-related risks - failure to obtain the necessary approvals, permissions, licenses, problems with defuelling, unavailability of waste disposal routes, etc.
- Internal project risks - failure to made the necessary decisions timely, failure to retain key staff, incidents, etc.
- External project risks - policy decisions, increases in regulatory stringency, failure of contractors, economic and financial difficulties, etc.

3.6.2 Risk Management

It is the responsibility of the operating organisation and the reactor administration to forecast to forecast the technical, legal, and economic/financial development. Five years before shut down of the reactor the administration must analyse and forecast the most probable developments of the next fifteen years or so and prepare the decision on the decommissioning using the results of this analysis. It would be useful to include in the preparation process the following:

- governmental institutions
- scientific and technical consultants
- financing institutions (for instance international operating banks)

It would be necessary to sign some insurance for technical risks (incidents, accidents, industrial safety, etc.)

The risk during the decommissioning must be managed by a corresponding project management, contracting with external contractors, financial and time management. It would also be useful to involve a legal consultant in the project management - for instance for problems in the field of licensing.

3.7 Sources of Funding

3.7.1 NPP Decommissioning Funding

With the purpose to ensure a steady centralised financing of decommissioning of NPP a all-branch uniform decommissioning fund was established at January 1, 1991 inside the Ministry of nuclear engineering and industry of the USSR (EFS) on the basis of submissions from the cost price of production at working NPP. The presence of a uniform fund for NPP decommissioning NPP allows to plan the necessary works. The EFS will finance all works connected with the NPP decommissioning including research, tests, design, constructional and other works connected to manufacturing of the equipment, to adaptations, to manufacturing of special tool for dismantling and other technological purposes. The EFS is not subject to use for other needs.

The submissions in EFS are made from the working NPP monthly during the life time of the units, beginning one month after the achievement of the nominal power of the NPP unit. The amount of submissions in EFS is 1.5% of the cost price of production of the operating NPP units.

It should be noted that the LNPP has established its own operating organisation (utility), independent on the central NPP utility Rosenergoatom. This is the reason, that the LNPP established its own decommissioning fund in 1997.

The situation for decommissioning funding in Kazakhstan is not as complex because the BN-350 NPP is operated by the MAEC who are owned by the Kazakhstan government. Currently it is assumed that the government will fund the decommissioning programme when the reactor shuts down in a similar manner to the current government funding for the feasibility studies being carried on the BN-350.

It is obvious that the EFS of the Russian Federation does not provide anywhere near the level of funding required for decommissioning as it has only been in existence for a few years - see section 2. Therefore the idea is to extend the life-time of the NPP units and to collect the amortisation sum in this period, which is not necessary for a replacement of the unit, because the unit has achieved the whole amortisation level, for decommissioning.[6]

[6] The overall amortisation was planned for the nominal life time, e.g. 30 years. A life time extension of 10 years would give 33% of the amortisation sum. International experience says that the decommissioning costs are approximately 15% of the price of a new unit.

Under the current circumstances this could be the only way to collect the necessary amount for decommissioning.

Up to now the cost estimate of the decommissioning projects has no uniform approaches. But the cost of the decommissioning of all NPP's and the timing diagrams of the necessity of using this fond are the base for the establishing this fond. So it is necessary to estimate the decommissioning cost for all units by the same way. For this purpose for instances in Germany the regulatory bodies demand an annual updated cost estimate for the decommissioning of a NPP unit. The cost calculations are carried out with the programme code STILLKO2. This program is acknowledged by the authorities. The results of this calculations are necessary for:

- planning funds and
- planning national or regional repositories for waste and spent fuel.

3.7.2 Research Reactor Decommissioning Funding

Funding sources for RR decommissioning is likely to remain one of the most influential concerns.

The funding for decommissioning of research reactors is currently difficult to establish. All institutes operating research reactors were originally State financed institutions in the former Soviet era. They received financing for the undertaking of research works, capital construction and other works, but these institutes did not accumulate any finance for others (off-schedule or unforeseen) work. In the conditions of transition to a market economy these institutes have found known independence and have received the status of operating organisations, but have not yet achieved the necessary financial security.

According to the Federal Law, recently accepted in Russia "On the use of atomic energy" the State has transferred on to the operating organisations of research reactors the duty to carry out decommissioning activities using their own forces or with other organisations.

Now that these organisations have economic independence by carrying out research and other applied works for external customers they could form the specialised funds for decommissioning of research reactors, stipulated by the Federal law. Realistically, the work of the institutes does not allow them to accumulate the required financing in these funds now. Moreover, up to the moment of the decommissioning of many research reactors there has not been enough time and consequently the appropriate accumulation in these funds will not be significant.

Obviously, it is very difficult to carry out the decommissioning of research reactors without the financial help of the state.

3.8 Financial Approval and Control

The decommissioning project management needs a correct financial planning and controlling. The procedure of financial approval and control is nearly the same as for constructions of new units, but must take into account, that some various kinds of co-workers and personnel will be involved in the decommissioning activities

- reactor personnel
- on-site personnel of external contractors
- home office staff of external contractors
- Regulatory Authorities

For all activities contracts must be prepared and agreed which are the basis for the financial control and approval. "Internal contracts" must be also agreed with the site facilities and/or departments which are involved in the operation, decommissioning, decontamination, engineering and dismantling.

It could be useful to involve an external independent financial Consultant in the project management.

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4. Conclusions and Recommendations

4.1 Conclusions

The following conclusions are identified:

- The countries of the CIS have inherited a significant legacy of nuclear reactors from the former Soviet Union with the current situation as follows:
  - In Russia four NPP units have already been shut down, 2 at Novovoronezh and 2 at Beloyarsk. In the short term (within the next 10 years or so) a further 10 NPP units are expected to shut down: Bilibino 1-4, Leningrad 1-2, Novovoronezh 3-4 and Kola 1-2.
  - In Ukraine three NPP units are shut down, all at Chernobyl NPP and the remaining one (unit 2) is expected to shut down in the near future.
  - In Armenia one unit has been shut down
  - In Kazakhstan the BN-350 fast reactor is expected to close within the next five years
  - A total of eight research reactors are known to be closed up to now, but many more are older than 25 years and are currently inactive. Assessments suggest that approximately 10 - 20 research reactors would be decommissioned in the next ten years
- Such estimations of when a NPP is expected to shut down can not be exact and are influenced by a balance of the anticipated end of design lifetime of 30 years with the pressures to extend such lifetimes to continue to produce power and support the countries economic situation. Similarly, research reactors are flexible by nature and can be refurbished and continue to operate well beyond the original design lifetime providing safety and funding conditions are met
- Decommissioning in the CIS is still relatively new and is constrained by the lack of funding and regulatory framework in which to carry out decommissioning activities. Funds for decommissioning NPPs have only been recently set up and do not have sufficient funds to cover the final decommissioning costs. This is exacerbated by the current financial difficulties the CIS currently face in adopting a market led economy from the original centrally led economies of the former Soviet Union. Funds for decommissioning research reactors are non-existent and the responsibilities have now been passed from the government to the operating organisations to finance the decommissioning. This drives the operating organisations to consider lifetime extension

of facilities in order to generate additional funds, some of which may be used for future decommissioning.

- The limited regulatory experience in the decommissioning field hinders the development of clear decommissioning practices in CIS countries which would assist in stabilising the decommissioning requirements for the different nuclear facilities. Clear benefits of economies can be achieved by considering generic decommissioning strategies for similar reactor designs but all must conform to the requirements in a structured regulatory regime.
- Radioactive waste management issues also constrain the decommissioning requirements of nuclear facilities. Most of the radioactive waste management facilities that are currently available are dealing with the accumulation of operational waste stored during the lifetime of the facility. The addition of wastes from decommissioning can not be met by the current systems in terms of storage or capacity of treatment facilities. The disposal of radioactive wastes remains unsolved with no final repository available for waste.
- The management of one waste stream, graphite, requires resolution at the international level with many countries having significant graphite waste volumes to be processed and treated.
- The management of spent fuel is dependant on the nuclear reactor type. Fuel from RBMK units have no disposal route through reprocessing and are currently stored on the NPP sites. Fuel from VVER units does have a disposal rout via reprocessing at Mayak but this is a costly option requiring funds to be made available. Fuel from research reactors varies in configuration and type and requires a number of reprocessing techniques. Some of these are available at sites such as Mayak but others are not and fuel is being stored on the site of the research reactor. Again, reprocessing, if available, is a costly option.
- Decommissioning of large, multi-unit NPP units is both technically complex and has significant implications for the local social and economic situation. In Russia decommissioning of older generation units on a site is linked to the construction of new units to replace the power generated and maintain the social and economic regime that exits to support the NPP site. It is not uncommon for a NPP site to support up to 50,000 (on average) local inhabitants who are linked directly or indirectly to the operation of the site.

- The location of research reactors can affect the pressures for decommissioning. Sites such as the Kurchatov Institute in Moscow are influenced by the local population concerns living close to the nuclear facilities.

4.2 Recommendations

The following recommendations are made:

- Decommissioning of nuclear facilities must have a clearly defined regulatory requirements. The collection and assessment of information related to decommissioning can prove beneficial in developing suitable decommissioning requirements. Consequently it is recommended that, in co-operation with CIS partners a framework is developed for the collection and evaluation of nuclear facility information relevant to decommissioning.
- Decommissioning of nuclear facilities must have a clearly defined waste management and disposal route. Although waste management can include the long term storage of wastes above the ground ultimately a final disposal site is required such as a deep geological facility. A national solution for a repository is required, possibly based on regional considerations if more than one is identified to serve the needs of the CIS. Such efforts should be developed with the co-operation of international collaboration to ensure best practice and modern methods are utilised.
- The management and disposal of graphite from nuclear facilities is a significant problem for a number of countries such as Russia, Ukraine, United Kingdom and France. Graphite from RBMK units is particularly susceptible to long term deterioration without suitable treatment or disposal. Solutions to the problem of graphite waste management should be applicable to these countries. Co-operation with interested parties including those from the CIS should be developed on an international level to spread best practice and experience.
- The collection and assessment of information related to decommissioning can prove beneficial in considering suitable decommissioning requirements. Consequently it is recommended that, in co-operation with CIS partners a framework is developed for the collection and evaluation of nuclear facility information relevant to decommissioning.
- Although large scale decommissioning of NPPs is not expected to commence in the short term, there will be a significant amount of decommissioning work related to Stage 1 activities and preparing reactor systems for future decommissioning strategies such as storage and surveillance. Consequently there is a requirement for the exchange of experience in decommissioning tools and techniques. In addition an exchange of

experience in developing decommissioning documentation has been identified to maintain best practice.

- The management of contaminated sodium is, again, a significant problem with many countries having significant quantities of sodium from fast reactor programmes. The treatment, conditioning and disposal of sodium is a complex requirement. Co-operation with interested parties including those from the CIS should be developed on an international level to spread best practice and experience.
- Techniques for decommissioning, particularly decontamination methods for research reactors in the CIS countries are limited. With the tendency to defer decommissioning later then techniques to remove or fix mobile activity are required.