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Main Features of Kola, Leningrad and Ignalina NPPs for Emergency Preparedness Purposes

Heikki Holmström
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December 2001

Abstract

Of the nuclear power plants situated in the Nordic and their neighbouring countries, the Ingalina, Leningrad and Kola plants are considered to pose the largest risks to the public.

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Key words

Nuclear power plant, RBMK, BWR, VVER, reactors, reactor design, design data, reactor core data, fuel data, layout, activity inventories, main features.

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Research Report

ENE4/32/01

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for Emergency Preparedness Purposes

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R e s e a r c h   R e p o r t

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Espoo, December 2001

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Distribution

Project Group of NKS/SBA-project

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CONTENTS

1 INTRODUCTION

Of the nuclear power plants situated in the Nordic and their neighbouring countries, the Ignalina, Lenigrad and Kola plants are considered to pose the largest risks to the public. The purpose of this report is to provide basic relevant information about the three plants for use in a case of a major nuclear accident or incident in any of them. The report could be used e.g. by authorities dealing with the resulting emergency measures to provide the public and the media with relevant information about the plant in question.

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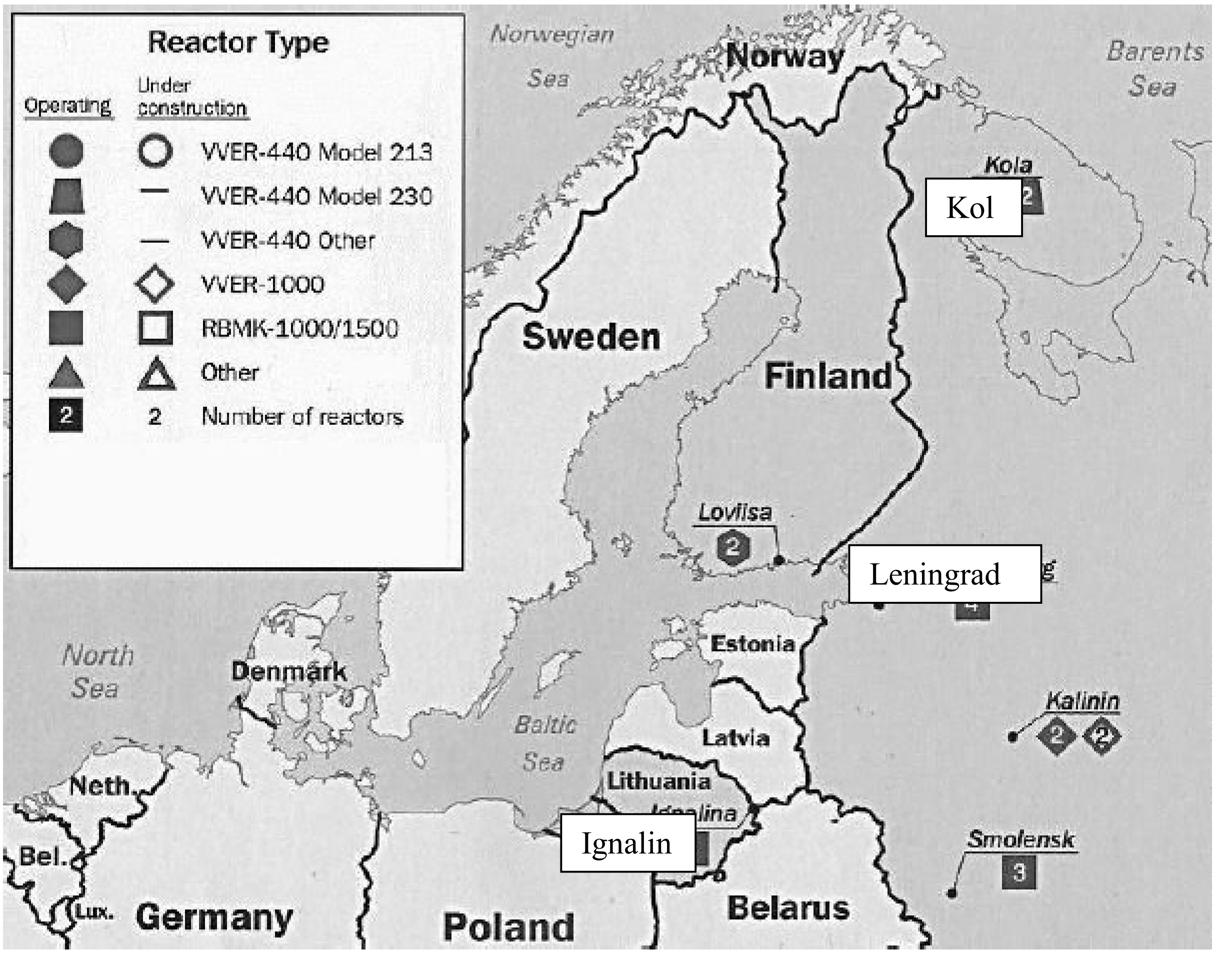


Fig.1.1. Map showing locations of the Kola, Leningrad and Ignalina NPPs (Source: http://www.lib.utexas.edu/maps/commonwealth/soviet_nuc96.jpg)

2 RBMK NPPS

2.1 Features of RBMK NPPs

2.1.1 General

Russian nuclear power plants are of two basic designs, generally known by the acronyms VVER and RBMK. The latter is a boiling-water cooled, graphite-moderated, channel-type reactor unique to the former Soviet Union. Leningrad is an RBMK-1000 NPP (electrical power 1000 MWe) and Ignalina an RBMK-1500 NPP (electrical power 1500 MWe).

The RBMK's were primarily designed during the 1950's and 1960's. These types of reactors have been constructed only in the former Soviet Union. There are 13 units in operation, 11 of them in Russia. The other two operating units are in the Lithuania. All the four RBMK units at Chernobyl in the Ukraine have been finally closed. One unit (Kursk 5) is still under construction in Russia. The electric power capacity of all operating RBMK units is 1000 MWe with the exception of the two Ignalina units in Lithuania.

The RBMK is known as a channel-type reactor. There is no actual reactor pressure vessel. Some 1600 pressure tubes provide channels passing vertically through a massive stack of graphite blocks acting as the moderator. Assemblies of slightly enriched uranium fuel are loaded on-line into the channels. The reactor is cooled by water flowing up the channels and through the fuel assemblies. The operating values such as temperatures, pressures, and specific power in the fuel are close to those of the western boiling water reactors.

The RBMK's have no Western-type containment structure. Instead, the units have a confinement around the main components of the primary circuit with a system of pressure suppression pools for steam condensation. The main differences between the RBMK's of different age are in the capacity of these confinement systems and of the emergency core cooling systems. The reactor can exhibit an instability that is known as a positive void coefficient. This means that if the voids formed by steam in the boiling water increase, then the nuclear reaction rate will also increase, the temperature will rise, and more steam voids will be produced. This design fault of RBMK's caused the Chernobyl accident, in combination with some other factors.

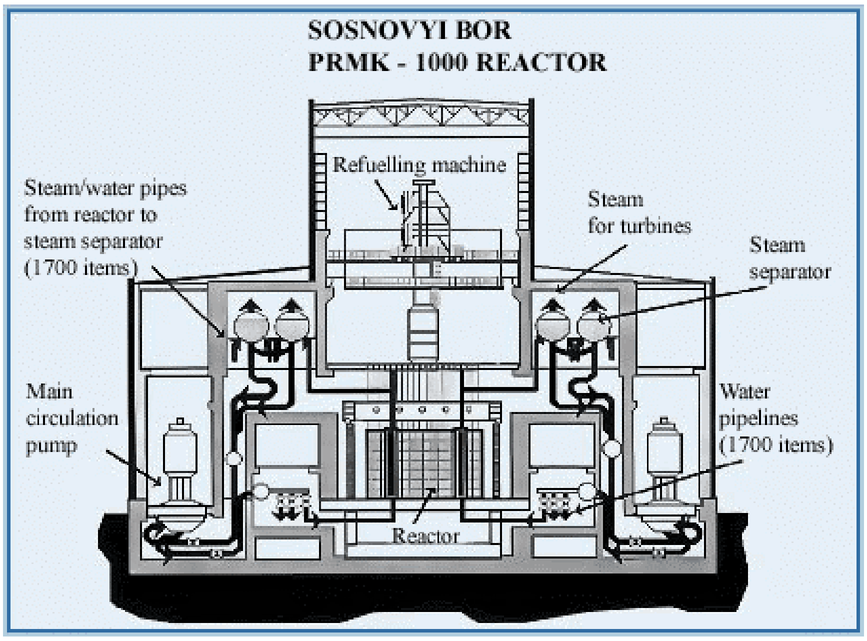


Fig. 2.1. The reactor building of an RBMK-1000 reactor (Leningrad nuclear power plant) (Source: www.stuk.fi/english/npp)

After the Chernobyl NPP unit 4 accident, several design modifications and upgrades have been performed so that a similar accident is no longer possible. For example, the instability due to the positive void coefficient has been decreased, the efficiency of the scram system has been increased, the operative safety margin is now displayed on the operator console, and the capacity of emergency core cooling systems and pressure suppression systems has been enhanced especially at the older units. Yet, other concerns remain, and they are focused on areas such as human errors, structural integrity, and fire protection. The fire in the Chernobyl NPP unit 2 in 1991 demonstrated the vulnerability of RBMK's to fires. The unit has been closed since then. There is a conviction among many international experts that none of the RBMK units is yet sufficiently safe.

2.1.2 Differences between BWR and RBMK reactors

The RBMKs could be considered as kind of boiling water reactors but designed from totally different principles than the Western type of boiling water reactors. One of the main differences is the way the neutrons are moderated. Western BWRs are normally water moderated whereas RBMKs are graphite moderated. RBMK thus represents unique design features with a graphite moderator and a very large core and a large load of low enriched uranium fuel.

The graphite moderator of the RBMK reactor plays a significant role in defining the characteristics of the reactivity feedback coefficients, and due to the large core size, the core power distribution is unstable, with the fuel load comprising several local critical masses. These special design features produce unique neutronics and complex reactivity control requirements.

Among the important safety design differences between the Leningrad and Ignalina RBMKs and Western BWRs the following items can me mentioned:

- Moderator type
- Power density
- Size of core
- Void coefficient
- Control rods
- Refuelling technique
- Passive safety systems
- Number of safety systems
- Boron injection system
- Active safety systems
- Containment system
- Filter/scrubber system

The graphite moderator of the RBMK reactor is exposed to a special ageing effect. Due to irradiation the graphite is accompanied by a creep or shrinkage effect, which causes a closure of the gap between the fuel channels and the graphite blocks. Thus, after about 15 years of operation the graphite blocks need to be bored out to enlarge the channel diameter - a very costly and complicated process.

The differences in volumetric power densities between RBMK reactors and Western BWRs are due to the size of the core. The core volume of a RBMK reactor is about 10 times the volume of a Western BWR with the same thermal power. The fuel specific power expressed as kW/kgU is about 22 for both type of reactors, whereas the core power density is about 50 W/cm3 for typical Western BWRs and 5-7 W/cm^3 for RBMK reactors.

The graphite moderator constitutes a large heat sink in case of a loss of coolant accident. E.g. in case of failure of the decay heat removal system the heat capacity of the graphite mass is assumed to accumulate most of the decay heat for at least 24 hours without leading to any fuel damage.

The coolant *void reactivity coefficient* of the RBMK reactor is positive under most operating conditions whereas this coefficient is negative for Western BWR reactors. The positive coefficient is due to the fact that the moderating effect of the water is relatively small since most of the moderation is caused by graphite. Thus a decrease of the coolant density by voiding is accompanied by a decrease in neutrons absorbed in the coolant and a corresponding increase in reactivity. In a Western BWR, the negative moderating effect of removing water is always greaterthan the positive absorber effect, so that the void coefficient is negative. The positive coolant void coefficient is supposed to have been an important contributor to the Chernobyl accident. From a regulation point of view it is desirable to have a negative void reactivity coefficient of small numerical value.

For most RBMK reactors the enrichment has been increased and additional absorbers have been installed in the core after the Chernobyl accident. In this way a less positive coolant void coefficient has been obtained because a smaller fraction of neutrons now is absorbed in the coolant, making the reactivity less sensitive to coolant density changes.

One characteristic difference between the control rods of RBMK reactors and Western BWRs is their direction of movement. The control rods of RBMK reactors are inserted from the top of the core, opposite to Western BWRs where control rods are inserted from the bottom. Thus, the RBMK way of movement utilizes gravity as a passive safety feature wheras the Western design utilizes the effect of faster response due to higher power density in the bottom of the core.

The number of control rods, their design and velocity of insertion have been changed for RBMK reactors after the Chernobyl accident. Each RBMK reactor has been equipped with 80 new absorber assemblies, which are left permanently in the core. The design of the local emergency control rods has been changed by eliminating water columns in the lower part of the rods and including larger absorbing sections, thus avoiding an initial positive reactivity insertion during operation (like at Chernobyl). Further the insertion time of the local emergency rods has been reduced from 18 to 12 seconds.

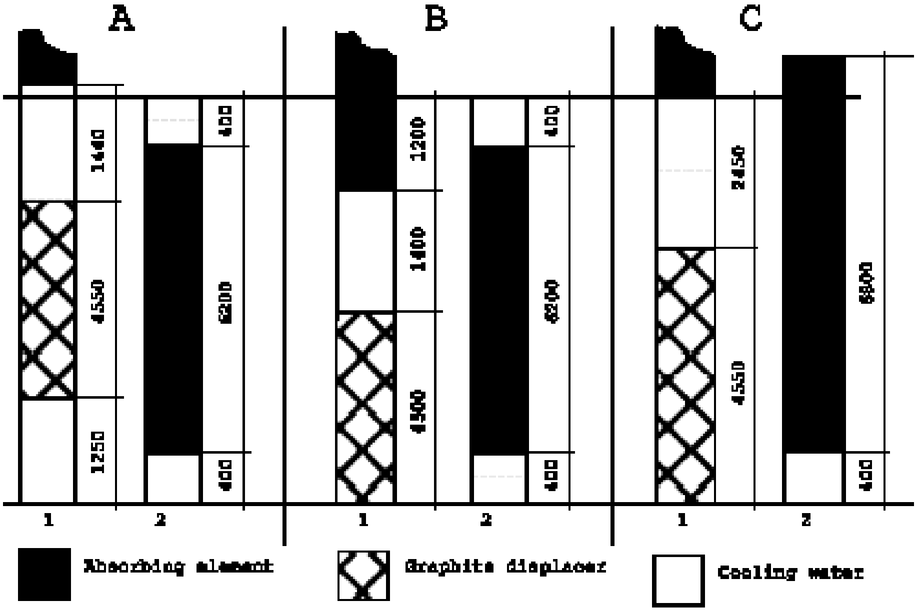


Fig. 2.2. Control rod positions at different levels in the core (Source: Ref. 1)

A. control rod of original design

B. control rod of original design partly inserted into core to eliminate the possibility of insertion of positive reactivity

C. improved control rod design
1 - rod withdrawn
2 - rod inserted

One special feature of the RBMK reactors is refuelling during power operation. The refuelling operation is remotely controlled, and the reactor hall is unoccupied during the operation. Normally two refuelling operations are made each day at full power and the whole operation takes about two hours.

In a Western BWR refuelling is carried out during the annual shutdown for maintenance and repair.

Both types of reactors are provided with emergency core cooling systems, but the application of redundancy and diversity is more consistent in Western reactors than is the case for the Leningrad and Ignalina nuclear power plants.

Western reactors are provided with a secondary diverse shutdown system, that is a boron injection system, which is to be used in case of a failure of the normal control rod shutdown system. The Leningrad and Ignalina NPPs have no secondary shutdown system.

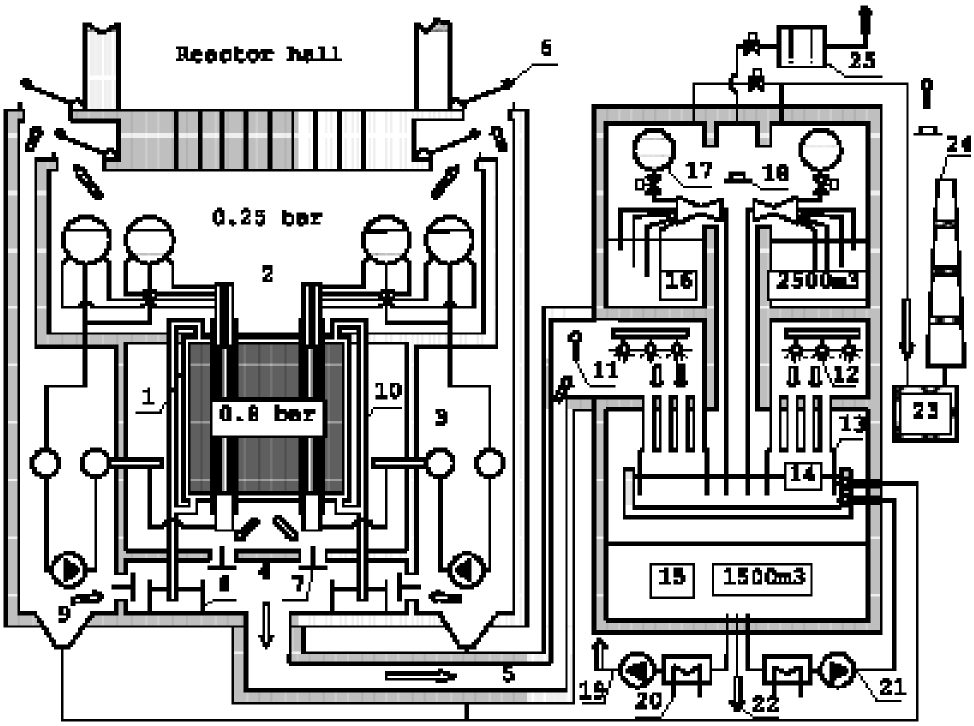


Fig. 2.3. Accident "localization system" of an RBMK reactor (Source: Ref. 1)

1. Reactor tank
2. Steam separator compartment
3. Pump compartment
4. Compartment below reactor
5. Corridor
6. Rupture disc
7. Relief valve
8. Water lock
9. Drainage
10. Relief pipes from the reactor tank
11. Thermoelement
12. Sprinkler
13. Bubbler
14. Bubblers water pool
15. Lower water pool
16. Upper water pool
17. Upper water tank
18. Pressure measurement
19. Cold water for sprinklers
20. NA-service water
21. Cold water for bubblers pool
22. Emergency core cooling water line
23. Carbon filters
24. Ventilation stack
25. Valves and rupture discs (membrane)

The lack of a pressure containment for the RBMK reactors is the most important design difference between Eastern and Western boiling water reactors from the safety point of view. The Ignalina NPP and units 3 and 4 of the Leningrad NPP are provided with a confinement system, an "*accident localization system"*. However, the design philosophy of this confinement is different from the Western philosophy. It is a building where, in case of a main coolant pipe break, the discharged steam and gas mixture is condensed by bubbling through a condenser-pool, purified and released to the atmosphere after a certain delay time. So it is not leaktight.

The overpressure protection system of the reactor tank also discharges to this *accident localization system* in case of a rupture of a fuel channel pressure tube. The original design basis of the protection system was a break of a single pressure tube, but the relief capacity from the reactor tank volume has been increased, so that it now can withstand simultaneous breaks of four fuel channels.

The units 1 and 2 at the Leningrad Nuclear Power Plant had originally no *accident localization system*, and the condensing capacity and the delay time of possible releases were smaller. However, apparently units 1 and 2 were provided with an *accident localization system* in 1997.

If the pressure in the reactor space exceeds the relief capacity, the upper biological shield will lift and a serious accident might occur. IAEA has stressed the necessity of increasing the relief capacity from the space so that the number of allowable simultaneous pressure tube breaks would be ten for all RBMK reactors. This backfit was apparently carried out at the Leningrad NPP in 1997.

The containments of Western BWRs are typically designed to withstand a pressure of 5 bar and also capable to withstand a crash of an airplane. Furthermore, if relief of steam or gases to the atmosphere should be necessary it will take place only after long delay times through filters and scrubbers.

2.2 Leningrad NPP

2.2.1 Description

The Leningrad nuclear power plant is located in the neighbourhood of the town Sosnovyi Bor on the Baltic coast about 70 km west of St. Petersburg and 240 km from Helsinki.

Fig. 2.4. Leningrad NPP (Source: www.laes.sbor.ru/new_lnpp/eng-flash/razdel/vizitka/photo/laes1.jpg)

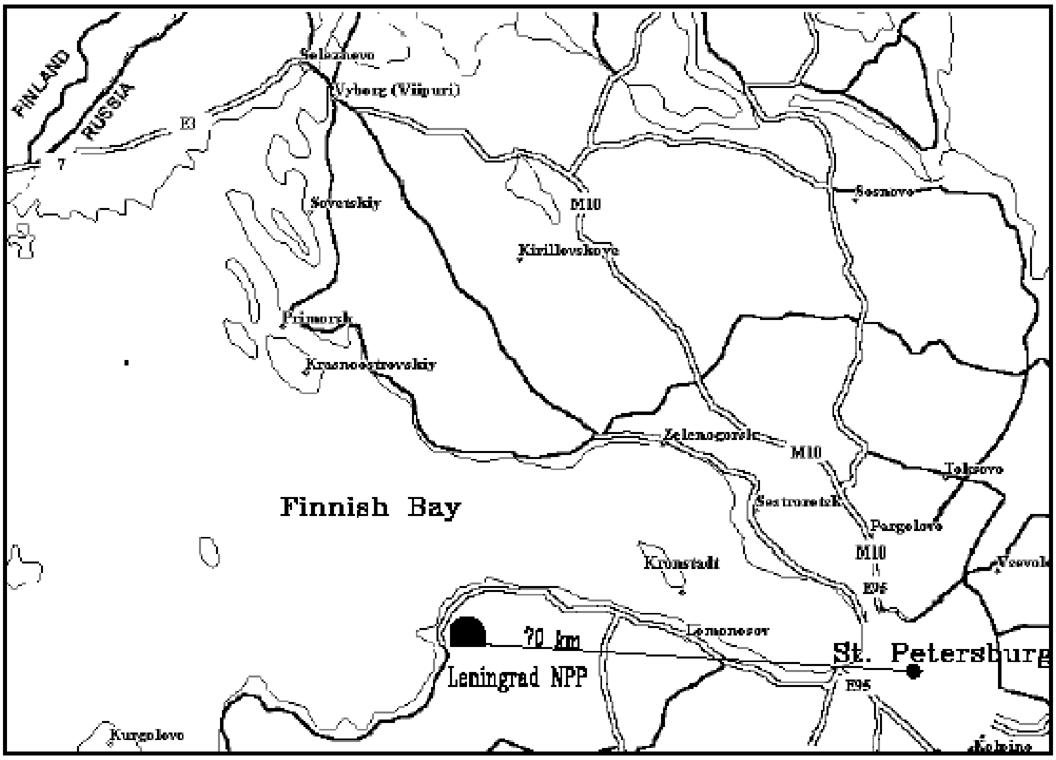


Fig. 2.5. Location of Leningrad NPP, 70 km West of St. Petersburg. (Source: Ref. 1)

The plant has four RBMK units. It has been built in two stages; the first two units were taken in operation in 1973 and 1975 and the second stage with units 3 and 4 in 1979 and 1981. The electrical output of each unit is 1000 MW. The main differences between the two stages are in the emergency core cooling systems and the confinement systems.

The RBMK reactors are graphite moderated. The graphite consists of blocks that are arranged in the form of columns and the blocks are penetrated by vertical channels, which provide locations for the fuel rods, control rods, graphite reflector coolant tubes and instrumentation.

Units 3 and 4 are same generation of RBMK reactors as Ignalina Nuclear Power Plant which is described in the next section.

Fig. 2.6. Leningrad NPP arrangement
(Source: www.laes.sbor.ru/new_lnpp/eng-htm/cont/proizv/tehnology/20b.htm)

1. Units 1 and 2
2. Units 3 and 4
3. Sea water pumping station, I Phase
4. Sea water pumping station, II Phase
5. Outlet channel, I Phase
6. Intake channel, I Phase
7. Intake channel, II Phase
8. Outlet channel, II Phase
9. Diesel building, unit 2
10. Spent fuel storage
11. Accounts Department
12. Training centre
13. Administrative building
14. Condensate cleanup
15. Diesel building, unit 1
16. Repair & construction shop
17. Nitrogen & oxygen shop
18. Storage facility
19. Component maintenance shop
20. Administrative building
21. Diesel building, II Phase
22. Information centre
23. Fire station
24. Print-house
25. Boiling facility

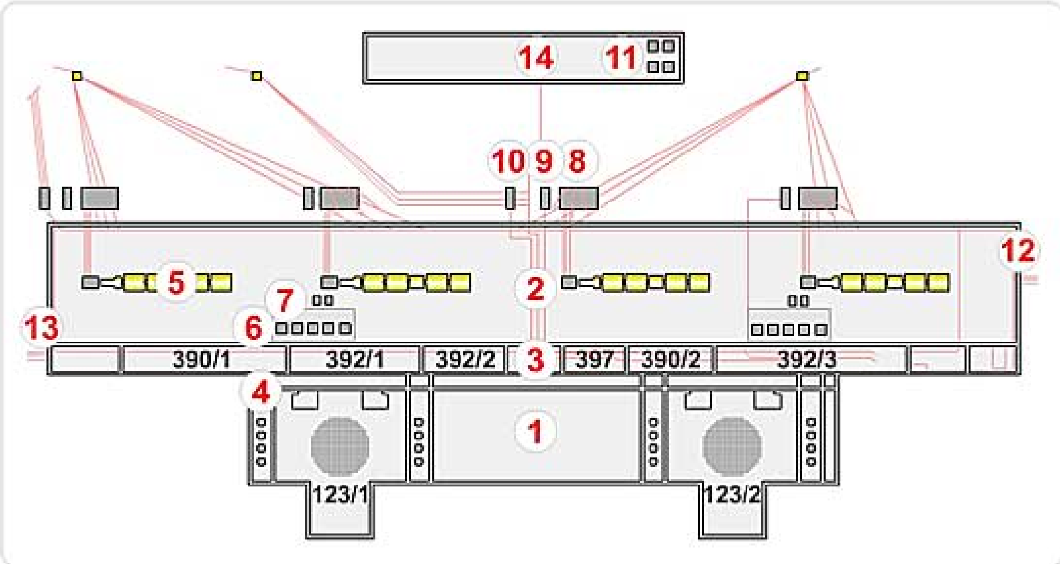


Fig. 2.7. Layouts of Units 1 and 2 of Leningrad NPP
(Source: www.laes.sbor.ru/new_lnpp/eng-htm/cont/proizv/tehnology/20a.htm)

1. Auxiliary building
2. Common turbine hall
3. Intermediate building
4. Main circulation pump
5. Generator
6. Main feed water pump
7. Auxiliary feed water pump
8. Main transformer
9. Auxiliary transformer
10. Start-up transformer
11. NA-pump, service water system
12. Cables to diesel building, unit 2
13. Cables to diesel building, unit 1
14. Sea water pumping station

123/2. Reactor hall of unit 2
123/1. Reactor hall of unit 1
392/1. Control room of unit 1
392/3. Control room of unit 2
392/2. Electrical equipment (SUZ, reactor instrumentation)
390/1. Electrical equipment (SKALA computer)
390/2. Electrical equipment (SKALA computer)
397. Central control room (external grid, fire detection)

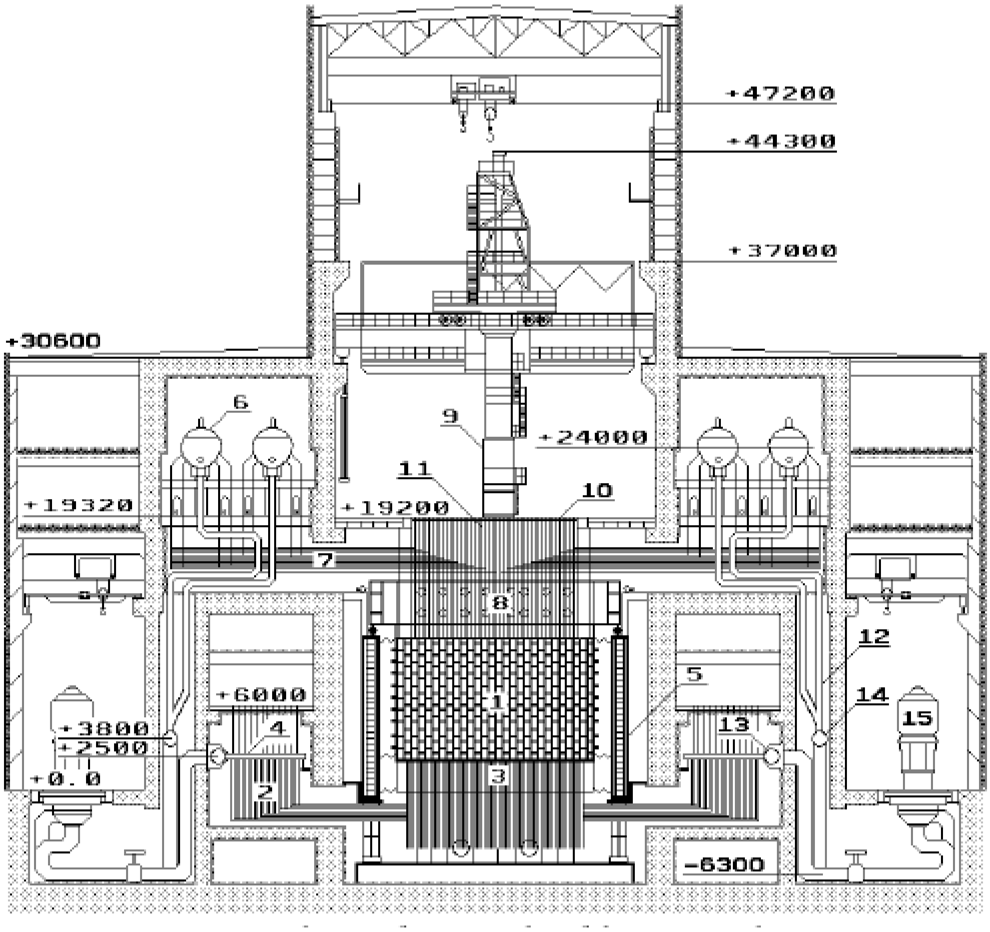


Fig. 2.8. Cross-sectional view of units 1 and 2 of Leningrad NPP (Source: Ref.1)

1. Graphite core
2. Lower pipelines
3. Lower biological shield
4. Distribution header
5. Biological side shield
6. Steam separator drum
7. Upper pipelines
8. Upper biological shield
9. Refuelling machine
10. Removable floor
11. Fuel channel ducts
12. Downcomers
13. Pressure collector
14. Suction collector
15. Main circulation pump

2.2.2 Summary of approximate design data for unit 1 of the Leningrad NPP

Main Data

Reactor type	RBMK	Pressure tube boiling water reactor
Net electrical output	1000 MW	

Reactor

Reactor thermal output	3200 MW
Number of circulation loops	2
Total coolant flow kg/s	10400
Pressure in a steam separator	70 bar
Steam flow kg/s	1500
Steam pressure at turbine inlet	65 bar
Steam temperature at turbine inlet	280 ^{0}C
Feedwater temperature	168 ^{0}C
Maximum thermal power in a fuel channel	3000 kW
Pressure in a pressure tube	
- at inlet	86 bar
- at outlet	75 bar
Temperature in a pressure tube	
- at inlet	270 ^{0}C
- at outlet	284 ^{0}C
Coolant flow through a pressure tube at maximum power	8 kg/s
Max velocity of steam/water mixture in a pressure tube	20 m/s
Max steam content in a pressure tube outlet mass	21 %

Reactor core data

Core diameter	11.8 m
Core height	7 m
Specific power	16.7 W/gU
Graphite mass in the core	1700 x 10^3 kg
Graphite temperature	700 ^{0}C
Maximum temperature of metal structures	330 ^{0}C
Minimum dryout margin	1.05

Fuel data

Total weight of uranium	192000 kg
Number of fuel assemblies	
- units 1 and 2	1693
- units 3 and 4	1661
Number of fuel rods per assembly	2*18
Fuel assembly diameter	79 mm
Fuel assembly length	6954 mm
Fuel rod diameter	13.5 mm
Lattice pitch	250 mm
Fuel enrichment	2.4 % ^{235}U
Maximum fuel temperature	1800 ^{0}C
Duration of operation of a fuel assembly at nominal power	1190 days
Average fuel burnup	22500 MWd/tnU

Pressure tube data

Pressure tube outer diameter	88 mm
Pressure tube wall thickness	4 mm
Average linear thermal power	146 W/cm
Maximum linear thermal power	350 W/cm
Maximum thermal flux on the surface of a fuel rod	83 W/cm2

Control rods

Number of control rods	
- units 1 and 2	191
- units 3 and 4	211
Type of control rods	annular boron carbide

Reactor circulation pumps

Number of main circulation pumps	8
Rated flow	2,2 m^3/s
Pressure after pump	90,5 bar
Pressure difference	20 bar
Nominal electrical power	5,5 MW
Speed	1000 rpm

Steam separating drums

Number of drums	4
Diameter of a drum	2.3 m
Length of a drum	30 m
Weight of a drum	200 x 10^3 kg
Pressure in a drum	70 bar

Turbine plant

Generator output	2 x 500 MW
Turbine shaft length	39 m
Turbine speed	3000 rpm
Pressure in the condenser	0,04 bar
Number of low pressure cylinders	4
Pressure in the high pressure inlet	65 bar
Temperature in the high pressure inlet	280 ^0C

2.3 Ignalina NPP

2.3.1 Description

The Ignalina nuclear power station is located in Lithuania, close to the borders of Belorussia and Latvia. The station is built near the town Ignalina and the distance to the capital Vilnius with 600 000 inhabitants is 130 km. Daugavpils in Latvia with 150 000 inhabitants is located 30 km from the plant.

Fig. 2.9. Ignalina NPP
(Source: www.iae.lt/inpp_en.asp?lang=1&subsub=8, f26.jpg)

The plant has two units with graphite moderated pressure tube boiling water reactors of similar type as the Leningrad Nuclear Power Plant (LNPP). The Ignalina Nuclear Power Plant (INPP) and units 3 and 4 at LNPP represent the second generation of RBMK development, while units 1 and 2 at LNPP represent the first generation.

The electrical output of each unit at the Ignalina Nuclear Power Plant is 1500 MW, but since the Chernobyl accident the allowable power of each unit has been reduced to 1250 MWe. The first unit was taken in commercial operation in 1984 and the second in 1987. The two units of the Ignalina Nuclear Power Plant comprise the only construction of RBMK type of reactors with a designed electrical output as high as 1500 MW.

The core dimensions of the Ignalina and Leningrad Nuclear Power Plants are the same as well as the amount of uranium in each core. However, Ignalina is designed to operate with 50 % higher power density in the core due to an increased heat transfer obtained by a rotational water flow in the uppermost half of the fuel assembly.

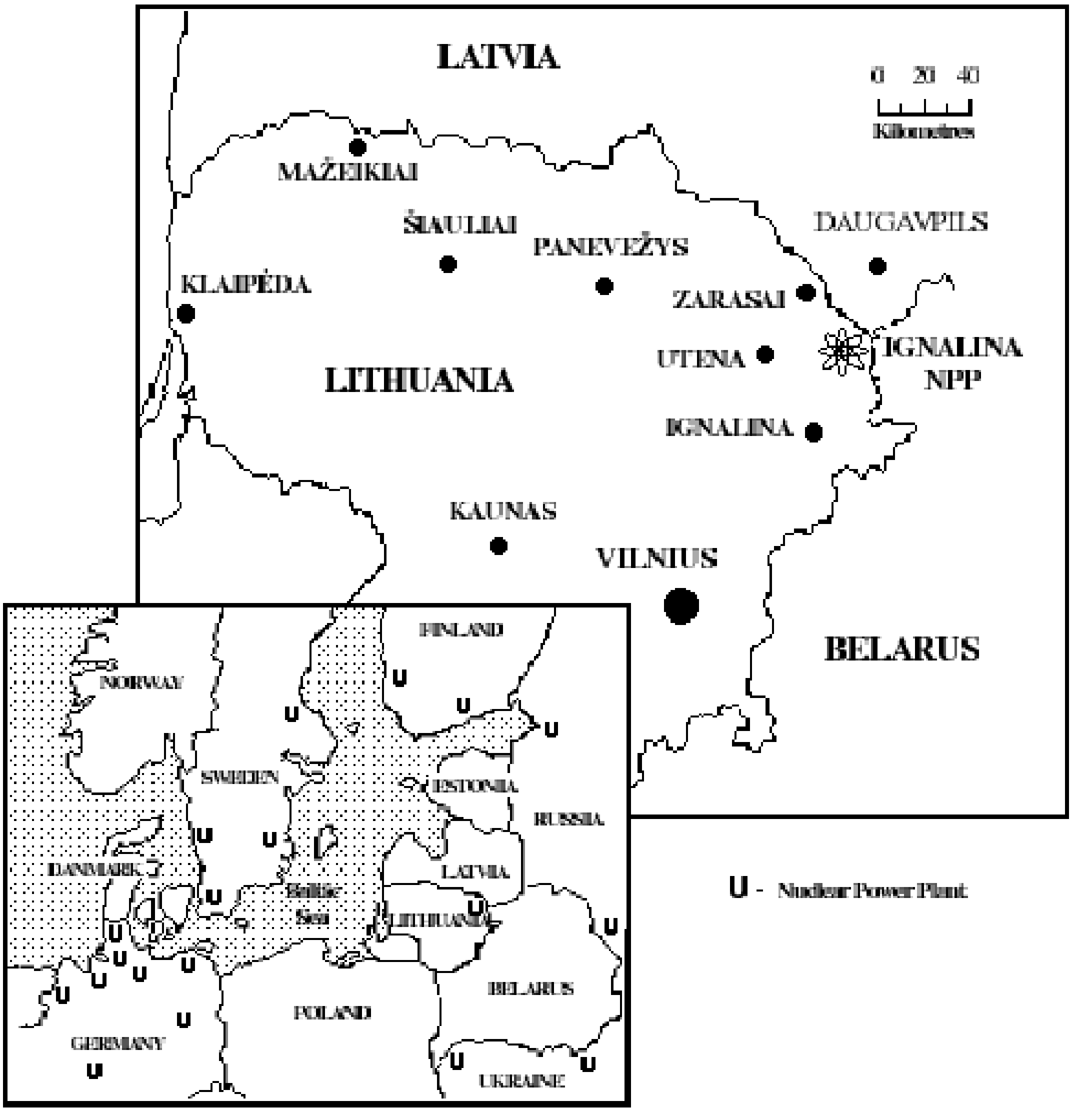


Fig. 2.10. Location of Ignalina NPP, near borders of Belorussia and Latvia (Source: www.lei.lt/insc/sourcebook/)

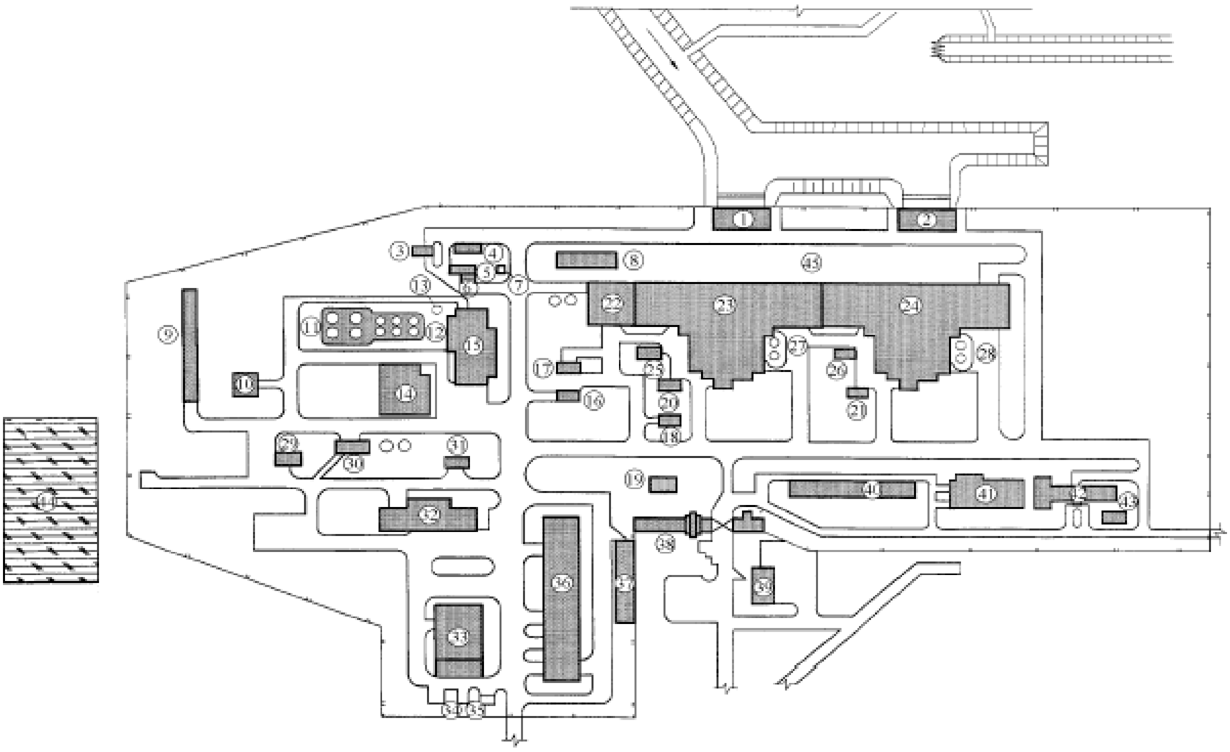


Fig. 2.11. *Ignalina NPP arrangement (*[Source](www.lei.lt/insc/sourcebook/)*: www.lei.lt/insc/sourcebook/)*

1,2 - service water pump stations
3 - acetylene bottle depot
4 - oil depot
5 - oil system equipment room
6- transformers equipment tower
7 - pump station for waste and liquid sewerage discharge
8 - hydrogen- and oxygenreceiving facility, low-activity waste storage
9 - low-level radwaste repository,
10 - medium- and high-activity waste storage
11 - operational shower- water reservoir
12 - drainage water tank
13 - venting stack of the radwaste reprocessing building
14 - bitumen storage
15 - liquid waste storage
16 - chemical water treatment building
17 - primary grade water tanks
18,19 - recreational facilities
20,21 - gas purification systems
22 - heat power station
23,24 - building plant units 1 and 2, respectively
25,26 - pressurised tank (accumulator) of the ECCS
27,28 – purified deminiralized water tanks
29 - car-washing facility
30 -bitumen depot
31 - special laundry
32 - chemical reagent depot
33 - equipment storehouse
34 - noble-gas reservoir depot
35 - reservoir facility with artificial evaporation
36 - repair building
37,38 - administrative buildings
39 – cafeteria
40 - diesel - generator building
41 - compressor and refrigeration station
42 - nitrogen and oxygen manufacture building
43 - liquid nitrogen reservoir
44 - 110/330 kV open distributive system.

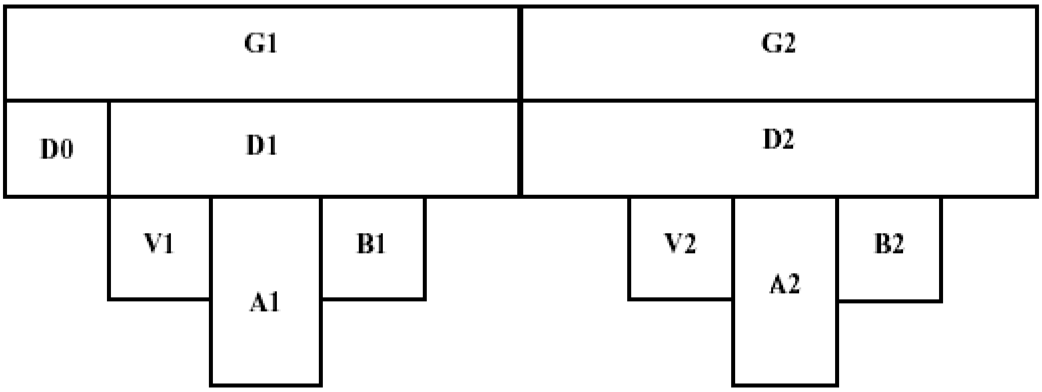


Fig. 2.12. General layout of Units 1 and 2 of Ignalina NPP (Source: www.lei.lt/insc/sourcebook/)

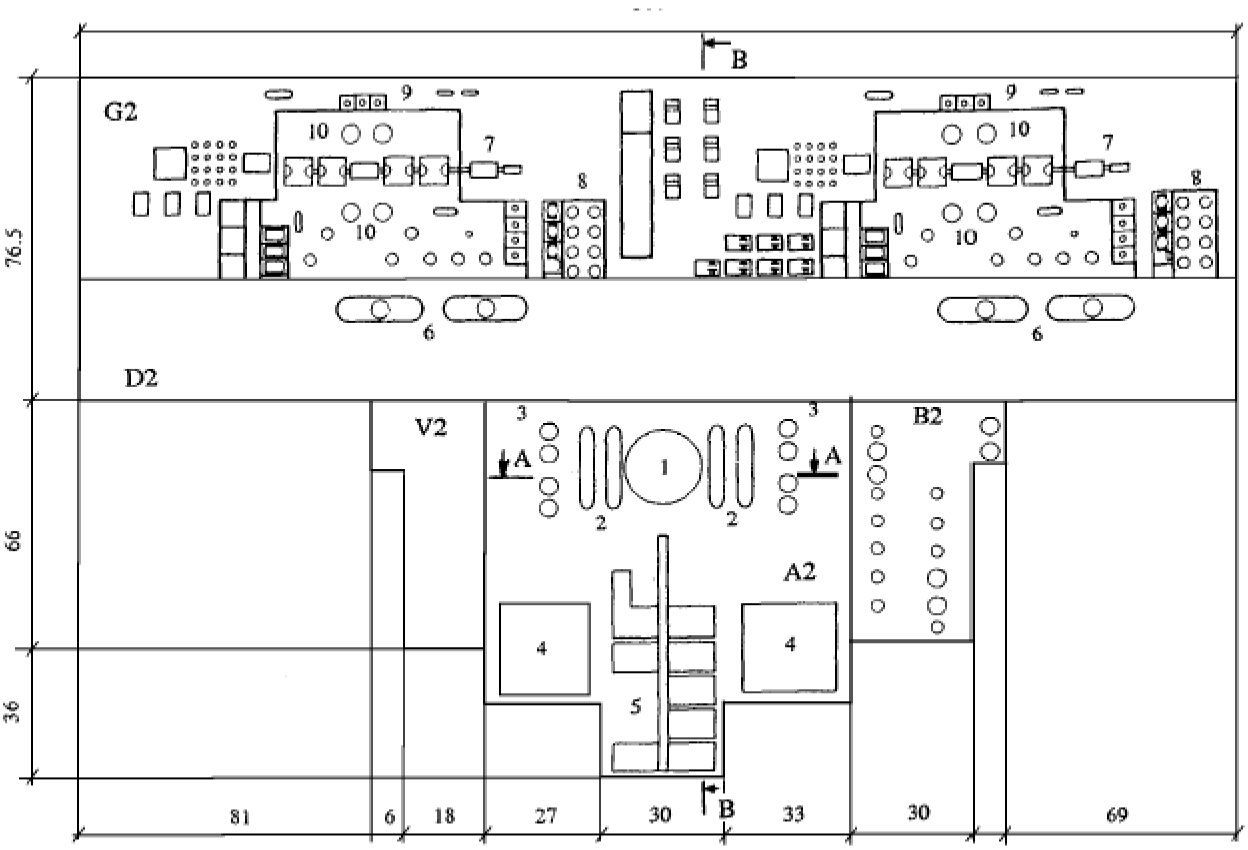


Fig. 2.13. Layout of Ignalina main buildings (Source: www.lei.lt/insc/sourcebook/)

1 - reactor
2 - pressure and suction headers
3 - main circulation pumps
4 - accident confinement system
5 - spent fuel compartment

6 - deaerators
7 - turbine generators
8 - condensate cleaning filters
9 - first stage condensate pumps
10 - separator - reheater

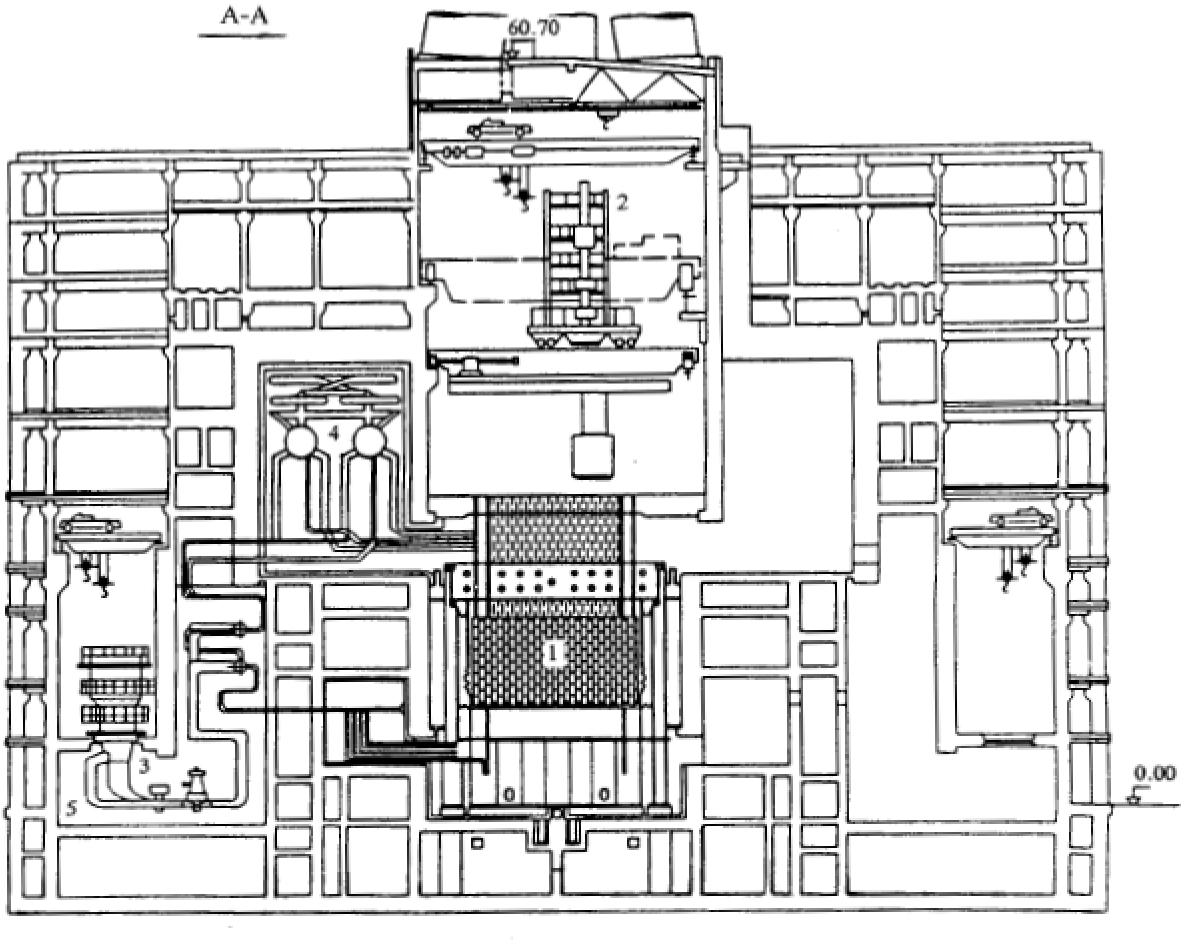


Fig. 2.14. Cross-section of one unit of Ignalina NPP (Source: www.lei.lt/insc/sourcebook/sob1.pdf)

2.3.2 Summary of approximate design data for unit 1 of the Ignalina NPP

Main data

Type	RBMK-1500
Thermal power, max	4800 MW (operating now at 83 % level)
Capacity, gross max	1500 MWe (operating now at 83 % level)
Capacity, net max	1440 MWe (operating now at 83 % level)

Reactor core data

Core diameter	11.8 m
Core height	7 m
Number of fuel channels	1661
Number of control rod channels	235
Reflector cooling channels	156
Square lattice pitch	0.25 m
Graphite mass in the core	1700 ton
Maximum graphite temperature	750 ^{0}C

Fuel data

Fuel material	UO_2
Fuel inventory	192 tU
Fresh fuel enrichment	2.0 wt%
Average linear heat rate	218 W/cm
Peak linear heat rate	485 W/cm
Rods per fuel element	18
Fuel pellet diameter	11.5 mm
Diameter of fuel rod	13.5 mm
Fuel elements per fuel assembly	18
Length of fuel element	3.4 m
Diameter of fuel element	79 mm
Channel outside diameter	88
Channel material	Zr/Nb
Average fuel burnup	21600 MWd/tU
Cladding material	Zr/1 % Nb
Cladding thickness	0.9 mm
Absorbing control rods, B_4C	211
Emergency rods, B_4C	24
Refuelling technique	On-load

Primary circuit data

Recirculation loops	2
Primary pumps	8
Steam drum separators	4
Pressure in steam generator	70 bar
Total coolant flow, max	13300 kg/s
Fuel channel inlet temperature	260 ^{0}C
Fuel channel outlet temperature	285 ^{0}C
Feed water flow rate	2400 kg/s
Feed water temperature	190 ^{0}C
Average steam content at core outlet	29 %

Turbine plant general

Turbines	2
Steam inlet temperature	280 ^{0}C
Speed	3000 rpm
Inlet turbine pressure	65 bar
Inlet temperature	280 ^{0}C
Steam flow	2445 kg/s
Moisture content inlet	0.5%
Number of high pressure cylinders	1
Number of low pressure cylinders	4
Turbine length	40 m

Generator general

Generator output	800 MW
Voltage	24 kV
Rotor cooling	hydrogen
Stator cooling	water

3 VVER NPPS

3.1 Features of VVER Reactors

3.1.1 General

As already earlier stated, Russian nuclear power plants are of two basic designs, generally known by the acronyms VVER and RBMK. The VVER is a pressurised-water reactor (PWR) with the acronym standing for water-cooled, watermoderated energy reactor. Kola is a VVER NPP.

There are (or have been) four (or five) generations of VVER reactors. The first two units of the Novovoronezh NPP, types VVER-210 and VVER-365, can be regarded as the prototypes of the VVER reactors (**0th** generation). The two units began operations in 1964 and 1970, and were closed in 1988 and 1990, respectively.

The **first** generation of VVER-440/230 model was developed in the 1960's. Currently there are 11 first generation VVER-440/230 units in operation. The two first units of the Kola NPP belong to this generation. Together with the RBMK's, this plant type causes the most concern about safety among Western experts. The design has many problems. It has no real containment building, the emergency core cooling capability as well as the redundancy and separation of safety equipment are rather limited, there are many deficiencies, especially in fire protection, and the reactor pressure vessels have problems with embrittlement. On the other hand, these reactors have some positive features from a safety point of view. Especially, the power densities in the reactor core are low, the safety margins are high, and the amounts of water in the primary and secondary sides are large, giving extra passive safety to the plant. The accident that occurred in the Greifswald Unit 1 (VVER-440/230) in December 1975, concretely demonstrated the vulnerability of the design to fires but also its passive safety due to its large coolant water volumes.

The **second** generation VVER-440/213 model was developed in the 1970's. There are 16 units of this model in operation. Two of these units are in Russia, namely, units 3 and 4 of the Kola NPP. The two Loviisa NPP units do not exactly belong to any of these generations, but they are perhaps closest to the VVER-440/213 model. The original Soviet design of the reactor units was adapted to western safety philosophy by several additional engineered safety features - including a containment building of the ice condenser type. Subsequently, as safety requirements developed in the international level,the safety systems were backfitted to meet the latest standards.

Several design deficiencies of the previous VVER-440/230's have been generally removed in this second generation. The containment has been upgraded, based on a system of suppression pools in a special bubbler-condenser tower. Yet, doubts exist on the operability of this containment design in accident situations. The emergency core cooling systems have been enhanced. The passive safety features are similar to the preceding generation. However, deficiencies remain in instrumentation, control, and fire protection.

The VVER-1000 (**third** generation) plants were designed during the years 1975–1985. There are 20 units in operation. Various future versions of the VVER-1000 power plant have been developed with acronyms such as VVER-91 and VVER-92. Overall, the operating pressures and temperatures as well as the safety concept of this model are similar to those of the western-designed pressurised-water reactor plants. Among the major improvements, the VVER-1000's have steel-lined concrete containment structures that conform with the western counterparts.

The VVER-640 (**fourth** generation) is a new design with improved safety. The primary circuit of this model is based on the VVER-1000 design with a lowered specific power and larger safety margins. New passive safety features have been added to the design. No VVER-640 units are in operation or under construction yet. Doubts have been presented on its economic viability, because the design and, consequently, the construction costs are similar to the VVER-1000 model, but the electrical power capacity has been reduced from 1000 MWe to 640 MWe.

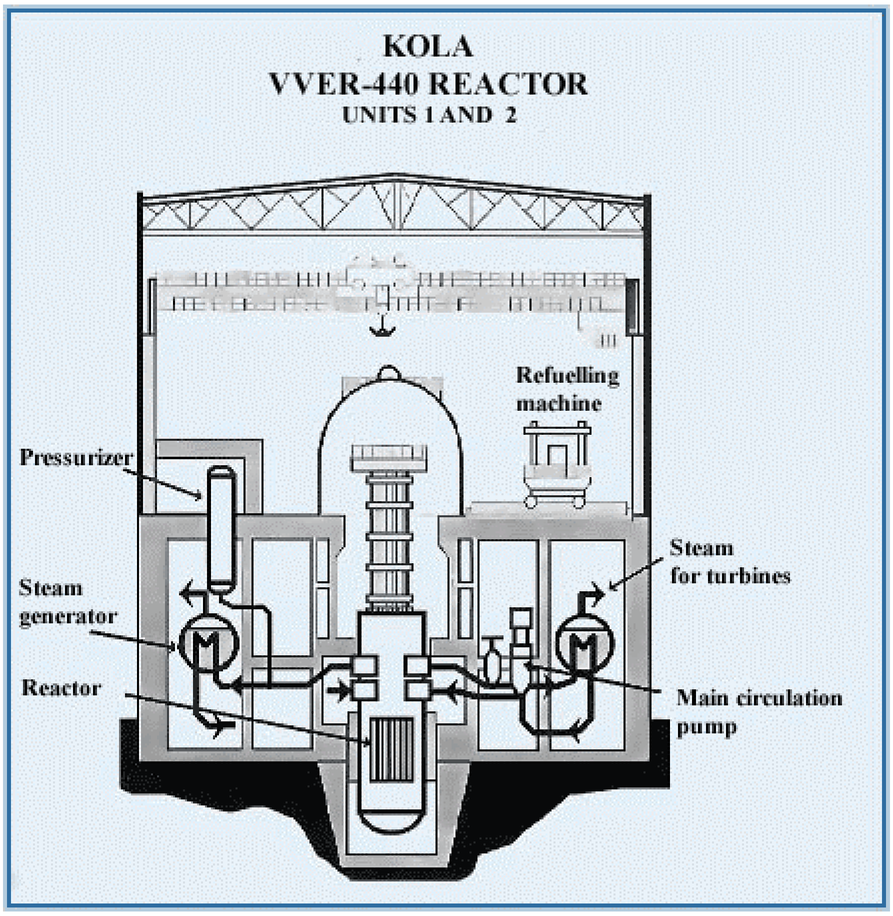


Fig. 3.1. The reactor building of a VVER-440/230 reactor (Kola NPP units 1 and 2). (Source: www.stuk.fi/english/npp)

3.1.2 Differences between Western PWRs and VVER-440 reactors

The VVER-440 are pressurized water reactors constructed from the same basic design principles as Western PWRs. Among the important safety design differences between the Kola reactors and the Western reactors the following items can be listed:

- Power density
- Water amount
- Number of loops
- Passive safety systems
- Number of safety systems
- Active safety systems
- Containment system
- Filter/scrubber system

The Kola plant has a low power density, which means a small probability for fuel failures.

The water inventory in the primary and secondary circuits of a VVER-440 is large compared to the core power and this has a positive effect on operating characteristics. Thermal transients in the core are effectively damped and natural circulation is sufficient to remove decay heat at shut-down from full power. In fact the natural circulation can be taken as a passive safety system.

The small gap between the fuel assemblies at the periphery of the core and the reactor vessel makes the vessel susceptible to radiation induced embrittlement by fast neutrons. This has been of very much concern for VVER reactors, where the gap is much smaller than is the case for Western PWRs.

Western plants are equipped with more safety systems than the Kola plant and *redundancy* and *diversity* have been applied to a greater extent. The safety systems are mostly relying on active components such as pumps and electrical valves.

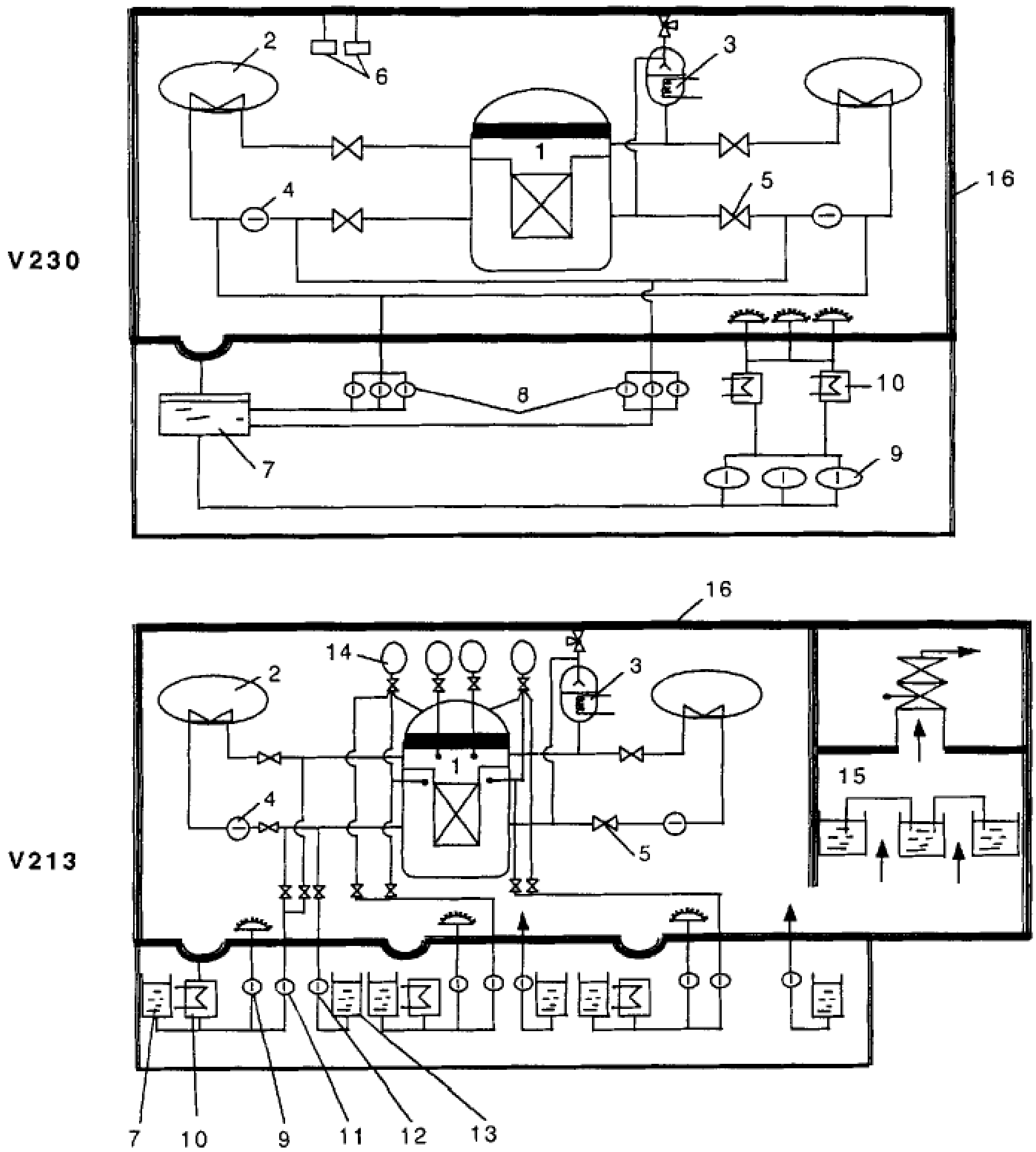
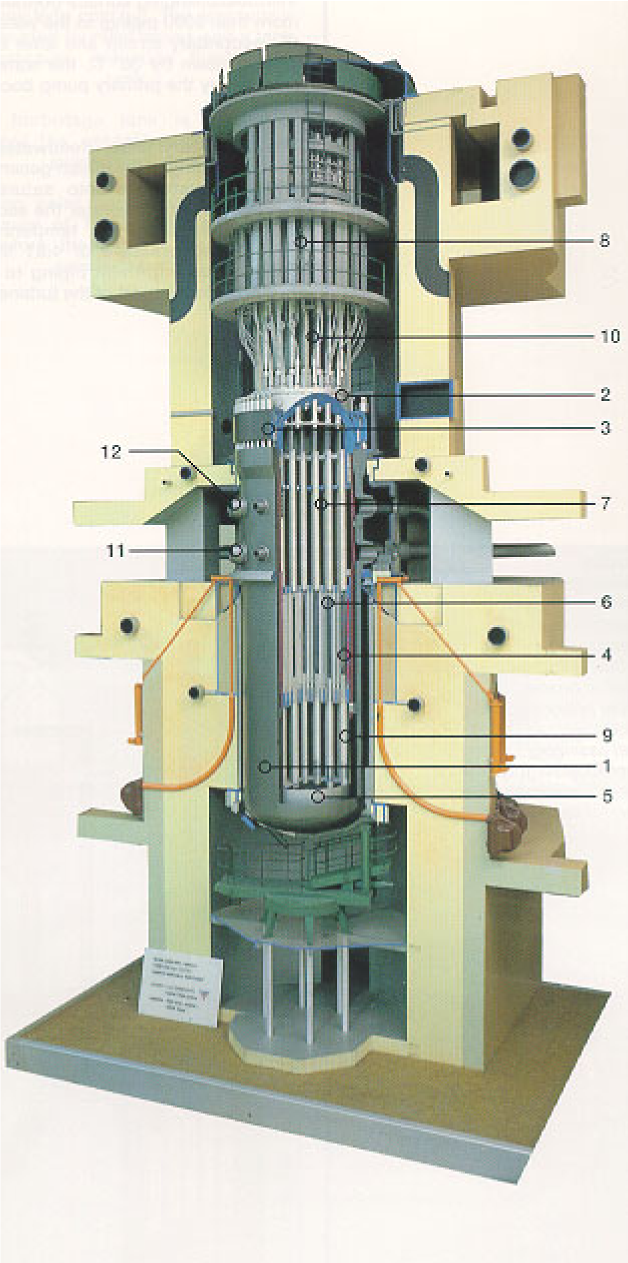


Fig. 3.2. Differences between VVER-440 reactor designs 230 and 213 (Source: Ref. 1)

1. Reactor
2. Steam generator
3. Pressurizer
4. Primary coolant pump
5. Shut-off valve
6. Pressure relief valve
7. Boric acid solution
8. High press. emergency pump
9. Sprinkler pump
10. Cooler
11. Low press. emergency pump
12. High press. emergency pump
13. Boric acid soultion tank
14. Hydraulic accumulator
15. Condenser bobler tower
16. Airtight compartment



1. Pressure vessel
2. Vessel cover
3. Vessel flange
4. Core barrel
5. Core barrel bottom
6. Reactor core
7. Protector tubes
8. Upper block
9. Protector tubes with buffers
10. Control assembly drives
11. Vessel inlet nozzle
12. Vessel outlet nozzle

Fig. 3.3. VVER 440/213 pressure vessel with internals
(Source: Information brochure of Dukovany NPP)

The main difference between the Kola and the Western PWRs is the lack of a proper containment function at the Kola plant. The two oldest units at Kola, type 230, have a leak tight concrete structure but it can only withstand an overpressure of about 0.8 bar before valves open to the atmosphere. Units 3 and 4, type 213, have an improved containment function, which can withstand an overpressure of about 1.5 bar due to the existence of a condenser bubbler tower. This is to be compared with a containment of Western design, which can withstand a pressure of about 5 bar and where venting is not directly to the atmosphere but often through stone filters or scrubbers with delay characteristics.

3.2 Kola NPP

3.2.1 Description

Fig. 3.4. Kola NPP (Source: www.insc.ru/main/Db/Kola/site/kol-site1.html)

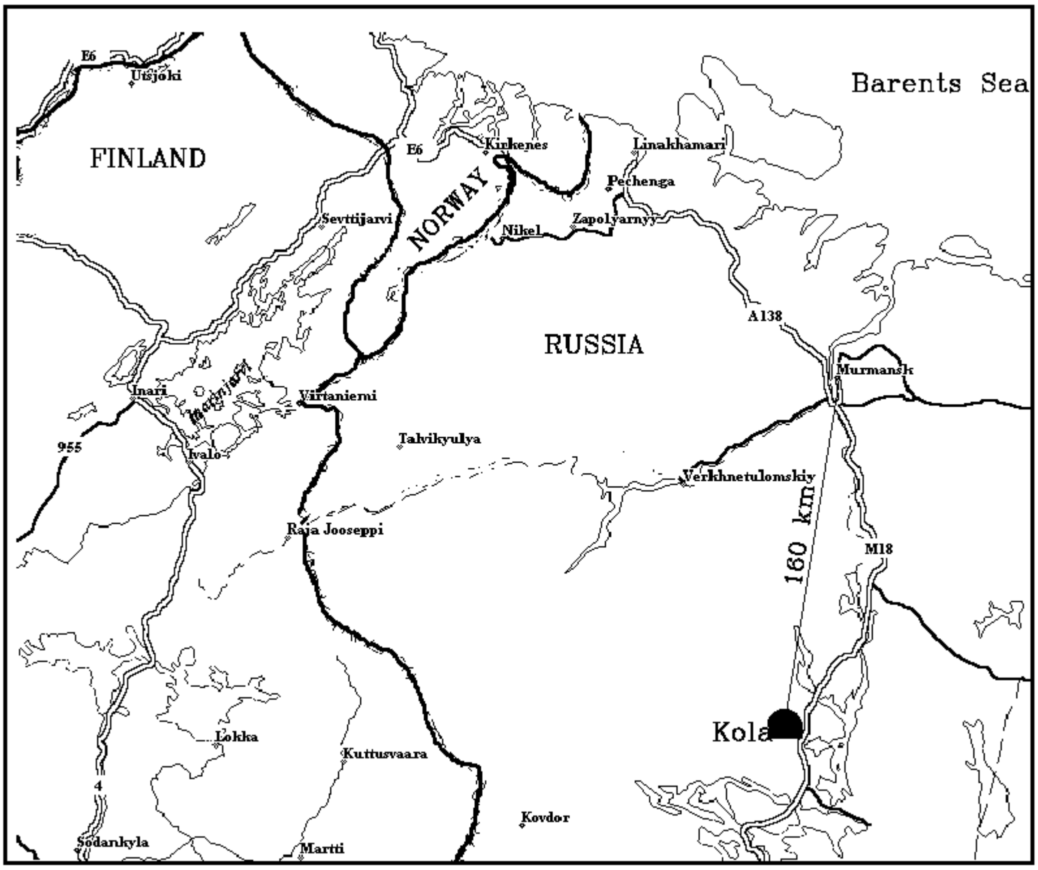


Fig. 3.5. Location of the Kola NPP, 160 km south of Murmansk. (Source: Ref. 1)

The Kola nuclear power plant is situated on the southern shore of Lake Imandra on the Kola peninsula in Russia (Fig. 3.5). As already mentioned above the plant has four VVER-440 units, Kola-1 and 2 of type 230 and Kola-3 and 4 of type 213. The two first units were commissioned in 1973 and 1974, and unit 3 and 4 in 1981 and 1984, respectively. The electrical power of each unit is 440 MW [9].

1. Main building
2. Ventilation tube
3. Auxiliary building
4. Transformers
5. 330 Kv electrical switchyard
6. 110 Kv electrical switchyard
7. Cooling water intake channel
8. Waste water offtake channel
9. Diesel fuel reservoirs
10. Diesel generator & compressor plant
11.
12. Changing rooms /General service building
13. Oil facility
14. Staff restaurant
15. Main entrance, reception
16. Main workshop
17. Security
18. Emergency power supply plant
19. Hydrogen recievers
20. Fire extinguisher charging post
21. Electrical switchyard workshop
22. Boiler house
24. Oxygen production
25. Waste water filtration plant
26. Oiled water filtration plant
27. Substation N70
28. Warehouses
29. Drinking water pumping

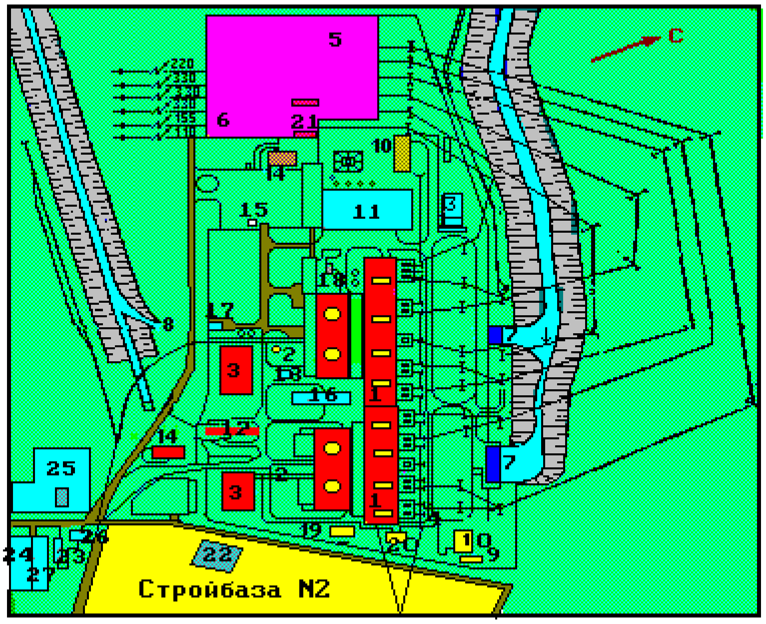


Fig. 3.6 Kola NPP site arrangement (Source: www.insc.ru/main/Db/Kola/site/kol-site8.html#eri-erfsp)

3.2.2 Summary of approximate design data for all Kola NPP units

Power

Thermal	1375 MW
Electrical	440 MW
Efficiency	31 %

Reactor plant

Coolant and moderator	H_2O
Fuel	UO_2
Cladding material	Zr 1% Nb
Number of fuel assemblies	313 (units 1,2), 349 (units 3,4)
Fuel configuration	Triangle
Number of fuel rods in a bundle	126
Fuel assembly active length	2420 mm
Average specific power	33 kW/kgU
Total fuel weight	37.4 tU
Number of control assemblies	37
Number of core screen assemblies	36
Number of reactor coolant loops	6
Operating pressure	123 bar
Average temperature	285 ^{0}C
Temperature difference	30 ^{0}C (268^{0}C -298^{0}C)
Reactor coolant flow rate, max	13 m^3/s

Reactor pressure vessel

Inside diameter	3560 mm
Maximum overall height	11800 mm
Wall thickness	-
Material	-
Design pressure	125 bar
Total weight	200 t

Steam generators

Numbers	6
Steam output per unit	125 kg/s
Steam temperature	255 ^{0}C
Steam pressure	46 bar
Feedwater temperature	225 ^{0}C
Overall length	12000 mm
Heat transfer area	2500 m^2
Diameter	3200 mm
Material	-
Tube material	-
Weight	145 t

Reactor coolant pumps

Numbers	6
Capacity	1.8-2.0 m^3/s
Discharge head	55 m (units 1,2), 40 m (units 3,4)

Power	2000 kW (units 1,2), 1400 kW (units 3,4)
Manufacturer	-

Pressurizer

Design pressure	125 bar
Design temperature	325 ^{0}C
Diameter inner	2400 mm
Wall thickness	-
Total height	-
Weight empty	-
Volume free	38 m^3
Water volume	-
Steam volume	-
Operating temperature	325 ^{0}C
Operating pressure	125 bar
Number of heaters	-
Total power	1620 kW

Secondary side

Main steam flow rate	750 kg/s
Main steam pressure	44 bar
Main steam temperature	255 ^{0}C
Condenser vacuum	0.03 bar
Feedwater temperature	220 ^{0}C

Turbine

No of turbines	2
Turbine speed	3000 rpm

Generator

Voltage	15 kV
Rated output	220 MVA

Emergency power supply

Diesel units	2
Output	-
Voltage	6 kV

4 ACTIVITY INVENTORIES OF KOLA, LENINGRAD AND IGNALINA NPPS

The total activity inventories of these nuclear power plants are listed in the tables 4.1 to 4.3. As a rough guidance the radionuclides can be categorised according to their chemical form and their volatilisation in increasing temperatures:

- Noble gases (Xe, Kr)
- Iodines (I)
- Caesium - Rubidium (Cs, Rb)
- Tellurium - Antimony (Te, Sb)
- Barium - Strontium (Ba, Sr)
- Ruthenium-type elements (Ru, Mo, Rh, Tc, Co)
- Lanthanide type elements and actinides (La, Nd, Y, Pr, Nb, Am, Cm, Pu, Np, Zr)

In a severe accident the release fraction of noble gases is in most cases rather large (close to 100%). The release fractions of the other elements depend strongly on the plant features and the severity of the accident sequence. By effective severe accident management systems existing in the Finnish and Swedish nuclear power plants the releases can be reduced to a level where early health effects can be avoided even in the close vicinity of the plant. For the reactors discussed in this report the release could be considerably larger.

Table 4.1. Nuclidewise total activity inventory (GBq) of the irradiated VVER-440 reactor core ***(Kola NPP)*** *(total amount of fuel 37.4 tU; enrichment of 4%;. four batch loading. specific power of 33 MW/tU; average core burnup of 30 000 MWd/tU)*

	Cooling time(h)				Cooling time(h)		
Nuclide	0	1	24	Nuclide	0	1	24
H 3	$6.2\cdot10^{5}$	$6.2\cdot10^{5}$	$6.2\cdot10^{5}$	I 134	$2.5\cdot10^{9}$	$1.8\cdot10^{9}$	$5.8\cdot10^{1}$
C 14	$1.5\cdot10^{3}$	$1.5\cdot10^{3}$	$1.5\cdot10^{3}$	CS 134	$3.6\cdot10^{8}$	$3.6\cdot10^{8}$	$3.6\cdot10^{8}$
KR 83m	$1.5\cdot10^{8}$	$1.4\cdot10^{8}$	6.4·105	I 135	$2.1\cdot10^{9}$	$1.9\cdot10^{9}$	$1.7\cdot10^{8}$
KR 85	$1.3\cdot10^{7}$	$1.3\cdot10^{7}$	$1.3\cdot10^{7}$	XE 135	$2.8\cdot10^{8}$	$4.1\cdot10^{8}$	$5.0\cdot10^{8}$
KR 85M	3.1·108	$2.7\cdot10^{8}$	7.7·106	CS 136	$4.6\cdot10^{8}$	$4.6\cdot10^{8}$	$4.4\cdot10^{8}$
KR 87	$6.0\cdot10^{8}$	$3.5\cdot10^{8}$	$1.3\cdot10^{3}$	CS 137	$1.3\cdot10^{8}$	$1.3\cdot10^{8}$	$1.3\cdot10^{8}$
KR 88	$8.5\cdot10^{8}$	$6.6\cdot10^{8}$	$2.4\cdot10^{6}$	BA 137m	$1.2\cdot10^{8}$	$1.2\cdot10^{8}$	$1.2\cdot10^{8}$
KR 89	$1.0\cdot10^{9}$	$2.1\cdot10^{3}$		BA 140	$2.0\cdot10^{9}$	$2.0\cdot10^{9}$	$1.9\cdot10^{9}$
SR 89	$1.3\cdot10^{9}$	$1.3\cdot10^{9}$	$1.2\cdot10^{9}$	LA 140	$2.4\cdot10^{9}$	$2.4\cdot10^{9}$	$2.3\cdot10^{9}$
SR 90	$1.1\cdot10^{8}$	$1.1\cdot10^{8}$	$1.1\cdot10^{8}$	CE 141	$2.0\cdot10^{9}$	$2.0\cdot10^{9}$	$2.0\cdot10^{9}$
Y 90	$1.4\cdot10^{8}$	$1.4\cdot10^{8}$	$1.3\cdot10^{8}$	CE 143	$1.8\cdot10^{9}$	$1.8\cdot10^{9}$	$1.1\cdot10^{9}$
SR 91	$1.4\cdot10^{9}$	$1.3\cdot10^{9}$	$2.5\cdot10^{8}$	PR 143	$1.8\cdot10^{9}$	$1.8\cdot10^{9}$	$1.8\cdot10^{9}$
Y 91	$1.6\cdot10^{9}$	$1.6\cdot10^{9}$	$1.6\cdot10^{9}$	CE 144	$1.3\cdot10^{9}$	$1.3\cdot10^{9}$	$1.3\cdot10^{9}$
ZR 95	$2.1\cdot10^{9}$	$2.1\cdot10^{9}$	$2.0\cdot10^{9}$	PR 144	$1.4\cdot10^{9}$	$1.3\cdot10^{9}$	$1.3\cdot10^{9}$
MO 99	$2.5\cdot10^{9}$	$2.5\cdot10^{9}$	$2.0\cdot10^{9}$	PR 144m	$1.6\cdot10^{7}$	$1.6\cdot10^{7}$	$1.6\cdot10^{7}$
TC 99m	$2.2\cdot10^{9}$	$2.2\cdot10^{9}$	$1.9\cdot10^{9}$	ND 147	$7.8\cdot10^{8}$	$7.8\cdot10^{8}$	$7.4\cdot10^{8}$
RU 103	$1.7\cdot10^{9}$	$1.7\cdot10^{9}$	$1.7\cdot10^{9}$	PM 147	$7.3\cdot10^{7}$	$7.3\cdot10^{7}$	$7.3\cdot10^{7}$
RU 105	$1.1\cdot10^{9}$	$9.6\cdot10^{8}$	$2.7\cdot10^{7}$	PM 148	$2.4\cdot10^{8}$	$2.4\cdot10^{8}$	$2.1\cdot10^{8}$
RH 105	$7.0\cdot10^{8}$	$7.1\cdot10^{8}$	$5.4\cdot10^{8}$	PM 148m	$1.9\cdot10^{7}$	$1.9\cdot10^{7}$	$1.9\cdot10^{7}$
RU 106	$3.9\cdot10^{8}$	$3.9\cdot10^{8}$	$3.9\cdot10^{8}$	PM 149	$9.5\cdot10^{8}$	$9.4\cdot10^{8}$	$7.0\cdot10^{8}$
PD 109	$6.2\cdot10^{8}$	$5.9\cdot10^{8}$	$1.8\cdot10^{8}$	PM 151	$2.3\cdot10^{8}$	$2.3\cdot10^{8}$	$1.3\cdot10^{8}$
AG 109m	$6.2\cdot10^{8}$	$5.9\cdot10^{8}$	$1.8\cdot10^{8}$	SM 153	$1.3\cdot10^{9}$	$1.3\cdot10^{9}$	$9.0\cdot10^{8}$
AG 110	$5.3\cdot10^{8}$	$1.1\cdot10^{5}$	$1.1\cdot10^{5}$	EU 156	$9.1\cdot10^{8}$	$9.0\cdot10^{8}$	$8.7\cdot10^{8}$
AG 110m	$8.4\cdot10^{6}$	$8.4\cdot10^{6}$	$8.4\cdot10^{6}$	U 237	$3.9\cdot10^{9}$	$3.9\cdot10^{9}$	$3.5\cdot10^{9}$
SB 124	$6.4\cdot10^{6}$	$6.4\cdot10^{6}$	$6.3\cdot10^{6}$	U 239	$1.3\cdot10^{10}$	$2.2\cdot10^{9}$	
TE 129	$3.2\cdot10^{8}$	$3.1\cdot10^{8}$	$4.0\cdot10^{7}$	NP 238	$3.1\cdot10^{9}$	$3.0\cdot10^{9}$	$2.2\cdot10^{9}$
TE 129m	$5.0\cdot10^{7}$	$5.0\cdot10^{7}$	$4.9\cdot10^{7}$	NP 239	$1.3\cdot10^{+10}$	$1.2\cdot10^{+10}$	$9.4\cdot10^{9}$
TE 131	$9.5\cdot10^{8}$	$4.6\cdot10^{8}$	$1.9\cdot10^{7}$	PU 238	$8.5\cdot10^{6}$	$8.5\cdot10^{6}$	$8.6\cdot10^{6}$
TE 131m	$1.5\cdot10^{8}$	$1.5\cdot10^{8}$	$8.6\cdot10^{7}$	PU 239	$6.5\cdot10^{4}$	$6.5\cdot10^{4}$	$6.6\cdot10^{4}$
I 131	$1.1\cdot10^{9}$	$1.1\cdot10^{9}$	$9.9\cdot10^{8}$	PU 241	$2.4\cdot10^{7}$	$2.4\cdot10^{7}$	$2.4\cdot10^{7}$
XE 131m	$1.3\cdot10^{7}$	$1.3\cdot10^{7}$	$1.3\cdot10^{7}$	PU 243	$9.0\cdot10^{8}$	$7.8\cdot10^{8}$	$3.1\cdot10^{7}$
TE 132	$1.5\cdot10^{9}$	$1.5\cdot10^{9}$	$1.2\cdot10^{9}$	AM 242	$1.5\cdot10^{7}$	$1.4\cdot10^{7}$	$5.2\cdot10^{6}$
I 132	$1.6\cdot10^{9}$	$1.5\cdot10^{9}$	$1.3\cdot10^{9}$	AM 244	$3.8\cdot10^{7}$	$3.6\cdot10^{7}$	$7.4\cdot10^{6}$
I 133	$2.2\cdot10^{9}$	$2.2\cdot10^{9}$	$1.0\cdot10^{9}$	CM 242	$8.1\cdot10^{6}$	$8.1\cdot10^{6}$	$8.1\cdot10^{6}$
XE 133	$2.2\cdot10^{9}$	$2.2\cdot10^{9}$	$2.2\cdot10^{9}$	CM 244	$1.5\cdot10^{7}$	$1.5\cdot10^{7}$	$1.5\cdot10^{7}$
XE 133m	$7.0\cdot10^{7}$	$7.0\cdot10^{7}$	$6.3\cdot10^{7}$	TOTAL	$2.4\cdot10^{+11}$	$1.0\cdot10^{+11}$	$5.9\cdot10^{+10}$

*Table 4.2. Nuclidewise total activity inventory (GBq) of the irradiated RBMK reactor core (**Leningrad NPP at Sosnovyi Bor**) (total amount of fuel about 90 tU; enrichment of 2.4%; discharge burnup of 12 500 MWd/tU; specific irradiation power density of 16.7 MW/tU).*

	Cooling time(h)				Cooling time(h)		
Nuclide	0	1	24	Nuclide	0	1	24
H 3	$6.4 \cdot 10^{5}$	$6.4 \cdot 10^{5}$	$6.4 \cdot 10^{5}$	I 134	$3.3 \cdot 10^{9}$	$2.4 \cdot 10^{9}$	$7.7 \cdot 10^{1}$
C 14	$1.4 \cdot 10^{3}$	$1.4 \cdot 10^{3}$	$1.4 \cdot 10^{3}$	CS 134	$7.1 \cdot 10^{7}$	$7.1 \cdot 10^{7}$	$7.1 \cdot 10^{7}$
KR 83m	$2.0 \cdot 10^{8}$	$1.9 \cdot 10^{8}$	$9.0 \cdot 10^{5}$	I 135	$2.8 \cdot 10^{9}$	$2.5 \cdot 10^{9}$	$2.3 \cdot 10^{8}$
KR 85	$1.4 \cdot 10^{7}$	$1.4 \cdot 10^{7}$	$1.4 \cdot 10^{7}$	XE 135	$8.2 \cdot 10^{8}$	$9.6 \cdot 10^{8}$	$7.3 \cdot 10^{8}$
KR 85M	$4.4 \cdot 10^{8}$	$3.8 \cdot 10^{8}$	$1.1 \cdot 10^{7}$	CS 136	$3.8 \cdot 10^{7}$	$3.8 \cdot 10^{7}$	$3.6 \cdot 10^{7}$
KR 87	$8.7 \cdot 10^{8}$	$5.1 \cdot 10^{8}$	$1.8 \cdot 10^{3}$	CS 137	$1.3 \cdot 10^{8}$	$1.3 \cdot 10^{8}$	$1.3 \cdot 10^{8}$
KR 88	$1.2 \cdot 10^{9}$	$9.6 \cdot 10^{8}$		BA 137m	$1.3 \cdot 10^{8}$	$1.2 \cdot 10^{8}$	$1.2 \cdot 10^{8}$
KR 89	$1.5 \cdot 10^{9}$	$3.1 \cdot 10^{3}$	$0.0 \cdot 10^{0}$	BA 140	$2.6 \cdot 10^{9}$	$2.6 \cdot 10^{9}$	$2.5 \cdot 10^{9}$
SR 89	$1.7 \cdot 10^{9}$	$1.7 \cdot 10^{9}$	$1.7 \cdot 10^{9}$	LA 140	$2.8 \cdot 10^{9}$	$2.7 \cdot 10^{9}$	$2.7 \cdot 10^{9}$
SR 90	$1.1 \cdot 10^{8}$	$1.1 \cdot 10^{8}$	$1.1 \cdot 10^{8}$	CE 141	$2.5 \cdot 10^{9}$	$2.5 \cdot 10^{9}$	$2.5 \cdot 10^{9}$
Y 90	$1.2 \cdot 10^{8}$	$1.2 \cdot 10^{8}$	$1.1 \cdot 10^{8}$	CE 143	$2.4 \cdot 10^{9}$	$2.4 \cdot 10^{9}$	$1.5 \cdot 10^{9}$
SR 91	$2.0 \cdot 10^{9}$	$1.9 \cdot 10^{9}$	$3.5 \cdot 10^{8}$	PR 143	$2.4 \cdot 10^{9}$	$2.4 \cdot 10^{9}$	$2.4 \cdot 10^{9}$
Y 91	$2.1 \cdot 10^{9}$	$2.1 \cdot 10^{9}$	$2.1 \cdot 10^{9}$	CE 144	$1.9 \cdot 10^{9}$	$1.9 \cdot 10^{9}$	$1.9 \cdot 10^{9}$
ZR 95	$2.7 \cdot 10^{9}$	$2.7 \cdot 10^{9}$	$2.6 \cdot 10^{9}$	PR 144	$1.9 \cdot 10^{9}$	$1.9 \cdot 10^{9}$	$1.9 \cdot 10^{9}$
MO 99	$2.7 \cdot 10^{9}$	$2.7 \cdot 10^{9}$	$2.1 \cdot 10^{9}$	PR 144m	$2.3 \cdot 10^{7}$	$2.3 \cdot 10^{7}$	$2.3 \cdot 10^{7}$
TC 99m	$2.4 \cdot 10^{9}$	$2.3 \cdot 10^{9}$	$2.0 \cdot 10^{9}$	ND 147	$9.9 \cdot 10^{8}$	$9.8 \cdot 10^{8}$	$9.3 \cdot 10^{8}$
RU 103	$2.0 \cdot 10^{9}$	$2.0 \cdot 10^{9}$	$2.0 \cdot 10^{9}$	PM 147	$3.4 \cdot 10^{8}$	$3.4 \cdot 10^{8}$	$3.4 \cdot 10^{8}$
RU 105	$1.2 \cdot 10^{9}$	$1.1 \cdot 10^{9}$	$2.9 \cdot 10^{7}$	PM 148	$1.6 \cdot 10^{8}$	$1.6 \cdot 10^{8}$	$1.4 \cdot 10^{8}$
RH 105	$1.1 \cdot 10^{9}$	$1.1 \cdot 10^{9}$	$8.2 \cdot 10^{8}$	PM 148m	$4.2 \cdot 10^{7}$	$4.2 \cdot 10^{7}$	$4.1 \cdot 10^{7}$
RU 106	$5.0 \cdot 10^{8}$	$5.0 \cdot 10^{8}$	$5.0 \cdot 10^{8}$	PM 149	$6.5 \cdot 10^{8}$	$6.5 \cdot 10^{8}$	$4.9 \cdot 10^{8}$
PD 109	$2.8 \cdot 10^{8}$	$2.7 \cdot 10^{8}$	$8.3 \cdot 10^{7}$	PM 151	$2.5 \cdot 10^{8}$	$2.5 \cdot 10^{8}$	$1.4 \cdot 10^{8}$
AG 109m	$2.8 \cdot 10^{8}$	$2.7 \cdot 10^{8}$	$8.3 \cdot 10^{7}$	SM 153	$3.4 \cdot 10^{8}$	$3.4 \cdot 10^{8}$	$2.4 \cdot 10^{8}$
AG 110	$4.9 \cdot 10^{7}$	$1.2 \cdot 10^{4}$	$1.2 \cdot 10^{4}$	EU 156	$1.1 \cdot 10^{8}$	$1.1 \cdot 10^{8}$	$1.0 \cdot 10^{8}$
AG 110m	$9.2 \cdot 10^{5}$	$9.2 \cdot 10^{5}$	$9.2 \cdot 10^{5}$	U 237	$4.3 \cdot 10^{8}$	$4.3 \cdot 10^{8}$	$3.9 \cdot 10^{8}$
SB 124	$5.1 \cdot 10^{5}$	$5.1 \cdot 10^{5}$	$5.0 \cdot 10^{5}$	U 239	$2.4 \cdot 10^{10}$	$4.1 \cdot 10^{9}$	$9.1 \cdot 10^{-9}$
TE 129	$4.1 \cdot 10^{8}$	$4.0 \cdot 10^{8}$	$5.1 \cdot 10^{7}$	NP 238	$8.0 \cdot 10^{7}$	$7.9 \cdot 10^{7}$	$5.8 \cdot 10^{7}$
TE 129m	$6.3 \cdot 10^{7}$	$6.2 \cdot 10^{7}$	$6.2 \cdot 10^{7}$	NP 239	$2.4 \cdot 10^{10}$	$2.4 \cdot 10^{10}$	$1.8 \cdot 10^{10}$
TE 131	$1.3 \cdot 10^{9}$	$6.0 \cdot 10^{8}$	$2.6 \cdot 10^{7}$	PU 238	$4.7 \cdot 10^{5}$	$4.7 \cdot 10^{5}$	$4.7 \cdot 10^{5}$
TE 131m	$2.0 \cdot 10^{8}$	$2.0 \cdot 10^{8}$	$1.2 \cdot 10^{8}$	PU 239	$5.1 \cdot 10^{5}$	$5.1 \cdot 10^{5}$	$5.2 \cdot 10^{5}$
I 131	$1.4 \cdot 10^{9}$	$1.4 \cdot 10^{9}$	$1.3 \cdot 10^{9}$	PU 241	$1.0 \cdot 10^{8}$	$1.0 \cdot 10^{8}$	$1.0 \cdot 10^{8}$
XE 131m	$1.6 \cdot 10^{7}$	$1.6 \cdot 10^{7}$	$1.6 \cdot 10^{7}$	PU 243	$4.4 \cdot 10^{7}$	$3.8 \cdot 10^{7}$	$1.5 \cdot 10^{6}$
TE 132	$2.0 \cdot 10^{9}$	$2.0 \cdot 10^{9}$	$1.6 \cdot 10^{9}$	AM 242	$3.0 \cdot 10^{7}$	$2.8 \cdot 10^{7}$	$1.0 \cdot 10^{7}$
I 132	$2.1 \cdot 10^{9}$	$2.0 \cdot 10^{9}$	$1.7 \cdot 10^{9}$	AM 244	$2.0 \cdot 10^{5}$	$1.9 \cdot 10^{5}$	$3.9 \cdot 10^{4}$
I 133	$3.0 \cdot 10^{9}$	$3.0 \cdot 10^{9}$	$1.4 \cdot 10^{9}$	CM 242	$1.2 \cdot 10^{7}$	$1.2 \cdot 10^{7}$	$1.2 \cdot 10^{7}$
XE 133	$3.0 \cdot 10^{9}$	$3.0 \cdot 10^{9}$	$2.9 \cdot 10^{9}$	CM 244	$6.0 \cdot 10^{4}$	$6.0 \cdot 10^{4}$	$6.0 \cdot 10^{4}$
XE 133m	$9.3 \cdot 10^{7}$	$9.3 \cdot 10^{7}$	$8.4 \cdot 10^{7}$	TOTAL	$3.0 \cdot 10^{11}$	$1.2 \cdot 10^{11}$	$7.1 \cdot 10^{10}$

*Table 4.3. Nuclidewise total activities (GBq) of the irradiated RBMK reactor core (**Ignalina NPP**) (total core weight about 90 tU; enrichment of 2.4%; discharge burnup of 12 500 MWd/tU; specific irradiation power density 26.0 MW/tU).*

	Cooling time(h)				Cooling time(h)		
Nuclide	0	1	24	Nuclide	0	1	24
H 3	$6,5\cdot10^{5}$	$6,5\cdot10^{5}$	$6,5\cdot10^{5}$	I 134	$5,1\cdot10^{9}$	$3,7\cdot10^{9}$	$1,2\cdot10^{2}$
C 14	$1,4\cdot10^{3}$	$1,4\cdot10^{3}$	$1,4\cdot10^{3}$	CS 134	$7,6\cdot10^{7}$	$7,6\cdot10^{7}$	$7,6\cdot10^{7}$
KR 83m	$3,1\cdot10^{8}$	$3,0\cdot10^{8}$	$1,4\cdot10^{6}$	I 135	$4,3\cdot10^{9}$	$3,9\cdot10^{9}$	$3,5\cdot10^{8}$
KR 85	$1,4\cdot10^{7}$	$1,4\cdot10^{7}$	$1,4\cdot10^{7}$	XE 135	$9,1\cdot10^{8}$	$1,2\cdot10^{9}$	$1,1\cdot10^{9}$
KR 85m	$6,9\cdot10^{8}$	$6,0\cdot10^{8}$	$1,7\cdot10^{7}$	CS 136	$4,8\cdot10^{7}$	$4,8\cdot10^{7}$	$4,6\cdot10^{7}$
KR 87	$1,3\cdot10^{9}$	$7,9\cdot10^{8}$	$2,8\cdot10^{3}$	CS 137	$1,3\cdot10^{8}$	$1,3\cdot10^{8}$	$1,3\cdot10^{8}$
KR 88	$1,9\cdot10^{9}$	$1,5\cdot10^{9}$	$5,4\cdot10^{6}$	BA 137m	$1,3\cdot10^{8}$	$1,3\cdot10^{8}$	$1,3\cdot10^{8}$
KR 89	$2,4\cdot10^{9}$	$4,8\cdot10^{3}$		BA 140	$4,1\cdot10^{9}$	$4,1\cdot10^{9}$	$3,9\cdot10^{9}$
SR 89	$2,7\cdot10^{9}$	$2,7\cdot10^{9}$	$2,6\cdot10^{9}$	LA 140	$4,3\cdot10^{9}$	$4,3\cdot10^{9}$	$4,2\cdot10^{9}$
SR 90	$1,1\cdot10^{8}$	$1,1\cdot10^{8}$	$1,1\cdot10^{8}$	CE 141	$4,0\cdot10^{9}$	$4,0\cdot10^{9}$	$3,9\cdot10^{9}$
Y 90	$1,2\cdot10^{8}$	$1,2\cdot10^{8}$	$1,2\cdot10^{8}$	CE 143	$3,7\cdot10^{9}$	$3,7\cdot10^{9}$	$2,3\cdot10^{9}$
SR 91	$3,2\cdot10^{9}$	$2,9\cdot10^{9}$	$5,5\cdot10^{8}$	PR 143	$3,7\cdot10^{9}$	$3,7\cdot10^{9}$	$3,6\cdot10^{9}$
Y 91	$3,4\cdot10^{9}$	$3,4\cdot10^{9}$	$3,3\cdot10^{9}$	CE 144	$2,5\cdot10^{9}$	$2,5\cdot10^{9}$	$2,5\cdot10^{9}$
ZR 95	$4,2\cdot10^{9}$	$4,2\cdot10^{9}$	$4,1\cdot10^{9}$	PR 144	$2,5\cdot10^{9}$	$2,5\cdot10^{9}$	$2,5\cdot10^{9}$
MO 99	$4,2\cdot10^{9}$	$4,1\cdot10^{9}$	$3,2\cdot10^{9}$	PR 144m	$3,0\cdot10^{7}$	$3,0\cdot10^{7}$	$3,0\cdot10^{7}$
TC 99m	$3,6\cdot10^{9}$	$3,6\cdot10^{9}$	$3,1\cdot10^{9}$	ND 147	$1,5\cdot10^{9}$	$1,5\cdot10^{9}$	$1,4\cdot10^{9}$
RU 103	$3,1\cdot10^{9}$	$3,1\cdot10^{9}$	$3,1\cdot10^{9}$	PM 147	$3,7\cdot10^{8}$	$3,7\cdot10^{8}$	$3,7\cdot10^{8}$
RU 105	$1,9\cdot10^{9}$	$1,6\cdot10^{9}$	$4,5\cdot10^{7}$	PM 148	$2,6\cdot10^{8}$	$2,6\cdot10^{8}$	$2,3\cdot10^{8}$
RH 105	$1,7\cdot10^{9}$	$1,7\cdot10^{9}$	$1,2\cdot10^{9}$	PM 148m	$4,9\cdot10^{7}$	$4,9\cdot10^{7}$	$4,9\cdot10^{7}$
RU 106	$5,9\cdot10^{8}$	$5,9\cdot10^{8}$	$5,9\cdot10^{8}$	PM 149	$1,0\cdot10^{9}$	$1,0\cdot10^{9}$	$7,8\cdot10^{8}$
PD 109	$4,4\cdot10^{8}$	$4,2\cdot10^{8}$	$1,3\cdot10^{8}$	PM 151	$3,9\cdot10^{8}$	$3,8\cdot10^{8}$	$2,2\cdot10^{8}$
AG 109m	$4,4\cdot10^{8}$	$4,2\cdot10^{8}$	$1,3\cdot10^{8}$	SM 153	$5,3\cdot10^{8}$	$5,2\cdot10^{8}$	$3,7\cdot10^{8}$
AG 110	$7,6\cdot10^{7}$	$1,4\cdot10^{4}$	$1,4\cdot10^{4}$	EU 156	$1,5\cdot10^{8}$	$1,5\cdot10^{8}$	$1,4\cdot10^{8}$
AG 110m	$1,1\cdot10^{6}$	$1,1\cdot10^{6}$	$1,1\cdot10^{6}$	U 237	$6,7\cdot10^{8}$	$6,6\cdot10^{8}$	$6,0\cdot10^{8}$
SB 124	$7,1\cdot10^{5}$	$7,1\cdot10^{5}$	$7,0\cdot10^{5}$	U 239	$3,7\cdot10^{10}$	$6,3\cdot10^{9}$	$1,4\cdot10^{-8}$
TE 129	$6,4\cdot10^{8}$	$6,2\cdot10^{8}$	$7,9\cdot10^{7}$	NP 238	$1,2\cdot10^{8}$	$1,2\cdot10^{8}$	$8,7\cdot10^{7}$
TE 129m	$9,7\cdot10^{7}$	$9,7\cdot10^{7}$	$9,5\cdot10^{7}$	NP 239	$3,7\cdot10^{10}$	$3,7\cdot10^{10}$	$2,8\cdot10^{10}$
TE 131	$1,9\cdot10^{9}$	$9,4\cdot10^{8}$	$4,1\cdot10^{7}$	PU 238	$4,3\cdot10^{5}$	$4,3\cdot10^{5}$	$4,4\cdot10^{5}$
TE 131m	$3,1\cdot10^{8}$	$3,1\cdot10^{8}$	$1,8\cdot10^{8}$	PU 239	$5,1\cdot10^{5}$	$5,1\cdot10^{5}$	$5,2\cdot10^{5}$
I 131	$2,2\cdot10^{9}$	$2,2\cdot10^{9}$	$2,0\cdot10^{9}$	PU 241	$1,0\cdot10^{8}$	$1,0\cdot10^{8}$	$1,0\cdot10^{8}$
XE 131m	$2,2\cdot10^{7}$	$2,2\cdot10^{7}$	$2,2\cdot10^{7}$	PU 243	$6,8\cdot10^{7}$	$5,9\cdot10^{7}$	$2,4\cdot10^{6}$
TE 132	$3,2\cdot10^{9}$	$3,1\cdot10^{9}$	$2,5\cdot10^{9}$	AM 242	$2,9\cdot10^{7}$	$2,8\cdot10^{7}$	$1,0\cdot10^{7}$
I 132	$3,2\cdot10^{9}$	$3,2\cdot10^{9}$	$2,6\cdot10^{9}$	AM 244	$3,2\cdot10^{5}$	$2,9\cdot10^{5}$	$6,1\cdot10^{4}$
I 133	$4,7\cdot10^{9}$	$4,6\cdot10^{9}$	$2,2\cdot10^{9}$	CM 242	$9,1\cdot10^{6}$	$9,1\cdot10^{6}$	$9,1\cdot10^{6}$
XE 133	$4,6\cdot10^{9}$	$4,6\cdot10^{9}$	$4,5\cdot10^{9}$	CM 244	$6,0\cdot10^{4}$	$6,0\cdot10^{4}$	$6,0\cdot10^{4}$
XE 133m	$1,4\cdot10^{8}$	$1,4\cdot10^{8}$	$1,3\cdot10^{8}$	TOTAL	$4,7\cdot10^{11}$	$1,9\cdot10^{11}$	$1,1\cdot10^{11}$

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The sources of the figures have been given in conjunction with the figures in parenthesis just after their titles, usually as direct links to the respective internet sites

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Abstract

Of the nuclear power plants situated in the Nordic and their neighbouring countries, the Ingalina, Leningrad and Kola plants are considered to pose the largest risks to the public.
The purpose of this report is to provide basic relevant information about these three plants for use in a case of a major nuclear accident or incident in any of them. The report could be used e.g. by authorities dealing with the resulting emergency measures to provide the public and the media with relevant information about the plant in question. The report can also be used for quick general familiarization with the plants in question.
The total activity inventories for all the plants are listed at the end of the report, in Chapter 4. The release of noble gases in close to 100% in most severe accidents, but the releases of other elements depend strongly on the plant features and the nature of the accident.
This report has been compiled from several sources. The main source has been an earlier NKS-report: "Design and Safety Features of Nuclear Reactors Neighbouring the Nordic Countries", TemaNord 1994:595, 1994. Only limited editing has been done. Sources of the figures are presented in parenthesis after the figure titles.

Key words: nuclear power plant, RBMK, BWR, VVER, reactors, reactor design, design data, reactor core data, fuel data, layout, activity inventories, main features

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