Nuclear Safety

Design and Safety Features of Nuclear Reactors Neighbouring the Nordic Countries

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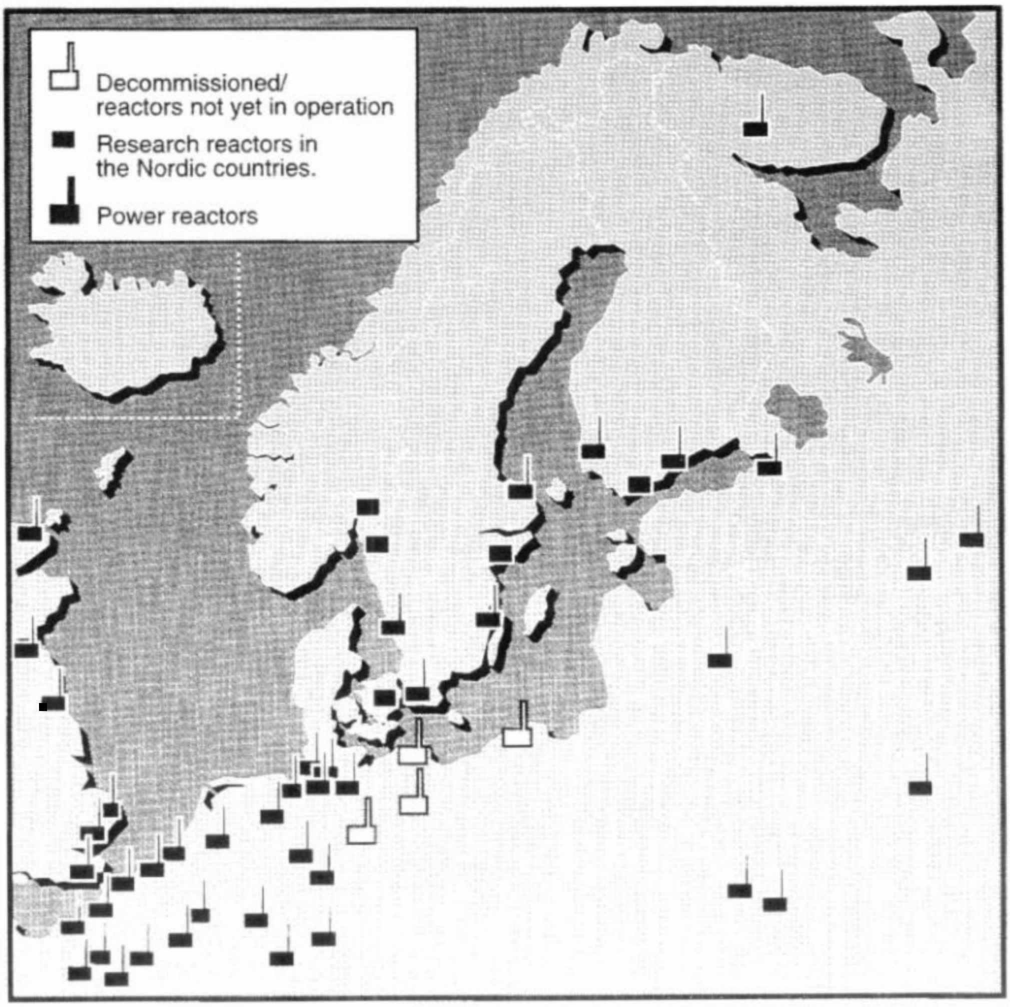

Design and Safety Features of Nuclear Reactors Neighbouring the Nordic Countries

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Final Report of the Nordic Nuclear Safety Research Project SIK-3

Edited by
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The Nordic Council of Ministers
was established in 1971. It submits proposals on co-operation between the governments of the five Nordic countries to the Nordic Council, implements the Council's recommendations and reports on results, while directing the work carried out in the targeted areas. The Prime Ministers of the five Nordic countries assume overall responsibility for the co-operation measures, which are co-ordinated by the ministers for co-operation and the Nordic Co-operation Committee. The composition of the Council of Ministers varies, depending on the nature of the issue to be treated.

The Nordic Council
was formed in 1952 to promote co-operation between the parliaments and governments of Denmark, Iceland, Norway and Sweden. Finland joined in 1955. At the sessions held by the Council, representatives from the Faroe Islands and Greenland form part of the Danish delegation, while Åland is represented on the Finnish delegation. The Council consists of 87 elected members - all of whom are members of parliament. The Nordic Council takes initiatives, acts in a consultative capacity and monitors co-operation measures. The Council operates via its institutions: the Plenary Assembly, the Presidium, and standing committees.

Abstract

The Chernobyl accident in 1986 revealed a remarkable lack of information in the Western countries on design and operation of Russian reactors. Therefore a Nordic project was started with the purpose of collecting information on the design and safety features of reactors neighbouring the Nordic countries. The nuclear power plants included are the RBMK reactors Ignalina and Leningrad, the VVER reactors Greifswald and Kola, the BWR reactors Brunsbüttel and Krümmel and the PWR reactors Stade and Brokdorf, all located within 100-450 km from the borders of a Nordic country. Marine reactors supposed to operate in the Nordic seas are also considered. Detailed reports for each power plant and marine reactors have been made.

Key words: RBMK reactors, VVER reactors, Ignalina NPP, Leningrad NPP, Kola NPP, Brunsbüttel NPP, Krümmel NPP, Brokdorf NPP, Stade NPP, ship reactors, submarine reactors, icebreaker reactors, nuclear safety criteria.

Summary

When the Chernobyl accident occurred in 1986 it appeared that knowledge available in the Nordic countries, i.e. Norway, Sweden, Finland, Iceland and Denmark, especially about the Eastern type of reactors was very poor. The lack of information applied to technical features, safety related factors, exact location etc. and also to the management set-up in the utilities and at the competent authorities. The information that was actually available was difficult to get hold of, so that the competent authorities could not easily utilize it.

In the SIK-3 project, information is collected in a systematic way about nuclear power plants located close to borders of the Nordic countries. The information includes design features and operational practices that are significant for plant safety, provided the data have been available. The data are presented in a uniform manner for all the reactors, so relevant information can easily be found in an emergency situation. Examples of information that is provided include map of site location, plant arrangement, safety criteria, comparison with similar reactor types, organization of the authorities etc.

The data of each plant are assembled in reports and are also available on a PC-database for easy updating. The information in these reports can be used for qualitative evaluations, but more ample information would be required in order to perform probabilistic safety studies. In case of a nuclear accident the information could be used to evaluate the progression of the accident and also to evaluate the potential consequences on the environment.

The nuclear power plants included in this project are the RBMK reactors Ignalina and Leningrad, the VVER reactors Greifswald and Kola, the BWR reactors Brunsbüttel and Krümmel and the PWR reactors Stade and Brokdorf, all of which are located within 100-450 km from the borders of a Nordic country.

The reports include essential facts and special safety features of each reactor type. However, it was not intended to include a thorough evaluation of the safety condition of each particular plant.

Some of the data are confidential and therefore restrictions have been imposed on the distribution of the detailed reports.

Nuclear powered icebreakers and submarines are known to operate in the seas close to the Nordic countries. Information has therefore been collected on the design and safety features of marine reactors to the extent the information has been available. The reports on ship reactors can be used as a basis for risk assessment in case of a future nuclear ship accident or in case of release of radioactive material near the sea bottom from disposed reactors or from sunken submarines.

The project has given rise to a contact net among the Nordic nuclear authorities and a knowledge base of, especially, the Eastern type of reactors. Therefore, in case of a nuclear accident in a neighbouring country, the Nordic nuclear authorities will be better prepared than was the case at the time of the Chernobyl accident.

Sammenfatning

Da Chernobyl ulykken skete i 1986, viste det sig, at den tilgængelige viden i Norden vedrørende især de østlige kernekraftreaktorer var utilstrækkelig. Den manglende viden gjaldt tekniske egenskaber, sikkerhedsrelaterede faktorer, placering samt organisationsstrukturen af både de østlige værker og de østlige nukleare myndigheder. De informationer, som fandtes i Norden, var vanskelige at få fat på, ligesom de var svære at udnytte for myndighederne.

I SIK-3 projektet er informationerne om kernekraftværkerne tæt ved de nordiske grænser samlet på en systematisk måde. Informationerne omfatter design egenskaber og driftsforhold, som har sikkerhedsmæssig betydning for anlægget, i det omfang oplysningerne har været til stede. Oplysningerne præsenteres på en ensartet måde for alle nukleare anlæg, således at de relevante detaljer hurtigt kan findes i tilfælde af en katastrofesituation. Eksempler på den information, som er inkluderet, er kort over placering af kernekraftværket, individuel placering af bygningerne, sikkerhedskriterier, sammenligning med reaktorer af lignende type, organisering af myndigheder o.s.v.

Oplysningerne om hvert anlæg er samlet i rapporter og er også tilgængelige på en PC-database for at lette fremtidige opdateringer. Informationerne i disse rapporter kan anvendes til kvalitative vurdereringer af sikkerheden på de enkelte anlæg, hvorimod egentlige risikostudier kræver adgang til mere detaljerede oplysninger. I tilfælde af at et uheld skulle indtræffe på et nabokernekraftværk, kan de udarbejdede rapporter anvendes til at vurdere, hvorledes uheldet kan tænkes at udvikle sig, herunder også vurdere mulige konsekvenser for omgivelserne.

De nukleare anlæg, som er behandlet i projektet, omfatter RBMK reaktorerne Ignalina og Leningrad, VVER reaktorerne Greifswald og Kola, BWR reaktorerne Brunsbüttel og Krümmel og PWR reaktorerne Stade og Brokdorf, alle placeret fra 100-450 km fra de nordiske grænser.

Rapporterne afspejler konstruktionsdata og specielle sikkerhedsegenskaber for hver reaktortype. Derimod har det ikke været hensigten i dette projekt at inkludere en egentlig vurdering af sikkerhedstilstanden for det enkelte anlæg.

Nogle af informationerne er fortrolige, og derfor har det været nødvendigt at pålægge restriktioner på distributionen af rapporterne.

Det vides, at isbrydere og undervandsbåde, som er drevet af nukleare reaktorer, opererer i de nordiske farvandene. Derfor er der i projektet indsamlet information omkring sikkerhedsforhold af disse marine reaktorer, i det omfang oplysningerne har været tilgængelige. Rapporterne om skibsreaktorer kan anvendes som grundlag for at bedømme risikoen for eventuelle fremtidige uheld med nukleare skibe. Ligeledes kan de anvendes i forbindelse med vurdering af frigørelse af radioaktivt materiale fra allerede sunkne skibe og undervandsbåde.

SIK-3 projektet har givet anledning til etablering af et kontaktnet mellem de nordiske nukleare myndigheder, samt opbygning af en database omkring specielt reaktorer af østlig oprindelse. I tilfælde af et fremtidigt uheld i et nabokernekraftværk vil de nukleare myndigheder i Norden derfor være langt bedre rustet, end tilfældet var, da Chernobyl katastrofen skete.

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Contents

1 Introduction

The purpose of this final report of the SIK-3 project, is to give an overview of the results obtained in the project with reference to the more detailed data reports.

The report has been aimed at readers with general interest in the subject, while more technically interested readers are referred to the data reports. Particular technical terms used in the text are printed in *italic* but not further explained in the present report.

The SIK-3 project is part of the Nordic NKS/SIK programme, carried out during 1990-1993. The programme also includes SIK-1, safety evaluation and SIK-2, severe accidents.

Reference 14 summarizes the achievements of the joint Nordic research programme NKS/SIK within reactor safety.

1.1 Background

The Chernobyl accident in 1986 revealed a remarkable lack of information in the Western countries on the design and operation of Russian reactors. In case of a nuclear accident such information is necessary to evaluate the progression of the accident and also to evaluate the potential consequences on the environment. In some Nordic countries the nuclear authorities were criticised for this lack of information concerning the Chernobyl type of reactors.

Therefore a Nordic project within the NKS, the Nordic Committee for Nuclear Safety Research, was initiated with the title: "Design and Safety Features of Reactors Neighbouring the Nordic Countries".

1.2 Objectives

The objectives of the SIK-3 project were:

- To collect, systematize and evaluate data on the safety of nuclear reactors within about 500 km from the border of a Nordic country, i.e. Norway, Sweden, Finland, Iceland and Denmark.

- To enable the nuclear safety authorities in the Nordic countries to respond to general safety related questions concerning a particular nuclear reactor. Such questions could come from politicians, the public or the media.

- To provide the Nordic nuclear safety authorities with diagrams of the important safety components of the nuclear plants as a basis for exchange of information with the plant personnel in an emergency situation.

1.3 Scope

The scope and limitations in the project have been the following:

- Marine reactors supposed to operate in the seas around the Nordic countries have been included.

- The reports made for each plant do not include an evaluation of the safety condition of the particular plant. They only state the facts and special safety features of each reactor type.

- Only power reactors have been treated in details. Research reactors and waste storage facilities have not been dealt with. Reactors within the Nordic countries have not been included.

- Most of the information for the different plants have been obtained by contact to the plant owner. Some of the data are confidential and therefore restrictions are imposed on the distribution of the detailed reports.

2 Neighbouring Nuclear Power Plants

2.1 Reactors considered

In table 1 are shown the neighbouring nuclear power plants considered with list of reactor type and distance from the Nordic borders. That comprises reactors within about 500 km from the borders of the Nordic countries.

Table 1. Reactors within 500 km from the border of a Nordic country.

Nuclear Power Plant	Reactor type	No of Units	Distance
Greifswald	VVER	6	80 km (DK)
Leningrad	RBMK	4	100 km (FIN)
Ignalina	RBMK	2	450 km (S)
Kola	VVER	4	120 km (FIN)
Brunsbüttel	BWR	1	100 km (DK)
Krümmel	BWR	1	200 km (DK)
Brokdorf	PWR	1	100 km (DK)
Stade	PWR	1	150 km (DK)
Marine reactors	(PWR)	-	-

***VVER** means Water Water Energy Reactor, (**V**odo-**V**odjanoj **E**nergeticheskij **R**eaktor), that is Pressurized Water Reactor of Russian type.*
***RBMK** means Reactor Large Power Channel type, (**R**eaktor **B**ol'shoj **M**oshchnost'i **K**anal'nogo tipa), that is Boiling-Water-Cooled, Graphite-Moderated, Channel type Reactor of Russian type.*
***BWR** means **B**oiling **W**ater **R**eactor of Western design.*
***PWR** means **P**ressurized **W**ater **R**eactor of Western design.*

During the project period it was decided to shut down all VVER units at the Greifswald Nuclear Power Plant. All other units shown in the table are operating.

The marine reactors are mainly of the pressurized water reactor type but they are much smaller in power than normal power reactors.

A map of the neighbouring reactors is shown in Fig. 1.

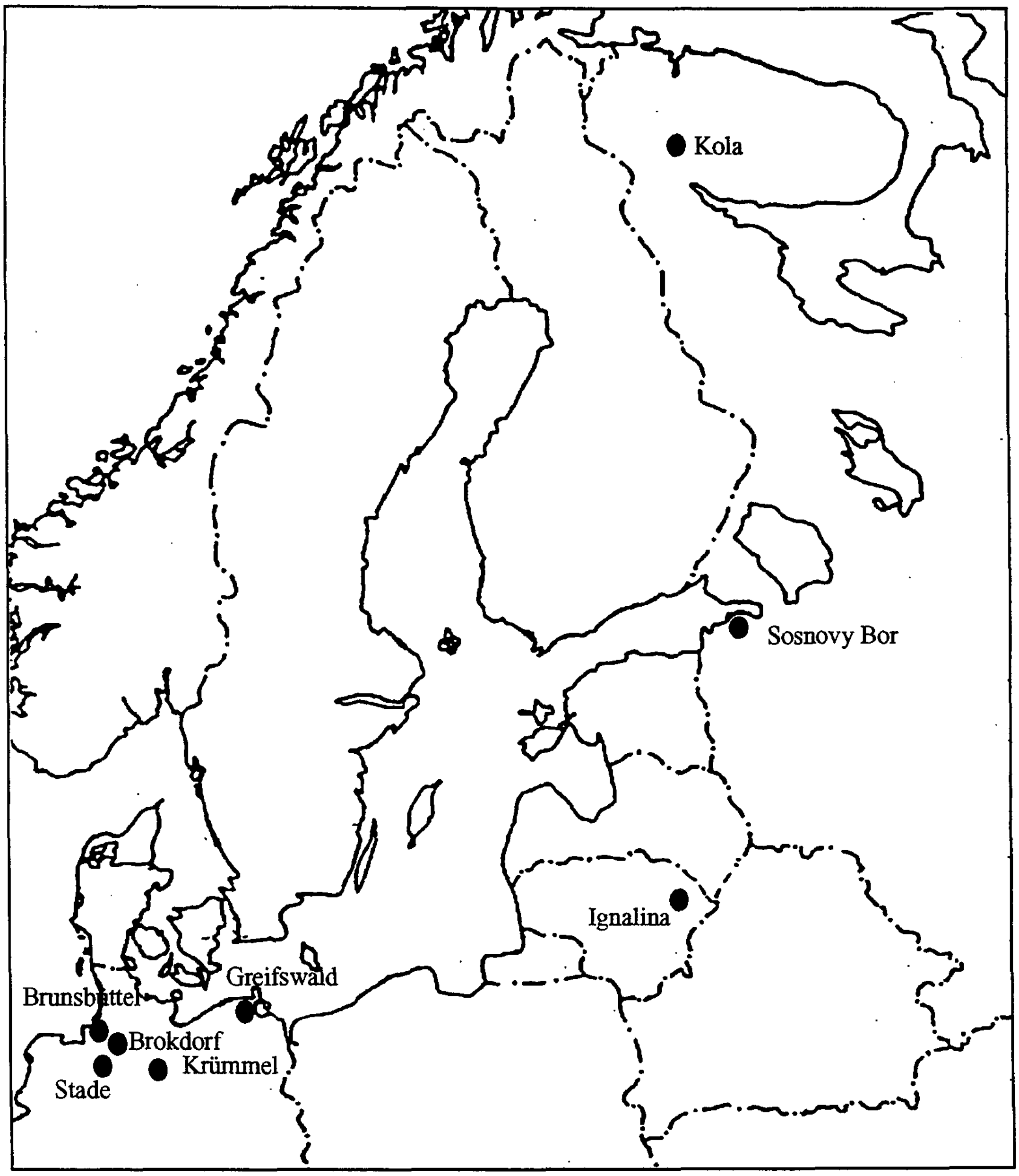

Figure 1. Map showing the nuclear power plants treated in the project.

2.2 Disposition of the data

The information collected for each plant is put into a common scheme to facilitate comparison of data for different plants. For that purpose a modified disposition from a Nordic proposal for organization of a nuclear safety report, (NARS) [1], was applied.

The main items in this disposition are listed below.

1. **Introduction**
2. **Summary of design data**
3. **Site and region**
4. **Safety criteria**
5. **Technical description and design evaluation**
6. **Fire protection**
7. **Plant performance during normal operation**
8. **Accident analyses**
9. **Radiation protection**
10. **Offsite dose assessment**
11. **Planning, organization and administrative control**
12. **Organization of the authorities**
13. **Probabilistic safety assessment**

Among the thirteen main items not all have been treated in detail. Item 5, "Technical description and design evaluation", constitutes the most comprehensive chapter in the reports, whereas item 8, "Accident analyses", often is very short and incomplete due to lack of information.

Appendix 1 shows the detailed disposition of the reports. Data for some of the items listed in Appendix 1 may be missing because the information was unavailable at the time of writing. However, the items are still included to assist in a future updating of the reports.

2.3 Validation of the data

In the beginning of the project period it was very difficult to get hold of information about the different reactors.

"Glasnost" and "Perestroika" in the Eastern countries were not implemented in practice and we were met with a lot of scepticism when we tried to contact the managers at the Eastern nuclear power plants.

However, during the project period this changed and we were invited to visit some of the plants. Thus, at Leningrad Nuclear Power Plant, close to Sosnovy Bor in Russia, one of the SIK-3 project group members spent two weeks, where he discussed and corrected the data in the report. At last, the final report was reviewed by the operating staff at the plant.

The same was the case at the Ignalina Nuclear Power Plant in Lithuania. A member of the project group payed several visits to the plant and met great hospitality and collaboration from the operating staff. This report was also reviewed and corrected by the operators of the plant.

Finally, at the Kola Nuclear Power Plant in the north westerly part of Russia we also got all the data we requested. Here the final report was also reviewed and approved by a member of the operating staff.

In fact, at all the three Eastern nuclear power plants the managers were very content with the prepared reports which provided them an English description of their power plants - useful to offer to all the Western delegations visiting their nuclear power plants.

In Germany, however, we did not meet the same kind of understanding when we asked for data of their nuclear power plants. At the Krümmel and Brunsbüttel Nuclear Power Plants located in Schleswig-Holstein by the river Elbe we succeded in paying a visit and were provided with satisfactory information and we got our final report reviewed and approved by the managers of the plant.

On the other hand, at the Stade and Brokdorf Nuclear Power Plants also located in Schleswig-Holstein by the river Elbe our request for information was not answered at all. Therefore the reports on Stade and Brokdorf are based on information obtained in the open literature and the content is not up-to-date.

3 Basic nuclear safety criteria

The primary risk source in a nuclear reactor is the large amount of radioactive material that is generated during operation, primarily the so-called fission products. Even if just a small fraction of these were released to the environment, this could cause severe danger to the biosphere. Therefore one of the main aims in nuclear safety work is to prevent release of radioactive fission products into the environment.

In order to prevent such a release a number of barriers are present between the primary risk source and the environment. The following barriers exist:

- Fuel matrix
- Fuel cladding
- Pressure boundary of primary coolant system including reactor vessel
- Reactor containment
- Filter

Furthermore, the nuclear process in the reactor core should be self-controlling or *inherently stable* during normal conditions, so that small pertubations in process parameters should always cause the reactor to return to normal conditions by itself.

During abnormal conditions the reactor should possess necessary shut-down capabilities.

3.1 Western safety criteria

Modern Western nuclear safety is based on the application of the *defence-in-depth* concept. Also *redundancy, diversity,* etc. are applied. Recently also passive or natural safety features have been discussed.

According to Western safety criteria, protective measures are realized at four different safety levels:

- Normal operation
- Transient conditions
- Design basis accidents
- Incidents beyond design basis

The Western safety concepts give priority to measures for *accident mitigation* and *accident management* as well as automatic actions of safety systems. In order to relieve the operators and to reduce the response frequency of protection systems, a progressive concept of protection by automatic control is applied.

The principal aim of all safety considerations is to ensure that the radioactive materials existing in a nuclear power plant are confined at all times. In other words a nuclear power plant must be designed and operated in such way that at all times during specified normal and upset operation and during the so-called design basis accidents the following conditions (*design goals*) must be fulfilled:

- The reactor can be safely shut down and kept shut down
- The residual heat can be removed
- The radiation exposure of personnel and radioactive releases to the environment must be kept as low as possible.

To achieve this design goal, the safety precaution principles were set up with a multiple level safety concept as follows:

- Assurance of normal operation with the least possible occurrence of abnormal operating conditions.
- Control of abnormal operating conditions that might occur by usage of engineered safety features.
- Assurance that design basis accidents stay within given limits and assurance of dose minimization by means of engineered safety features.

Furthermore the so-called "single failure criteria" must be fulfilled, that is the safety systems must comply with the design criteria even under the assumption of a single component failure in one of the safety systems.

When analyzing emergency conditions, the following criteria are applied:

- With the reactor at rated power, a maximum diameter pipe break with a two-way free outflow of coolant, a so-called guillotine break, is postulated to be a design basis accident.

As to the fuel, the following design limits are applied during normal operation:

- The number of failed fuel rods with gas leakage must be less than 1.0 % of the total number of fuel rods
- The number of failed fuel rods resulting in direct contact between fuel and coolant must be less than 0.1 % of the total number of fuel rods

During accident conditions the following limits are applied:

- The maximum cladding temperature must be less than 1200 °C
- Local depth of oxidation of fuel cladding must be less than 17 % of the original thickness

3.2 Eastern safety criteria

Modern Eastern nuclear safety is based on the same principles as Western, that is the application of the *defence-in-depth* concept. Also *redundancy, diversity*, etc. are applied but the Eastern reactors covered in this project only have the following barriers:

- Fuel matrix
- Fuel cladding
- Pressure boundary of primary coolant system including reactor vessel
- Confinement

According to Eastern safety criteria, protective measures are realized at three different safety levels:

- Normal operation
- Upset conditions
- Design basis accidents

As to the fuel the following design limits are applied during normal operation:

- The number of failed fuel rods with gas leakage must be less than 1.0 % of the total number of fuel rods
- The number of failed fuel rods resulting in direct contact between fuel and coolant must be less than 0.1 % of the total number of fuel rods

When analyzing design basis accidents the following criteria are applied:

- With the reactor at rated power, break of a pipe with a diameter of 500 mm and a two-way free outflow of coolant, a so-called guillotine break, is thought to be a design basis accident.
- (For some of the oldest Russian reactors the pipe break diameter was limited to 32 mm - the pipes then had flow reducing orifices)

During accident conditions the following limits are applied:

- The maximum cladding temperature must be less than 1200 °C
- Local depth of oxidation of fuel cladding must be less than 18 % of the original thickness

3.3 Discussion of the differences between Western and Eastern safety criteria

The safety criteria defined above are not complete, several other criteria exist for the different systems, but those listed are the main criteria.

There seems to be no big differences between Western and Eastern safety criteria, the main one being the lack of full containment for many of the Russian reactors built so far, but this is going to change for new reactor constructions.

In general there has been a different approach to safety in the West and the East.

In the West, safety design has often been demonstrated through tests and experiments in pilot plants. This demonstration has fulfilled two goals, partly it has shown the functioning of the system in question and partly it has helped verifying computer codes developed for analysing the safety of nuclear power plants.

The economy of the new safety features has also played a role in the Western safety approach. Several design calculations have been made in order to minimize the cost and still fulfil the objectives of the safety systems.

In the former Soviet Union safety design was often based on calculations rather than experiments. However, often the systems were designed from a conservative point of view, that is pipe dimensions, number of pumps, size of vessels etc. were bigger than necessary. In this way compensation was made for some of the uncertainties in the calculations and for the lack of experiments and verification of codes.

4 Pressurized Water Reactors

In the following sections the design data of each PWR reactor treated in the project will be summarized. These are the Western type of PWRs, Stade and Brokdorf, and the Eastern VVER plant at Kola.

Finally, some characteristic differences between the Western and Eastern type of pressurized water reactors will be mentioned.

4.1 Stade PWR

The Stade nuclear power plant is located at the river Elbe about 35 km north west of Hamburg, Fig. 2. The pressurized water reactor has an electrical power of 670 MW and it has been operating since 1972 [13].

Like other German PWR reactors Stade has a double containment with an inner spherical steel shell and an outer hemispherical concrete structure. The space between the two shells is kept below atmospheric pressure by a ventilation system and any minor leakage flow from the inner containment is filtered before reaching the environment.

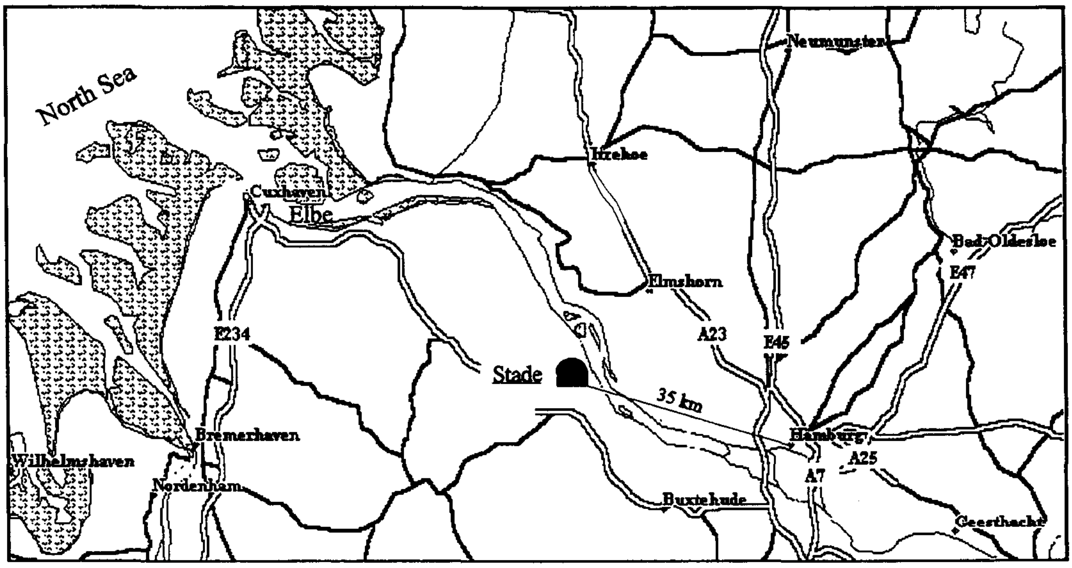

Figure 2. Location of Stade Nuclear Power Plant, 35 km north west of Hamburg

Table 2. Summary of design data of the Stade Nuclear Power Plant

Overall plant

Thermal output	1892 MW
Electrical output	662 MW
Net electrical output	630 MW
Net overall efficiency	33.8 %

Reactor plant

Coolant and moderator	H_2O
Fuel	UO_2
Cladding material	Zircaloy-4
Enrichment	3.29 % U235
Number of fuel elements	157+3
Fuel configuration	15 X 15
Fuel rod diameter	10.75 mm
Fuel assembly overall length	3655 mm
Average specific power	33.8 kW/kgU
Number of control assemblies	49
Absorber material	Ag80In15Cd5
Number of reactor coolant loops	4
Operating pressure	154 bar
Reactor coolant flow rate	44000 t/h
Reactor coolant temperature	281 ^{0}C/308 ^{0}C
Reactor inlet/outlet	284 ^{0}C/312 ^{0}C

Reactor building

Spherical steel containment	48 m
Wall thickness	23/30 mm
Design pressure	3.8 bar
Test pressure	4.24 bar
Thickness of outer concrete shell	600 mm

Reactor pressure vessel

Inside diameter	4080 mm
Maximum overall height	10400 mm
Wall thickness	197+7 mm
Material	22NiMoCr37
Nozzle diameter inlet/outlet	700 mm
Total weight	279 t

Steam generators

Number	4
Steam output per unit	249 kg/s
Overall height	15600 mm
Diameter	2900/3500 mm
Wall thickness	50/61 mm
Tube material	Incoloy 800
Weight	160 t

Reactor coolant pumps

Numbers	4
Capacity	4.07 m^3/s
Discharge head	64 m
Power	3200 kW
Motor speed	1490 rpm
Weight	61 t
Design temperature/pressure	350 ^{0}C/175 bar

Secondary side

Main steam flow rate	998 kg/s
Main steam pressure	51 bar
Main steam temperature	265 ^{0}C
Maximum moisture content	0.25 %
Condenser vacuum	0.032 bar
Feedwater temperature	207 ^{0}C

Turbine

Three-cylinder single-shaft	
Turbine speed	1500 rpm

Generator

Rated output	780 MVA

Emergency power supply

Diesel units	3
Output	2200 KVA
Voltage	11 kV

4.2 Brokdorf PWR

The Brokdorf nuclear power plant is located at the river Elbe about 70 km north west of Hamburg, Fig. 3. The pressurized water reactor has an electrical power of 1380 MW and it has been operating since 1986 [12].

Brokdorf was the fourth nuclear power plant to be built at the riverside of Elbe. The three previous were Stade, Brunsbüttel and Krümmel. Brokdorf represents the third generation of German PWR development.

The construction of Brokdorf was delayed by 4 years due to massive opposition against nuclear energy in the middle of the seventies.

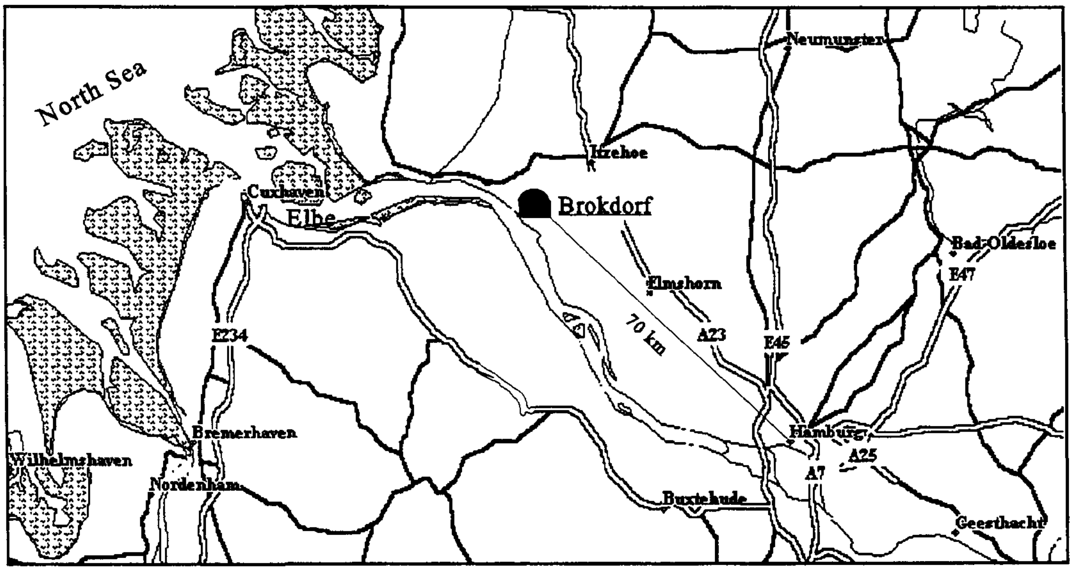

Figure 3. Location of Brokdorf Nuclear Power Plant, 70 km north west of Hamburg.

Table 3. Summary of design data of the Brokdorf Nuclear Power Plant

Overall plant

Type	PWR/KWU
Commercial operation	22.12.86
Thermal output	3765 MW
Electrical output	1380 MW
Net electrical output	1307 MW

Reactor plant

Coolant and moderator	H_2O
Fuel	UO_2
Cladding material	Zircaloy
Number of fuel elements	193
Fuel configuration	16 X 16
Number of fuel rods in a bundle	236
Fuel assembly active length	3900 mm
Average specific power	36.4 kW/kgU
Total fuel weight	103 t
Number of control assemblies	61
Number of reactor coolant loops	4
Operating pressure	158 bar
Reactor coolant flow rate	18800 kg/s

Reactor building

Spherical steel containment	56 m
Material	Aldur 50/65 D
Wall thickness	30/60 mm

Design pressure	5.3 bar
Test pressure	6.63 bar
Design temperature	145 °C

Reactor pressure vessel

Inside diameter	5000 mm
Maximum overall height	12668 mm
Wall thickness	142-250 mm
Material	22NiMoCr37
Design pressure	175 bar
Total weight	585 t

Steam generators

Number	4
Steam output per unit	515 kg/s
Steam temperature	284.5 °C
Steam pressure	68.65 bar
Overall height	20100 mm
Heat transfer area	5400 m^2
Diameter	4570 mm
Material	20MnMoNi55
Tube material	Incoloy 800
Weight	539 t

Reactor coolant pumps

Numbers	4
Capacity	5.39 m^3/s
Discharge head	89.6 m
Power	7300 kW
Manufacturer	KSB

Pressurizer

Design pressure	175 bar
Design temperature	362 °C
Diameter inner	2600 mm
Wall thickness	135 mm
Total height	14380 mm
Weight empty	140 t
Volume free	65 m^3
Water volume (Op.)	38 m^3
Steam volume (Op.)	27 m^3
Operating temperature	346 °C
Operating pressure	157 bar
Number of heaters	102
Total power	2000 kW

Secondary side

Main steam flow rate	1926 kg/s
Main steam pressure	66.75 bar
Main steam temperature	280 °C
Condenser vacuum	0.032 bar
Feedwater temperature	211 °C

Turbine

Turbine speed	1500 rpm

Generator

Voltage	27 kV
Rated output	1640 MVA
Cooling media	H_2O/H_2O
Cos Fi	0.8434

Emergency power supply

Diesel units	4
Output	5000 kW
Voltage	10 kV

4.3 Kola PWR

The Kola nuclear power plant is situated on the southern shore of Lake Imandra on the Kola peninsula in Russia, Fig. 4. 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].

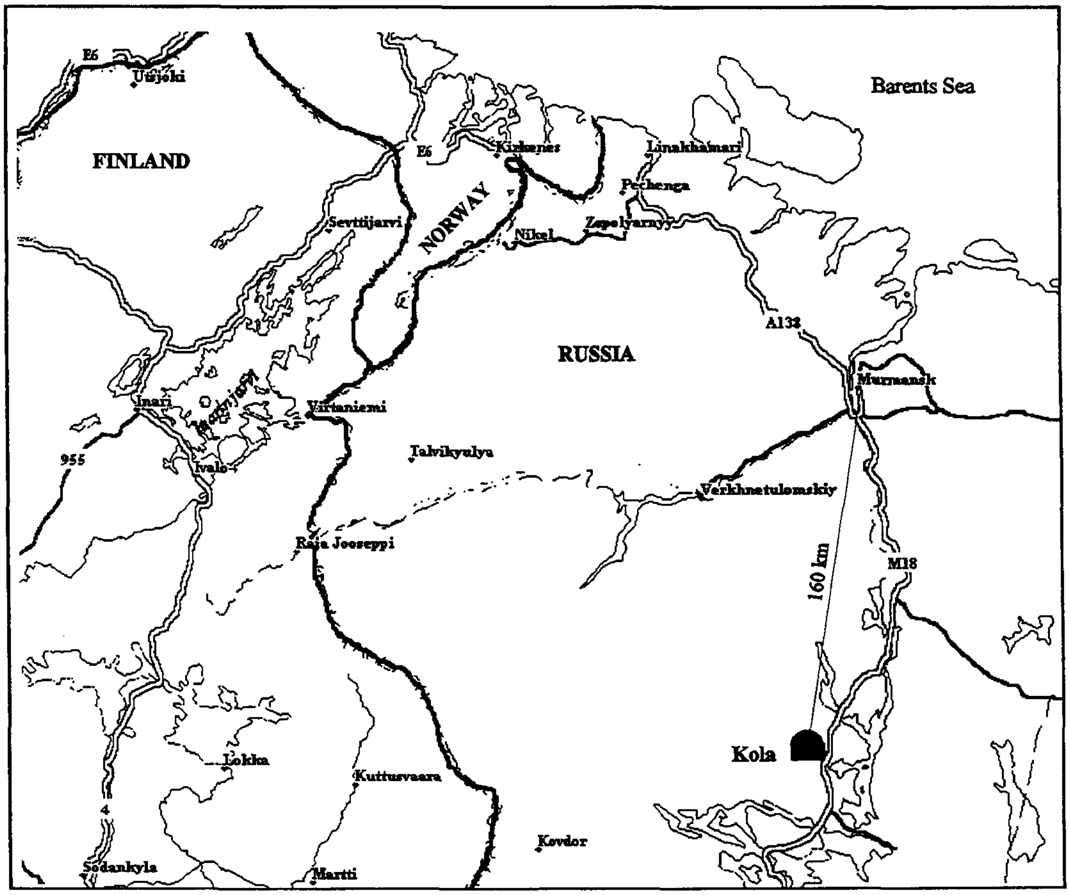

Figure 4. Location of Kola Nuclear Power Plant, 160 km south of Murmansk.

The essential difference between type 230 and type 213 is improved safety features of the latter model. Thus the containment function is much improved in Kola-3 and 4 compared to unit 1 and 2. The existence of a condenser building with a considerable free volume in addition to the steam condensing capabilities constitutes a significant improvement in the containment function. The aim of this design improvement has been to prevent uncontrolled releases of radioactivity during normal operation and in case of an accident to limit these releases to an acceptable level.

Below is shown a summary of the main design data which is valid both for model 230 and model 213 if not stated otherwise.

Table 3. Summary of design data of the Kola Nuclear Power Plant

Power	
Thermal	1375 MW
Electrical	440 MW
Efficiency	30.6 %
Reactor plant	
Coolant and moderator	H_2O
Fuel	UO_2
Cladding material	Zr 1% Nb
Number of fuel assemblies	349
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	41 t
Number of control assemblies	37
Number of core screen assemblies	36
Number of reactor coolant loops	6
Operating pressure	125 bar
Reactor coolant flow rate	10.8 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 ^0C
Steam pressure	44 bar
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	1.56 -1.97 m^3/s
Discharge head	51 m
Power	1400 kW
Manufacturer	-

Pressurizer

Design pressure	125 bar
Design temperature	325 ^0C
Diameter inner	2400 mm
Wall thickness	-
Total height	-
Weight empty	-
Volume free	38 m^3
Water volume	-
Steam volume	-
Operating temperature	325 ^0C
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 ^0C
Condenser vacuum	0.03 bar
Feedwater temperature	220 ^0C

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

4.4 Characteristic differences between Western and Eastern PWR 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 Stade and Brokdorf 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.

The Stade and Brokdorf 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.

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 leaktight concrete structure but it can only withstand an overpressure of about 0.8 bar before valves open to the atmosphere. Unit 3 and 4, type 213, have an improved containment function which can withstand an overpressure of about 1.5 bar thanks 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. In Fig. 5 the design of VVER type 213 is shown compared with the older VVER type 230.

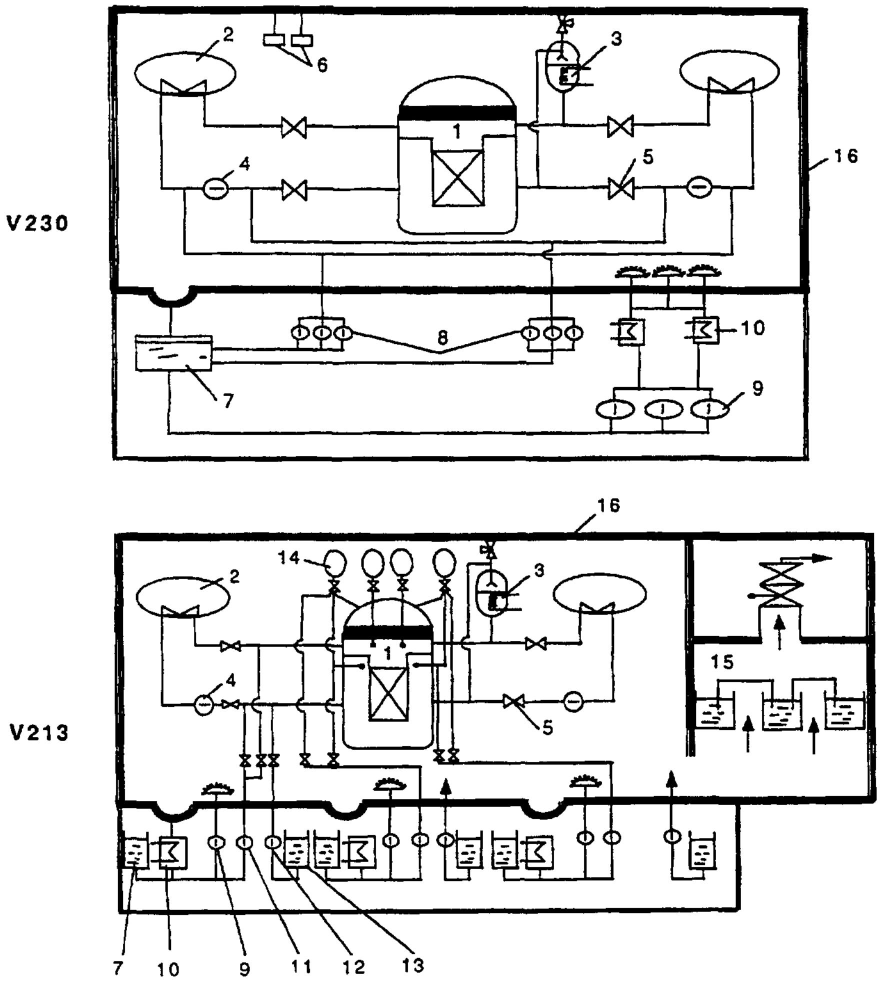

Figure 5. Differences in design between VVER-440/230 and VVER-440/213.

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

5 Reactors Cooled By Boiling Water

In the following sections the design data of each of the reactors cooled by boiling water will be summarized. These are the Western type of boiling water reactors, Brunsbüttel and Krümmel in Germany and the Eastern type of boiling water graphite reactors, the so-called RBMK reactors Leningrad in Russia and Ignalina in Lithuania.

Some characteristic differences between the Western and Eastern types of reactors cooled by boiling water will be mentioned.

5.1 Brunsbüttel BWR

Brunsbüttel nuclear power plant is located in Schleswig-Holstein by the river Elbe in the north-western part of Germany, Fig. 6. The plant has an electrical power of 806 MW and it has been operating since 1977 [10].

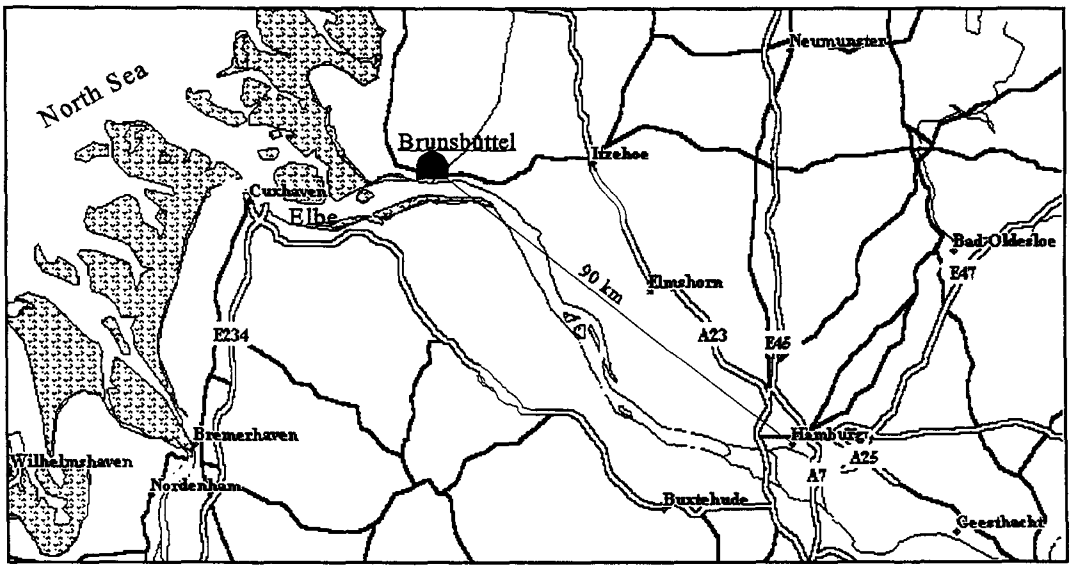

Figure 6. Location of Brunsbüttel Nuclear Power Plant, 90 km north west of Hamburg.

The plant is a single unit, direct cycle light water moderated and boiling water cooled reactor. The reactor was the first of a new design, where internal recirculating pumps were introduced instead of the older design with external or partly external pumps. Since then the internal recirculating pumps have been adopted in all subsequent BWRs in Germany.

The reactor containment is of spherical shape and fabricated from steel. The *pressure suppression system* within the containment introduced in this design was also a new safety feature. The purpose of this system is to limit the pressure buildup in case of a pipe break to within the design limits.

Table 4. Summary of design data of the Brunsbüttel Nuclear Power Plant

Main data

Type	BWR
Thermal power	2292 MW
Capacity, gross	806 MWe
Capacity, net	770 MWe
Efficiency, gross	35.1%
Operator	KKB
NSSS supplier	AEG
Constructor	KWU
Architect Engineer	KWU
Date of order	March, 1970
Start of construction	April, 1970
First criticality	June, 1976
Commercial operation	February, 1977

Containment general

Containment, type	pressure/suppression
Containment, material	steel, spherical
Containment diameter	27 m
Containment wall thickness	18-30 mm
Containment height	34 m
Design pressure	4.25 bar
Design temperature	135 °C
Design temperature condenser chamber	95 °C
Containment dry well air volume	3769 m^3
Containment wet well air volume	2362 m^3
Containment water volume	2300 m^3

Reactor plant general

Reactor vessel:	
diameter	5.58 m
height	20.7 m
weight	525 t
wall thickness	139+4 mm
Vessel material	ASTM A-508 Cl II
Cladding material	SS
Design pressure	87.3 bar
Coolant	H_2O
Moderator	H_2O
Internal recirculating pumps	8
Pump speed	400-2000 rpm
Speed regulation	frequency variation

Total coolant flow	35672 ton/hr
Coolant flow per pump	1570 kg/s
Inlet temp	277 °C
Outlet temp	285 °C
Outlet pressure	69.6 bar
Steam flow	4464 t/h
Moisture content	0.2%
Inlet pressure	73 bar
Feed water inlet temp	215 °C
Feed water flow	1240 kg/s

Reactor core data

Core diameter	3.97 m
Core height	3.66 m
Number of fuel assemblies	532
Number of fuel rods per FA	8x8, watercross
Number of control rods	129
Control rods absorber material	B_4C
Control rods material	SS
Normal operation	electromechanical
Insert time	122 sec. 3 cm/s
Scram	hydraulic
Insert time	2.7 sec. 140 cm/s

Fuel data

Fuel material	UO_2
Fuel inventory	104 tU
Specific power	24.1 KW/kgU
Power density, avg	50.6 KW/l
Average heat transfer	45.7 W/cm^2
Heat transfer area	4792 m^2
Burnup initial core	21000 MWD/tU
Burnup replacement core	40000 MWd/t
Cladding material	Zr-2
Cladding thickness	0.85 mm
Channel material	ZR-4
Refueling	22-25%
Refueling frequency	12 months

Turbine plant general

Turbines (One high-pressure and two low-pressure)	1
Turbine generator supplier	KWU
Speed	1500 rpm
Inlet turbine pressure	67 bar
Inlet temperature	282 °C
Steam flow to turbine	1140 kg/s

Moisture content	0.45%
Exhaust pressure	0.04 bar
Condenser, heat transfer area	2x24300 m^2
Condenser, cooling water flow	120000 m^3/h
Cooling water temp	11-21 °C
Total cooling water flow	130000 m^3/h
Cooling methods	river water
Dump capacity	93%
Gross effeciency	35.2%
Net effeciency	33.6%

Generator general

Supplier	KWU
Voltage	27 kV
Frequence	50 Hz
Output	1006 MVA
Power factor	0.80
Generator stator cooling	H_2O
Generator rotor cooling	H_2O

5.2 Krümmel BWR

The Krümmel nuclear power plant is located in Schleswig-Holstein by the river Elbe in the northern part of Germany. The plant is on the northern side of Elbe between the towns of Geesthacht and Lauenburg, Fig. 7. The major city near Krümmel is Hamburg 35 km away with about 1.2 million inhabitants [11].

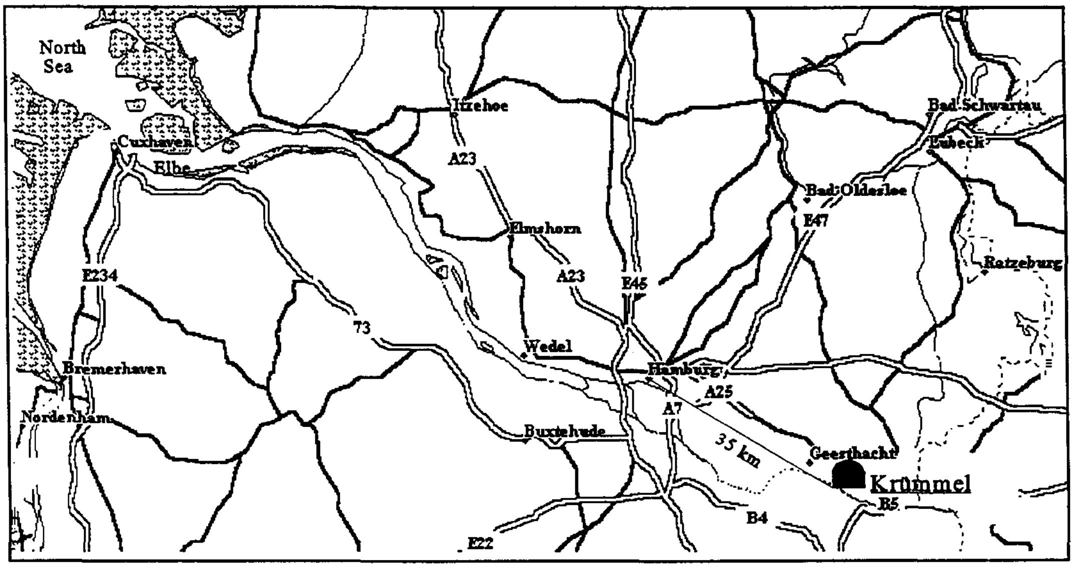

Figure 7. Location of Krümmel Nuclear Power Plant, 35 km south east of Hamburg.

The plant has an electrical output of 1316 MW and it has been operating since 1984. It is a single unit, direct cycle light water moderated and boiling water cooled reactor. The walls of the reactor building are made of reinforced concrete, 1.5 m thick. They protect the containment and the safety equipment against any conceivable external loadings such as airplane crash, chemical explosions etc.

The containment is spherical and made of 30 mm thick steel as was the case with Brunsbüttel. It is also equipped with a hydrogen control system and during operation the containment is filled with nitrogen to eliminate the possibility of a hydrogen-oxygen explosion in case of an accident.

Table 5. Summary of design data of the Krümmel Nuclear Power Plant

Main data

Type	BWR
Thermal power	3690 MW
Capacity, gross	1316 MWe
Capacity, net	1260 MWe
Efficiency, gross	35.7 %
Operator	KKK
NSSS supplier	AEG
Constructor	KWU
Architect Engineer	KWU
Date of order	January, 1972
Start of construction	January, 1974
First criticality	September, 1983
Commercial operation	March, 1984

Containment general

Containment, type	pressure/suppression
Containment, material	steel, spherical
Containment inner diameter	29.6 m
Containment wall thickness	25-30 mm
Design pressure	5.1/4.6 bar
Design temperature	150/170 °C
Containment water volume	3770 m^3

Reactor plant general

Reactor vessel,	
diameter, inner	6.78 m
height, outer	22.38 m
weight	790 ton
wall thickness	163+4 mm
Vessel material	ASTM A508II
Cladding material	SS
Design pressure	87.3 bar
Operating pressure	70.6 bar
Coolant	H_2O
Moderator	H_2O

Internal recirculating pumps	10
Pump speed	600-2000 rpm
Speed regulation	frequency variation
Total coolant flow	55600 ton/hr
Coolant flow per pump	1540 kg/s
Inlet temp	279 °C
Outlet temp	286 °C
Outlet pressure	70.6 bar
Inlet pressure	72 bar
Steam outlet flow	2000 kg/s
Steam temperature	286 °C
Feed water inlet temp	215 °C

Reactor core data

Core diameter	4.99 m
Core height	3.71 m
Number of fuel assemblies	840
Number of fuel rods per FA	63, 8x8
Number of control rods	205
Control rod absorber material	B_4C
Control rod material	SS
Normal operation	electromechanical
Insert time	112 sec, 3 cm/s
Scram	hydraulic
Insert time	2.5 sec, 150 cm/s

Fuel data

Fuel material	UO_2
Fuel inventory	155.8 tU
Specific power	23.7 KW/kgU
Power density, avg	50.9 W/cm^3
Average linear heat rate	160 W/cm
Peak linear heat rate	440 W/cm
Max heat transfer	112 W/cm^2
Average heat transfer	46 W/cm^2
Heat transfer area	7710 m^2
Linear power density	18.6 KW/m
Fresh fuel enrichment	1.95 wt%
Burnup initial core	17500 MWD/tU
Burnup replacement core	29000 MWd/t
Cladding material	Zr-2
Cladding thickness	0.85 mm
Channel material	Zr-4
Channel size	140x140 mm
Channel wall thickness	3 mm
Number of spacers	7
Spacer material	Zr-4
Fuel rod diameter	12.5 mm
Fuel pellet diameter	10.57 mm
Fuel pellet height	11 mm

Refueling	25%
Refueling frequency	12 months

Turbine plant general

Turbines	1
One high-pressure and three low-pressure	
Turbine generator supplier	KWU
Speed	1500 rpm
Inlet turbine pressure	67 bar
Inlet temperature	282 °C
Steam flow	2000 kg/s
Moisture content	0.2%
Number of steam lines	4
Exhaust pressure	0.044 bar
Moisture content	10%
Condenser, heat transfer area	3x20000 m^2
Cooling water flow	63 m^3/s
Cooling water temp	11-21 °C
Cooling methods	river water
Dump capacity	69%
Gross effeciency	35.6%
Net effeciency	34.1%

Generator general

Supplier	KWU
Voltage	27 kV
Output	1530 MVA
Frequence	50 Hz
Power factor	0.86
Generator rotor cooling	H_2O
Generator stator cooling	H_2O

5.3 Leningrad RBMK

The Leningrad nuclear power plant is located in the neighbourhood of the town Sosnovy Bor on the Baltic coast about 70 km west of St. Petersburg and 240 km from Helsinki [3], Fig. 8.

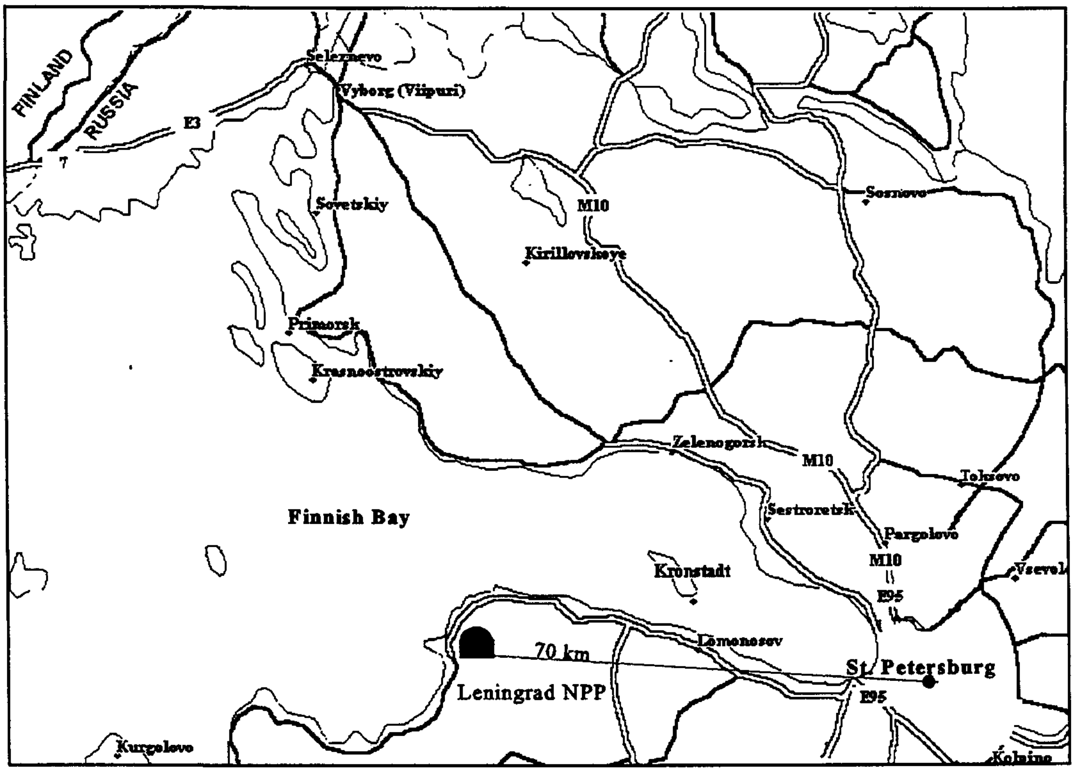

Figure 8. Location of Leningrad Nuclear Power Plant, 70 km west of St. Petersburg

The plant has four units with graphite moderated pressure tube boiling water reactors, a type of reactor which only has been constructed in the former Soviet Union. The Leningrad nuclear power plant 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-type 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.

A cross-sectional view of unit 1 and 2 of the Leningrad nuclear power plant is shown in Fig. 9.

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

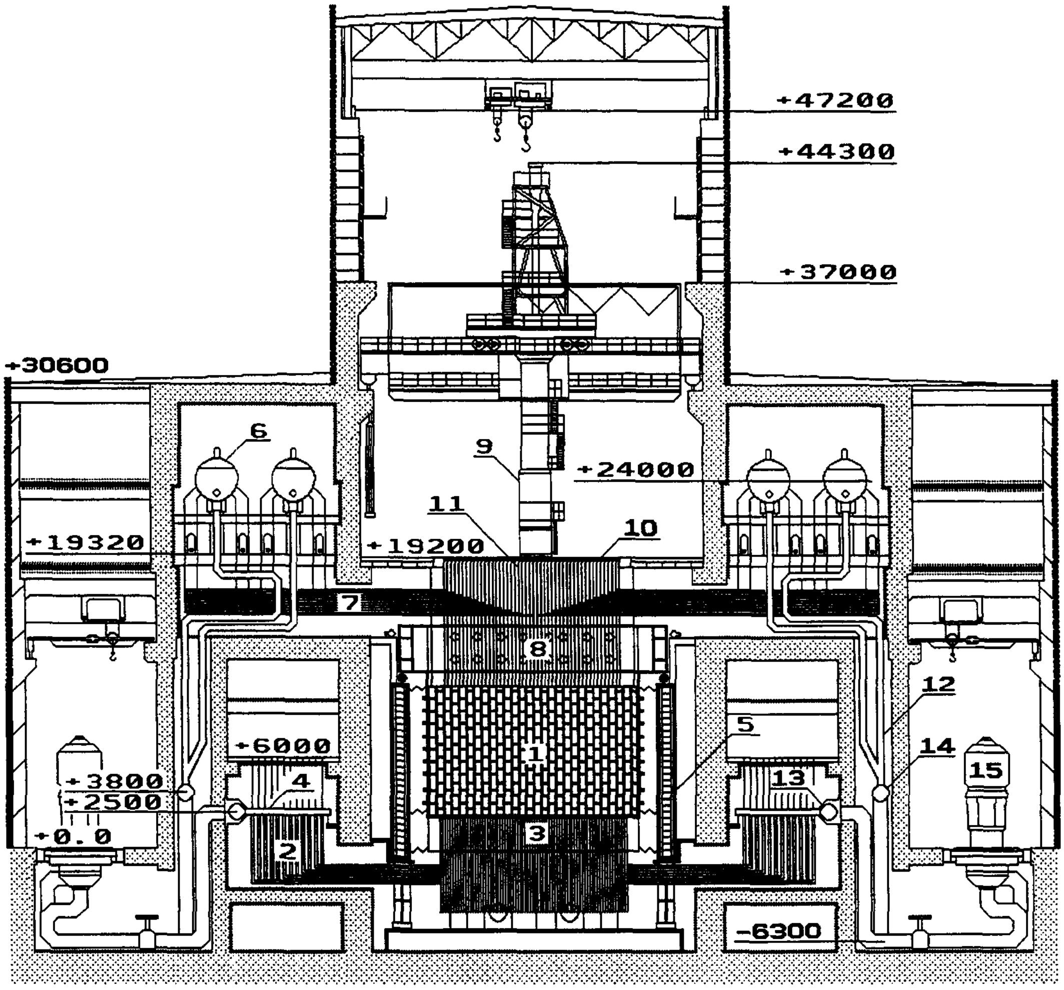

Figure 9. Cross-sectional view of unit 1 and 2 of the Leningrad NPP

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 - Refueling machine
10 - Removable floor
11 - Fuel channel ducts
12 - Downcomers
13 - Pressure collector
14 - Suction collector
15 - Main circulation pump

Table 6. Summary of design data for unit 1 of the Leningrad Nuclear Power Plant

MAIN DATA		
Reactor type	RBMK	Pressure tube boiling water reactor
Net electrical output	MW	1000
Reactor		
Reactor thermal output	MW	3200
Number of circulation loops		2
Total coolant flow	kg/s	10280
Pressure in a steam separator	MPa	7
Steam flow	kg/s	1610
Steam pressure at turbine inlet	MPa	6.5
Steam temperature at turbine inlet	°C	280

Feedwater temperature	°C	168
Maximum thermal power in a fuel channel	kW	3000
Pressure in a pressure tube		
• at inlet	MPa	8.75
• at outlet	MPa	7.5
Temperature in a pressure tube		
• at inlet	°C	270
• at outlet	°C	284
Coolant flow through a pressure tube at maximum power	kg/s	5.9
Maximum velocity of steam/water mixture in a pressure tube	m/s	20
Maximum steam content in a pressure tube outlet	mass %	27

Reactor core data

Core diameter	m	11.8
Core height	m	6.7-7
Specific power	W/gU	16.7
Graphite mass in the core	kg	1700×10^3
Graphite temperature	°C	600 - 750
Maximum temperature of metal structures	°C	350
Minimum dryout margin		1.35

Fuel data

Total weight of uranium	kg	192000
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	mm	79
Fuel assembly length	mm	6954
Fuel rod diameter	mm	13.5
Fuel enrichment	% ^{235}U	1.8 - 2.4
Maximum fuel temperature	°C	1800
Duration of operation of a fuel assembly at nominal power	days	1190
Average fuel burnup	MWd/tnU	19500

Pressure tube data

Pressure tube outer diameter	mm	88
Pressure tube wall thickness	mm	4
Average linear thermal power	W/cm	146
Maximum linear thermal power	W/cm	350
Maximum thermal flux on the surface of a fuel rod	W/cm^2	83

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	m^3/s	1.9 - 2.9
Pressure after pump	MPa	9.05
Pressure difference	MPa	1.8
Nominal electrical power	MW	4.4

Steam separating drums		
Number of drums		4
Diameter of a drum	m	2.3
Length of a drum	m	30
Weight of a drum	kg	200×10^3
Pressure in a drum	MPa	7.0

Turbine plant		
Generator output	MW	2 x 500
Turbine shaft length	m	39
Turbine speed	rpm	3000
Pressure in the condenser	kPa	4
Number of low pressure cylinders		4
Pressure in the high pressure inlet	MPa	6.5
Temperature in the high pressure inlet	^0C	280

5.4 Ignalina RBMK

The Ignalina nuclear power station is located in Lithuania, close to the borders of Byelorussia and Latvia. The station is built near the town Ignalina and the distance to the capital Vilnius with 600 000 inhabitants is 130 km [4]. Daugavpils in Latvia with 150 000 inhabitants is located 30 km from the plant, Fig. 10.

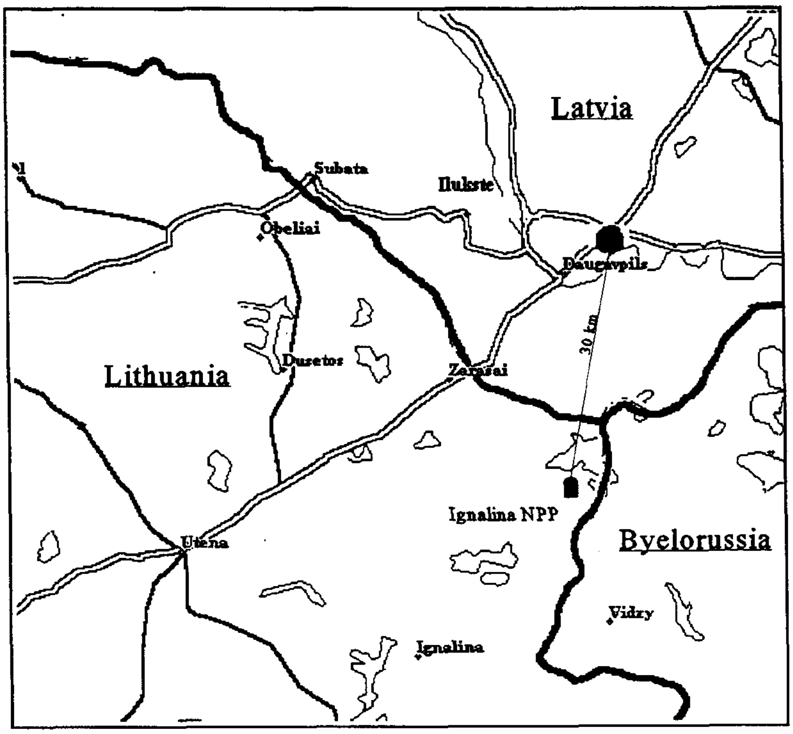

Figure 10. Location of Ignalina Nuclear Power Plant

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.

Table 7. Summary of design data for unit 1 of the Ignalina Nuclear Power Plant

Main data	
Type	RBMK-1500
Thermal power	4800 MW
Capacity, gross	1500 MWe
Capacity, net	1440 MWe
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 °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	21500 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
Primary pressure	70 bar
Total coolant flow	11100 kg/s
Fuel channel inlet temperature	260 °C
Fuel channel outlet temperature	284 °C
Feed water flow rate	2305 kg/s
Feed water temperature	190 °C
Maximum steam content at core outlet	29 %

Turbine plant general

Turbines	2
Steam inlet temperature	280 °C
Speed	3000 rpm
Inlet turbine pressure	65 bar
Inlet temperature	280 °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

5.5 Characteristic differences between BWR and RBMK reactors

The RBMKs could be considered as a 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 the Brunsbüttel and Krümmel 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/cm^3 for Brunsbüttel and Krümmel 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 the 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 greater than 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, see Fig. 11, thus avoiding an initial positive reactivity insertion during operation as was the case at Chernobyl. Further the insertion time of the local emergency rods has been reduced from 18 to 12 seconds.

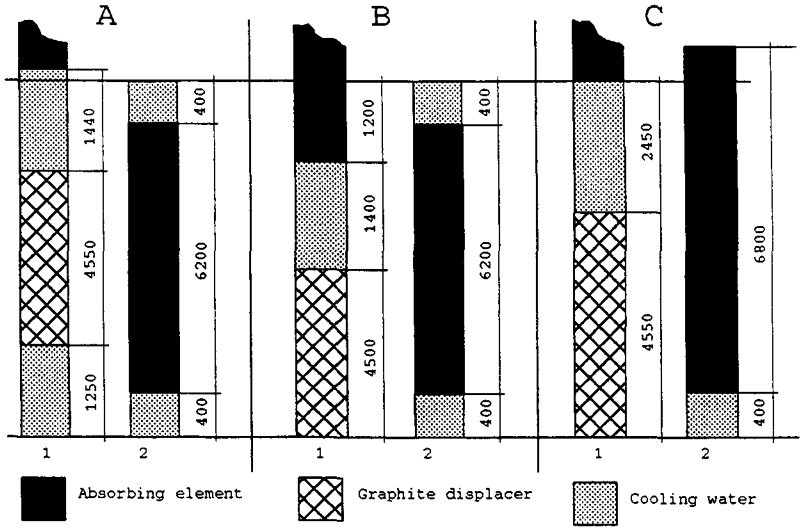

Figure 11. Control rod positions at different levels in the core

A - rod of original design
B - rod of original design partly inserted into core to eliminate the possibility of insertion of positive reactivity
C - improved rod
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.

At a Western BWR refuelling is carried out during the annual shut-down 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 the Krümmel and Brunsbüttel reactors than is the case for the Leningrad and Ignalina nuclear power plants.

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

The lack of a pressure containment for the RBMK reactors is from a safety point of view the most important design difference between Eastern and Western boiling water reactors. The Ignalina Nuclear Power Plant and units 3 and 4 of the Leningrad Nuclear Power Plant are provided with a confinement system, a so-called *accident localization system*. However, the design philosophy of this confinement is different from the Western philosophy. It is not a leaktight building around the reactor but it is a building where the discharged steam and gas mixture in case of a main coolant pipe break is condensed by bubbling through a condenser-pool, purified and released to the atmosphere after a certain delay time, Fig. 12.

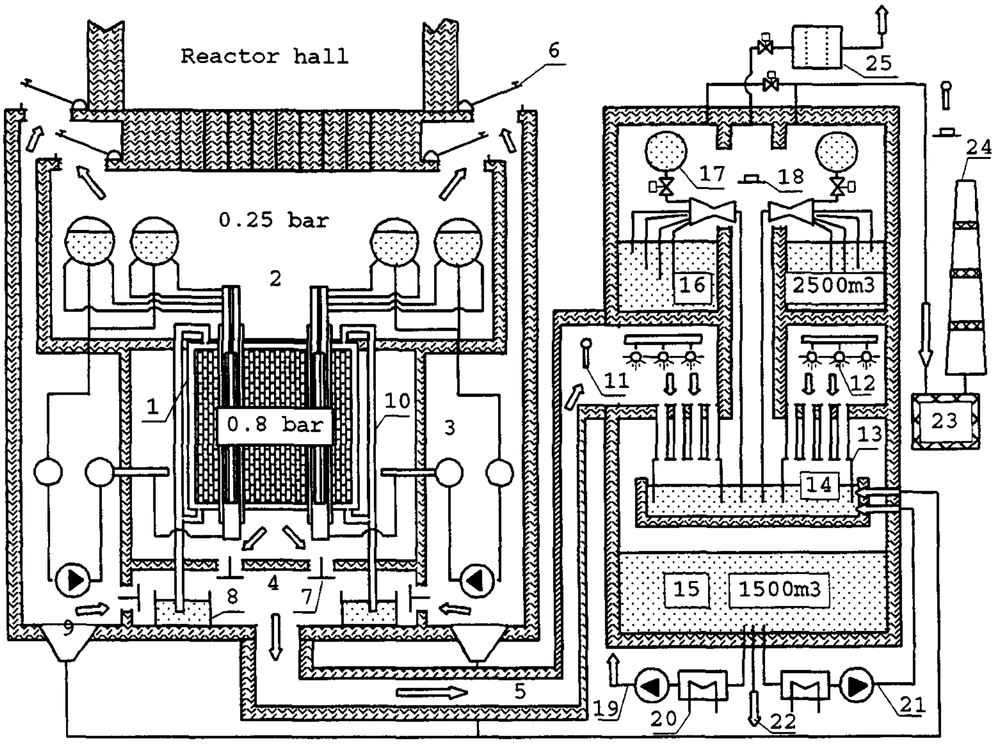

Figure 12. Accident localization system of RBMK reactor

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 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 have no *accident localization system*, so the condensing capacity and the delay time of possible releases are smaller. According to the backfitting plans for units 1 and 2 they will be provided with an *accident localization system* in 1995.

IAEA has stressed the necessity of increasing the relief capacity from the reactor space, so plans are underway to increase the number of allowable simultaneous pressure tube breaks to ten for all RBMK reactors. If the pressure in the tank space exceeds the relief capacity, the upper biological shield will lift and a serious accident might occur.

The containments of the Brunsbüttel and Krümmel BWRs are 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.

6 Marine reactors

The number of nuclear powered vessels and especially submarines operating in international waters is so large, that the authorities and the public are showing increased concern of the potential risk; some of these vessels are also operating near the Nordic coasts. More than 400 of them are military submarines, about 80 of which are continuosly present in the North Atlantic. In 1989 Russia had six nuclear-propelled icebreakers and one combined icebraker-cargo ship in operation, Table 8 [5].

Table 8. Nuclear powered vessels in use or in order (parenthesis) in 1989.

	Sub-marines	**Aircraft carriers**	**Cruisers**	**Ice-Breakers**	**Cargo**	**Sum**
France	12 (4)	1 (1)				13 (5)
Russia	214 (16)	(2)	2 (2)	6	1	224 (20)
UK	23 (3)					23 (3)
USA	157 (12)	5 (2)	9			171 (14)
Total	**406 (35)**	**6 (5)**	**11 (2)**	**6**	**1**	**431 (42)**

The safety authorities and the public in the Nordic countries are becoming increasingly concerned about the potential risks involved, as evidenced by the recent Russian submarine accidents in the North Atlantic.

6.1 Nuclear propulsion plants

A typical layout of a propulsion plant with a pressurized water reactor is shown in Fig. 13.

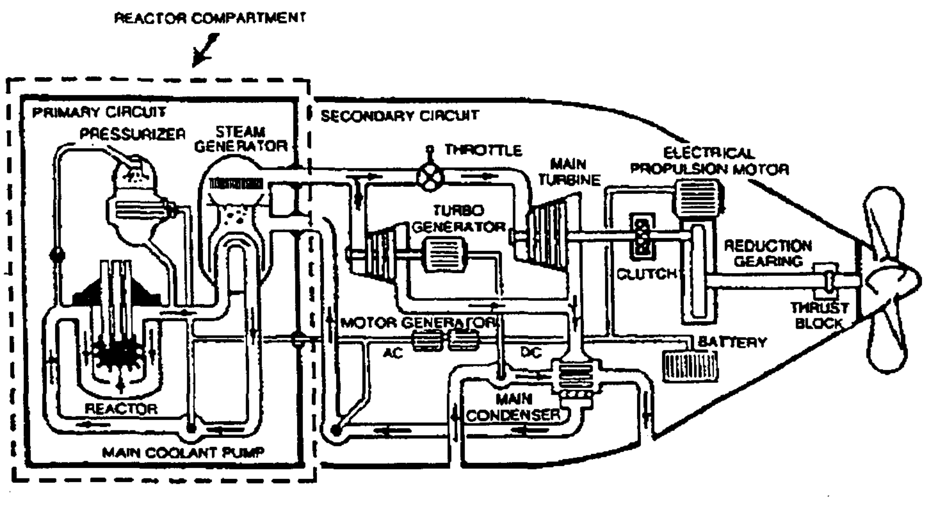

Figure 13. Layout of a dispersed PWR nuclear propulsion plant.

The primary circuit of the nuclear propulsion plant consists of the nuclear fuel core, control rod systems, pressure tank, pressurizer and steam generators. The secondary circuit comprises steam turbines, steam condenser and auxiliary systems. A mechanical clutch and gear system transfers the power to the propeller shaft.

The design shown above is a so-called dispersed loop design in contrast to the design shown in Fig. 14, where the steam generators and main coolant pumps are integrated in the pressure tank.

Since weight and space requirements are of great concern for submarines, the integrated design is mostly applied in this case.

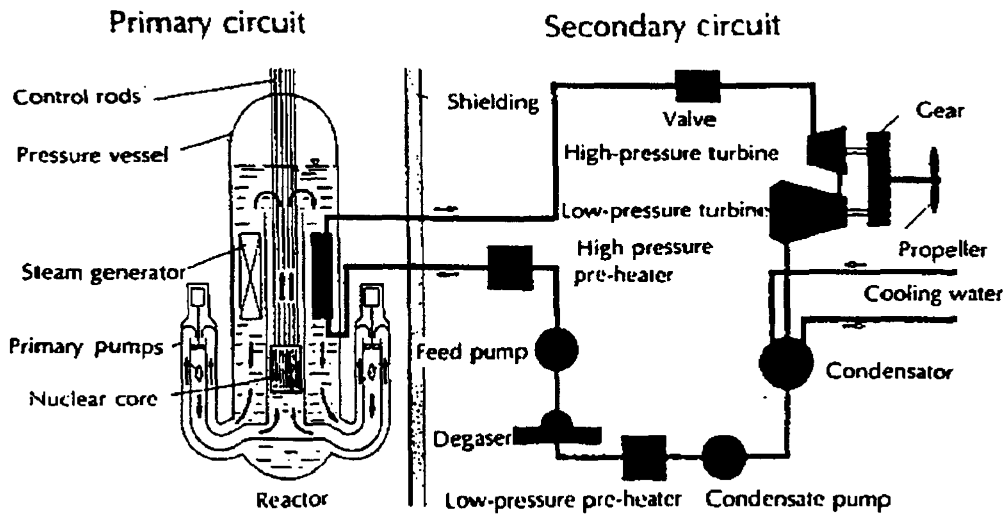

Figure 14. Layout of an integrated nuclear propulsion plant

6.2 Nuclear powered submarines

It is hardly surprising that the first practical application of nuclear propulsion of maritime vessels was in submarines. The use of nuclear power permits the submarines to move submerged for almost any period of time.

The Russian type of submarines are often provided with two nuclear propulsion plants of the PWR type, while the Western types rely on a single PWR plant. Most submarines are equipped with containments as well as emergency core cooling systems.

The type of information needed for safety assessment of nuclear submarines is classified and not available through the open literature. One has to rely on generic information relating to the design, operation and safety assessments of civilian nuclear power plants, land-based and marine-based. To some extent this might be adequate, but vital information on reactor fuel design and operation is lacking.

To perform a risk assessment of a sunken submarine the following information would be needed:

- General design, construction and layout of nuclear submarines
- Design and layout of nuclear plants
- Design, mechanical strength and function of all protective fuel barriers
- Vulnerability of these protective barriers to collisions or accidents, in particular to shocks from explosive charges onboard
- Power level and operating history of the core of the nuclear plant prior to damage

According to the *defence-in-depth* principle the following four safety barriers exist:

- The first barrier is the fuel cladding. In the reactor core there may be more than 10 000 pins or plates and if the cladding should crack the fission products will be released to the primary system. The amount of release depends on fission product volatility and fuel temperature.
- The second barrier is the walls which contain the primary coolant under pressure i.e. the walls of the pressure vessel. It is believed that the pressure vessel itself can withstand considerable shocks, whereas the tubes connecting the pressure vessel with steam generators, pressurizer, etc. are vulnerable. No information is available on how the primary system in submarine plants is designed to resist external shocks. The choice between a loop and an integrated design is influenced by the system's ability to resist shocks.
- The third barrier is the containment structure, that is a tank which contains the entire primary system. It can either be a "full pressure load" design with full-scale blow-down of the coolant in the primary system or it can be a "pressure suppression system".
- The fourth barrier - the safety enclosure or the submarine hull - is intended to prevent the release of radioactive substances to other parts of the vessel. The limits of the safety enclosure are longitudinal and transverse bulkheads of the reactor compartment and the hull.

Fig. 15 illustrates the *defence-in-depth* principle with the four barriers.

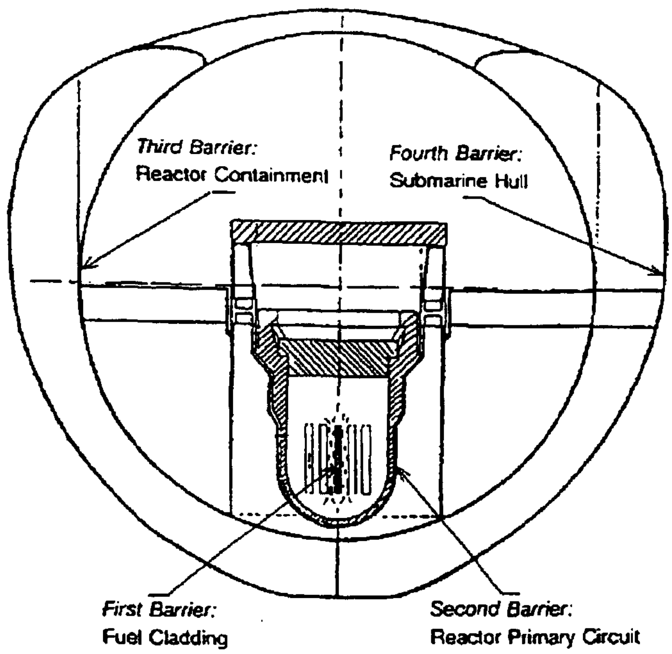

Figure 15. The four physical barriers between the fission products and the environment.

6.3 Civilian nuclear ships

Three countries have built and operated nuclear powered merchant ships, United States, Germany and Japan. The three ships were NS Savannah, NS Otto Hahn, and NS Mutsu. They were all supplied with pressurized water reactors with cluster type fuel elements containing slightly enriched UO_2 and a containment tank around the primary circuit. Data for the three ships are listed in table 9 [6].

Table 9. Data for nuclear merchant ships

	Savannah	**Otto Hahn**	**Mutsu**
Type of ship	Cargo + passenger	Ore carrier	Special cargo
Start of construction	1958	1963	1968
Initial criticality	1961	1968	1974
Full power	1962	1968	1990
Retired	1971	1979	1992
Length, m	182	172	130
Beam, m	23.8	23.4	19.0
Depth, m		145	132
Dead weight, t		15000	2430
Cargo, t	9400	14000	2400
Gross tonnage	15600	16900	8200
Shaft horsepower	22000	10000	10000
Service speed, kn	20	16	16.5
Gross thermal power, MW	76	38	36
Core diameter, cm	157.6	112	114.6
Core height, cm	167.6	115	104
Number of fuel elements	32	12+4	32
Lattice pitch, square, cm	24.7	26.8	17.96
Fuel enrichments, %	4.4	3.7	4.0
Core loading, kg U	7112	2622	2440
Power density, kW/l	23	33	33.5
Average burnup, MWd/tU	7300	7260	5530
Fuel materiale	UO_2	UO_2	UO_2
Number of control rods	21	12	12
Number of coolant loops	2	3	2
Reactor pressure, bar	123	63.5	110

The first civilian nuclear ship in the world was the Soviet icebreaker NS Lenin. Construction started in 1956 and operation in late 1959. NS Lenin was provided with 3 stern screws and the hull was divided into sections by 11 transverse waterproof bulk-heads.

Initially the icebreaker was supplied with 3 identical reactors designated OK-150, a pressurized water reactor chosen because it allows a compact design and because of the negative temperature coefficient of PWRs. The thermal power of the reactors was 90 MW.

According to Western intelligence reports NS Lenin experienced a nuclear related accident around 1966-67. As a consequense of this accident, the original power plant was removed and replaced by two KLT-40 plants, which also are based on pressurized water reactors and have a thermal power of 135 MW. The same type of plant has subsequently been installed in all later Russian icebreakers. The NS Lenin icebreaker was retired in 1989.

KLT-40 is provided with emergency core cooling systems capable of supplying water to the primary system in case of a major leak and it also has a containment.

All Russian nuclear icebreakers are operated by the Murmansk Artic Shipping Company and the KLT-40 plants have been in operation for more than 110 000 hours, corresponding to 125 operating years. The icebreakers have been able to operate continuously for 400 days in the Artic with availability around 76-79 % and only one scram in average per year.

The construction of a Russian icebreaking transport/container ship, NS Sevmorput started in 1984 and the ship was finished in 1988. A substantial amount of information about this ship is available since a Russian safety report has been published in English.

NS Sevmorput has a high strength steel containment with a pressure suppression system, designed to cope with the consequences of a main coolant pipe break. Should the ship sink, a pressure equalizer system will flood the containment until the pressure is the same on both sides of the containment wall.

7 Concluding remarks

Design and safety features for eight nuclear power plant sites, neighbouring the Nordic countries within 100-450 km, have been collected and systematized during the SIK-3 project. The project has provided improved knowledge of especially the Eastern type of reactors, for the benefit of the nuclear authorities within the Nordic countries.

The uniform presentation of the data of each plant is designed to facilitate easy access to the information when needed. It should be stressed, that an evaluation of the safety condition of a particular plant on basis of the reports is beyond the scope of the project. The reports only state the facts and special safety features of each reactor type, providing quick information in emergency situations.

When the project was initiated scepticism was present among some of the participants as to the usefulness of its outcome. However, this doubt disappeared as assistance to the Eastern countries from the West was intensified.

As a matter of fact, the reports on Ignalina and Leningrad nuclear power plants have been utilized by EBRD, the European Bank for Reconstruction and Development, in connection with establishing an assistance program for the RBMK reactors in the former Soviet Union.

The contact net among the Nordic nuclear authorities has also been improved during the project, useful if a nuclear accident involving Nordic countries should occur.

A contact net between the Nordic nuclear authorities and the Eastern nuclear power plant operators has also been established and bilateral agreements will help to sustain and improve this net. In general the hospitality met from the Eastern nuclear power plant operators was better than from most of the Western operators.

However, improvements and backfitting of the Eastern nuclear power plants take place at a fast rate, so an updating of the reports will be needed within few years. Also the rather sparse treatment of the Stade and Brokdorf nuclear power plants ought to be improved in the future.

Finally, the project has made accessible some of the design characteristics of nuclear powered ships operating in the seas close to the Nordic countries.

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Appendix 1 : Disposition of the reports

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Appendix 2 : Rules of distribution for SIK-3 reports

The SIK-3 reports about reactors in neighbouring countries may contain information that was obtained confidentially. This implies some restrictions on their distribution.

The reports are printed in a limited number - preferably in the normal publication series of the authors' institution, but with a NKS cover or front page. Limitations concerning distribution are described in the introductory remarks of each report, taking into account the following rules:

- The reports are available for all participants within the NKS program according to an approved distribution list

- The reports are mainly for use of reactor safety authorities in the Nordic countries and for their national advisory bodies and contributing utilities.

- Provided that the nuclear plant organization described in the report has scrutinized the report, it may under certain circumstances be exchanged with authorities or utilities in third countries for their own restricted use. The same possibility exists if further distribution is desired, e.g. to international organizations.

- Such exchanges must be initiated by one of the Nordic reactor safety authorities, and it can only take place after consultation with the corresponding authorities in the other Nordic countries.

Design and Safety Features of Nuclear Reactors Neighbouring the Nordic Countries

There are several nuclear power plants in operation close to the borders of the Nordic countrie, and in the sea surrounding the Nordic region there is a consideable number of nuclear submarines and other vessels. The authorities responsible for nuclear safety need to have easy access to information about all these reactors. The present report contains an overview of individual reports about reactors neighbouring the Nordic countries, produced in a joint Nordic project of the NKS-programme.

The Nordic Committee for Nuclear Safety Research - NKS
organizes pluriannual joint research programmes. The aim is to achieve a better understanding in the Nordic countries of the factors influencing the safety of nuclear installations. The programme also permits involvement in new developments in nuclear safety, radiation protection, and emergency provisions. The three first programmes, from 1977 to 1989, were partly financed by the Nordic Council of Ministers.

The 1990 - 93 Programme
Comprises four areas:

*	Emergency preparedness	(The BER-Programme)
*	Waste and decommissioning	(The KAN-Programme)
*	Radioecology	(The RAD-Programme)
*	Reactor safety	(The SIK-Programme)

The programme is managed - and financed - by a consortium comprising the Danish Emergency Management Agency, the Finnish Ministry of Trade and Industry, Icelands's National Institute of Radiation Protection, the Norwegian Radiation Protection Authority, and the Swedish Nuclear Power Inspectorate. Additional financing is offered by the IVO and TVO power companies, Finland, as well as by the following Swedish organizations: KSU, OKG, SKN, SRV, Vattenfall, Sydkraft, SKB.

ADDITIONAL INFORMATION is available from
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