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The Norwegian Assistance Program for Increased Reactor Safety in Eastern Europe

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Norwegian Radiation Protection Authority
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Kola Nuclear Power Plant, Ignalina Nuclear Power Plant, Leningrad Nuclear Power Plant, Safety at Nuclear Power plants, WWER-440 reactors, RBMK reactors assistance program.

Abstract

The report gives a summary of the Norwegian assistance program at the Kola Nuclear Power Plant, Ignalina Nuclear Power Plant and Leningrad Nuclear Power Plant and discusses the general safety of the plants.

Referanse

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Sammendrag

Rapporten gir et sammendrag av det norske bistandsprogrammet ved Kola kjernekraftverk, Ignalina kjernekraftverk og Leningrad kjernekraftverk og diskuterer generell sikkerhet ved kraftverkene.

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Summary

For several years Norway has focused on issues related to international nuclear safety. Consequently, under the Norwegian Plan of Action for Nuclear Safety, Norwegian governmental authorities have been actively involved in bilateral co-operation efforts to improve safety at Kola Nuclear Power Plant, Leningrad Nuclear Power Plant and Ignalina Nuclear Power Plant.

Norway`s major involvement began in 1993 at the Kola NPP, and has included projects within several different areas of nuclear safety with a total budget of 124 million NOK[1]. In this report, the projects have been grouped as follows (UD-1999):

1. Reliability of core cooling and emergency power supply;
2. Component reliability and primary circuit reliability;
3. Improved instrumentation and control;
4. Operational safety;
5. Safety studies.

The involvement in Ignalina and Leningrad NPP started 1996 and 1997, respectively. The accumulated budget for the Norwegian efforts at Leningrad NPP is 13.8 million NOK with focus on the following two areas:

1. Training of personnel and prevention of human error;
2. Component reliability and primary circuit integrity.

The Norwegian monetary contribution related to projects at Ignalina NPP is 11 million NOK, with main efforts dedicated to the following two areas:

1. Security and physical protection of the plant;
2. Fire safety

In the early phase of the projects, difficulties were encountered concerning tax exemption and indemnity for the delivery of equipment to Kola NPP. Matters improved successively, following the signing of the Norwegian-Russian Framework Agreement in 1998. Another positive change is the involvement of Russian contractors, who now contribute to the supply of considerable parts of the equipment and services and give a tighter co-operation between Russian and Western suppliers. The feedback from the beneficiaries has generally been positive throughout the project periods.

Introduction

In Norway, the governmental Plan of Action for Nuclear Safety Issues is the major instrument of co-operation with the East-European countries and Russia on nuclear safety issues and problems related to radioactive pollution from past and present nuclear activities. The overall goal in the plan is to protect the public health, the environment and national economic interests from radioactive pollution and pollution due to chemical warfare agents in Russia and other countries in the Eastern Europe. The plan of action identifies four key areas (UD-1999):

1. Safety of nuclear installations;
2. Handling, storage and deposition of radioactive waste and spent nuclear fuel;

[1] 1 NOK = approx. 8 EURO

3. Radioactive pollution in the northern territory;
4. Weapon-related environmental hazards.

All projects to increase nuclear safety at nuclear power plants in Eastern Europe are carried out within the framework of the first key area i.e. safety of nuclear installations.

The first visit of a Norwegian delegation to the Kola NPP took place autumn 1992 and resulted in the start-up of the first phase assistance project in 1993. The project is now in its fourth phase, and so far 124 million NOK[2] for upgrading nuclear safety through bilateral projects has been allocated. In addition, a support to the Nuclear Safety Account administrated by the European Bank of Reconstruction and Development (EBRD/NSA), has been made. This makes Norway the major contributor to enhanced safety at the Kola NPP.

	Year	**Budget**
Phase 1	1993-1995	20 million NOK
Phase 2	1996-1998	40 million NOK
Phase 3	1998-2000	40 million NOK
Phase 4	2001-2002	24 million NOK
EBRD/NSA	1994	2 million EURO
Total	**1993-2002**	**140 million NOK**

The assistance projects at Leningrad NPP and Ignalina NPP were initiated in 1996 and 1997, respectively. From the start-up of these assistance projects, a total of 13 million NOK has been allocated for safety upgrades at Leningrad NPP and 11 million NOK at Ignalina NPP.

[2] 1 US$ = approx. 9 NOK

The majority of the safety issues addressed in the assistance programs are technical. Simultaneously, a large number of non-technical issues are recognised to be of vital importance to reactor safety. Examples of these factors are plant management, operational and emergency procedures, operational practices, training, safety culture and interface with regulatory authorities. The inclusion of such non-technical issues is a relevant factor when planning balanced projects. For example, the delivery of a certain type of equipment to a plant, is usually made in conjunction with the provision of required training that not only focuses on the use of the specific instrument, but also on test metrology and maintenance of the equipment. Spare parts are usually an important part of all equipment deliveries.

In all of the bilateral projects, the Norwegian Radiation Protection Authority (NRPA) has been responsible for overall project management and co-ordination with other donor countries. The practical implementation of the projects has been handled by others. The Institute for Energy Technology/OECD Halden Reactor Project (IFE), with special competence in the reactor field, has been responsible for implementing most of the nuclear technology projects, both at Kola and Leningrad NPPs. At Kola NPP, many infrastructure projects and projects including technology of a wider industrial character were implemented by Kværner Kimek A/S during the first phase of the co-operation and by Storvik & Co A/S during the second and third phases. At Ignalina NPP, all the project have been implemented by Swedpower AB. In addition, a large number of other subcontractors, both Russian and from other Western countries, have been involved.

Following the initial phase of the projects, development towards utilising more Russian-produced supplies, equipment and services can be noted. In addition to the economic benefits of this approach, it also facilitates the licensing process from the Russian authorities. Futhermore, it ensures easier access to maintenance and repair for the beneficiary of the assistance, who thereby becomes less dependent on the donor nation.

Many of the assistance projects have been performed in co-operation with other donor countries, especially Sweden and Finland. A close co-operation with other counties has proved to be a cost-effective way of performing projects when each individual country has limited resources and of optimising the use of national expertise. Close international co-operation has also proven useful for sharing experiences and avoiding duplicate deliveries.

For many of the projects, a stable management framework and long-term involvement have resulted in a step-by-step upgrading of equipment, competence and monitoring routines. For the projects at Kola NPP, the experiences gained during one phase of the project can be used later. Operational experiences from the use of a particular piece of equipment can be applied when planning future projects.

In the planning of projects, on of the constraints has been the enhancement of safety without necessary prolonging the lifetime of the plant. In general, priority has been given to reactors still having several years left of their design lifetime and to projects that improve the safety of more than one reactor. In recent years therefore, most of the assistance to Kola NPP has been focused on reactors3 and 4. Additionally, new projects are planned according to priorities made by the Kola NPP and in understanding with the local department of the Russian reactor safety authority, «Gosatomnadzor».

Although nuclear safety at these plants has improved significantly over the last few years, this does not mean that all of the safety questions at the NPP have been resolved. Modern society demands a higher level of safety compared to the requirements at the time these reactors were built. There is still much important work to be done in order to enhance the safety level of older Russian reactors.

2 Safety issues and reactor design

2.1 The WWER reactor design

The WWER-reactor type is a light water cooled and moderated pressurised water reactor, and can be described as an East European variant of the pressurized water (PWR) reactor, the most widespread reactor type used in the world today. The development of the “WWER family” began in the mid 1950s and the reactor designs are commonly grouped into three generations (NEI-1997). Safety concerns with this reactor type are mainly in connection with the first generation of WWER, of which 18 units were built. Of these, seven units were closed down in the 1980s, including the two smaller prototype reactors at Novovoronish in Russia (PRIS-database, IAEA-1992, IAEA-1994, IAEA-1996a). The other eleven first generation WWER reactors, all WWER-440 subtype 230, have (to a different degree) all been modernised and upgraded with respect to safety, including the two oldest reactors at Kola NPP.

The two younger reactors at Kola NPP, both WWER-440 subtype 213, are second-generation WWER reactors. In total, 17 units of this type were built and one has been shut down. With the exception of the two units at Loviisa in Finland, all of these reactors are located in Eastern Europe. There is an international consensus that these reactors have some safety deficiencies, although the second generation is considered to be safer than the first generation (IAEA-database)

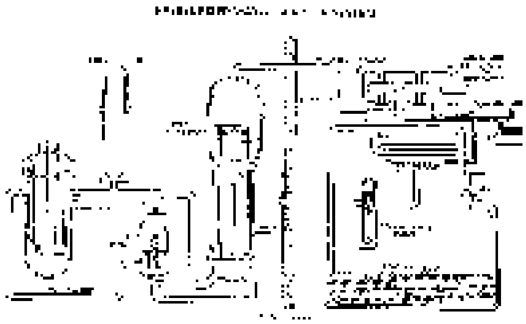

Figure 2-1 Principal diagram of the cooling system of a "standard" PWR.

The principal design of the WWER cooling system is quite similar to the PWR illustrated in figure 2.1, even if some components (especially the steam generator) have a slightly different layout. The reactor core is pressurised and cooled by water circulating between the reactor core and the steam generator. Within the secondary loop (in the steam generator) water is heated to boiling point and the steam is conducted to the turbo generators producing the electricity. The steam in the secondary loop is then condensed in the condensers and heat is transferred to an ultimate heat sink.

Generally, reactor safety is dependent on the control of reactivity in the reactor core, the cooling of the reactor core, the transportation of heat from the reactor core to an outer heat sink, and the emission barriers (Stokke 1997). The control of reactivity in the reactor core of a WWER-440 reactor is not very different to the methods used in western PWRs and is to a certain extent self-regulating since the reactivity of the core decreases if the temperature in the core rises and cooling water evaporates. Control of the reactivity is achieved through the correct positioning of control rods in the reactor core. In an emergency situation, the reactivity of the core can be reduced by the injection of boron water into the primary cooling circuit.

The reactors are equipped with six steam generators each. In the event of a failure or leakage, the steam generators can be individually isolated after the reactor has been shut down. This makes the system less vulnerable if one of the main circulation pumps should fail. However, as with western NPPs, the system is vulnerable if cooling of several of the main circulation pumps should be lost simultaneously, a situation that might arise if electricity from the outside power supply source is lost.

Loss of coolant (LOCAs) would generally be the most probable initiating event of a serious accident on WWER reactors, at least if only internal factors are considered. A probabilistic safety assessment (PSA) for Units 4 at Kola NPP indicates that more than 65% of core melt frequency is attributable to LOCA as the initiating event (Kola-1999). To retain the integrity of the primary circuit, it is important that the coolant pumps do function with no leakages and breakages in the cooling system. Since the cooling circuits operate at different pressures it is essential to avoid leakage between them.

As with all other power reactors, the WWER reactor core must be cooled for some time after the reactor has been shut down due to the residual heat caused by the high radioactivity in the core. A positive feature of the WWER-440 is the large amount of water contained in the primary cooling loop. Thus, the WWER-440 can withstand a loss of coolant for some time before the core begins to melt. At least three incidents involving WWER-440 reactors operating without coolant for a considerable time following a power failure have been described. However, the natural circulation were sufficient to avoid damage to the fuel (Stokke 1999):

- 1977, at Greifswald (in the former GDR): a fire in the turbine hall led to a power failure and LOCA for 6 hours;
- 1982 at Metsamor in Armenia: a fire led to a power failure and LOCA for four hours;

- 1993 at the Kola NPP, a storm led to a power failure and LOCA for 2.5 hours.

The components in the primary cooling loop are subject to constant high pressure. At the same time the metal in the reactor tank is prone to become brittle as a result of constant «bombardment» with neutrons. Compared to most western PWRs,, this is a problem of particular significance for the WWER-440 due to the more energetic neutron spectrum caused by the short distance between the tank and the reactor core. This in turn decreases the ability of the reactor tank to withstand a rapid drop in temperature, a problem which could arise when using non-preheated emergency cooling water. In the WWER-213 model, the likelihood of this problem is smaller because these reactors have 8—10 millimetres of stainless steel lining which weakens the neutron spectrum on the tank. This is not the case for the older 230 model (Stokke-1993). Some operators, amongst them Kola NPP, have taken several measures to reduce problems of tank embrittlement, and among others things have changed the loading programs in order to reduce the neutron flux.

In the event of a reactor accident, radioactive substances must penetrate several barriers before they are released into the environment. First, the substances must be released from the uranium fuel and the cladding surrounding it. Secondly, the radioactive material must penetrate the reactor tank. These barriers are practically the same in the WWER and the western PWR. Western reactors, however, are equipped with an airtight reactor containment intended to prevent further emissions of radioactivity to the atmosphere. The WWER-440/230 reactor is also equipped with a confinement to prevent the release of radionuclides, but this was not designed to withstand the same gauge pressure as the PWR. Furthermore, the WWER-440/230 has nine valves opening at a gauge pressure of 50-65 kPa — allowing potential radioactive pollutants to be released to the outside environment (Stokke 1993, p 38). The WWER-440/213 design is presumed to have a much better capability of containing possible emissions because the radioactivity must pass through a condensation tower before it reaches the atmosphere. In the condensation tower a significant part of the emissions will be washed out and retained as they «bubble» through a high column of water.

The WWER-440 reactors were designed according to former Soviet requirements and have, especially in the unmodified version, several shortcomings when evaluated against modern standards of nuclear safety. The quality of the materials is poorer, especially for older units, and is rarely specified to the same degree as is required for more modern plants. In certain systems the degree of redundancy is poorer compared to modern plants, and in some cases there is a lack of physical separation between redundant systems. The instrumentation of the WWER-440 for managing and monitoring the process itself is insufficient as well as the man-machine interface which has developed significantly in the West. In many cases lower design standards have been accepted with respect to external dangers such as fire, flooding and earthquakes.

2.2 Kola Nuclear Power Plant

The Kola Nuclear Power Plant is located near Polyarny Zory, a town in the southern part of the Kola Peninsula, approximately 200 km from the Norwegian border (Kola NPP web-page: http://www.kolanpp.ru/english/index.html).

Figure 2-2: A view of Kola NPP (Photo NRPA)

The four reactors at the Kola NPP represent two generations of WWER technology. The two oldest reactors, units 1 and 2, are subtype 230, while units 3 and 4 are subtype 213. There are many common features between the reactors with respect to construction and safety philosophy, but there are marked differences in the safety systems between the two reactor types. In several important areas, the 213-subtype design is regarded to be safer compared with the 230 subtype.

Table 2.1: *Specification of units 1-4 at Kola NPP with respect to reactor type, construction year and start of operation.*

	Type	Construction start	Operaton start
Unit 1	WWER 440-330	1970	1973
Unit 2	WWER 440-230	1973	1975
Unit 3	WWER 440-213	1976	1982
Unit 4	WWER 440-213	1977	1984

Upgrading nuclear safety at the Kola NPP has received high priority throughout the 1990s, and significant sums have been invested both by Russian authorities and through western assistance. The top priorities in the modernisation program have been as follows (Antonov 1999, p 29):

- Diagnostics on condition of metals in main equipment and pipelines, implementation of non-destructive testing of metal;
- Replacement of old and obsolete equipment;
- Reduction of probability for common-cause failure;
- Improvement of manuals for normal operation and emergency operation;
- Upgrading of important systems of safety to improve their reliability.

During a speech to representatives from the Nordic countries in September 2001, the chief engineer of Kola NPP, Mr. V. Omelchuk summarised the most important safety improvements for the reactors during the period from 1989-2001:

- Diagnostics and non-destructive testing systems for all four units;
- Process equipment upgrades (SGSV, PSV, MIV, gas-removal, reactor cooling systems, primary pressure control, vessel lifetime) for all units;
- Control and protection systems (NFMS, RPS, ESFAS, APR, CS, ECR, I&C etc) for all four units;
- Reliable power supply (Mobile DG, DC system, stand-by power supply buildings etc) for all four units;
- Upgrade of confinement for all four units;
- Reduction of common cause failures (communications, fire safety, heavy load transportation) for all four units;
- Improved personnel training (full-scale simulator) for units 3 and 4;
- Accident analyses (PSA, safety assessment, fire safety, seismic studies etc.) for units 3 and 4.

In the period 1989—2001, a total of 113 million USD has been spent on safety upgrades at units 1 and 2, and 40 million USD for upgrades at units 3 and 4. Of the total upgrade cost of 152 million USD, 33 million USD originate from technical assistance programs, while the rest comes from Kola NPP's own funds. The donors are EBRD (13.8 million USD), Norway (9.3 million USD), the United States (6.5 million USD), TACIS (1.7 million USD), Sweden (0.8 million USD) and Finland (0.8 million USD).

Kola NPP has planned safety improvement programs for the years 2000—2005 with a total budget of 48 million USD. The programs are primarily targeted towards reactors 1 and 2. The main focus in this program is:

- Complementary emergency feedwater system (all units);
- Upgrades of diesel generator control system (all units);
- Replacement of roof covering;
- ECCS and upgrade of sprinkler system (units 1 and 2);
- Improvement of the reliable power supply for group 2 consumers by means of two additional DG cells (units 1 and 2);
- Upgrade of service water system (units 1 and 2);
- Fire protection of metallic structures in turbine hall (all units);
- Improved tightness of unit 1 and 2 confinement and installation of leakage valves at ventilation system (unit 2);
- Accident location system with jet condensers (units 1 and 2);
- LRW treatment system;
- In-depth safety analyses reports (units 1 and 2);
- PSA (unit 3);
- SG compartment sumps (units 3 and 4);
- Inspection and replacement of obsolete equipment (units 3 and 4).

2.2.1 INES events at Kola NPP

Events concerning nuclear safety at nuclear power plants are classified according to the international INES scale (International Nuclear Events Scale). The lower levels (1-3) are classified as incidents, while the upper level (4-7) are considered accidents (IAEA 2001). Events with no safety significance are classified below scale/level 0 and are termed deviations. Deviations and events in the lower end of the scale occur from time to time at all operating NPPs in the world. Events ranked according to the INES scale should be used with great care as an indication of the safety at any NPP, but the number of safety significant events at Kola NPP has decreased markedly since 1993 and is today at the level comparable to a typical Western NPP.

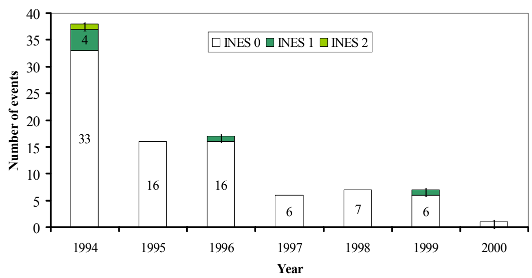

Figure 2-3: Nuclear events at Kola NPP in the period 1994—2000 (Minatom-1998, Minatom-2001).

2.3 The RBMK reactor design

The RBMK reactor is another former Soviet reactor type. Seventeen RBMK reactors were built in Eastern Europe, including the four units at the Chernobyl plant that are now closed down (NEI 1997, PRIS database). These reactors were commissioned in the period between the early-to-mid 1970s to the early 1980s. All the reactors are based on the same construction design, but the later units were constructed to a higher safety standard and are referred to as second-generation units. The main advantage of the second-generation RBMK is the accident confinement system and the advanced functionality of the core cooling system. A third generation of RMBK reactors also exists; however, only one unit was built. Only minor differences exist between the second and third generation RBMK (LEI 1997, p 15).

The RBMK is a boiling water reactor type and its construction is basically different from most other reactor types as being water-cooled and graphite-moderated. The design has both positive and negative features (NEI 1997, IAEA 1996b):

<u>Advantages</u>

- Due to the low core density, the RBMK can withstand a loss of coolant without sustaining damage to its nuclear fuel (in the event of loss of power, for example) for a longer period than other comparable reactors so long as the reactor is shut down;
- The fuel can be replaced during reactor operation;
- Due to the graphite moderator, the reactor can run on fuel with a lower enrichment.

<u>Disadvantages</u>

- The RBMK often lacks adequate reactor containment, but some do have an accident localisation system;
- The reactors have a positive void coefficient[3], i.e. the core effect will increase if cooling water is lost due to vaporisation;
- Limited number of accident mitigation systems;
- Limited redundancy and physical separation of important safety systems;
- Limited capacity for steam suppression in the graphite stack;
- Inadequate fire protection.

In the aftermath of the Chernobyl accident, technical and organisational changes were planned and implemented to improve the operational safety of all RBMK reactors. These changes had the following objectives (Almanas 1997 p. 191):

- Reduction of the positive steam reactivity coefficient to less than 1 β;
- Re-designing of the control rods to increase the prompt shutdown reactivity;
- Installation of programs designed to calculate the effective reactivity reserves and to display the result to the operator;
- Elimination of the possibility to disconnect the emergency protection system while the reactor is in operation;
- Modification of technical specifications for pump operation to ensure that even

[3] For most RMBK reactors, the positive void reactivity coefficient has been reduced by changes in core configuration and the introduction of burnable poison.

at low power a sub-cooling margin is maintained at the reactor inlet.

Reduction of the positive steam reactivity coefficient was accomplished through major changes in the configuration of the reactor core, i.e. by installation of a number of absorber rods in combination with an increase in fuel enrichment. The number of control rods was simultaneously increased. Greater efficiency of the shut down system was achieved by improving the system for inserting the core protection rods, and a new design system for fast scram was installed on all RBMK reactors.

2.4 Ignalina Nuclear Power Plant

Ignalina NPP is located on the banks of Lake Drūkšiai near the town of Visaginas in the north-eastern corner of Lithuania (LEI 1997, p. 11). The power plant has two second generation RBMK-1500 reactor units. The designed thermal effect of each unit is 4800 MW and the designed electrical effect is 1500 MW. The maximum thermal effect has been reduced to 4200 MW for safety reasons. The reactors are among the largest in the world, and the design power is 50% higher than that of all other RBMK units (Ignalina NPP web-page: http://www.iae.lt).

Figure 2-4: An overview of Ignalina NPP.

Table 2.2: Specification of units 1 and 2 at Ignalina NPP with respect to type of reactors, construction year and start of operation.

	Type	Construction start	Operation start
Unit 1	RBMK-1500 (2nd generation)	1977	1983
Unit 2	RBMK-1500 (2nd generation)	1978	1987

After Lithuania achieved national independence, a safety improvement program (SIP) was initiated in order to increase and maintain operational safety at Ignalina NPP until permanent closure. Due to the economic situation in Lithuania, a grant agreement with the European Bank for Reconstruction and Development (EBRD) was signed providing a grant fund of 33 million ECU. These funds supported 20 projects in three areas (Almanas 1997 p. 191):

- Operational safety;
- Near-term technical safety improvements;
- Provision of services.

The key projects included the installation of a low flow and low reactivity trip system, a full scope simulator, fire protection equipment, the replacement of safety valves and motor gate valves and various types of equipment for inspection of the primary circuit. At the same time safety upgrades were achieved through several bilateral co-operative ventures, in which Sweden has made the largest contribution to addressing important safety issues within non-destructive testing, fire safety, quality assurance, physical protection and upgrades of communication systems.

In the periode 1997-1999, the safety improvement program (SIP) was continued and extended by a SIP-2 program. The SIP-2 program was more long-term in character but activities were concentrated within three main areas:

- Design modifications;
- Management and organization development;
- Safety analyses.

2.4.1 INES events at Ignalina NPP

The number of events according to the INES scale is given in Figure 3-2 for the period 1994—2000. In more recent years a reduction in the number of events with safety significance (INES > 0) may be noted. The number of events without any safety significance (INES 0) appears to be at a stable level. However, a direct comparison between the number of INES 0 events at Ignalina and INES 0 events occurring in other NPP's might have very limited value due to differences in registration and reporting procedures.

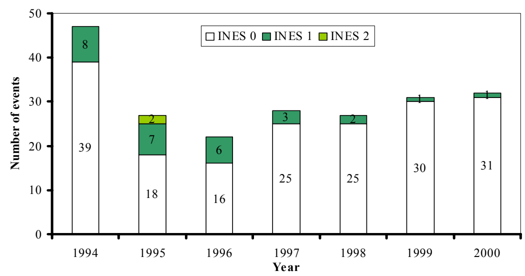

Figure 2-5: Nuclear events at Ignalina NPP in the period 1994 to 2000 (Vatesi 1999, Vatesi 2000).

In its National Energy Strategy (1999) Lithuania has expressed the intention to close down unit 1 of Ignalia NPP by the end of 2005 (LME-1999). Closure of unit 2 will be addressed in the next national energy strategy to be issued in 2004.

2.5 Leningrad Nuclear Power Plant

Leningrad NPP is situated near the town Sosnovy Bor at the Gulf of Finland, approximately 100 kilometres from St. Petersburg. The plant has four RBMK-1000 reactors, each with an electrical effect of 925 MW (Leningrad NPP web-page: http://www.laes.sbor.ru).

Figure 2-6: Entrance of Leningrad NPP (Photo: NRPA)

Table 2.3: Specification of units 1-4 at Leningrad NPP with respect to reactor type, construction year and start of operation.

	Type	Construction start	Operation start
Unit 1	RBMK-1000 (1^{st} generation)	1970	1973
Unit 2	RBMK-1000 (1^{st} generation)	1970	1975
Unit 3	RBMK-1000 (2^{nd} generation)	1973	1979
Unit 4	RBMK-1000 (2^{nd} generation)	1975	1981

The reactors at Leningrad NPP were also upgraded significantly with respect to safety after the 1986 Chernobyl accident. Important upgrades include the following (Antonov 1999, pp. 69-70):

- Modernisation of the emergency core cooling system;
- Replacement of all fuel channels on the two oldest reactors;
- Replacement of drum separator internals;
- Improved monitoring of material and components with modern non destructive testing techniques;
- A full-scope training simulator.

2.5.1 INES events at Leningrad NPP

Figure 3-4 indicates a small increase in the number of registered INES 0 events during the period 1994 to 2000, even through the total number not can be considered especially high. An INES 0 event is without safety significance by definition and is therefore termed a "deviation". It is difficult to draw any conclusions about the safety level at a given nuclear power plant purely based on the number of INES 0 events simply because the absence of safety significance makes the number of INES 0 events a safety indicator of limited value. Simultaneously, it is also known that the number of INES 0 events might be affected by changes in the procedures for reporting and ranking deviations from normal operation.

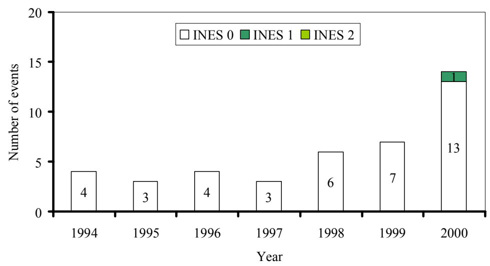

Figure 2-7: Nuclear events at Leningrad NPP in the period 1994 — 2000 (Minatom 1998, Minatom 2001)

3 Kola NPP : Description of the projects

3.1 Reliability of core cooling and emergency power supply

Kola NPP, like all other NPPs in the world, is dependent on a reliable power supply system even when reactors are shut down. The electricity is needed for preserving important core cooling functions and for steering and control functions. In addition, general electricity needs for working light and communication systems must be met. Under normal circumstances this electricity will be supplied from the external net. In generally this is a critical factor for all NPPs with respect to vulnerability in event that the electricity supply from the external net should fail. Kola NPP has eleven large diesel generators for the backup power supply. During a storm in 1993, a serious incident happened occurred as a result of problems with the unit 1 diesel generator following loss of electricity from the external net.

3.1.1 Mobile emergency power plant.

In a worst-case scenario, more than one of the 11 stationary diesel generators may become inoperable due to a common cause failure. A realistic scenario might be a fire where the lack of fire protective walls between reactor 1 and 2 diesel generators can cause simultaneous generator inoperability. Another problem is the distribution vulnerability to failure of the power distribution between the different units or switchboard rooms being to failure.

This situation was early identified in the assistance project as a problem with respect to safety and measures to improve systems for back-up electricity were given high priority.

Figure 3-1: Mobile emergency power plant (Photo: NRPA)

<u>Phase 1</u>

In order to ensure the availability of electric power to the cooling system, in 1995Norway supplied an operable mobile emergency power plant located on two trailers. Trailer 1 holds the diesel generator, a transformer, an electrical distribution panel, cable drums and electronic control and surveillance equipment. Trailer 2 is divided into two sections, a control room with distribution boards and control/monitoring equipment and a room for cable drums and equipment for connection to the existing system at the Kola NPP. Since the risk of a common cause failure is greatest for units 1 and 2, the mobile emergency power plant is located close to these units. For use at unit 3 or 4, the mobile plant can be moved within an hour, giving sufficient time to prevent severe accidents as a result of a cooling failure.

The mobile emergency power plant has several connecting points, located both inside and outside the NPP. This makes it possible to run the pumps even if the internal power distribution net at Kola NPP should fail.

<u>Phase 2</u>

In Phase 2 the fire protection at the mobile plant was enhanced. Extra cables and a cable drum with a load capacity of 500 kW were delivered. The higher load was regarded as especially important in that it permitted the running of periodic tests, including when the reactors are in operation. The cables will increase preparedness through facilitating more rapid connection to the mobile emergency power plant.

3.1.2 Complementary Emergency Feedwater System (CEFWS).

As mentioned above, efficient cooling of the core and removal of heat to an ultimate heat sink is a pre-condition for safe reactor operation, even when the reactor is in shut down mode.

"Feedwater" is a commonly used term to denote the water injected into the steam generators on the secondary circuit to replace evaporated water as the reactor cooling water in the primary circuit is cooling. To prevent damage to the steam generators, which can lead to a leakage of coolant from the primary to the secondary circuit, it is of vital importance that there is sufficient water level in the steam generators at all times. The steam produced is normally used for electricity production, but can alternatively be dumped trough the safety valves. This water is not radioactively contaminated.

In view of the importance of the decay heat function, all reactors at Kola NPP are equipped both with a regular feed water system and an emergency feedwater system. To be prepared for emergency scenarios such as a large fire in the turbine hall where both these systems can be out of operation, Kola NPP decided to install an extra feedwater system independent of those already in place. This was initially a part of the EBRA/NSA project at the Kola NPP, but was dropped because it could not be completed before the end of 1998. However, the project for a complementary emergency feedwater system was included in the Norwegian assistance program of 1998.

The complementary emergency feedwater system (CEFWS) is a fully autonomic system. This means that it is produces its own electricity. Pumps, steering and control systems are completely independent of all other systems on the plant.

The PSA performed by Kola NPP related to internal events for unit 4 indicates an approximately 30 % reduction in the probability of a core melt at reactors 3 and 4 with the installation of CEFWS. The installation of a similar system at the Loviisa NPP in Finland reduced the probability of a core melt by more than 90 %, a figure that also took into account major external events such as fire and flooding.

Figure 3-2 One of the diesel engines/pumps for the CEFWS at Kola NPP.(photo: Storvik)

Phase 3

Based on an initiative from the Finnish reactor safety authorities STUK, Norway, Finland and Sweden decided to cooperate in the delivery of important components to a complementary emergency feedwater system for reactors 3 and 4 at the Kola NPP. The NPP itself contributed to the project by covering the expenses for design, civil construction, water tanks, and the installation of mechanical and electrical equipment.

The contribution from the Nordic countries included the delivery of three diesel pump units, electrical actuated valves and steering and control systems. The expenses were shared between the Nordic countries whereby Norway covered 50-60 % of the costs and the remaining balance was evenly divided between Sweden and Finland.

3.1.3 Upgrade of stationary diesel generators

As mentioned earlier, all nuclear power plants are equipped with large diesel generators to provide backup electricity so as to sustain important safety functions in the event of a loss of external power. At the Kola NPP there are 11 diesel generators, of which five diesel generators provide backup electricity for reactors 1 and 2, and six generators for reactors 3 and 4. The diesel generators are in good condition from a mechanical point of view, but the steering and control systems are obsolete and are thereby less reliable. Occasionally, some of the diesel units have failed to start, as was the case during the INES 3 incident in 1993.

Phase 3

During a prestudy prior to phase 3, a feasibility study was carried out concerning

the upgrade of certain DG units. Due to the implementation of the CEFWS project, the main project was postponed for economical reasons.

Phase 4

At present, a project for upgrading the DG units for reactors 3 and 4 is under preparation. This will be a co-operation between Norway and Sweden, where Norway assists in upgrading of the DG for reactors 3 and 4 and Swedish assistance is devoted towards the DG for reactors 1 and 2.

3.2 Component reliability and primary circuit integrity

The main components of the primary circuit consist of a pressure vessel, pressurizers, cooling pumps, steam generators and pipes. During operation all components of the primary circuit are pressurized. For purposes of reactor safety it is of crucial importance to preserve the integrity of the primary circuit and to avoid loss of core cooling due to leakage of cooling water.

Several factors might contribute to a leak in the primary circuit such as defects in material, poor water quality and mechanical stress. Vibrations in the pumps or other rotating components might cause the latter. The same situation applies to the secondary cooling circuit. A regime for safe reactor operation will have programs for monitoring the degradation of safety of important components as well as the factors contributing to the degradation.

3.2.1 Control of water chemistry.

Corrosion and degrading of materials in the cooling circuits is highly dependent on the water chemistry. Chemical factors commonly monitored are acidity (pH), oxygen, chlorides, fluorides and sulphates. In pressure water reactors, it is common to add boron into the primary circuit to control the reactivity in the reactor core. Other chemicals commonly added to the cooling water are ammonia, hydrazine and potassium hydroxide in order to regulate the pH and optimise the water chemistry. In a PWR, the steam generators represent an interface between the primary and secondary cooling circuit. To avoid chemical degradation of the inventory of the steam generators, it is necessary to monitor the water chemistry in the secondary cooling circuit as is routinely done in the primary circuit. A rapid change in water chemistry might be a first warning of leakage between the cooling circuits.

The background for the first Norwegian projects on water chemistry was the unsatisfactory equipment and routines at Kola NPP for monitoring the water quality. Better equipment for such monitoring was a priority task for the management as well. The Kola NPP management expressed the need to establish good routines for deviation control and correction. In cooperation with the Kola NPP, it was agreed to implement a step-by-step upgrading of equipment and competence in the chemistry laboratory, which would include both equipment and training.

Phase 1

In the first phase, equipment at the central chemistry laboratory was the main priority. A modern ion-chromatograph was provided for swift, reliable and accurate analysis of several different impurities in the cooling water. Spare parts and chemicals for the first two years were also included since such goods are difficult to obtain in Russia.

Prior to installation of the equipment at Kola NPP, chemists from the plant participated in several training programs at IFE-Halden, which covered operation, maintenance and quality assurance. Routines were introduced for comparing measured values against predefined standards. Study tours for chemists from Kola NPP to western plants were also organized.

Figure 3-3 Chemists from the Kola NPP being trained at IFE-Halden (Photo: IFE)

Phase 2

In order to avoid chemical degradation in the secondary circuit, certain chemical substances have to be continuously monitored to ensure compliance with specified limits. As part of Phase 2, equipment was installed enabling continuous monitoring of sodium, conductivity, cation-conductivity and pH in the steam generators at reactor 1. Sensors for measuring the oxygen content in the feedwater at reactor 1 were simultaneously installed. Training of personnel in usage and maintenance was also provided.

In addition, a unit for anion analysis in the cooling water, identical to the equipment delivered in Phase 1, and an ion-chromatograph for extended monitoring of impurities was delivered. Furthermore, extensive training of personnel from Kola NPP, quality assurance, maintenance programs, spare parts and consumption parts were also included in phase 2.

Phase 3

The main objective of Phase 3 was to extend online monitoring of the chemical composition of the cooling water inside the steam generators to include reactors 2, 3 and 4 with respect to water chemistry monitoring. In addition, instrumentation for on-line monitoring of the oxygen content in the cooling water of reactors 2, 3 and 4 was delivered. Phase 3 also contained some training, chemicals, spare parts and different equipment for the ion-chromatographs.

Phase 4

The main content in this phase is the delivery of instruments for surveillance of TOC (Total Organic Carbon) in the cooling circuits with respect to water chemistry. This type of analysis is common at Western nuclear power plants, but is not prevalent in Eastern Europe. Kola NPP does not have the equipment required for performing this kind of analysis. In order to increase the quality and capacity at the chemical department, eight instruments for monitoring oxygen and hydrogen in the cooling circuits will be delivered. In future it is expected that Kola NPP will be able to provide its own spare parts and chemicals, but phase 4 will also include some spare parts as well as additional training of personnel.

3.2.2 Vibration monitoring

Vibration monitoring of rotating machinery is carried out by instrumentation that monitors vibration in the machinery. Vital information can then be obtained, both from the frequency spectrum itself and from the development of the spectrum over time. Vibration monitoring gives users information about a beginning failure of machinery and machine elements at an early stage. This information can be crucial to prevent breakdown in machinery. In the worst case, vibrations being transferred to the pipeline system can lead to the disruption of pipes and a loss of cooling water. Machine diagnosis is a further development of vibration monitoring where measured data are analysed to provide early warning of developing errors in each of the machine components. With machine diagnosis, regular maintenance can be carried out based on diagnostic information and priority given to what is really needed at any given time instead of more or less unplanned repairs. This gives enhanced safety without any «unpleasant surprises». It also permits an optimal use of existing maintenance resources.

Figure 3-4: Test of vibration monitoring equipment in the machine diagnostic laboratory at Kola NPP (Photo IFE)

<u>Phase 1</u>

Vibration monitoring was installed at 756 measuring points on 60 medium-sized pumps and components. Only fixed measuring points were selected in order to avoid errors connected to the use of mobile sensors. Portable data collectors ambulating between the measuring points accumulated data. The results of the vibration measurements were transferred to IFE-Halden every month for analysis and recommendations. Prior to the installation of a vibration monitoring system at Kola NPP, Russian specialists from KNPP's «Diagnostic Laboratory» received comprehensive training at the IFE-Halden reactor.

<u>Phase 2</u>

In this phase, the vibration monitoring system was further developed to provide permanent data collection from the main circulation pumps at reactors 3 and 4.

<u>Phase 3</u>

The main objective of Phase 3 was to enable Kola NPP to perform and implement in-house analysis of the collected data from the vibration monitoring system. In order to prepare for a realistic implementation, the system was extended to include a Russian version of the analysis program SPADE and a local analysis network consisting of two workstations and a new system for the power supply to the diagnosis laboratory. This phase also included upgraded portable monitoring instruments and additional equipment to complement equipment delivered earlier for permanent monitoring of the main circulation pumps at reactors 3 and 4.

Phase 4

During Phases 1-3, Kola NPP implemented a strong regime for vibration monitoring as a result of assistance with new and modern equipment and the training of operators. Spare parts and continuation of training has thusbeen a natural and logical priority in Phase 4. However, the equipment that Kola NPP currently has for vibration monitoring in turbines is also obsolete. It is recognized that in many cases vibrations may cause imbalance in rotating components. This situation will be addressed during Phase 4 by replacement of obsolete equipment with more modern equipment, and Phase 4 will therefore include equipment for balancing rotating components.

3.2.3 Inspection equipment for components and materials.

Safety analyses performed for the Kola NPP and other WWER-440 reactors indicate that pipe rupture may be a realistic cause of a loss of coolant accident (LOCA). Brittleness in the reactor tank is a considerable risk factor as well. Inspections of materials and components have high priority at KNPP and are regularly performed at several key control points, including pipes, bolts and welding seams, as well as certain welding points typical to the WWER-440 reactor tank. Several methodologies and instruments are utilised for this work including ultrasound, eddy currents, visual inspections and x-rays.

Phase 2

As part of Phase 2, two modern ultrasonic instruments with several sound heads of various dimensions for the inspection of welding seams were delivered. One of the instruments has mainly been used in inspection work at the plant, while the other is used for training and further development of inspection routines and verification of quality requirements.

Another part of Phase 2 was a theoretical study of the quality of the materials used in the pressurized tanks at reactors 1 and 2. The Russian Kurchatov Institute carried out this work. The study encompassed calculations of neutron influence on material structure and characteristics.

Figure 3-5: Ultrasonic inspection (Photo: IFE)

Phase 3

This phase consists of the complementation of equipment for ultrasound inspections and a digital ultrasound instrument for particularly demanding inspections. It also includes two analog instruments for routine work and three instruments for monitoring wall thickness in the heat transfer loop. The delivery also contains an assortment of magnetic heads and other accessories. Phase 3 includes a large number of measuring probes for eddy current analysis applicable for the monitoring of the heat exchangers. Monitoring sensors for checking bolts and nuts are also included.

Another problem at Kola NPP reactors is related to the different components and pipes used during construction of the plant. The materials used were produced at different times and with materials of different quality and different composition. Although records do exist in which the different material qualities are specified, a priority in Phase 3 has been the procurement of a portable material analyser for the purposes of material quality control. This in conjunction with maintenance and repair will ensure that the correct qualities and welding solutions are utilised.

As a part of Phase 3, Norway elected to partially finance a mechanical fracture study of the material used in the reactor 2 tank. The main goal of these studies was to investigate to what degree neutron radiation has affected resistance to cracking of the material in the reactor tank and in the welding seams. The conclusions indicated that the reactor tank was in better condition than expected.

For routine checking of tightness in the welding seams, gaskets etc. during repair or maintenance, a portable helium leak-seeking system was delivered.

3.2.4 Mechanical workshop.

It was discovered that KNPP lacked important equipment for mechanized maintenance, hence due to the radiation hazard, high quality manual maintenance work could not be carried out. Consequently in Phase 2 of the Norwegian assistance program, KNPP proposed equipment for an advanced mechanized workshop in order to improve the quality of welding operations in areas of high radiation. This would alleviate the difficulties associated with necessary maintenance work as well as improve radiation protection and reduce radiation doses to KNPP personnel. The proposal was accepted, and machines and equipment for maintenance and mechanized elimination of defects in welding seams were delivered in 1997.

3.2.5 Visual inspection of reactor pressure vessel.

The reactor pressure vessel and its internals are subjected to constant high pressure, temperature transients and corrosive elements in a high radiation environment. Especially neutron irradiation will cause defects in materials, and regular inspections are therefore essential to assure the integrity of the pressure vessel and its internals. Such inspections must be performed by remote cameras operable at radiation levels as high as 3000 Gy/h.

The need for such equipment was evident and planned for in Phase 4 of the assistance program. The inspection equipment supported by the program is capable of and suitable for inspections of components within the pressure vessel in areas that are difficult to access as well as certain structures of the vessel itself. This includes inspections inside openings with a diameter down to 20 mm, inspection of narrow locations where it is difficult to operate a camera and inspections from different angles and directions. The transmitting part of the equipment has a manipulator that can be mounted on the fuel-handling machine and four special purpose camera packages. The receiving part of equipment is operated from the cabin of the fuel-handling machine. This has control units for the manipulator and the cameras as well as a control console with a video monitor. A system for image archiving and review is also included.

3.3 Improved instrumentation and control

In a complex technical environment such as a NPP an important safety aspect is the so-called "human factor". A multitude of factors surrounding the human being in such an environment can affect human behaviour, thinking and actions, which in turn strongly can influence safety. There are several examples of incidents and accidents at nuclear power plants that have been caused by human error. In several cases human error has been a contributing factor to an incident even when technical failure proves to be the main cause. A classic example is the 1979 Three Mile Island accident in the United States where misinterpretation of information in the control room led to the total melting of the reactor core.

Figure 3-6 Control room of Kola NPP (Photo: NRPA)

Instrumentation is generally a weak point at older WWER reactors. At the same time the man/machine interface is an area in which the research of recent years has resulted in major improvements in nuclear safety worldwide.

3.3.1 Safety Parameter Display System (SPDS)

SPDS is a support system used to assist operators in the control room to monitor critical safety functions. The system simplifies and enhances the control room function and reduces the possibility of human error. With the help of SPDS, safety-related information is presented in a short and lucid form on a display screen, giving a better overview of plant safety.

<u>Phase 2</u>

Norwegian and Finnish authorities financed the SPDS system for reactors 1 and 2 on a approximately 50/50 basis. Implementation of the system was carried out as a co-operative effort between IFE-Halden and the Finnish company IVO International. Due to problems concerning exemption from nuclear liability, the system could not be finished during Phase 2, but had to be finalised during phase 3.

<u>Phase 3</u>

Along with finishing the SPDS for reactors 1 and 2, extensive testing for Year 2000 compatibility was carried out, and courses arranged in use and maintenance of the system. Preliminary work for a similar project at reactors 3 and 4 was also performed during Phase 3.

<u>Phase 4</u>

The SPDS for reactors 3 and 4 is similar to the SPDS for units 1 and 2 and will be delivered during Phase 4. The collaboration with the Finnish reactor safety authority, STUK is continued in this project. On the technical level there is a significant difference between this SPDS and the earlier system for units 1 and 2. The latter system has a redundant computer system that will ensure operation of the system, even in the event of a computer failure.

Figure 3-7: On-site acceptance test of the SPDS for units 1 and 2 (Photo: IFE)

3.3.2 TV-monitoring of refuelling operation

Each of the fuel elements in the four reactors at Kola NPP is replaced every third year on average. The normal scheme is to replace a third of the fuel elements every year during power outages. Once the fuel has been removed from the reactor tank, it is kept in the spent fuel pool for a considerable time until the heat production caused by high radioactivity has decreased.

There are several critical moments during refuelling. First of all, the fuel elements to be exchanged must be identified and verification of the fixing devices with respect to reliability. In parallel, verification is needed to ensure that the elements are undamaged. In order to avoid nuclear criticality, it is also important to ensure that the removed elements are correctly placed in the intermediate storage facility. Because of the heavy radiation this is an operation which requires the use of remotely operated underwater cameras.

Phase 1

An underwater camera for monitoring work in the basin for reactors 1 and 2 was installed in 1996 and is functioning as expected.

Phase 2

On the basis of the positive experience gained from Phase 1, another camera for reactors 3 and 4 was installed during Phase 2.

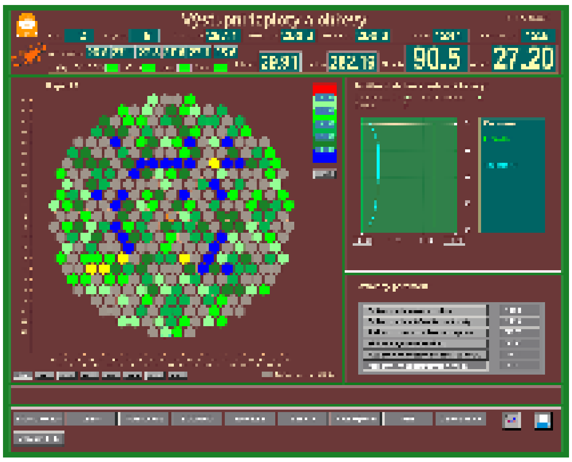

Figure 3-8 A screen image from a SCORPIO-WWER system showing status of reactor core (Photo: IFE)

3.3.3 Core surveillance – SCORPIO.

The reactor core is the main component of a NPP and good core surveillance is obiously closely linked to the safety of the plant. It is especially important to ensure that the fuel is not overloaded, a condition that might result in a fuel failure. The application of new types of fuel to improve refuelling economy demands more accurate core surveillance. The main objectives for increased core surveillance are:

- Better stipulation of the core status in relation to technical specifications and fuel operational limits;
- Realtime calculations of effect distribution by the same physics models applied to core design and safety analyses;
- Precalculated consequences of reactor manoeuvres (control rods, boron, temperature) relevant to important safety parameters before the actual manoeuvres are made.

Phase 4

Phase 4 will include a SCORPIO-WWER system installed in two machines (one on-line and one off-line) in the reactor physics department at Kola NPP. The system will be constructed to allow installation in the control room at a later stage. The core model to be used is the Kurchatov model BIPR.

3.4 Operational safety

3.4.1 Radio/Telecommunications.

Generally at large processing plants, rapid communication between key personnel in crisis and accident situations is an important safety factor. According to the former procedures at Kola NPP, this contact was established by telephone; however, this was recognized to be unsatisfactory, and consequently improved internal communication was identified as a topic for assistance in the program.

Phase 1

The first phase included a system consisting of 300 pagers and a system of radiotelephones with 18 handheld units. These units can also be mounted in cars and communication with key personnel within the 30 km zone can be obtained. Furthermore, 11 emergency power batteries were delivered to ensure a safe power supply to the communication centre.

Phase 2

In the second phase, attention was diverted towards upgrading the old radio communication equipment at the plant itself. For infrastructure serving mobile communications, a system of seven repeater stations and five control boards was selected. In addition, 56 radiotelephones with various accessories similar to those in Phase 1 and 50 pagers for the paging system were also included.

Phase 3

In Phase 3, the radio communications project is continued and extended. A new function to be added is the possibility for control room personnel to overrule on-going calls and make prioritised calls directly to ambulatory personnel. The pager system will also be replaced in order to avoid false alarms. Equipment for testing, error recovery and logging the course of events will also be included. Fire fighting personnel have also received radio equipment.

3.4.2 Computer system for preventive maintenance.

Technical documentation including design documents is a necessity both for safety upgrades and maintaining the existing functionality. In the early 1990s, Kola NPP lacked a system for assuring the quality of the storage and update of such documents. Documentation was archived in manuals kept in different locations of the plant. Often, the original technical drawings were in bad condition due to presence of natural ammonia in the paper which dissolved the ink.

The quality assurance of technical documentation can easily be linked to other systems such as a system for preventive maintenance. Such systems contain functions for failure reporting, prioritising maintenance tasks and documenting maintenance.

Phase 1

Here the project aimed at gathering parts of the technical documentation in a maintenance system delivered by SAMA Software. This system has possibilities for a total solution for continuous maintenance at the KNPP.

Phase 2

In this period of the project efforts were concentrated on acquiring software to organice preventive maintenance and educate KNPP personnel in the use of the software. Similar systems are also in use at non-nuclear facilities such as Norwegian offshore installations, where preventive maintenance routinely is carried out.

3.4.3 Spectroscopy equipment

Any normal reactor operation will lead to some emission of radioactive substances. Although such emissions are considered not to cause any unacceptable radiological consequences, the licensing authorities normally require the NPPs to monitor their emissions. Several types of instruments are used in this monitoring, including gamma and alfa-beta spectrometers. Such spectrometers are not only useful for measuring radioactivity levels, but they can also identify and quantify even small concentrations of radioactive nuclides.

Figure 3-9: HPGe detector used for gamma spectroscopy (Photo: IFE)

Phase 2

Two HpGe gamma spectrometers were provided for, one to be used inside the plant to analyse cooling water and filters etc and while the other would be used mainly to monitor environment samples from the surrounding area of the plant. Equipment for logging, storage of data, standards for calibration, spare parts, consumer parts and training were also included.

Phase 3

With the addition of a third gamma spectrometer and an alpha-beta spectrometer, Phase 3 represented a strengthening of the measuring and analysing capacity. These instruments will mainly be used in the environmental monitoring nearby the plant.

3.4.4 Kola NPP representative at the OECD Halden Reactor Project .

In order to ensure effective co-operation with Kola NPP during the implementation of the assistance program, a representative from Kola NPP stays at the Institute of Energy Technology in Halden. The representative has daily contact with management and experts at Kola NPP and works primarily with the practical implementation of the assistance program. Important tasks are carried out in co-operation with IFE personnel and include preparations of technical specifications and solutions for the projects to be performed as well as administrative tasks covering customs clearance and shipments of equipment. The Kola specialist participates in courses at IFE-Halden, gaining insight and knowledge of safety philosophy as it is applied in the West. Representatives from Kola NPP have been present in Halden throughout most of the period from 1995 to the present.

3.5 Safety analyses

3.5.1 Probabilistic Safety Analyses (PSA)

Probabilistic Safety Analysis (PSA) is considered to be the best way to assess detailed information of the safety level at a nuclear power plant. Results from a PSA (level 1) are expressed in terms of probability of the reactor core melting. Given the contribution from different technical components to the probability of a core melt, PSA is a useful tool for planning and making priorities for safety upgrades. The same applies to the use of PSA results to the overall risk of failure. Another result of interest is the total core melt frequency as an indicator for the total safety of the plant.

A PSA is an extensive and complex task and represents a considerable workload, tens of man-years. The work is usually organized with an independent and external group to review the results. Western nuclear experts are commonly used for this purpose in analyses at Eastern nuclear power plants.

Phase 3

As a part of its safety improvements, Kola NPP will deliver a PSA analysis of reactors 2 and 4. The Norwegian assistance program and the Swedish and American authorities have supported this work. For the PSA on reactor 2, Norway is financing about 5.6 man-years of analytical work to be performed by external Russian technical support organizations. In addition, specialists from IFE-Halden have been participating in the review process of the human factor part of the PSA.

3.5.2 Fire hazard analyses (FHA).

Fire safety at WWER-440 reactors, including the reactors at the Kola NPP, is generally insufficient compared to present standards. Several cases of fire in nuclear power plants have occurred, some of which constitute a severe threat to safety.

Fire hazard analyses of nuclear power plants are complicated tasks. A major output of a FHA is information as to how fire can spread between the different rooms of a plant. The amount of data to be applied in these calculations is extensive, including qualitative and quantitative data of flammable material in every room, details of the ventilation system and building details such as the construction material of walls, floors and doors. Possible routes for fire spread are then compared with the location of important safety equipment. Rooms that will resist a potential fire are then ranked according to the number of redundant systems to ensure the safe shutdown of the reactor.

In certain rooms, such as the turbine hall, specific methods must be utilised. Several different fire scenarios are then applied in calculations to give an indication of the extent and severity of destruction the fire will cause to the systems located in the turbine hall.

<u>Phase 2</u>

In phase 2, Norwegian consultants examined the fire safety in five areas and made recommendations for dedicated fire protection measures. The areas under study were: the diesel generator station, the turbine hall, the control room (reactors 3 and 4), the cable room (reactors 3 and 4) and the room housing the main circulation pumps (reactors 3 and 4).

<u>Phase 3</u>

Several of the Nordic countries assisted in the conduction of a fire hazard analysis for unit 4 at Kola NPP. Norwegian assistance took form as a database server for the collection and analysis of the data. In addition, Norway has financed the data collection and analytical work on the preliminary fire zone definition and an extensive analysis of the turbine hall, both of which was performed by an external Russian technical support organisation.

4 Ignalina NPP: Description of projects

The first Norwegian involvement in safety projects at Ignalina NPP began in 1996. Up to 2001, Norway has assisted in three different bilateral projects at Ignalina NPP with a total contribution of approximately 12.5 million NOK. These projects have been implemented as common projects in collaboration with Swedish International Projects (SIP). Sweden`s long experience in working with safety upgrades at Ignalina NPP contributes greatly to cost-effectiveness.

4.1 Security and physical protection

Nuclear power plants are vulnerable to sabotage and terrorism. Furthermore, nuclear power plants have storage facilities for fuel and radioactive material and other hazardous material that must be protected. In the assistance program concerning Ignalina NPP, physical protection has been a major topic and several projects have been devoted to improving the physical protection of the plant.

Figure 4-1: The perimeter surrounding Ignalina NPP, including some of the (later replaced) systems for physical protection (Photo: NPRA)

4.1.1 Increased physical protection of perimeter

Ignalina NPP is surrounded by a perimeter buffer zone to protect the plant from unauthorised entrance. Since many of the alarm systems and TV surveillance at the perimeter of the plant were obsolete, a project was implemented to replace these systems. In addition, a general upgrade of the fences and other physical constructions designed to prevent unauthorised entrance was carried out by the plant themselves.

Figure 4-2: A representative of INPP (Mr. Velishkovsky) in front of the monitors for the perimeter surveillance system.

4.2 Fire safety

Fire is an event with a relatively high probability to take place in a nuclear power plant. There are several examples of fire in nuclear power plants. The management at Ignalina NPP has been engaged in efforts to improve the fire safety at the plant and several project devoted to these matters have been supported, especially by Sweden. This work includes an extensive fire hazard analysis and several projects to improve both passive and active fire protection systems.

4.2.1 Door project

In order to improve passive fire protection, approximately 600 fire doors were replaced through a common Norwegian-Swedish project. Some of these doors are also a part of the physical protection of the plant and are equipped with alarms and locking systems. The replacement of the fire doors as a first priority was based on the internal fire hazard analysis.

Figure 4-3: Example of a fire door to be replaced (Photo NRPA).

4.2.2 Extended fire safety

The project to further increase the general fire safety is a continuation of earlier projects in this field. This particular project started up early spring 2000 and the main content is:

- Installation of smoke and fire detectors in the diesel generator building;
- Installation of smoke detectors and temperature sensors in the main circulation pump room;
- Replacement of flammable floor coverage in a number of different rooms;
- A new fire alarm central in the main control room.

Figure 4-4: Final inspection of new floor covering.

5 Leningrad NPP: Description of projects

The assistance projects at Leningrad NPP are of more recent date compared to the Norwegian involvement at Kola and Ignalina NPP. It started with a pre-study, mainly carried out over the course of 1998 that prepared for the start-up of the first projects in early 1999. So far, 13.0 million NOK has been allocated to safety upgrades at Leningrad NPP. The main funds have been directed towards the development and implementation of a refuelling machine simulator.

5.1 Training of personnel and prevention of human error

Human error is generally recognised as one of the main causes of incidents and accidents in the nuclear industry; hence the training of personnel is commonly recognised to be a key factor in reducing the risk of human error. In recent years, Leningrad NPP has built up a training centre to educate personnel in the operation of the plant. It is believed that improved competence in the personnel will pay off by reducing the probability for "human error". This training centre includes a full-scope simulator provided by the United States government.

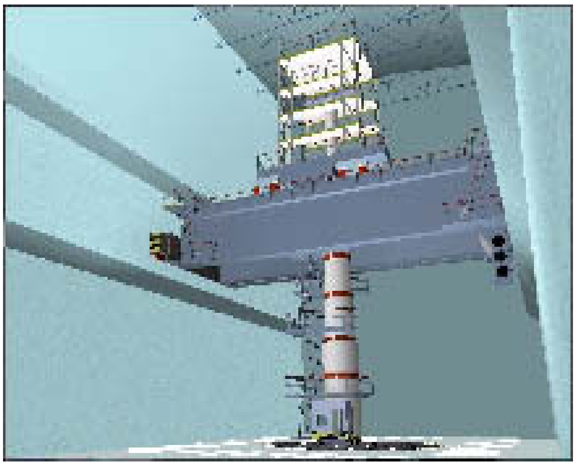

Figure 5-1: Virtual reality picture from the refuelling machine simulator (Picture: IFE).

5.1.1 Refuelling machine simulator

A characteristic feature of the RBMK and the CANDU[4] reactor type, is that these reactor types are designed for changing fuel during operation. An advantage is therefore that the need for periodic power outages is limited compared with other reactor types. Another feature of these reactor types is their suitability and capacity to produce plutonium of high quality. With respect to present RBMK operation, however these qualities are of less importance. The outage periods are often motivated by maintenance and safety upgrades and there seems to be no shortage with respect to plutonium in Russia, it is rather a problem of the opposite with large amount of excess plutonium.

Changing the fuel in a RBMK reactor is considered to be a critical operation as it involves interference with the primary circuit under pressure and at operation temperature. Error occurring during fuel change can result in leakage, loss of pressure, temperature gradients and other mechanical problems. It can influence the reactivity of the core as well. Such a change in reactivity will come in addition to the challenges in maintaining a steady power output

[4] CANDU is a trademark for "Canada Depleted Uranium", a Canadian designed heavy water moderated reactor type.

during normal operation. Due to the complexity of the refuelling process, this operation is performed according to strict procedures. In practice, the fuel is usually changed during the night shift, and only when the reactor has been running stably for a longer period and when no other conflicting activities are taking place. Naturally, the personnel participating in fuel change operations must be well trained. Thus, a system of quality control for maintaining and improving competence will contribute to better safety.

During the spring of 2000, a training simulator for the refuelling machine was a part of the Norwegian assistance program. This simulator uses an approach of "virtual reality" for visualising all major components and operations in the refuelling process. Prior to this, Leningrad NPP had no means of training operators of the refuelling machine.

During the year 2001, further developments in the scope of this simulator began. The aim is to extend the use of the simulator to include other persons participating in the fuel change such as crane operators. Extending the scope also includes a connection between the refuelling simulator and the full scope simulator. This gives the possibility to practice and exercise in co-operation with control room personnel. The system will be extended to include some built-in malfunction scenarios for training procedures for operations in non-normal and emergency situations.

5.2 Component reliability and primary circuit integrity

As described in an earlier chapter above concerning Kola NPP, a wide range of quality control techniques are commonly utilised in nuclear power plants to ensure an acceptable quality of components and welds. In recent years, Leningrad NPP has developed the systems and the capacity for performing such inspections.

5.2.1 Eddy current equipment

In a nuclear power plant heat exchangers represent an interface between different cooling circuits operating at different pressures. A risk might exist for water leakage into a circuit with lower operation pressure, and in a worst case a pressure drop in the cooling system could affect the cooling function of the plant. "Eddy current" is a non destructive testing (NDT)-technique commonly used for inspections of heat exchangers and such instrumentation was part of the Norwegian assistance project. The delivery of equipment also included training courses in use of the equipment and in the manufacture of measuring probes. The Finnish reactor safety authority STUK, with its long history and experience in co-operation with Leningrad NPP, carried out the practical implementation of the project.

5.2.2 Qualification of NDT inspectors

In recent years, Leningrad NPP has developed the skills to perform various measurements to monitor the degradation of components in the primary circuit. Improvements in instrumentation have been followed by a diversity of education and training programmes. In the co-operation with Leningrad NPP, four candidates have been certified to perform ultrasonic testing according to the EN473/NORDTEST certification scheme. This international certification includes examination both on theoretical and practical issues. Leningrad NPP reports that this certification scheme has resulted in improved integrity of the test personnel.

6 Experience and discussion

The basic Norwegian interest in the involvement on safety upgrades at nuclear power plants in Eastern Europe is motivated by the necessity to bring these reactors to internationally acceptable safety standards. Recognising that closure of these plants was not realistic a near future and that improving of safety was likely to be costly, international involvement was considered necessary in view of the economic situation in Russia and Lithuania.

Considerable changes have taken place in East Europe in the entire period of Norwegian participation in nuclear safety projects. These changes have affected the implementation of the assistance projects, their focus and the resource allocations. At an early stage of the co-operation, prior to the signing of the Norwegian-Russian framework agreement in 1998, questions of customs liability and nuclear indemnity became major areas of concern. Several projects at Kola NPP were delayed as a result of such complications.

From the beginning of the involvement in Russian nuclear power plant safety, the growth of a better and stronger licensing and inspection authority, the Gosatomnadzor (GAN), can be noted. This development is considered to be of significant importance for nuclear safety in Russia, but it also presents challenges to the implementation of projects involving Western equipment. This is because Russia has its own national design standards and documentation must be provided in Russian and accepted in order to obtain a licence to use the equipment in a nuclear power plant.

The rather complex and expensive licensing procedures for equipment purchased outside Russia makes it preferable to utilise Russian-made equipment wherever possible. Similar advantages apply to Russian subcontractors, often very competitive in price compared to western contractors, especially for consulting services. Another important consideration is a more cost-effective availability of spare parts and services. Furthermore, from a long-term perspective the use of domestic goods and services in Russia will make the receiver less dependent upon the donor.

The main criterion of success for a nuclear safety project is the net contribution to the improvement of nuclear safety. In general, positive developments in nuclear safety can be noted through a summary of the involvement in a diversity of projects. Among strategies giving positive effect have been the following:

- Selection of projects based on priorities set by the receiving organisation. This ensures project integration with the plant's own internal safety programmes;
- The issues addressed are generally recognised as important to nuclear safety, and the majority of the topics are considered by IAEA to be issues of high safety concern;
- Good mechanisms for contact and feedback from the host countries. The indications are that the projects have been well received and the systems and equipment are in active use;
- An organisational structure whereby the overall project management is performed by an independent organisation without commercial interest in the projects and where the implemented is done by competent companies and organisations;
- Clear contracts and agreements with all involved parties in which responsibilities are clearly stated. These agreements are to a large degree standardised and important factors such as reporting, translations and licensing are taken into account;

- Active co-ordination and co-operation between the Nordic institutions and authorities involved in the assistance programs devoted to better nuclear safety in Russia and East-Europe. This contributes to cost-effectiveness for all parties involved;
- The use of safety diagnostic tools and assessment methodologies such as PSA and FHA. The PSA for unit 2 at Kola NPP have to a large degree verified prioritised safety issues addressed in the Norwegian assistance projects. For some projects, PSA have been used to document the safety significance of a single project;
- The step-by-step approach, which makes adjustments and corrections more flexible during project phases;
- The transfer of knowledge, build-up of local competence and sharing of safety philosophy. These elements address the human factor, which is probably the most important factor for safety in such technical environments. In addition such elements strengthen confidence between the involved partners.

The feedback has generally been positive, emphasising specific and definite results with short implementation times. The strategy to adopt an incremental approach, combined with a long-term and stable commitment from the Norwegian government has been fundamental for the good results that have been achieved.

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List of abbreviations

CANDU	pressurized heavy water moderated and cooled pressure type reactor
CEFWS	complementary emergency feedwater system
DG	diesel generator
EBRD	European Bank for Reconstruction and Development
ESFAS	emergency safety features actuation system
IE	internal event
I&C	instrumentation & control
FHA	fire hazard analyses
LOCA	loss of coolant accident
NDT	non destructive testing
NPP	nuclear power plant
NRPA	Norwegian Radiation Protection Authority
PSA	probabilistic safety assessment
PWR	pressurized water reactor
SIP	Swedish international projects – nuclear safety
SPDS	safety parameter display system
SGSV	steam generator safety valve
STUK	Säteiltyurvakeskus (Finland, radiation and nuclear safety authority)
RBMK	boiling water cooled graphite moderated pressure tube reactor type
WWER	water cooled, water moderated energy reactor
β	reactivity equivalent to the delayed neutron fraction

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