8. SAFETY ANALYSIS

8.1 Background and scope

The original design of the standard series VVER 1000 model 320 reactor was applied in 1978 at Zaporozhe Unit 1. Subsequently, Units 2 to 5 at Zaporozhe NPP, Unit 3 at South Ukraine NPP, Unit 3 at Rivne NPP and Unit 1 at Khmelnitsky NPP were constructed to a similar design. At the time, the design met the regulatory documents that were valid in the former Soviet Union. These standards included OPB-73 (General provisions for assuring safety during design, construction and operation of NPPs, State Committee for Utilisation of Atomic Energy, Ministry of Power and Electrification and Ministry of Public Health of USSR, Moscow, 1973) and associated standards available at the design stage and in parallel with the development of OPB-82.

During construction of these Units, several modifications were introduced to comply with the developing regulations. More recently, Zaporozhe Unit 6 was completed and commissioned in 1996 according to an upgraded design. In 1995, the government of Ukraine reaffirmed its commitment to resume the building of two VVER 1000 i.e. Rivne Unit 4 and Khmelnitsky Unit 2. These units had been between 80 and 90% complete when their construction was halted by the Ukrainian moratorium on nuclear energy in 1990, following the accident at Chornobyl Unit 4.

In terms of technical features, design and construction, VVER 1000 NPP's are more similar to Western PWRs than are any other reactors of Russian design. Generally speaking, the design of the model 320 is consistent with international safety practice. The overall safety objective is to ensure that, for all accidents taken into account in the design, the radiological consequences if any, would be very limited. Therefore, a “defence in depth” (Section 8.2.4) consisting of three leak tight barriers, the fuel cladding, the primary circuit and the containment, has been placed between the fuel and the environment. The use of reliable structures, components and systems, and of redundant engineered safeguard systems, contributes to the achievement of accident prevention which is the first priority of designers.

However, assessments based on the experience of the general designer KIEP and the other organisations involved in the design, have revealed some cases of insufficient reliability of equipment or unsatisfactory quality of manufacture, as well as some deviations from up-to-date regulatory documents or discrepancies with international practice. A modernisation programme has therefore been necessary to achieve a safety level in line with Western safety objectives and practices for both design and operational safety aspects.

A summary of the modernisation programme is given in the “Project Presentation” included in the document package made available to the public. The Project Presentation provides a short description of the safety upgrading measures which aim at compliance with Ukrainian rules and with Western safety objectives and practices. Those measures address the identified safety issues requiring solution. Some of the most characteristic of those issues are listed below as an illustration.

- The potential for primary to secondary circuit leakage caused by possible failure of the steam generator collector, a potential accident specific to the horizontal steam generators which are used with VVER 1000.
- The unreliable insertion of control rods; after an assembly has been in the reactor for three years of operation, the drop time for the control rods exceeded the maximum design value.

- Pressure vessel embrittlement surveillance, a safety concern with respect to maintaining and monitoring the reactor pressure boundary integrity. The specimen containers have been placed in such a way that the vessel ageing follow-up is not optimum.

- Low Power and Shutdown (LPS) conditions accidents which should be analysed comprehensively, according to generic observations from probabilistic safety assessments (PSA) results on different plants worldwide.

- Some shortcomings which reflect deviations from current international practices resulting from lessons from the Three Mile Island (TMI) accident.

- Possible containment sump clogging during a large break loss of coolant accident (LB LOCA) which can be solved by replacement of the thermal insulation of the primary circuit.

- Hydrogen removal from the containment atmosphere under design basis accident conditions which can be solved by installation of a system of detection and after burning.

The following important safety issues need further investigation before final solutions can be developed and implemented, or before compensatory measures can be defined when direct solutions are not feasible.

- Systematic protection of equipment against dynamic loads such as jet forces and pipe whipping. In addition to some pipe restraints and support reinforcements, the Leak Before Break (LBB) concept will be reviewed in order to define the extent to which it will be applied to avoid costly upgrading.

- Fire protection and fire-fighting capability. An overall fire vulnerability analysis will be achieved to define and implement necessary additional improvements.

Some electrical and mechanical equipment has to be improved or replaced in order to upgrade the plant availability, due to quality discrepancies in manufacturing.

The importance of human factors in operation of VVER 1000 makes all operational aspects very significant to safety. Some improvements are necessary in the field of operational aspects. Even if a plant has been designed to be safe, the normative basis in Ukraine does not provide systematic coverage of all issues relevant to safe operation.

The safety analysis report has to be completed to fill the gap between the original design and the most recent standards in force in Ukraine, namely OPB-88 and its associated codes, and to take account of the modernisation programme. It is also necessary to apply international practices and experience feedback from Western NPPs in order to correct the weak points that have previously been identified through a series of works on VVER 1000 model 320 reactors by international or Western organisations such as Riskaudit, IAEA and WANO.

IAEA has produced a generic assessment of design and operational safety issues of the VVER 1000s [8.1] and has managed two specific reviews of the modernisation programme at Rivne Unit 4 [8.2] and Khmelnitsky Unit 2 [8.3]. The main conclusions of these reviews are provided in Section 8.3.

A detailed and independent safety evaluation of Rivne Unit 4 and Khmelnitsky Unit 2 was performed by Riskaudit GRS/IPSN International [8.4] within the framework of a contract with the European Commission; in their respective countries, GRS (Germany) and IPSN (France) are the independent technical safety organisations that support national regulatory authorities.

The upgrading measures of the modernisation programme are described in Section 8.4. These measures aim to improve the safety level of R4/K2 to a safety level in line with Western safety objectives and practices, one of the requirements for a loan grant. They have been defined to:

- eliminate deviations from the most recent Ukrainian regulations; and
- apply feedback from international practices and experience from Western PWRs.

8.2 Safety approach

The following paragraphs consist of excerpts from the IAEA report "Basic Safety Principles for Nuclear Power Plants" n° 75 - INSAG-3, the objectives and principles of which were used by the promoters of the modernisation programme to achieve an updated plant which will comply with the current Ukrainian rules and which will reach a safety level in line with western safety objectives and practices for both aspects of design and operational safety.

8.2.1 General nuclear safety objective

The general objective of nuclear safety is to protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.

8.2.2 Radiation protection objective

The objective of radiation protection is to ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is kept as low as reasonably achievable and below prescribed limits, and to ensure mitigation of the extent of radiation exposures due to accidents.

8.2.3 Technical safety objective

The technical safety objective is to prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, event those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

Accident prevention is the first safety priority of both designers and operators. It is achieved through the use of reliable structures, components, systems and procedures in a plant operated by personnel who are committed to a strong safety culture.

However, in no human endeavour can one ever guarantee that the prevention of accidents will be totally successful. Designers of nuclear power plants therefore assume that component, system and human failures are possible, and can lead to abnormal occurrences, ranging from minor disturbances to highly unlikely accident sequences. The necessary additional protection is achieved by the incorporation of many engineered safety features into the plant. These are provided to halt the progress of an accident in the specific range of accidents considered during design and, when necessary, to mitigate its consequences. The design parameters of each engineered safety feature are defined by a deterministic analysis of its effectiveness against the accidents it is intended to control. The accidents in the spectrum requiring the most extreme design parameters for the safety feature are termed the design basis accidents for that feature.

Attention is also directed to accidents of very low likelihood but more severe than those considered explicitly in the design (accidents "beyond the design basis"). Some of these severe accidents could cause such deterioration in plant conditions to the extent that proper core cooling cannot be maintained, or that damage occurs to fuel for other reasons. These accidents would have a potential for major radiological consequences if radioactive materials released from the fuel were not confined adequately. As a result of the accident prevention policy, they are of low probability of occurrence.

Since the possibility, however small, exists that such accidents could occur, other procedural measures are provided for managing their course and for mitigating their consequences. These additional measures are defined on the basis of operating experience, safety analysis and the results of safety research. Attention is given in design, siting, procedures and training to controlling the progression and consequences of accidents. Limitation of accident consequences requires measures to ensure safe shutdown, continued core cooling, adequate confinement integrity and offsite emergency preparedness. High consequence severe accidents are therefore extremely unlikely because they are effectively prevented or mitigated by defence in depth.

In the safety technology of nuclear power, risk is defined as the product of the likelihood of occurrence of an accident and its potential radiological consequences. The technical safety objective for accidents is to apply accident prevention, management and mitigation measures in such a way that overall risk is very low and that no accident sequence, whether it is of low probability or high probability, contributes to risk in a way that is excessive in comparison with other sequences.

8.2.4 Strategy of defence in depth

Defence in depth is singled out amongst the fundamental principles since it underlies the safety technology of nuclear power. All safety activities, whether organisational, behavioural or equipment related, are subject to layers of overlapping provisions, such that were a failure to occur it would be compensated for or corrected without causing harm to individuals or to the public at large. This concept of multiple levels of protection is the central feature of defence in depth, and it is repeatedly used in the specific safety principles that follow.

Two corollary principles of defence in depth are defined, namely, accident prevention and accident mitigation. These corollary principles follow the general statement of defence in depth.

The defence in depth concept provides an overall strategy for safety measures and features of nuclear power plants. When properly applied, it ensures that no single human or mechanical failure would lead to harm to the public, and even combinations of failures that are only remotely possible would lead to little or no harm. Defence in depth helps to ensure that the three basic safety functions (i.e. controlling the power, cooling the fuel and confining the radioactive material) are preserved, and that radioactive materials do not reach people or the environment.

Defence in depth is implemented primarily by means of a series of barriers which should in principle never be jeopardized, and which must each be violated in turn before harm can occur to people or the environment. These barriers are physical, providing for the confinement of radioactive material at successive locations. The barriers may serve operational and safety purposes, or may serve safety purposes only. Power operation is only allowed if this multibarrier system is not jeopardized and is capable of functioning as designed.

8.2.5 Safety assessment and verification

Safety assessment includes systematic critical review of the ways in which structures, systems and components might fail, and identifies the consequences of such failures. The assessment is undertaken expressly to reveal any underlying design weaknesses. The results are documented in detail to allow independent audit of the scope, depth and conclusions of the critical review. The safety analysis report prepared for licensing contains a description of the plant sufficient for independent assessment of its safety features. It includes information on the features of the site that the design must accommodate. It provides detailed information on the major features of systems, especially of those systems used in reactor control and shutdown, cooling, the containment of radioactive material and particularly the engineered safety features. It describes the analysis of the limiting set of design basis accidents and presents the results.

The safety analysis report and its review by the regulatory authorities constitute a principal basis for the approval of construction and operation, demonstrating that all safety questions have been resolved adequately or are amenable to resolution.

Methods have been developed to assess whether safety objectives are met. These methods are applied at the design stage, later in the life of the plant if changes to plant configuration are planned, and in the evaluation of operating experience to verify the continued safety of the plant. Two complementary methods, deterministic and probabilistic, are currently in use. These methods are used jointly in evaluating and improving the safety of design and operation.

In the deterministic method, design basis events are chosen to encompass a range of related possible initiating events which could challenge the safety of the plant. Analysis is used to show that the response of the plant and its safety systems to design basis events satisfies predetermined specifications both for the performance of the plant itself and for meeting safety targets. The deterministic method uses accepted engineering analysis to predict the course of events and their consequences.

Probabilistic analysis is used to evaluate the likelihood of any particular sequence and its consequences. This evaluation may take into account the effects of mitigation measures inside and outside the plant. Probabilistic analysis is used to estimate risk and especially to identify any possible weaknesses in design or operation that might cause excessive contribution to risk. The probabilistic method can be used to aid in the selection of events requiring deterministic analysis.

The process is repeated in whole or in part as needed later in the plant's life if ongoing safety research and operating experience make this possible and advisable.

8.3 The IAEA generic evaluation and reviews

8.3.1 The IAEA generic evaluation

In March 1996, IAEA produced a report on: "Safety issues and their ranking for VVER 1000 model 320 nuclear power plants" [8.1].

The objective of the IAEA study was to present the safety issues in VVER 1000/320 NPPs i.e. deviations form current recognised safety practices in design and operation judged to be significant by their impact on plants' defence in depth.

The report presents those issues according to their impact on the main safety functions, with each described individually. The report presents the safety issues by area and lists individual issues and their ranking. Altogether, 84 safety issues (71 in the design area and 13 in the operational area) were identified, of which:

- 11 were in Category III which means defence in depth is insufficient, immediate corrective action was necessary;
- 38 were in Category II, which means defence in depth is degraded, action was needed to resolve the issue;
- 22 were in Category I, which means departure from international practices, to be addressed as part of actions to resolve higher priority issues, and
- 13 issues were not ranked because they belong to the operational area and because the ranking criteria are difficult to be used.

Issues in Category IV are of the highest safety concern. Defence in depth is unacceptable and immediate action is required to overcome the issue. No Category IV issue was identified.

The report stated that "the main safety concept of those reactors is similar to the PWR units designed at the same time in other countries... the basic safety concept of defence in depth is realised by general design criteria including the use of redundancy, diversity, independence and fail-safe design". The main safety improvements (listed in full in the project presentation) were considered to be the following and are consistent with the safety issues identified in Section 8:

- to consider the loss of the steam generator integrity;
- to improve the reliability of the insertion of control rods;
- to better monitor the integrity of the reactor coolant pressure boundary;
- to improve the in-service inspection and diagnostic system;
- to develop a reliable safety and safety related equipment qualification system;
- to improve fire protection and fighting capability; and
- to prepare for each plant a comprehensive safety analysis report, which should lead to the preparation of a complete safety analysis report.

That IAEA report also stated that "much of the back-fitting and upgrading work recognised as being required has been or is being performed".

The Ukrainian NRA requested Energoatom and the plants to implement the recommendations made by the IAEA.

8.3.2 The "IAEA review of the modernisation programme at RIVNE NPP Unit 4, from 2 to 12 October 1995" dealing with revision 0 of this programme

The IAEA draft report referenced TC project RER/9/035 September 1995 served as the basis for the review. The contents of this version are practically identical to those of the IAEA final report published later in March 1996 [8.1].

Consequently, the conclusions drawn up during this review are also valid by comparison with the IAEA final report.

The following excerpts from the IAEA official mission report [8.2] are relevant.

- "Out of 71 safety issues in the design area applicable to the VVER 1000/ 320 reactors, 56 issues have been addressed by the proposed measures in the modernisation programme of the Rivne Unit 4".
- "The discussions indicated that in the case of 13 of the 15 unaddressed safety issues, the intent of the IAEA recommendations related to the individual issues is or will be met by actions independent of the modernisation programme".
- “Out of the two remaining issues, the first is related to systems (power operated valves on the Emergency Core Cooling System injection lines) and, the second is related to electrical power supply (ground fault in “Direct Current circuits

8.3.3 The “IAEA review of the modernisation programme at KHMELNITSKY NPP Unit 2 from 10 to 14 June 1996” dealing with revision 1 of this programme

The final IAEA report [8.1] served as the basis for the review. This version contains the same issues as the earlier September 1995 edition. The following conclusions of the review [8.3] for Rivne Unit 4 have been met.

- "Out of 71 safety issues in the design area applicable to the VVER 1000/320 reactors, 68 issues have been addressed in revision l of the modernisation programme of the Khmelnitsky Unit 2".
- "The discussions indicated that in the case of one of the three unaddressed safety issues, the intent of the IAEA recommendations related to the individual issue is or will met by actions independent of the modernisation programme".
- "Out of the 2 remaining unaddressed issues, the first is related to systems (power operated valves on the ECCS injection lines) and the second is related to electrical power supply (ground fault in "Direct Current Circuits")”.

8.3.4 Comments on the results of the two reviews and results following the issue of modernisation programme revision 2, which is the applicable last revision

IAEA did not review the modernisation programme revision 2 (i.e. the last applicable revision). IAEA conclusions drawn up for revision l therefore have to be extrapolated to revision 2. Revision l better complies with the IAEA recommendations than did revision 0. As there is no fundamental difference between R4 and K2, the conclusions of the review of K2 (revision l) are also valid for R4.

Revision 2 complies with the Riskaudit assessment (see Section 8.4) and therefore is more in line with Western safety objectives and practices than was revision 1. Moreover, NRA requested that deviations to IAEA safety issues which remained in revision 2 should be substantiated and, if it was not possible to eliminate the deviation, that a solid argumentation should be developed. The authors of the programme performed a comprehensive analysis presented at the beginning of the modernisation programme to explain how revision 2 met almost entirely the IAEA recommendations. In addition, the authors provided assessments why two issues, not of high safety relevance, remain unaddressed in revision 2:

- IAEA issue "system-8" about control of valves on the emergency core cooling system:

  Control of valves on emergency core cooling system.

- IAEA issue "electrical power-6" about ground faults in the direct current circuit:

  Ground faults in direct current circuit.

For issues concerning the operational safety which are not addressed in the modernisation programme, and noting that setting up emergency procedures based on a symptom-oriented approach is planned in the framework of the programme, the authors state that "issues are being solved or are to be solved in the frame of operational activity of each NPP or in the branch programme". Indeed, Ukraine is developing a generic or branch programme dealing with some VVER 1000 model 320 common safety issues and staff from Kmelnitsky and Rivne NPPs are planning activities to improve safety culture. IAEA publication 75-INSAG-4 1991 demonstrates that safety culture is a fundamental concept that involves all persons and organisations at all levels in nuclear engineering, management, operation and control. The modernisation programme is therefore only one element of the steps necessary to achieve improved safety culture. It is the intention of the project parties to promote safety culture during implementation of the project and during future operation of K2/R4 according to IAEA principles.

8.4 Summary result of the "completion and upgrading" project and of the Riskaudit assessment

The modernisation programme revision 1 was produced in July 1996 by PMG [8.5]. It was then assessed by Riskaudit. Taking into account the recommendations made by Riskaudit in its first assessment, PMG subsequently produced the modernisation programme 2 which was reviewed by Riskaudit. Riskaudit report 120 [8.4] is included in the document package made available to the public. Appendices 1 and 2 of that report present ‘Compilation of the result, evaluation of the modernisation programme RIVNE 4 and KHMELNITSKY 2 Units’ and ‘Safety issues for VVER-1000 comparison IAEA (Issue Book) - RIVNE 4 and KHMELNITSKY 2 modernisation programme’.

The modernisation plan measures recommended by PMG are grouped in ten categories. The most significant of them are:

- core design;
- pressurised components;
- electrical systems;
- instrumentation and control;
- containment;
- internal and external hazards, system analysis;
- accident analysis;
- operational safety with a specific complementary upgrading programme, the so-called “operational programme ” developed by the NPP operators; and
- radiation protection.

Riskaudit was asked to cover the following tasks :

- definition of safety objectives to be met after implementation of the modernisation programme;
- safety evaluation of the modernisation programme;
- assessment of utility (i.e. Rivne and Khmelnitsky NPP) reports on the existing status of the NPPs, including quality of construction and qualification of equipment;
- licensing procedures; and
- conclusion on the safety concept.

Safety objectives were proposed by Riskaudit and approved by NRA. Those objectives are "totally in line with the ones used in Western Europe (and accepted by national safety authorities) as well as with the Basic Safety Principles edited by the IAEA (INSAG 3)" Their goal is to ensure that "for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor, and the likelihood of severe accidents with serious radiological consequences would be extremely small".

The safety evaluation which was performed by Riskaudit [8.4] aimed at:

- verifying the completeness of the proposed modernisation measures;
- checking the acceptability of the measures; and
- assessing the implementation schedule of proposals made by PMG.

Riskaudit confirmed that NRA had provided the necessary documentation concerning the legislative basis, which allowed Riskaudit to verify that Ukrainian regulatory practice was quite in line with Western practices.

In particular, Riskaudit concluded as follows:

"To the condition that Riskaudit recommendations will be taken into account and that all proposed and recommended measures will be properly implemented:

- The construction, management and operation of the plants will be in line with the fundamental principles set out in IAEA documents. These include in particular the IAEA Safety Series No. 75-INSAG-3 and the Nuclear Safety Standards (NUSS) Codes of Practices."

This conclusion should be interpreted in the context of the recognised requirement for a full scope BDBA study (Section 8.6.1).

8.5 Design basis accident consequence assessment

8.5.1 Source term

The maximum design basis accident for K2 was estimated by Kyivenergoproekt [8.6]. In accordance with the design basis and standards, this design basis accident is assumed to be a double-ended rupture at the reactor end of the main coolant pipe (Dequiv. = 850 mm). The relevant source term is consistent with that used for DBA anlaysis of Western PWRs.

In the accident scenario, the following failures were taken into account:

- external power outage at the outset of accident;
- sticking in the extreme top position of the most efficient control element;
- coincidental reactor coolant pipe break with a single failure of an ECCS active component (one of the HP and LP coolant pumps), e.g. caused by a diesel generator startup failure; and
- failure of the mechanical part in one passive element (ECCS tank).

For the assessment of radiological consequences, assumptions were made for a 100% fuel cladding failure with subsequent release of fission products into the primary coolant and containment. Radioactive gas efflux into the environment is a function of containment tightness and time history of overpressure.

Initial data for the analysis were as follows:

- reactor coolant pipe rupture was identified in different compartments of the containment;
- heat removal to walls and components was not considered;
- borated water storage tank at DBA, water t° = 50-60 °C;
- emergency boric acid storage tank, V = 630 m^3;
- containment air volume V = 60 000 m^3;
- number of operating spray systems at DBA (a sprinkler system is used to reduce pressure inside containment at RCP breaks) = 1 (out of 3);
- spray system actuation set point on pressure in the containment = 0.3 kgf/cm^2 (0.029 MPa);
- time from actuation signal to start the pump unit (containment pressure 0.3 kgf/cm^2 (0.029 MPa) to attainment of stable downstream flow of spray solution = 95 s;
- cooling water, t° = 33 °C; and
- free containment air volume where steam air mixture is condensed by the spray system, V = 40 000 m^3.

Table 8.1 summarises the source term corresponding to such an accident. It was assumed that fission product release from the fuel element gas gap into the containment is 100 % (a 100% cladding failure).

Table 8.1
Release to environment for design basis accident [8.4]

Physical and chemical form	Radionuclides	Release through dump valve (TBq)
Noble gases	Kr-85m	10.1
	Kr-87	27.0
	Kr-88	24.5
	Xe-133	153
	Xe-135	23.6
Iodine	I-131	3.2
	I-132	0.11
	I-133	0.86
	I-134	0.056
	I-135	0.28
Aerosol	Sr-90	0.011
	Cs-137	0.037
TOTAL		243

8.5.2 Consequence assessment methodology

Two sets of calculations were performed to investigate the consequences associated with the accident scenario described above. The consequence analysis was carried out by CEPN, Fontenay-aux-roses, France, using the PC COSYMA computer program [8.7] for probabilistic assessment.

In the first set of calculations, a deterministic evaluation was carried out of the dose to a hypothetical individual located at the centre-line of the released plume of radionuclides, under "worst-case" assumptions regarding atmospheric conditions. This dose was then compared with the intervention levels at which emergency actions would need to be taken (Section 5), should such an accident take place.

A second set of calculations considered, in a probabilistic manner, the potential occurrence of a wider range of consequences, including both individual and collective dose, according to statistical sampling of possible weather conditions. The calculations were performed on a grid similar to that used in the analysis of doses from normal operations (Section 7), using the same basic population and food production statistics.

The models used in the consequence assessment bear some resemblance to those used to assess the impact of normal operations (Section 7), since they reflect the same atmospheric phenomena and environmental processes leading to radiological exposure. The main difference is that the accident event itself and the resulting dispersion processes are simulated through time, whereas in normal operations, environmental concentrations are assumed to achieve steady-state or average conditions.

Dispersion of the released plume was evaluated using a Gaussian plume model. For the probabilistic calculations reported here, the plume trajectory was assumed to follow the wind direction, which was updated from the assumed start of the accident, according to a meteorological database representative of conditions at Khmelnitsky NPP.

8.5.3 Design basis accident consequences

Deterministic calculations were performed assuming "worst-case" dispersion conditions, in order to obtain a pessimistic estimate of potential individual doses. The assumed conditions are given in Table 8.2.

Table 8.2
Assumed conditions for a deterministic individual dose assessment

Pasquil Gifford stability category	F
Mixing height	100 m
Wind velocity	2 ms^{-1}
Precipitation	0 ms^{-1}
Release height	0 m
Distance from release point	3 km
Exposure pathways	Cloudshine Groundshine Inhalation Resuspension Skin and clothing

The calculated committed effective doses and doses to the thyroid at 3 km from the plant, as a function of time, for the hypothetical most-exposed individual (assumed to be standing in the open on the centre line of the plume) are given in Table 8.3. The distance of 3 km was chosen as most representative given the presence of the exclusion zone and the population distribution around the site. For comparison with Table 8.3, the lower intervention level for the implementation of protective countermeasures (recommended by ICRP and included in Ukrainian regulatory documents, Section 5) would not be exceeded at the boundary of the 3 km zone, even after 50 years of accumulated exposure. The accumulated doses are well within the requirements applied to DBA elsewhere in Europe and those specified in Ukrainian law (i.e. 50 mSv, Section 5). Nevertheless, it is recommended that a complete deterministic evaulation of the radiological consequences of the DBA, including the ingestion pathway, is completed according to the most recent ICRP recommendations (Section 9.3.2).

Table 8.3
Individual committed dose equivalent from design basis accident

Time	Committed dose equivalent (mSv)	
	Effective dose	Thyroid
1 day	0.01	0.073
7 days	0.034	0.42
1 year	0.066	0.86
50 years	0.070	0.86

Results of probabilistic assessments are summarised in Table 8.4 as mean and peak values of the distributions of total potential collective dose and predicted potential numbers of fatalities. More than 90% of the total potential collective dose to the population within 200 km of the plant, after a 50-year integration period, arises as a result of ingestion pathways.

Table 8.4
Mean peak values of consequences from design basis accident out to a 200 km radius

Health effects	Mean	Peak
Early fatalities	0	0
Collective dose (man.Sv)	1.2	6.2
Total hereditary effects	0.012	0.062

According to these calculations (and consistent with the deterministic calculations of individual doses) accumulated doses are well within the requirements applied to DBA elsewhere in Europe and those specified in Ukrainian law (i.e. 50 mSv, Section 5); there would be no early deaths as a result of the DBA considered. Moreover it is very unlikely that delayed deaths could be observed in any post-accident epidemiological analysis.

8.6 Beyond design basis accident consequence assessment

8.6.1 Source term

In compliance with standards and specifications effective in Ukraine and similar to Western practices, a radiological safety analysis must consider beyond design basis accidents.

A full scope BDBA study for VVER-1000/320 has not yet been performed. The list of BDBAs to be assessed for K2/R4 will be given in the EAP. According to the results of the assessments in terms of source terms, possible design or operational prevention and/or mitigation measures will be defined for implementation in the framework of the modernisation programme, as well as corresponding evaluations of radiological consequences, if necessary.

Preliminary analyses were made for a group of BDBAs for which management measures are being implemented on operating VVER-1000/320 plants, including those that provide for prevention of fuel melting. From BDBAs that have already been considered, an accident allowing major leakage from the primary to the secondary circuit was chosen as the most representative accident. The following represents the results of a preliminary scenario provided by Kyivenergoproekt [8.6].

This scenario concerns a primary to secondary leak (Dequiv. = 100 mm which corresponds to a steam generator header failure) with an open dump valve on a damaged steam generator.

In such a case an instantaneous leak of the SG collector (Dequiv. = 100 mm) is assumed to cause an abrupt pressure drop in the primary loop that, in turn, actuates the emergency reactor protection system in response to low pressure in the circuit, closes turbine stop valves and shuts the turbine driven feed water pump off.

Pressure in the defective SG after closure of turbine stop valves increases to open the dump and upon the failure of the latter to close, further decreases to 8.0 kgf/cm^2 (0.79 MPa) within 30 minutes. At the 100 second, the steam generators are filled with water and the water steam mixture starts to flow in the steam lines. The maximum estimated discharge of the mixture through the dump valve of a defective SG is up to 600 tonnes.

Staff actions (starting at minute 10) aim at a fast cooling down of the reactor with the use of dump valves of the non-defective steam generators to provide reactor cooldown to 100 °C by minute 40.

In the process of the assumed accident, no additional cladding failures occur and the fission product content in the coolant leakage is determined by the design clad failure rate (1 % of gas gap releases and 0.1 % due to direct contact of fuel with coolant) and the spike release of radionuclides from such fuel elements.

Fission products, mixed with the coolant, are released through the dump valve into the atmosphere. During discharge, part of the outgoing coolant turns to steam which carries away some coolant as a finely divided fog.

Table 8.5 provides the source term derived for the above so far assessed BDBA as supplied by Kyivenergoproekt [8.6] (though future re-evaluation is possible as mentioned above.

8.6.2 Consequence assessment methodology

The methodology used for the assessment of the hypothetical most representative BDBA was the same as that used for the DBA (Section 8.5.2).

8.6.3 Beyond design basis accident consequences

As for DBA consequences, deterministic calculations were performed assuming "worst-case" dispersion conditions, to obtain a pessimistic estimate of the potential individual dose. The assumed conditions are given in Table 8.2.

The calculated committed effective doses and doses to the thyroid at 3 km from the plant, as function of time, for the hypothetical most-exposed individual are given in Table 8.6. The lower intervention level for the implementation of protective counter-measures (50 mSv for thyroid) would not be reached at the boundary of the 3 km zone.

Table 8.5
Release to environment for beyond design basis accident

Physical and chemical form	Radionuclides	Release through dump valve (TBq)
Noble gases	Kr-85m	13.7
	Kr-87	40.7
	Kr-88	51.8
	Xe-133	92.5
	Xe-135	23.7
Molecular iodine	I-131	13.3
	I-132	34.0
	I-133	25.9
	I-134	22.9
	I-135	19.2
Organic iodine	I-131	0.67
	I-132	1.70
	I-133	1.30
	I-134	1.15
	I-135	0.96
Aerosol	Sr-90	0.0019
	Ru-106	0.0037
	Cs-134	0.070

Physical and chemical form	Radionuclides	Release through dump valve (TBq)
	Cs-137	0.96
	La-140	0.048
	Ce-144	0.052
TOTAL		345

Table 8.6
Individual committed dose equivalent from design basis accident

Time	Committed dose equivalent (mSv)	
	Effective dose	Thyroid
1 day	0.099	0.99
7 days	0.24	3.2
1 year	0.42	5.1
50 years	0.53	5.3

Results of probabilistic assessments are summarised in Table 8.7. According to these calculations, there would be no early deaths as a result of the most probable BDBA. Where delayed deaths from cancer are predicted, the total number would be extremely small compared with the size of the exposed population and it is unlikely that they could ever be observed in any post-accident epidemiological analysis.

Table 8.7
Mean peak values of consequences from beyond design basis accident

Health effects	Mean	Peak
Early fatalities	0	0
Collective dose (man.Sv)	17.7	129
Total hereditary effects	0.18	1.3

The projected impacts of the BDBA are very low.

The source term corresponding to the "most representative" BDBA and the methodology for assessing consequences will have to be confirmed as part of the safety analysis report that will be submitted to NRA for approval prior to commissioning the plant.

8.7 Conclusions

The proposals for completion and upgrading of K2 and R4, both being the latest VVER 1000 model 320, have been reviewed by IAEA and Riskaudit.

The evaluation of the Modernisation Programme and of its associated ‘branch’ and ‘operational’ programmes leads to the following conclusions.

- The construction, management and operation of the plants will be in line with the fundamental principles set out in IAEA documents, in particular IAEA Safety Series No. 75 (INSAG 3) and the Nuclear safety Standards Codes of Practice.
- Each level of the defence in depth concept will be increased significantly.
- The upgraded plants will be able to achieve a safety level in line with Western safety objectives and practices for both design and operational safety aspects.
- The proposed measures, complemented by those recommended by Riskaudit, are considered to be complete and adequate to cope with internationally recognised safety deficiencies for this type of plant.
- The schedule for modernisation is acceptable from the safety point of view.
- After implementation of corrective measures for weak points already identified, after completion of the proposed plants for inspection and after correction of corresponding weak points, the quality status of the plants will be in line with the quality achieved in Western plant.

The consequence assessment of the design basis accident i.e. a double-ended rupture at the main primary coolant pipe (Dequiv = 850 mm) and for the beyond design basis accident highlights the fact that the accumulated doses are well within the requirements applied to design basis and beyond design basis accidents elsewhere in Europe and those specified under Ukrainian law.

In compliance with Ukrainian and Western standards, a radiological safety analysis considered beyond design basis accidents. The hypothetical scenario concerns a major primary to secondary leak (Dequiv = 100 mm corresponding to a steam generator header failure). The lower intervention level for implementation of protective countermeasures (50 mSv for the thyroid) is not reached at the boundary of the 3 km zone.

8.8 References

8.1 IAEA Safety issues and their ranking for VVER-1000 model 320 nuclear power plants. IAEA.EBP.VVER-05, Vienna, March 1996.

8.2 IAEA review of the Modernisation Programme at Rivne NPP Unit 4 from 2 to 12 - October 1995. IAEA - VVER - SC – 151, March 1996.

8.3 IAEA review of the Modernisation Programme at Khmelnitsky NPP Unit 2 from 10 to 14 June 1996. IAEA - VVER - SC 178, February 1997.

8.4 Riskaudit GRS/IPSN International. Final Assessment Report for the Loan Approval Procedure. Support to the Ukrainian regulatory authorities in licensing activity related to the completion and safety upgrading of R4 and K2 units and the safety upgrading of Zaporozhe 6. Report No. 120, December 1997.

8.5 PMG. Modernisation programme 1. Project Management Group, July 1996.

8.6 Khmelnitsky NPP - Data for Environmental Impact Assessment - UKK000IR Kyivenergoproekt l996

8.7 PC COSYMA - An accident consequence assessment package for use on a PC - CEC report EUR-14916, 1993.