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Comments on the Safety of Khmelnitsky Unit 2 and Rivne Unit 4 in the Frame of the Public Participation Procedure of EBRD

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3.1 Executive Summary

The Ukrainian government has requested financing from the European Bank for Reconstruction and Development (EBRD) and other sources for the modernization, upgrade, and completion of two VVER-1000 model 320 nuclear power plants (NPPs) located at the Khmelnitsky and Rivne sites; the specific units to be completed are Khmelnitsky Unit 2 (K2) and Rivne Unit 4 (R4).

As part of a Public Participation Process which is one of the preconditions for funding according to EBRD guidelines, EBRD has issued for public comment Environmental Impact Assessments (EIAs) for completion of K2 and R4. The EIAs were prepared by Mouchel Consulting Ltd. (an environmental consulting firm with headquarters in West Byfleet, United Kingdom) for the project sponsor (Energoatom, the Ukrainian national nuclear power utility).

There is a lack of project documentation made available in this Public Participation Procedure (PPP) which is not in compliance with international practice. Additional documentation, including references in the EIAs, was requested from Energoatom and was not provided by 2 November 1998.

3.1.1 Project Sponsor and EIA Claims for K2/R4 Safety and Risk Levels

The current report by the Institute of Risk Research (IRR) reviews the K2/R4 EIAs from a safety perspective. The project sponsors claim:

(a) K2 and R4 will be completed to an "*internationally acceptable safety level*";

(b) K2 and R4 will have a safety level comparable to that of similarly aged, but recently relicensed, western plants;

(c) routine discharges of radioactivity from two RBMK units operating at Chernobyl would significantly exceed those arising from operation of K2/R4;

(d) the RBMK reactor is inherently less safe than is the VVER reactor, posing an increased core damage frequency (CDF) for the upgraded K2/R4 units will be close to the value for recently re-approved pressurized water reactors (PWRs) and significantly lower than the corresponding value for the RBMK;

(e) VVERs have a strong leaktight containment, are stable reactors, and physically cannot generate an explosive Chernobyl-type accident. Thus the overall safety level of an RBMK can never be equivalent to that of a VVER-1000/320;

(f) continued operation of Chernobyl Unit 3 has many implications for the final entombment of Unit 4; and

(g) the EIAs have analyzed the "*most representative*" beyond design basis accident (BDBA), and the lower intervention level for implementation of counter-measures would not be reached at the boundary of the 3 km zone using worst-case dispersion conditions.

Of these claims, only claim "e" is partially substantiated in the documentation reviewed. The VVER-1000/320 design is a stable reactor and has a containment. However, as to the portion of the claim that the overall level of safety of the RBMK can never approach that of the VVER-1000/320, this is not substantiated. The reason for this is the high percentage of VVER-1000/320 core damage frequency which is comprised of scenarios which **bypass** the containment, thus negating its value as a risk reduction mechanism. **None of the other claims above are substantiated in the EIAs.**

There are no risk estimates presented in the EIAs, either for the VVER-1000/320 design generally or for K2 or R4 specifically. Similarly, there are no risk estimates presented either for the RBMK design generally or for any of the Chernobyl units specifically. There are thus no risk comparisons between K2/R4 and Chernobyl, nor between K2/R4 and any western PWRs. The EIAs do not define what they mean by "*similar vintage*", nor are any specific recently relicensed EU PWRs identified for a risk comparison. In addition, no comparison is made of the proposed K2/R4 designs with any consistent set of western regulatory criteria, including the International Atomic Energy Agency (IAEA) Nuclear Safety Standards (NUSS). Thus, the EIA statements regarding risk comparability with RBMK reactors and western NPPs as well as comparability with western safety standards are without a basis.

The accident which was analyzed in the EIAs and which is claimed to be a BDBA is in fact a design basis accident (DBA). As analyzed, this accident (which involves a steam generator collector failure with an effective diameter of 100 mm, together with failure of a steam relief valve (BRU-A) to reclose) is fully mitigated by operator action, with core damage limited to typical DBA parameters (i.e. no fuel melting, no severe core damage).

There are a large number of BDBAs, some of which are known from previous VVER-1000/320 probabilistic safety assessments (PSAs) to result in severe core damage together with containment failure or containment bypass. (Indeed, if the analyzed accident is modified by assuming ineffective operator action, it will proceed to core damage under containment bypass conditions.) There is no basis for asserting that the accident actually analyzed is in any way representative of BDBA accidents.

The EIAs do not analyze any Chernobyl replacement options other than K2/R4 completion or construction of fossil-fired units. Other options are readily identifiable, but not discussed. From the nuclear risk reduction point of view one among such options is the use of the same amount of money required for K2/R4 completion to modernize and upgrade Ukraine's **existing** operating VVER-1000/320 units. Such an option should make up for the lost capacity arising from final closure of Chernobyl, would significantly reduce overall risks of a catastrophic nuclear accident in Ukraine, and should additionally result in more of the money being expended in Ukraine and Russia instead of being spent outside these countries in western Europe and the United States.

3.1.2 Most Significant Safety Issues Not Adequately Addressed in the K2/R4 Modernization and Upgrade Program

Based on review of additional project specific documentation (most of them not made available in the frame of the PPP) analysis and discussion by the Institute of Risk Research (IRR) has identified the following as the most safety significant issues which are not adequately addressed in the K2/R4 modernization and upgrade program:

- **Steam generator collector failure** – important due to high Core Damage Frequency (CDF) contribution, high containment bypass frequency (large release frequency), continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), failure to implement an automatic safety system response to the initiating event, failure to replace the steam generators, lack of symptom-oriented Emergency Operating Procedures (EOPs), limited Emergency Core Cooling System (ECCS) water inventory, and lack of adequate compensatory measures at the time of startup.
- **Steam generator tube rupture** – important due to high CDF contribution, high containment bypass frequency (large release frequency), inadequate NDE, continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), lack of symptom-oriented EOPs, limited ECCS water inventory, and lack of adequate compensatory measures at the time of startup.
- **Qualification of atmospheric dump valve (BRU-A) for water and two-phase flow** – important due to containment bypass implications in the event of a steam generator collector failure or a steam generator tube rupture.

- **Probabilistic safety assessment (PSA)** – important because the modernization program is almost completely deterministic, because the upgrade program ignores a number of recommendations from MOHT[1] based on PSA results for VVER-1000/320 reactors, and because the PSAs for K2 and R4 are not scheduled to be completed until after startup.
- **Emergency operating procedures (EOPs)** – important because the existing procedures and the ones which will be in place at startup are event-oriented procedures instead of symptom-oriented EOPs as recommended following the TMI-2 and Chernobyl accidents, because of the high CDF contribution of human errors with event-oriented EOPs, and because of the importance of human actions in mitigating containment bypass accidents which dominate CDF for the VVER-1000/320 design.
- **Fire prevention/fire protection** – important due to lack of previous fire hazards analysis, the lack of fire PSA for VVER-1000/320 (except Temelin, for which the results are not publicly available), and the lack of coverage of fire in K2/R4 PSA until after startup.
- **Seismicity** – important due to low Peak Ground Acceleration (PGA) level for design (0.05g) compared with seismic hazard at 10,000 year return interval (0.17g), lack of seismic qualification of Essential Service Water (ESW) system (multi-unit concurrent accident risk), lack of seismic qualification of ventilation and fire protection water pumps, and lack of seismic PSA/seismic margin analysis until after startup.
- **Geology** – Monitoring of karst phenomena and karst water, consequences of possible accidents for groundwater safety areas (emergency preparedness) and impact of karst activity on pile foundations of R4 not addressed in the MP for K2 and R4. Necessary paleoseismic and seismotectonic studies not included in the MP for K2/R4.
- **Loss of heat sink** – important due to multi-unit concurrent accident potential, dependent failure potential, implications for spent fuel pool severe accidents, possible high CDF contribution from loss of ESW, and lack of improvements in the modernization program.
- **Conservation/requalification and qualification of equipment** – the quality of the requalification program and equipment qualification addressed in the MP is questionable because of the poor quality of conservation measures during the construction halt and the non-availability of large parts of the manufacturing and construction documentation. Furthermore a complete program to qualify equipment under extreme environmental conditions and seismicity is still pending is not planned to be implemented before start-up. This is a deviation from international acceptable practice (e.g. equipment qualification is a precondition for licensing in US plants).
- **Three Mile Island (TMI) requirements** – Their implementation is a precondition for obtaining an operating license for US plants. Not all TMI issues are addressed in the MP. Some are planned to be implemented after start-up.
- **Reactor Core** – it is unclear if the measures to solve control rod jamming addressed in the MP deal with the root causes of this issue. Further studies and operational feedback are necessary. Automatic control of Xenon oscillations and power distribution are not specified in the MP and will be implemented after start-up.
- **Design Base Accident (DBA) and Beyond Design Base Accident (BDBA)** – a more comprehensive spectrum of accidents (including reactivity accidents) should be analyzed than proposed in the MP before start-up.
- **ECCS sump screen clogging** – proposed measures for avoiding concerning the analysis of insulation material behavior under Loss Of Coolant Accident (LOCA) conditions and the implementation of a selected technical solution to ensure residual heat removal under LOCA appear to be appropriate but not sufficient.

[1] MOHT is an association of the following organizations: Atomenergoproject, OKB Gidopress, Kurchatov Institute, VNIIAES, Zarubejatomenergostroy, Rosenergoatom, et al.

- **Logistic and infrastructural preconditions** – due to the lack of financial and industrial resources, the Ukrainian supporting infrastructure is not favorable for the further development of nuclear power. This issue is not addressed in the MP.
- **Presence of fail-open pneumatic containment isolation valves** – the designs use fail-open pneumatic valves as containment isolation valves, which is at variance with western safety criteria and which results in a greatly increased risk of an interfacing LOCA given failure of the pneumatic system.
- **Additional safety issues** – reactor coolant pump (RCP) seal failures, emergency battery discharge time, replacement of 6 kV switchgear, reactor pressure vessel embrittlement, reactor vessel head leak monitoring system, replacement of I&C, containment structure and containment bypass accidents, containment ultimate capacity, man-induced hazards, extreme weather conditions, spent fuel storage, leak before break application to secondary piping, pipeline break impact inside reactor building

3.1.3 Conclusions and Recommendations

- There is no basis in the EIAs to assert comparability of the K2/R4 safety level with western NPPs. In contrary there is significant evidence that K2/R4 exceeds IAEA INSAG-3 targets for CDF and LRF, the latter by a large margin.
- There is no basis in the EIAs to assert risk superiority of K2/R4 over Chernobyl Unit 3.
- The EIAs fail to analyze a severe BDBA; the accident analyzed is a DBA, not a BDBA.
- The EIAs fail to examine any project alternatives except replacement of Chernobyl by K2 and R4 or by fossil-fired units.
- Significant safety issues have been identified by IRR which are not adequately addressed in the K2/R4 Modernization and Upgrade Program.
- A comprehensive treatment of these issues is a precondition to reach the minimum acceptable safety level, formulated by IAEA in the INSAG-3 targets for core damage frequency and frequency of large releases, which is one of the EBRD guidelines for funding.
- Before making a decision to fund the completion of K2/R4, EBRD must – in compliance with its own guidelines (acceptable safety level) – require that Energoatom demonstrates how significant safety issues not adequately addressed in the K2/R4 Modernization and Upgrade Program will be resolved before start-up. Without solving these significant safety issues K2/R4 will not reach the minimum acceptable safety level formulated by IAEA in INSAG 3.
- Completion of K2 and R4 on basis of the present Modernization Program should not be funded by EBRD.
- Instead, it is recommended to assess possible alternatives on the basis of safety and risk comparisons.
- From the risk reduction point of view one readily identifiable and reasonable nuclear option is to fund a program of PSAs, reliability and safety upgrades for the existing eleven operating VVER-1000/320 units in Ukraine.

3.2 Preface

This report was commissioned by the Austrian Government as part of the Public Participation Process for preparing public comments on the Environmental Impact Assessment (EIA) documents prepared by the European Bank for Reconstruction and Development (EBRD) concerning the proposed modernization, upgrade, and completion of VVER-1000/320 nuclear power plants (NPPs) designated as Khmelnitsky Unit 2 (K2) and Rivne Unit 4 (R4).* Khmelnitsky Unit 2 would be completed as part of a two-unit block; Unit 1 is already in operation. Rivne Unit 4 would also be completed as part of a two-unit block; Unit 3 is already in operation, as is a two-unit block of VVER-440/213 reactors. The modernization, upgrade, and completion of K2 and R4 has been proposed by the Ukrainian government (in accordance with a Memorandum of Understanding executed between the Ukrainian government and the G-7/European Union) as an alternative to continued operation of two RBMK-1000 units at the Chernobyl** station (Units 1 and 3).

The Chernobyl station is the site of the disastrous accident in April 1986 at Unit 4, which resulted in a large release of radioactivity to the environment and the abandonment of a 30-kilometer radius around the station. Unit 2 (a first-generation RBMK NPP) at the Chernobyl station was shut down following a fire in the turbine building on 11 October 1991; Unit 3 (a second-generation RBMK) remains in operation; Unit 1 (also a first-generation RBMK) is potentially operable, but has been shut down since 30 November 1996 as part of Ukraine's response to the G-7/European Union Memorandum of Understanding.

The VVER-1000/320 design proposed for K2 and R4 employs a four-loop pressurized water reactor (PWR) housed in a prestressed concrete containment. The VVER-1000/320 design shares design principles similar to western PWRs (such as defense-in-depth and containment), however there are some very significant differences in execution of these design principles as well as some notable departures from specific western regulatory criteria (including International Atomic Energy Agency Nuclear Safety Standards) which raise safety concerns about the generic design.

The RBMK design at Units 1, 2, and 3 of the Chernobyl station is a vertical pressure tube reactor, cooled by boiling light water and moderated by graphite, with the primary system housed in a reactor building incorporating a pressure suppression confinement system (Unit 3 only). Safety concerns have also been identified for the RBMK design, particularly the lack of a containment as well as safety system deficiencies (particularly in first-generation RBMKs such as Chernobyl Units 1 and 2).

The current report by the Institute of Risk Research (IRR) of the Academic Senate of the University of Vienna is focused **primarily** on ***safety issues*** associated with the proposed commissioning and operation of K2 and R4. **Secondarily**, the report addresses some public comment ***process*** issues related to the review of the K2 and R4 EIAs. The report also addresses potential options to completion of K2 and R4 which were not discussed in the EIAs.

3.3 Introduction

3.3.1 EBRD Public Participation Process

The Ukrainian government has requested financing from the European Bank for Reconstruction and Development (EBRD) for the modernization, upgrade, and completion of two VVER-1000/320 nuclear power plants (NPPs) located at the Khmelnitsky and Rivne sites. The specific units to be completed are Khmelnitsky Unit 2 (K2) and Rivne Unit 4 (R4). The specific proposal is to complete and upgrade K2 and R4 in exchange for permanent closure of the Chernobyl nuclear station. This proposal arose out of a 20 December 1995 Memorandum of Understanding executed between the G-7/European Union and the government of Ukraine.[2]

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* The Rivne station is also known as "Rovno". The spelling as "Rivne" is consistent with the usage in the EIA for that facility.

** Previously, the spelling was "Chernobyl" or "Tschernobyl". In recent years, the above spelling was adopted internationally at Ukrainian request, and is used here for that reason.

[2] The G-7 countries are Canada, France, Germany, Italy, Japan, the United Kingdom, and the United States. The countries of the European Union (EU) are Austria, Belgium, Denmark, Finland, France, Germany, Greece, Ireland, Italy, Luxemburg, the Netherlands, Portugal, Spain, Sweden, and the United Kingdom.

As part of its policies and procedures, EBRD has issued for public review and comment Environmental Impact Assessments (EIAs) on the completion of K2 and R4. The EIAs were prepared by Mouchel Consulting Ltd. (based in West Byfleet, United Kingdom, with an office in Kiev, Ukraine), for Energoatom, which is the Ukrainian national nuclear utility. The EIAs (Mouchel 1998a; Mouchel 1998b) were made available for public review and comment by 15 December 1998 without EBRD comment or endorsement.

The EBRD has issued **Environmental Procedures** which contains an annex concerning "Environmental Screening Categories". Under the provisions of the annex, the K2/R4 project is an "A" level operation (EBRD 1998: Annex 4). As such, it is the responsibility of the project sponsors (i.e., Energoatom) to "*provide sufficient environmental information to the Bank to enable its Board of Directors to make a decision*" (EBRD 1998: Section 3).[3] The EBRD **Environmental Procedures** specifically state that with respect to "A" level operations (such as K2/R4), the project sponsor "*must ensure through a thorough appraisal that all key issues, and the role of the public in the appraisal, have been identified*".[4] Further, EBRD states (EBRD 1996a: 8), "*The public requires adequate information on the environmental aspects of an operation in order to comment.*"

The current report by the Institute of Risk Research (IRR), prepared for the Austrian Government, reviews the K2/R4 EIAs from a safety perspective. The report also addresses some EIA process issues related to the EIAs and their handling by the EBRD and the project sponsors.

Especially it has to be stressed that there is a lack of relevant project documentation which has been made available in the Public Participation Procedure (PPP). Additional documentation, including references in the EIA, was requested from Energoatom and was not (as of 2 November 1998) been provided to IRR (see Attachment 4). Access to some of the relevant documents which are the basis for this report (see references quoted in this report), were obtained from sources outside the PPP. Among other open questions caused by this lack of information is that the status of the Modernization Programme (MP) for K2/R4 and other upgrading programmes for Ukrainian NPPs and their possible interference is unclear.

According to international practice the project documentation should be made available in the frame of a PPP. IRR notes that in other countries (specifically the United States), it is **legally required** that documentation cited in environmental impact analyses (EIAs) be made publicly available at the same time that draft EIA documentation is issued for public comment.[5]

3.3.2 Role of the EIAs

The EIAs are required by EBRD policies and procedures to provide sufficient information to the EBRD to enable its Board of Directors to make a sound decision. In addition, the EIAs are required to provide a thorough appraisal of all key issues associated with the project. Adequate information on the environmental aspects of a project is a prerequisite for the public in order to prepare comments.

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[3] If information is to be "*sufficient*", in the view of IRR the information must be adequate to demonstrate the truth of the matter asserted. For example, it is not enough to **assert** that the safety level of K2/R4 will be equivalent to that of western NPPs. In order for the information to be "*sufficient*", the information must **demonstrate** that this equivalence exists. If information is relied upon that is not part of the EIAs, this information must be made publicly available for review in order that its validity can be tested by reviewers of the EIAs. Thus, it is not enough to reference a safety analysis report on K2 or R4; the report referenced must be available for review if it is relied upon for the truth of matters asserted in the EIAs.

[4] "*Thorough*" means "*carried out completely and carefully*" or "*painstakingly careful*" (**Collins English Dictionary and Thesaurus**, 1995).

[5] One of the authors of the IRR report (S. Sholly) knows this to be a fact based on his personal experience as a preparer of the accident analysis section of environmental impact statements for USDOE facilities.

The EBRD Board of Governors will use the final EIAs, as well as an assessment of how well the final EIAs respond to the public comments on the draft EIAs, as part of their basis for deciding on whether to fund the proposed project.

According to the EIAs (Mouchel 1998a: 0.1; Mouchel 1998b: 0.1):

> *The objective of the EIA is to provide Energoatom's possible financial partners (e.g., EBRD, EURATOM) with an assessment of the extent to which environmental and radiological impacts associated with the proposed project have been addressed to date, or will be addressed during further development of the project. The EIA also provides a basis for the continuing public consultation process.*
>
> *The EIA is based on an earlier study, supplemented by information that has been obtained, or comments that have been provided by, various parties subsequent to that earlier study. The EIA work is ongoing, particularly concerning the development of an Environmental Action Plan (EAP) that will be covenanted into the project financial and legal documentation.*
>
> *The EIA provides a factual account of the legislative background, the existing site, the proposed project, radiological protection arrangements, nuclear safety issues, and potential discharges of radioactive and non-radioactive materials to the environment. For the identified potential discharges, it provides the results of an assessment of their radiological and environmental impacts, taking into account both normal operation and abnormal conditions. These impacts are compared where possible with those that might arise from the base case alternative, i.e., maintaining the Chernobyl NPP in operation. Measures are identified to mitigate possible environmental and radiological impacts. An Appendix provides a summary of public consultation activities that have been undertaken to date.*

3.3.3 Rationale for K2/R4 Completion – Replacement of Chernobyl Units 1 and 3

The rationale stated by the project proponents for completion of Khmelnitsky Unit 2 and Rivne Unit 4 is that their capacity is needed to replace Chernobyl Units 1 and 3 which are to be shut down under the terms of a Memorandum of Understanding negotiated between the G-7/European Union and the government of Ukraine. This issue and related claims made by the project proponents, are addressed in more detail in the sections which follow.

3.3.3.1 Project Sponsor Claims for K2/R4 Safety

The project sponsor (Energoatom) and its agent (Mouchel)[6] have made several claims regarding the safety of Khmelnitsky Unit 2 and Rivne Unit 4. These claims are cited and discussed below:

- **CLAIM: The project will complete Khmelnitsky Unit 2 and Rivne Unit 4 to an "*internationally acceptable safety level*" (Energoatom 1998: 3).**

  DISCUSSION: This claim has not been substantiated, and cannot be substantiated until a probabilistic safety analysis (PSA) has been performed. Such an analysis for K2 and R4 is not scheduled to be performed until after startup.

  Energoatom identifies no **specific** internationally acceptable safety level (either a set of consistent regulatory criteria or a set of probabilistic safety targets). At a bare **minimum**,

[6] Mouchel is acting (at least indirectly, if not in fact) as an agent of Energoatom. The EIAs were commissioned by the European Commission (EC), which is one of the parties to the Memorandum of Understanding between G-7/EU and the government of Ukraine. Energoatom is a Ukrainian state-owned and state-operated monopoly. The EIAs were not paid for by EBRD, and EBRD must recognize the advocacy nature of Mouchel's relationship vis-a-vis Energoatom. The EIAs were paid for with TACIS funds; TACIS is an EC program.

Energoatom[7] should have provided a comparison with the IAEA NUSS series of standards, as well as a comparison with the INSAG safety targets (i.e., core damage frequency < 10^{-4} per year and large release frequency < 10^{-5} per year). No such comparison has been provided, nor has a comparison with any other comprehensive set of western safety standards (e.g., German KTA, French, or USNRC) been provided either by Energoatom or by Mouchel in the EIAs. Neither Energoatom nor Mouchel has no plans to perform a probabilistic safety assessment, from which a comparison to the INSAG safety targets could be made, until **after** Khmelnitsky Unit 2 and Rivne Unit 4 are commissioned. No comparison has been provided of the Khmelnitsky Unit 2 or Rivne Unit 4 design to PSA study results on other VVER-1000/320 units.

It is well recognized that the VVER-1000/320 generic design has a number of safety deficiencies. The IAEA has characterized eleven of these issues as "*Category III*", meaning the issues are "*of high safety concern*", that "*defence in depth is insufficient*", and that "*immediate corrective action is necessary*" (IAEA 1996a: 10, 32).

Although the IAEA indicates that some of the safety deficiencies were identified, at least in part based on PSA studies, an issue-by-issue review of the IAEA safety deficiency list shows clearly that only **one** out of the **eighty-four** safety deficiencies identified by the IAEA mentioned PSA as a basis. Moreover, even the basis for this issue (AA-8, *Accidents under low power and shutdown (LPS) conditions*) cited **generic** observations of PSA studies made for different plant types worldwide, rather than the results of **VVER-1000/320** PSA studies.

The IAEA, in 1996, stated that final PSA results for VVER-1000/320 units were "*not available at WWER-1000 NPPs*" (IAEA 1996a: 119). **Clearly**, the IAEA safety deficiency list does **not** include insights from VVER-1000 PSA studies. (It should be noted that later studies from IAEA cite such PSA results, and several such PSA studies are known to have been completed or are under preparation at this time.) Thus, it cannot be asserted or assumed that the identification of VVER-1000 safety issues in IAEA-EBP-WWER-05 (IAEA 1996a) takes into consideration **VVER-1000/320** PSA studies; clearly, the contrary is true – the identification of VVER-1000/320 safety issues in that report was, save a single exception, entirely **deterministic** in nature, and **none** of the issues derives from a VVER-1000/320 PSA result.

The IAEA has acknowledged that "*PSA results are an important base for the assessment of the measures directed to upgrading the safety*", and has recommended that a Level 1 PSA should be performed as a **minimum** for all VVER-1000/320 NPPs (IAEA 1996a: 119). **Neither the IAEA listing, nor the safety improvement programmes based on the IAEA listing, can be confidently said to have identified the risk dominant contributors for VVER-1000/320 units and dealt with these risk contributors in such a fashion that the level of risk is consistent with that from western PWRs.**

Achievement of some form of comparability with western safety objectives is itself an illusory goal in any event. It is beyond reasonable dispute that achievement of conformance with western safety criteria is **not** a guarantee of a particular level of risk (e.g., achievement of a quantitative safety target, such as core damage frequency – CDF – or large release frequency – LRF). The example of U.S. Nuclear Regulatory Commission (USNRC) regulations is particularly instructive. Essentially all NPPs in the United States have been licensed to the same basic set of criteria (the General Design Criteria contained in 10 CFR Part 50, Appendix A). Notwithstanding this, there is a remarkable spread of quantitative safety outcomes.

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[7] The Nuclear Safety Standards (NUSS) are the basic safety standards promulgated by IAEA. Although Mouchel **asserts** comparability of the K2/R4 design to these standards, nowhere in the EIAs is a point-by-point comparison of the K2/R4 designs with the NUSS standards either documented or referenced. The assertion of comparability to the NUSS standards seems to be based on the use of broad, general design objectives as set forth in INSAG-3. Conformance to these **objectives** does **not** guarantee conformance to the NUSS standards, nor does it guarantee achievement of any particular level of risk (i.e., achievement of the INSAG safety targets of a CDF of < 10^{-4} per year and a large release frequency of < 10^{-5} per year) (IAEA 1988).

Internal event probabilistic safety assessment (PSA) results are available for U.S. NPPs as a result of the USNRC "Individual Plant Examination" (IPE) program instituted in response to Generic Letter 88-20. These results show as span in internal events core damage frequency from 1.1×10^{-7} per year to 4.3×10^{-4} per year for 91 plants for which results were available in November 1993 (Sholly 1993). These values span a range of a factor of nearly 4,000.

While differences in methodology and assumptions across these PSA studies account for some of the variation, clearly the achievement of conformance to a given set of western safety criteria is no guarantee of a particular level of protection against core damage from internal events. In addition, conformance to a given set of western safety criteria is no guarantee of meeting the INSAG safety targets since of 91 individual units examined using internal events PSA methodology, twelve (13.2%) exceeded the INSAG CDF safety target, and twenty-four (26.4%) exceeded the INSAG large release frequency safety target.

Given the divergences from western safety criteria noted in the IAEA report, and given that the safety improvement programme for K2 and R4 is not probabilistically based, one can scarcely expect better results for K2 and R4. **Resolution of the IAEA-EBP-WWER-05 list of safety issues does not and cannot provide any guarantee of meeting the IAEA INSAG safety targets for CDF or LRF.** The fact of the matter is that meeting **any** existing set of deterministic regulatory criteria cannot provide such a guarantee. Only by supplementing the deterministic review with a PSA can this be achieved.

The type of spread in CDF results seen above for internal events also applies to PSA results for external events, as a recent draft USNRC publication summarizing fire PSA results indicates (USNRC 1998b: App. B). The fire CDF values for US NPPs ranged from 8.1×10^{-8} per year to 2.3×10^{-4} per year. (Of course, it must be recognized that there was as significant change in fire protection regulations in the US following the 1975 Browns Ferry fire; the new regulations were put into effect in 1979.) Seismic PSA results also show a very large spread, and some individual units show accident sequences among the dominant contributors to core damage arising from such diverse external initiators as external flooding, internal flooding, hurricanes, tornado missiles, and others.

- **CLAIM: After completion the two units will have a safety level similar to that of similarly aged but recently re-licensed, western plants (Energoatom 1998: 6). The project would allow the safety of the plant to be comparable to that achieved in the European Union for NPPs recently re-approved by national safety authorities (Mouchel 1998a: 0.6 & 11.1; Mouchel 1998b: 0.6 & 11.1).**

DISCUSSION: These claims have not been substantiated.

Both Khmelnitsky Unit 2 and Rivne Unit 4 were originally designed with the Soviet Union's general rule OPB-73. During the construction period, OPB-82 and OPB-88 came into effect (Mouchel 1998a: 4.1; Mouchel 19988b: 4.1). No systematic evaluation of conformance of K2 or R4 with OPB-73, OPB-82, or OPB-88 has been provided in the EIAs. Thus, not only has there been no assessment of the K2/R4 design against western safety criteria, there has not even been an assessment of the designs against their original design intent as set forth in OPB-73 and OPB-82. In fact, the IAEA safety issue document (IAEA 1996a) contains numerous examples of cases where the VVER-1000/320 design **does not** comply with these basic Russian standards.

Moreover, although the EIAs stated that the safety level of K2 and R4 will be "*comparable to that achieved in the European Union for NPPs recently re-approved by national safety authorities*", the EIAs do **not** identify either a particular NPP or group of NPPs from which to make a comparison with K2 and R4. In addition, the EIAs themselves do not document such a comparison.

Further, the EIAs do **not** identify any specific EU nuclear safety authority whose standards for re-licensing could be used as a basis for comparison with K2 and R4, nor do the EIAs document such a comparison. Finally, the EIAs do **not** identify the time period within which the EU NPP re-approvals by nuclear safety authorities occurred (was this in the 1970s, 1980s, 1990s, or when?).

Accordingly, the statement in the EIAs cited above is hopelessly vague and without substance. All the reader has is Mouchel's opinion that K2 and R4 will be "*comparable*" (whatever that means in the nuclear safety context) to some unidentified EU NPPs which were re-approved for operation at some unidentified time by some unidentified national safety authorities when the NPPs were compared with some unidentified set of safety criteria.

In addition, it should be noted that the both the IAEA and the MOHT consortium have identified a wide variety of safety issues for VVER-1000/320 units. Indeed, the IAEA identified Category III issues (that is, those which are of **high** safety concern, for which defense-in-depth is **insufficient**, and for which **immediate corrective action** is necessary; IAEA 1996a: 10) in most categories of plant safety (e.g., controlling power in normal operation, controlling power in shutdown, maintaining the integrity of the primary coolant system, decay heat removal via the secondary system, residual heat removal in a primary to secondary leak, residual heat removal after design basis and beyond design basis accidents, containment integrity, classification and qualification of components, electrical power supply, and internal hazards). Despite the **broad nature** of the recognized deficiencies, even the project sponsors concede that the upgrading and modernization programmes being proposed for Khmelnitsky Unit 2 and Rivne Unit 4 "***essentially** concerns electrical and control command equipment*" (Mouchel 1998a: 0.3; Mouchel 1998b: 0.3; underlining emphasis added). It is not clear that either Mouchel or the project sponsors understand the broad nature of the safety deficiencies in the VVER-1000/320 design based on deterministic reviews, nor is it clear that either Mouchel or the project sponsors understand that probabilistic evaluations of potential safety deficiencies are as yet still missing and could result in the identification of **additional** safety issues for which the existing upgrade programmes are inadequate. (Indeed, as discussed later, the MOHT consortium – unlike the IAEA – **has** in fact identified PSA-based upgrades for VVER-1000/320 units, many of which are **not** included in the K2/R4 modernization programme.)

Mouchel has also failed to acknowledge that in many cases re-licensing of NPPs in the west – along with periodic safety reviews or PSRs – **incorporate** the use of probabilistic safety assessments (PSAs) in the process, rather than defer such analyses until after the re-licensing process is complete. Canada, Finland, Germany, Italy, Sweden, and the United Kingdom (all EU countries) include PSA in some way in the licensing process. Other countries which employ PSA or which plan to include the United States and the Netherlands (also part of the EU) (OECD 1992: 15). The majority of countries in the OECD which perform periodic safety reviews consider PSAs as part of these reviews (OECD 1992: 27). Indeed, full scope PSAs are required in the course of PSRs in Germany, the Netherlands, and Switzerland, and were also major factors in such reviews in Sweden and the United Kingdom (for older units). Spain uses PSAs to examine the safety level of plants in operation, and the France expects to use PSAs as tools for future reviews (OECD 1992: 30). In contrast, PSAs for K2 and R4 are not planned to be completed until after startup.

In short, Mouchel has not identified any **specific** western or international safety standards which it has used as a basis for comparison with the safety of K2 and R4, nor has Mouchel **demonstrated** that such standards have been complied with in the upgraded designs of K2 and R4.

[Even farther from this is the EBRD's statement, in its "Project Summary" document for K2 and R4 (which was available on the EBRD World Wide Web at the following URL, http: //www.ebrd.org/english/opera/Psd/psd1998/k2r4.htm, on 30 September 1998), "*Succesful implementation of this project would also provide an internationally acceptable benchmark for safety levels of nuclear power units with VVER 1000 type reactors.*" **There is simply no basis for this statement**, and it indicates that whomever at EBRD prepared this project summary document has little or no understanding of how the K2/R4 upgrade programme was developed nor of western safety standards and safety analysis practices in general. Neither the importance of PSA and severe accident insights in safety upgrade programmes, nor the complete lack of such insights in the development of the K2/R4 upgrade programme, are apparently understood by EBRD.]

Finally, Mouchel fails to acknowledge that the MOHT consortium identified two groups of proposed upgrades based on PSA and severe accident insights for VVER-1000/320 reactors, most of which are **not** included in the K2/R4 upgrade programmes. The items recommended by MOHT are as follows (MOHT 1996: Part 1/A2, pp. 2-4; Part 3, pp. 132-139):

- Upgrade pressurizer relief valves for two-phase and water flow to allow for performing feed and bleed (i.e., removal of decay heat from the primary system by “bleeding” primary coolant from the pressurizer relief valves and providing makeup with the high pressure injection system). [Included in the upgrade programme as technical measure 13411, with implementation **before** startup; KIEP 1996: 67-68.]
- Development and implementation of additional means of steam generator supply from reliable sources (due to the limited capacity of the emergency feedwater system, which in the basic design is limited to 8-10 hours of heat removal). [Included in the upgrade programme as technical measure 13311, but only **after** startup; KIEP 1996: 63-64.]
- Provision of an additional common diesel generator. [Included in the upgrade programme as technical measure 24411, with implementation **before** startup; KIEP 1996: 203-204.]
- Provision of an automated algorithm of protective actions in case of large primary-to-secondary leak. [**Not** included in the upgrade programme.][8]
- Upgrading of the suction pipelines of the primary circuit heat removal system to improve its reliability. [Included in the upgrade programme **before** startup.]
- Introduction of a passive decay heat removal system (SPOT; consisting of a heat exchanger to dump steam to the environment via natural circulation with the primary circuit intact). [**Not** included in the upgrade programme.]
- Implementation of additional, higher pressure (15 bar), hydroaccumulators to provide for extended (several hours) passive injection in case of failure of active emergency core cooling system. [**Not** included in the upgrade programme.]
- Modernization of the area under the reactor vessel to accommodate core melt accidents, and provision of additional borated water inventory to flood the area under the reactor vessel in case of core melt. [**Not** included in the upgrade programme.]
- Implementation of filtered vented containment for severe accidents. [**Not** included in the upgrade programme.]
- Installation of passive hydrogen recombiners for severe accident hydrogen loads. [Included in the upgrade programme as technical measure 16211, but only analysis is included **before** startup; equipment installation is **after** startup; KIEP 1996: 118-119.]
- Implementation of containment penetration room leakage collection system, processed through the filtered venting system. [**Not** included in the upgrade programme.]

  Of these **eleven** measures, only **four** are included in the upgrade programme but only **two** will result in plant modifications before startup. Thus, only two of the eleven recommended measures will be implemented before startup.

  The following observations can be made concerning the consideration of PSA and severe accident issues in the upgrade programme:

- Only **one** of the 147 measures in the upgrade programme cites PSA results as **part** of its basis (technical measure 12411 concerning the development of procedures to manage primary to secondary leakage). Although this measure is scheduled to be implemented before startup, it is known that the emergency operating procedures will be **event-oriented** rather than **symptom-oriented** as recommended following the TMI and Chernobyl accidents), thus the effectiveness of even this measure can be questioned. (In addition, it should be observed that MOHT recommended that primary-to-secondary leak-

[8] This would avoid primary reliance on operator action to respond to this initiating event, replacing operator action with an automatic system response and placing the operators in a confirmatory role, following emergency operating procedures to ensure that the automatic system responded to the event. This would lower the CDF from SG collector failures.

age be remedied by automatic safety system response, rather than reliance on operator procedures; see above.)

- Of the 84 safety issues cited by the IAEA, only **one** refers to PSA results (and even in this case the PSA results are PSAs of designs **other than VVER-1000/320**, representing generic results from a variety of reactor designs).
- Of the eleven PSA-related and severe-accident-related recommendations in the MOHT consortium report for VVER-1000/320 reactors, only four are included in the upgrade programme and two of these will not be implemented until **after** startup.

Clearly, there was very **minimal** consideration of severe accidents and PSA results – that is, very **minimal** consideration of **risk** – in the development of the K2/R4 upgrade programme. Since performance of PSAs for K2 and R4 is deferred until **after** startup, there is no significant technical basis for concluding or assuming that the risks posed by K2 and R4 are either acceptable or somehow consistent with the risks posed by reactors currently operating or recently re-approved for operation in the EU.

Finally, the use of the RiskAudit report (RiskAudit 1997a) cannot be bootstrapped into a conclusion of risk or safety comparability between K2 and R4 and EU NPPs. This is because RiskAudit also did not consider PSA and severe accident issues since RiskAudit took as a basic assumption that all essential design and operational safety weak points for the VVER-1000/320 design had been previously identified by IAEA (IAEA 1996a), RiskAudit, and WANO, **none** of which considered VVER-1000/320 PSA or severe accident study results, and because RiskAudit performed no such studies of its own (RiskAudit 1997a: 8-10).

At the very least, if Mouchel wishes to assert a level of safety comparable to recently relicensed EU plants, Mouchel should perform the required review to either a specific nation's re-licensing standards or at least to the IAEA criteria for periodic safety review (IAEA 1994b). No attempt at such a review is contained in or referenced in the EIAs prepared by Mouchel.

- **CLAIM: The routine discharges of radioactivity from two RBMK units operating at Chernobyl would significantly exceed those arising from operation of two VVER-1000 units (Mouchel 1998a: 0.7; Mouchel 1998b: 8).**

DISCUSSION: This claim has not been substantiated, and is demonstrated below to be misleading.

Mouchel indicates that the RBMK releases would be in excess of the VVER normal releases as follows (Mouchel 1998a: 10.9):

Radioactivity Category	RBMK/VVER
Long-lived radionuclides	28
Iodine-131	9
Noble gases	11

However, Mouchel **fails** to point out that there are **no** public residents within 30 km of Chernobyl as a result of the exclusion area established due to the Unit 4 accident in 1986. Compared with the doses at 0-5 km, the doses at 30 km would be lower; judging from Table 7.3, the reduction would be a factor of 200-300. Thus, even if the releases from Chernobyl are 30 times larger, the dose to the **nearest** resident would be **lower** at Chernobyl than at either Khmelnitsky or Rivne. In addition, as indicated by Mouchel, even if the Khmelnitsky or Rivne doses were increased by a factor of 30 they would still be small relative to the limits applied to members of the public (Mouchel 1998a: 10.9).

Thus, the fact that the normal operational releases of radioactivity is larger for the RBMKs operating at Chernobyl site compared with a VVER-1000/320 operating at Khmelnitsky or

Rivne has no significance. In fact, the **dose** to the nearest member of the public (which is the environmental impact resulting from the releases) would in fact be **lower** for Chernobyl. Thus, rather than constituting an advantage for the project proponents, this factor constitutes a disadvantage compared with the base case. Certainly this issue provides no justification for completion of K2 and R4 as an alternative to continued operation of Chernobyl.

- **CLAIM: The RBMK reactor is inherently less safe than is the VVER reactor. The no-change option therefore would result in an increased risk of a catastrophic accident leading to widespread contamination (Mouchel 1998a: 0.7; Mouchel 1998b: 0.8).**

DISCUSSION: This claim has not been substantiated.

Mouchel acknowledges that in the nuclear power industry, “risk” is defined as the ”*product of the likelihood of occurrence of an accident and its potential radiological consequences*” (Mouchel 1998a: 8.4). **However, nowhere in the EIAs prepared by Mouchel are either the frequency or consequences of severe accidents estimated for either VVER-1000/320 or RBMK reactors.** Moreover, no such documents are cited by Mouchel, nor has Mouchel itself performed no such analyses.

In fact, some RBMK and VVER-1000/320 PSA results are available. The internal events PSAs of the Temelin (VVER-1000/320) and Ignalina (RBMK) units are available. External events analyses remain to be performed and/or issued publicly. The Temelin PSA is relevant because the Temelin design is a VVER-1000/320 like K2 and R4, only the upgrades being implemented for Temelin are more extensive than for K2 and R4. The Ignalina PSA is relevant for Chernobyl even though the Ignalina design is an RBMK-1500 (third generation RBMK) and Chernobyl Units 1 and 3 are RBMK-1000 (first and second generation RBMKs, respectively), because the risk comparison contained in the EIAs is for the upgraded condition of Chernobyl.

The Temelin PSA calculated an internal events core damage frequency (CDF) of 7.9×10^{-5} per year (IAEA 1996b: 113). Two observations are pertinent. **First**, man-made and natural phenomena hazards (such as fires, flooding, aircraft crash, turbine missiles, and earthquakes) and shutdown events will have to contribute a CDF of less than 2.1×10^{-5} per year in order that the total CDF from all causes is kept below the INSAG safety target of 1.0×10^{-4} per year. This is considered by IRR to be an unlikely outcome for Temelin (fires and earthquakes will likely contribute at least this much if not more to CDF). More importantly, since K2 and R4 will not be upgraded to the same extent as Temelin, it is quite likely that the CDF for K2 and R4 will be higher for internal events, and probably for external events as well.

Second, even with the more substantial upgrading of Temelin, containment bypass events contribute about 83% of the internal events CDF (steam generator collector failure, steam generator tube rupture, and interfacing LOCA involving the low pressure piping of the ECCS outside containment), for a total large release frequency of almost 6.6×10^{-5} per year. This is substantially above the INSAG safety target of 1.0×10^{-5} per year for large release frequency. Given the situation for K2/R4, this value will almost certainly be exceeded since the initiating event frequency will be higher (due to the failure to replace the condensers with titanium condensers as was done at Temelin) and the operator error rates will be higher (due to the use of event-oriented EOPs at commissioning of K2/R4, in contrast with Temelin where symptom-oriented EOPs will be implemented at commissioning).

The Ignalina PSA calculated an internal events CDF of 3.2×10^{-5} per year (ES-Konsult AB 1996: 30). Even if it is assumed that **all** Ignalina core damage events proceed to a large release (which is likely), the large release frequency for Ignalina internal events is a factor of a little over two **less** than the large release frequency for Temelin internal events. (The Ignalina result also exceeds the INSAG safety target of 1.0×10^{-5} per year for large release frequency.)

Clearly, neither the Temelin nor Ignalina results directly represent K2/R4 or Chernobyl. **However**, the results do illustrate that one **cannot** simply **assume** – as Mouchel has done – that simply because the Chernobyl units lack a containment the frequency of a large release accident is **necessarily** larger at Chernobyl than at the upgraded K2/R4 reactors.

Moreover, although it is tragic it must be acknowledged that the consequences of a large release accident at Chernobyl will necessarily be less than in 1986 simply because there is no one living within 30 km of the site anymore. The same cannot be said of either the Khmelnitsky or Rivne sites, where 250,700 persons (Mouchel 1998a: 3.9) and 134,680 persons (Mouchel 1998b: 3.8) (respectively) living.

Finally, there are significant differences in the physical characteristics of severe accident releases from VVER-1000/320 and RBMK units. RBMK units (as illustrated by the Chernobyl Unit 4 accident) release less than half (actually, about one-quarter) of the total release in the first day, with the remainder of the release coming over a period of a week or more. This provides substantial opportunity to mitigate the release (indeed, the total Chernobyl release would have been greater but for heroic mitigation actions undertaken by plant staff and emergency response organizations). In addition, the release is likely to be elevated due to energy input considerations (the Chernobyl release height was in excess of 1000 meters). In contrast, the bulk of large release PWR accident sequences tend to be **complete** within 2-4 hours of the time of vessel failure. In addition, the release heights are substantially lower (10-200 meters) than the RBMK releases.

All told, the PWR large release source terms pose more hazard to close in populations than an RBMK large release source term. This characteristic is exacerbated in the K2 and R4 risk comparison with Chernobyl since there is no offsite population within 30 km of Chernobyl whereas there are substantial offsite populations within 30 km of both K2 and R4.

- **CLAIM: The core damage frequency for the upgraded K2/R4 units will be close to the value for recently re-approved PWRs and significantly lower than the corresponding value for the RBMK (Mouchel 1998a: 10.10; Mouchel 1998b: 10.10).**

  DISCUSSION: This claim has not been substantiated.

  No CDF results for any plants are cited by Mouchel. Mouchel does not identify the "*recently reapproved PWRs*" for which the comparison is asserted, nor the CDF for these plants. Mouchel does not identify the CDF for K2/R4 (and **cannot** since PSAs for these units will **not** be completed until **after** startup) or for any other VVER-1000/320 units. Mouchel does not identify the CDF for the RBMKs at Chernobyl or for any other RBMK. CDF results for internal events for Temelin (VVER-1000/320) and Ignalina (RBMK-1500) cited above indicate that the CDF for a VVER-1000/320 may be **higher** than for an RBMK for internal events. No external events CDF results are available, and thus no inferences can be made regarding CDF comparisons between VVER-1000/320 and RBMK units.

  On what basis has Mouchel made its statement? Does the statement regarding K2/R4 CDF consider the plant status at startup, when many upgrade programme components will not yet be implemented, or does the statement reflect the plant status once all the upgrade programme measures have been fully implemented? Did Mouchel consider the Temelin PSA results, the Kozloduy PSA results, the Novovoronezh PSA results, or the Ignalina PSA results? Certainly there is no indication that they have.

  In short, Mouchel is **speculating**. Not only is there **no basis** for this speculation, there is **substantive** information indicating that the **speculation** is **incorrect**.

- **CLAIM: PWRs and VVERs have a strong leaktight containment, are stable reactors, and physically cannot generate an explosive Chernobyl-type accident. Therefore, the overall safety level of an RBMK can never be equivalent to that of a VVER 1000 (Mouchel 199a: 10.11; Mouchel 1998b: 10.11).**

  DISCUSSION: This claim is only partially substantiated.

  It is true that the K2 and R4 units will have a full containment. **However**, these containments are designed such that the bottom of the containment is located **above** grade, and in the event of a severe accident where the bottom of the containment is penetrated by core debris a containment bypass/failure condition will exist because the compartments below the containment are not pressure retaining and not leaktight. In addition, the containment leak rate for K2 and R4 is three times that for comparable US PWRs (0.3% per day vs. 0.1% per day), thus the degree of leak tightness in comparison with western PWRs is less.

While the VVER-1000/320 design cannot generate a Chernobyl-type accident (explosive release), it is nonetheless possible for PWRs generally to experience severe accidents involving containment bypass or containment failure in which release fractions of the same order as those in the Chernobyl accident (INSAG 1986; USNRC 1998):

Accident Sequence	Kr & Xe	I	Cs	Sr	Ru	La	Ce
Chernobyl Unit 4 accident (INSAG 1986: 34)[9]	1.0	0.20	0.13	0.04	0.029	??	0.028
Surry NPP station blackout with a steam explosion (RSUR-1)	1.0	0.35	0.31	0.06	0.006	0.006	0.01
Station blackout, containment leak (RSUR-2)	1.0	0.06	0.03	0.003	0.001	$<1\times10^{-10}$	$<1\times10^{-10}$
Interfacing LOCA (RSUR-4)	1.0	0.12	0.12	0.025	$<1\times10^{-10}$	0.0003	0.004

Finally, it is not possible on the basis of current information presented in the EIAs or reviewed by IRR for this report to conclude that the overall safety level of an RBMK "*can never be equivalent to that of a VVER 1000*". Such a conclusion could **only** be drawn after PSAs are performed on the VVER-1000/320 and RBMK units of interest. This has not been done by Mouchel. Clearly, on a **deterministic** basis one would prefer an NPP with a containment (provided the CDF for the NPP is not dominated by containment bypass events) to an NPP without a containment. However, the valid point of comparison is not containment vs. no containment, but rather risk vs. risk. That is, the bottom-line issue is which design poses the greatest **risk**, not which design has a **containment**.

- **CLAIM: Continued operation of Chernobyl Unit 3 has many implications for the final entombment of Unit 4 (Mouchel 1998a: 10.12; Mouchel 1998b: 10.12).**

  DISCUSSION: **This claim has not been substantiated.**

  This claim is simply stated in both EIAs without elaboration, without citation, and without analysis. The USDOE sponsored a study of the risk of a serious accident at Chernobyl Unit 3 resulting from the collapse of damaged structures of the adjacent Unit 4 reactor. The study concluded that the risk of such an accident was "*very small*" and that this issue required no special consideration apart from other issues at the plant, that no special preventive measures were needed, and that no further detailed studies were needed (USDOE 1997). Although the study was reported in draft form, the study summary states that the assessment conclusions were not expected to change. The study was performed by the Chernobyl Center for Nuclear Safety, Radioactive Waste, and Radioecology in Slavutich, Ukraine. Fourteen staff members from the Scientific and Technical Center of the Ministry of Environmental Protection and Nuclear Safety of Ukraine participated in performing the assessment, and the assessment was reviewed by representatives from the Interbranch Agency of the Ukrainian Scientific and Technical Center, the Industrial Association of the Chernobyl Nuclear Power Plant, and the Nuclear Regulatory Administration of Ukraine, in addition to US technical experts under USDOE sponsorship.

- **CLAIM: The EIAs have analyzed the "*most representative*" beyond design basis accident (BDBA), and the lower intervention level for implementation of counter-measures would not be reached at the boundary of the 3 km zone using worst-case dispersion conditions (Mouchel 1998a: 8.15-8.18).**

  DISCUSSION: **This claim has not been substantiated**.

[9] Other authorities report higher release fractions for Iodine and Cesium. For example, releases of 20-40 percent of the Cesium inventory and 50-60 percent of the Iodine inventory are reported (Sweet 1996).

The EIAs acknowledge that no full-scope analysis of beyond design basis accidents (BDBAs) has been completed for a VVER-1000/320 unit. The EIAs then indicate that "*preliminary*" analyses had been made of an unidentified group of BDBAs for which management measures were being implemented on operating VVER-1000/320 units. From this unidentified set of accidents, a fully mitigated steam generator collector failure (100 mm equivalent diameter) with a stuck open atmospheric dump valve was selected as "*most representative*" (Mouchel 1998a: 8.15; Mouchel 1998b: 8.15). No criteria were identified by Mouchel for this selection, nor was the list of accidents from which the selection was made identified by Mouchel.

The accident selected is **not** associated with severe core damage or core melt. Rather, the accident parameters, including the source term, indicate that it is a design basis accident (DBA). Indeed, the IAEA has concluded that this accident **should be** considered to be a DBA for the VVER-1000/320 design.

A whole host of potential BDBAs are obvious for VVER-1000/320 plants, based on general design considerations, PSAs of other PWRs, and available PSA information on VVER-1000/320 units other than K2 and R4. The IAEA itself has identified the following BDBAs for VVER-1000/320 units:

- Total loss of heat sink for greater than 30 hours (IAEA 1996a: 21, 124).
- Reactor vessel failure due to pressurized thermal shock (IAEA 1996a: 47, 114-115).
- Multiple steam generator tube rupture (IAEA 1996a: 54).
- Steam generator collector failure with an equivalent diameter greater than 100 mm (IAEA 1996a: 55, 116).
- Main steam line break with return to power (recriticality) (IAEA 1996a: 113).
- Boron dilution at shutdown conditions (IAEA 1996a: 117, 120).
- Loss of primary coolant at shutdown conditions (IAEA 1996a: 117).
- Loss electrical power at shutdown conditions (IAEA 1996a: 117).
- Spent fuel cask drop (IAEA 1996a: 121).
- Anticipated transients without scram (ATWS) (IAEA 1996a: 122).
- Station blackout (total loss of electrical power) (IAEA 1996a: 123).

RiskAudit also identified BDBAs for the VVER-1000/320 design at Rivne Unit 3 (RiskAudit 199410/72-73):

- Anticipated transients without scram (ATWS).
- Main steam line break with rupture of one or more steam generator tubes.
- Main steam line break with steam generator collector cover break.
- Complete loss of feedwater.
- Complete loss of heat sink.
- Complete loss of low pressure injection.
- Complete loss of high pressure injection.
- Complete loss of containment spray.

Separately, RiskAudit also identified need to analyze station blackout as well as the BDBAs listed immediately above specifically for K2/R4 (RiskAudit 1994b: 9-30, 9-31). The Ukraine nuclear regulatory authority reportedly identified the following BDBAs for analysis for K2/R4 (RiskAudit 1994b: 9-32, 9-33):

- Anticipated transients without scram (ATWS).
- Station blackout.
- Small LOCA with HP ECCS failure.
- Small LOCA with HP ECCS and LP ECCS failure.
- Small LOCA with station blackout.

- Medium LOCA with HP ECCS failure.
- Medium LOCA with HP ECCS and LP ECCS failure.
- Large LOCA with HP ECCS failure.
- Large LOCA with HP ECCS and LP ECCS failure.
- Large LOCA with containment spray system failure.
- Loss of all feedwater (including EFW).
- SG collector failure with stuck-open BRU-A.
- Multiple steam line break (isolable and non-isolable).

Other BDBAs identifiable from VVER-1000/320 PSA studies (Temelin, Kozloduy, Balakovo) include the following (Kujal 1994; IAEA 1996b):

- Station blackout (loss of offsite power and failure of emergency AC and/or DB power).
- Failure of RCP seal cooling system, failure of high pressure injection (either at the outset of the accident or due to failure to depressurize the primary coolant system and stop the break flow outside containment prior to exhausting the borated water inventory to the secondary side of the plant; interfacing LOCA), and failure of containment sprays (due to loss of borated water inventory outside containment).
- A “classic” interfacing LOCA as first identified in the 1975 WASH-1400 report (failure of isolation valves between the reactor coolant system and the low pressure injection system, resulting in piping failure outside the containment and blowdown of primary coolant outside containment, with inability to recirculate the coolant causing failure of high pressure injection, low pressure injection, and containment sprays).
- Loss of decay heat removal during shutdown with the reactor coolant system in a reduced inventory condition.
- Steam generator tube rupture with failure of the operators to depressurize the reactor coolant system and stop the break flow outside containment prior to exhausting the borated water inventory to the secondary side of the plant (interfacing LOCA).
- Large LOCA with failure of low pressure injection.
- Medium LOCA with failure of high and low pressure injection.
- Small LOCA with failure of high pressure injection.
- Loss of all feedwater (main, auxiliary, and emergency) and failure of bleed and feed cooling).
- Steam generator tube rupture, failure of high pressure injection.

There is clearly no shortage of potential BDBAs for the VVER-1000/320 design, many of which result in core damage and some of which do so under containment bypass conditions. Clearly, many of the accidents listed above have the potential to result in worse consequences than the accident selected as “*most representative*” of BDBAs in the K2/R4 EIAs.

The K2/R4 upgrade programme includes an analysis of BDBAs, however the measure (Item 19211) includes only the analysis of station blackout, anticipated transients without scram (ATWS), and loss of all feedwater before startup. Analyses for all other BDBAs is deferred to **after** startup (KIEP 1996: 161/316). Thus, there is **simply no basis** for selection of the analyzed accident above as the “*most representative*” of BDBAs. And, as indicated previously, the selected accident is a DBA, not a BDBA.

3.3.3.2 Project Cost Estimates

There have been a number of cost estimates for completion, modernization, and commissioning of Khmelnitsky Unit 2 and Rivne Unit 4. These estimates are discussed below in chronological order:

- **EBRD Study of Economic Aspects of Nuclear Generation and Safety Improvements in Eastern and Central Europe, June 1993, 0.92 billion USD**

  This study forecast costs to complete and upgrade of 460 million USD each for Khmelnitsky Unit 2 and Rivne Unit 4 (EBRD 1993).

- **USDOE/Minatom Study, July 1994, 0.98 billion USD**

  A study prepared by USDOE and Minatom in July 1994 estimated the costs of upgrading and commissioning K2 and R4 as 485 million USD and 495 USD, respectively, or a total of 0.98 billion USD (ANL 1995).

- **Energoatom Project Presentation, August 1988, 1.25 billion USD**

  In a project presentation to the EBRD, the project sponsor estimated the overall project cost, including physical and price contingencies, was estimated at 1.250 billion United States dollars (USD). This cost estimate was stated to include the following costs: (a) the completion programme; (b) the repair and replacement programme; (c) the modernization programme, both before and after commissioning; (d) the first fuel load; (e) the tests and commissioning; (f) engineering activities; (g) project management; (h) licensing and certification; and (i) miscellaneous costs like customs, insurance, and the financial engineer for the banks (Energoatom 1998: 12).

- **EBRD, September 1998, 1.725 billion USD**

  In the EBRD Project Summary Documents for K2 and R4 at EBRD's web site on 30 September 1998, the EBRD estimated the cost of completing both units and providing support to the Ukraine Nuclear Regulatory Authority at 1.725 billion USD (EBRD 1998).

3.3.4 Current K2/R4 and Chernobyl Status

Construction of Khmelnitsky Unit 2 and Rivne Unit 4 was halted by a suspension of nuclear construction in Ukraine in 1990. The construction status of the two units is identified in the EIAs as 80% to 90% complete (Mouchel 1998a: 8.1). As of 1990, the designs of K2 and R4 were similar to the then-existing operating VVER-1000/320 units at Zaporozhe NPP (Units 2-5), South Ukraine NPP (Unit 3), Khmelnitsky NPP (Unit 1) and Rivne NPP (Unit 3).

Subsequent reviews of the VVER-1000/320 design identified a number of safety and operational issues which required resolution (e.g., IAEA 1996a). A programme of modernization and upgrades has been identified for implementation at K2 and R4 (see Section 3.3.5, below). Implementation of this programme is apparently awaiting funding by EBRD and other international sources (e.g., Euratom, U.S. Export-Import Bank, etc.).

The Chernobyl station consists of four RBMK-1000 NPPs. Units 1 and 2 are first-generation RBMKs, while Units 3 and 4 are second-generation RBMKs. Chernobyl Unit 4 was destroyed in the 1986 accident. Chernobyl Unit 2 has been shut down since a turbine hall fire and partial collapse of the turbine building on 11 October 1991. There has been discussion of rehabilitating the unit and renewing its operation (Mouchel 1998a: 10.2, 10.8 & 10.13). Unit 1 has been shut down since 30 November 1996 (NEI 1997: 1). Chernobyl Unit 3 remains in operation.

3.3.5 K2/R4 Modernization and Upgrade Programme

According to Energoatom, the completion programme for Khmelnitsky Unit 2 and Rivne Unit 4 includes the following completion, rehabilitation, and modernization work (Energoatom 1998: 8-10):

- **Completion** – the completion of the units according to the original design. This work includes outstanding construction work such as completion of plant installation; completion of partial installations; electrical wiring; instrumentation and control equipment; plant cleanliness; and equipment commissioning and functional testing. See Table 3.3.
- **Rehabilitation** – replacement of deteriorated equipment or its repair to the status suitable for operation. This work includes inspections to determine the equipment and civil works needing restoration to a state suitable for commissioning and startup, including repair and replacement tasks such as refurbishing, repainting, surface preservation, and repairs. See Table 3.4.
- **Modernization** – upgrading of safety, quality of operation, and the availability of the units. This work includes the modernization programme, waste processing facilities at the plants; installation of a full-scope simulator at Khmelnitsky; and improvements to the switchyard at Rivne. The modernization programme includes 148 measures, 144 of which are common to the two units and two items each which are specific to Khmelnitsky Unit 2 or Rivne Unit 4. The modernization programme is intended to eliminate deviations from current Ukrainian national safety norms; to improve the reliability of safety-related equipment by upgrading the design quality, manufacture, and installation; and to improve operation quality. See Table 3.5; note that this table is subdivided into measures to be completed before and after commissioning and commercial operation.

It is important to recognize that the modernization measures are nearly entirely deterministic in their basis. No probabilistic safety assessment (PSA) of K2 or R4 is available, and PSAs on other VVER-1000/320 units have not been extensively used to define the modernization measures. Programs for safety improvement which do not include PSA insights may focus on issues which are not the most relevant from the risk reduction point of view, and this could have a significant impact on plant safety by leaving high frequency or high risk accident scenarios which are not addressed adequately by the modernization program.[10]

Table 3.3: K2/R4 Measures to Complete to Original Design

Construction and Buildings	Complete civil works, surface protection, finishing, and roofing.
Mechanical Equipment	Lay out equipment liaisons, pipes and valves, thermal insulation, and align machines.
Electrical Equipment	Lay out electrical wiring, cables, and connections.
Instrumentation and Control (I&C)	Install, calibrate, and tune instrumentation.
Pre-Commissioning	Perform nuclear cleanliness, equipment tests, and pre-service inspection before commissioning.
Commissioning	Perform overall plant tests, fuel loading, physical tests, and on-load tests.

[10] This is a paraphrase of an observation from an IAEA publication which identifies safety issues for RBMK units. The IAEA states specifically (IAEA 1995a: 47), "*Programmes for safety improvements which do not include insights from PSAs may focus on issues which are not most relevant from the risk reduction point of view. This has a significant impact on plant safety.*" IAEA recommended the performance of a plant-specific PSA as a medium priority issue for RBMKs; IAEA defines "medium" as (IAEA 1995a: 8), "*Issues that reflect insufficient defense-in-depth and have a significant impact on plant safety. Short term actions might be necessary to improve safety as applicable to each specific NPP, until the issue is fully resolved.*"

Table 3.4: K2/R4 Measures in Rehabilitation (Repair and Replacement) Programme

Construction and Buildings

- Repair and preserve containment pre-stressing cables, their protective caps, and cover joints.
- Finish containment liner painting.
- Control containment hatches tightness.
- Complete metalwork painting in reactor building and turbine hall.
- Repair spalled concrete structures.
- Replace epoxy paint on floors in reactor building auxiliary rooms.

Mechanical Equipment

- Paint external surfaces.
- Repair valve manual actuation, gate tightening surfaces, paint external surfaces on pipes.
- Repair metal burn-through, ribs, dents, scratches, metal drops, nicks, and cracks on mechanical equipment.
- Realign turbine shaft.
- Paint the internal surface of the condenser vacuum system ejector housing; of valve bodies and bonnets; and of bearings and stems.
- Adjust pump electrical motors external fittings and bearings.
- Preserve electric motors in grease – replace anti-reverse, impeller fixture, rubber seals of primary pumps.
- Repair surface metal defects of steam generator, pressurizer basis metal, ECCS tanks and parts of main circulation pipeline, repair flange threaded jack of reactor vessel, manometer connecting pipe to blowdown separator, repair dentin upper unit cover.
- Replace rubber seal, stop washers and splints in diesel generators, seals and working liquid of snubbers, connecting pieces, coupling nuts and rubber seal of pumps, disc sealing surfaces of fast acting valves.

Instrumentation and Control (I&C) and Electrical Equipment

- Provide manufacturer's documentation for main power circuit and outdoor switchyard, 750 kV.
- Repair parts of main power circuit and outdoor switchyard, 750 kV.
- Replace deficient parts of I&C, electrical systems, electric cables, sensors, regulation units, and power supply units.
- Replace deficient parts of fuel-loading machine.

Table 3.5: K2/R4 Modernization Programme Measures (1 of 7)

MEASURES TO BE COMPLETED *BEFORE* COMMISSIONING

REACTOR CORE

Improve Reactivity Control

- Replace neutron flux control system with modern instrumentation to improve reactivity control at all levels of power.
- New fuel loading strategy to optimize use of fuel and to reduce the neutron fluence on the reactor pressure vessel.
- Implement devices for sub-criticality control of the reactor core at shutdown.

Prevent Excessive Drop Time of Control Rods

- Measures to limit fuel bending.
- Introduce heavy weight control rods.
- Replacement of control rod drive mechanisms.

MAJOR UNIT COMPONENTS

Mitigate Risk of Reactor Vessel Embrittlement

- Heating of tanks of emergency injection systems up to 20°C.
- Heating emergency injection accumulators up to 55°C.
- Improve the reactor vessel neutron flux monitoring to enable its irradiation to be more effectively controlled.
- Relocate vessel metal specimens and modify the correspondent vessel surveillance program.

Prevent Rupture of Main Feedwater and Steam Lines

- Lay out fixed rigid support of steam and feedwater pipelines at 28.8 m level.
- Analyze rupture mechanisms of steam and feedwater pipelines at 28.8 m level.

Reinforce Strength of Components

- Recalculate strength of pipes significant to safety and modify their supports if necessary.
- Perform strength calculation of the reactor vessel head.

Prevent Spillage of Radioactive Water Outside the Reactor Containment

- Develop procedures to control leakage from primary to secondary circuit in the steam generators.
- Implement a detection system for primary circuit leakage.
- Ensure tightness by periodic in-service inspection of ECCS suction lines at the bottom of the sumps.
- Prevent radioactive release through main coolant pump heat exchangers.

Implement Diagnostic Systems for Inspection of Reactor Components

- Determination of residual lifetime of main primary circuit and turbine components.
  In-service inspection of reactor pressure vessel by television or ultrasound.
- Primary circuit coolant leakage diagnosis.
- In-core noise diagnosis system.

Table 3.5: K2/R4 Modernization Programme Measures (2 of 7)

MEASURES TO BE COMPLETED *BEFORE* COMMISSIONING (cont.)

Improve Preventive Maintenance and In-Service Inspection

- Implement an automatic control system of the secondary circuit chemistry to mitigate corrosion in the steam water circuit.
- Replace steam generator blowdown system to limit corrosion in the steam generators.
- Implement an annealing machine for conditioning the vessel main joint gasket before laying out.

Improve Safety of the Units

- Replace safety valves on the low pressure injection lines used for residual heat removal at shutdown and implement procedures to prevent over-pressure at cold shutdown.
- Replace steam generator safety valves.
- Qualify pressurizer safety valves to implement feed and bleed procedures.
- Prevent insulation material from clogging the sumps in case of LOCA by replacing the insulation material by a metallic type.
- Install a hydrogen removal device from primary circuit during cooldown and cold shutdown.
- Replace air conditioners in the reactor building.
- Install a position indicator on the pressurizer safety valves.
- Install sealed valves 1600 nominal diameter for isolation of ventilation system of reactor compartment.

Improve Main Cooling Pump Seals

- Upgrade thermal barriers to improve operational reliability and safety of main cooling pumps.
- Modify auxiliary water makeup circuit to increase the time of interruption of makeup water to the main cooling pump seals without damage.

ELECTRICAL SUPPLY COMPONENTS

Improve Power Supply Reliability

- Install an additional diesel generator set.
- Replace inverters of the emergency power supply.
- Increase battery discharge time.
- Analyze additional sources of power for safety systems.
- Improve emergency diesel generator reliability.
- Replace electrical and fire protection devices of the switchboards.
- Replace high voltage transformers bushings 750 kV.
- Analysis of external power grid.
- Replace deficient electrical wiring.
- Implement a multiple channel system recording voltage perturbation in the generator.

Table 3.5: K2/R4 Modernization Programme Measures (3 of 7)

MEASURES TO BE COMPLETED *BEFORE* COMMISSIONING (cont.)

INSTRUMENTATION AND CONTROL (I&C) COMPONENTS

Improve I&C Efficiency

- Reinforce immunity of I&C components against electromagnetic interference (EMI).
- Segregate the impulse lines of the three primary pressure sensors against common mode failure.
- Detect the presence of a gas-steam volume inside the reactor vessel after an accident.
- Improve the steam generator water level control.
- Replace boron meters by up-to-date Boron 10 concentration measuring devices.
- Redesign cable racks of temperature monitoring measurement system in the core.

CONCRETE CONTAINMENT

Control the State of the Pre-Stressed Concrete Containment and Civil Structures

- Implement equipment for control of containment vacuum test.
- Implement a diagnostic system of forces in pre-stressing cables.
- Implement diagnostic systems for containment state assessment.
- Exchange sealed cable penetrations.
- Calculate containment reliability using modern codes.

HAZARDS AFFECTING UNIT INTEGRITY

Internal Flooding

- Assess risk of internal flooding in reactor building compartments.
- Assess risk of internal flooding in the machine hall.

Fire Hazards

- Perform systematic fire hazard analysis to improve fire protection if necessary.
- Protect the cable bundles with fire resistant coating.
- Perform an analysis of the possibility of ensuring reactor shutdown in case of fire in the cable compartment under the Main Control Room and Emergency Control Room and 6 kV switchboards.
- Implement a fire extinguishing system with backed up power supply for water distribution.
- Replace fire resistant doors.
- Install fire protection valves in air conduits.
- Improve fire retardant coating on turbine hall roof.
- Install gas fire fighting in electronic equipment compartments.

Analyze Hazards External to the Units

- Analyze aircraft crash on reactor building.
- Analyze risk of airborne toxic gases external to the plant on personnel in Main Control Room and Emergency Control Room.
- Analyze risk of shock wave loads on reactor building in case of explosion external to the plant.

Table 3.5: K2/R4 Modernization Programme Measures (4 of 7)

MEASURES TO BE COMPLETED *BEFORE* COMMISSIONING (cont.)

ACCIDENT ANALYSIS

Perform Analysis of Design Basis Accidents

- Prepare a list of design basis accidents.
- Carry out the analysis of design basis accidents not taken into consideration in the former safety analysis report.
- Carry out the analysis of design basis accidents using modern codes.

OPERATIONAL SAFETY

Quality Assurance

- Develop general quality assurance (QA) program based on IAEA recommendations and ISO 9001

Operating Procedures

- Improve operating procedures for safety related reactor systems.
- Improve technical instructions of reactor equipment and systems in normal operation.
- Develop accident procedures.
- List nuclear hazardous works in a regulatory document.

Radiation Protection and Personnel Protection

- Enhance the function of the existing radiation protection department.
- Implement a computerized access management system.
- Implement an automatic radiation monitoring system around the site.

MEASURES TO BE COMPLETED *AFTER* COMMISSIONING

REACTOR CORE

Improve Power Control of the Core

- Introduce a more effective core control strategy to mitigate Xenon oscillations and control power distribution.
- Optimize accurate engineering margin factors to improve power control.
- Introduce new control rods with burnable absorber.
- Load new fuel containing uranium and gadolinium.

Improve Diagnostics on Fuel

- Implement a device inside the refueling machine mast to detect fuel rod deficiency.
- Determine the correlation between damaged fuel and activity of reference isotopes in primary coolant.

Table 3.5: K2/R4 Modernization Programme Measures (5 of 7)

MEASURES TO BE COMPLETED *AFTER* COMMISSIONING (cont.)

Spent Fuel Management and Refueling

- Develop equipment for completing fuel loading procedures in case of loss of power.
- Enlarge storage capacity of spent fuel pool to provide space for full core discharge unloading in an emergency.
- Implement equipment to transport the spent control rod clusters from the reactor and for their repository at the NPP site.

MAIN UNIT COMPONENTS

General

- Develop equipment qualification.
- Develop a system for monitoring neutron flux on the reactor vessel to determine remaining vessel life time.
- Implement systems by using leak-before-break concept to detect signs of small leakage indicating imminence of large primary high energy line break.
- Increase means of steam generator makeup water.
- Implement automatic control system for primary coolant chemistry in normal operation.
- Install displacement indicators for piping.
- Optimize maintenance of pressurizer safety valves.
- Implement a Technical Support Centre (TSC) for assisting operators in emergency situations.
- Implement television cameras for closed premises.
- Implement a data storage device monitoring process parameters in case of emergency situation.
- Install a seismic monitoring and recording device.
- Analysis air conditions inside safety system rooms with cold weather external to the plant.
- Develop accident procedures.
- Elaborate regulations for metal surveillance of equipment and pipes.
- Replace the radiation monitoring system inside the unit.
- Implement an appliance to tighten the bolts of the main coolant pump main seal.
- Implement a mockup for quality control of O-rings of main coolant pump seals.
- Implement a hydraulic adjustment procedure for the main coolant pumps.
- Suppress leakage isolation devices in impulse lines of the steam generator water level measurement.
- Replace pump seals, fasteners, and pump shaft of feedwater pumps.
- Upgrade pump body, sealing, and coupling sleeves of booster pump PTA 3800-20.
- Replace valves gland package.

Improve Turbine Drain Lines

- Suppress useless pipe bends in turbine drain lines.
- Replace sections of drain lines with potential intensive erosion by stainless steel pipes.

Implement Diagnostic Systems

- Implement system to detect vibrations in components.
- Determine criterion for preventing plugging of steam generator tubes.
- Implement system to detect defects in the main coolant pump.

Table 3.5: K2/R4 Modernization Programme Measures (6 of 7)

MEASURES TO BE COMPLETED *AFTER* COMMISSIONING (cont.)

Reinforce Strength of Components

- Perform strength calculation of the air duct weld of the reactor top head.
- Perform strength analysis of makeup nozzle thermal shield.

Improve Efficiency of Pumps

- Upgrade anti-reverse device of main coolant pumps.
- Implement procedure for identifying defects in body components of main coolant pumps.
- Implement a temperature monitoring system for main coolant pump motors.
- Introduce an oil lubricated thrust bearing and a sleeve coupling for high head auxiliary injection pump CN-150-110.
- Upgrade impeller and sleeve coupling for emergency cooling pump CNR-800-230.
- Introduce an oil lubricated thrust bearing and coupling for spray pump CNCA-700-140.

Improve Efficiency of Valves

- Upgrade sealing assemblies of fast cut off valves on the main steam lines and their maintenance and repair.
- Upgrade maintenance and repair of fast cut off valves on main steam lines.
- Replace steam generator feedwater control valves.

ELECTRICAL SUPPLY COMPONENTS
Improve Electrical Component Reliability

- Replace 6 kV switches.
- Assess residual lifetime of cables and replace them.
- Install additional battery backed up emergency lighting fixtures.

Implement Diagnostic Systems

- Implement diagnostic system on windings of turbine generator stator.
- Implement diagnostic system on windings of 6 kV motor stator.

INSTRUMENTATION AND CONTROL (I&C) COMPONENTS
Modernize I&C Parts

- Replace monitoring device on generator process parameters.
- Replace in core instrumentation and control, computer, and software.
- Replace turbine control system.
- Replace power unit display and control computer Titan 2 (mainly before implemented).
- Replace obsolete I&C instrumentation of the plant by modern devices.
- Develop a generalized safety parameter display system.

Table 3.5: K2/R4 Modernization Programme Measures (7 of 7)

MEASURES TO BE COMPLETED *AFTER* COMMISSIONING (cont.)

Introduce Diagnostic Systems for Improving Maintenance

- Implement computerized network for diagnosis.
- Implement a vibration diagnostic system for rotating machines (K2 will be before commissioning).
- Implement loose parts diagnostic system in primary circuit.
- Implement noise diagnosis system for steam generator headers.
- Implement monitoring of functional parameters: reactivity balance, control rod sticking, number of fuel cycles, heating-cooldown rate, load escalating and dropping rate.
- Implement back pressure valve diagnosis system.
- Implement air operated valves diagnosis system.

CONCRETE CONTAINMENT

- Perform analysis of leak-tightness of concrete containment, especially at penetrations, in order to ensure isolation of reactor in case of emergency.
- Analysis of state of the pre-stressing cables and fittings, and pre-stressed structures of the concrete containment.
- Calculate containment reliability using modern codes.

HAZARDS AFFECTING UNIT INTEGRITY

- Physically protect shut-off valves with barriers against internal missiles.
- Replace combustible petroleum oil by non-flammable agent in main coolant pump lubrication system.
- Implement automatic hydrogen dumping device from generator housing to protect the generator against hydrogen explosion.
- Implement smoke prevention system in rooms and corridors of the reactor building.
- Install hydrogen detection and ignition devices inside the concrete containment.

ACCIDENT ANALYSIS

- List the beyond design basis accidents to be analyzed.
- Perform analysis of hydrogen accumulation inside the reactor plant.
- Carry out Level 1 and Level 2 probabilistic safety analysis.

OPERATIONAL SAFETY

- Improve verification and testing procedure of safety-related reactor systems.
- Improve maintenance and repair procedures for reactor equipment.
- Develop a computerized recording of events on NPP equipment.
- Develop maintenance and repair procedures for main feedwater pumps.
- Develop maintenance and repair procedures for booster pump PTA 3800-20.
- Develop maintenance and repair procedures for valves gland package.

Completion of a Level 1 and Level 2 PSA for K2 and R4 (acknowledged as “safety significant measure” by the project sponsor) is not scheduled to take place until after the units have been commissioned and have entered commercial operation (Energoatom 1998: 28).

3.3.6 Summary Design Information

3.3.6.1 Generic VVER-1000/320 Design

Information for this section of the report has been compiled from a number of sources (IAEA 1996a; Mouchel 1998a; Mouchel 1998b; RiskAudit 1994).

VVER is an acronym for Vodo-Vodyannoy Energeticheskiy Reactor (water-cooled, water-moderated, reactor). The VVER-1000/320 design is the third standard VVER design (the earlier models were the VVER-400/230 and VVER-440/213), which was developed between 1975 and 1985 (NEI 1997: 6) in accordance with former Soviet regulation OPB-73 (Mouchel 1998a: 4.1). The first VVER-1000/320 unit was Zaporozhe Unit 1 (Mouchel 1998a: 8.1).

Worldwide VVER-1000/320 unit operating experience at the end of 1998 will be almost 219 reactor-years as demonstrated in Table 3.6.

The VVER-1000/320 design is a pressurized water reactor employing four primary coolant loops, each containing hot and cold leg piping, a water-cooled, shaft-seal reactor coolant pump (RCP), and a horizontal steam generator (SG). The primary piping has an inside diameter of 850 mm. The reactor core is housed in a smooth-bottom reactor pressure vessel (RPV), and consists of sintered, low enriched uranium fuel in fuel rods clad in a zirconium-niobium alloy. The rods are contained in fuel assemblies in a hexagonal geometry. Control rods containing boron carbide are used to control reactor power and shut down the reactor.

The reactor coolant pump seals are cooled by an independent closed-loop system cooled by the essential service water system. This system, designated ZUP, consists of three pumps (backed by diesel generators), two heat exchangers, one surge tank, and a valve header system (RiskAudit 1994: 2/16). Emergency seal injection can be provided by the makeup pumps (IAEA 1996a: 59).

The reactor primary system is housed in a prestressed concrete, large dry containment. The containment is 45 meters in diameter and 53 meters high. The side walls are 1.1-1.2 meters thick. The design pressure range of the containment is 0.2 to 5 bars at an internal temperature of up to 150°C (RiskAudit 1994: 2/15). The design basis leak rate at maximum pressure and temperature conditions is 0.3% volume per day (RiskAudit 1994: 7/29).

Safeguards systems are in a three independent train arrangement house in three separate rooms. Each room contains one train of high pressure injection (HPI), low pressure injection (LPI), and containment spray system (CSS) pumps. There are four passive accumulators for injection into the core in the event of a large pipe break. The HPI, LPI, and CSS all take suction from a common borated water storage tank which is integrated with the containment sump. That is, there is no switchover from injection to recirculation phase of operation – all of these pumps operate in what is (effectively, compared with western PWRs) a recirculation mode. The sump is located at El. +6.6 meters, and is filled with 630 m^3 of borated water (RiskAudit 1994: 11/73).[11]

The thermal power of the reactor is 3,000 MWt. The gross electrical output of the plant is 1,000 MWe, produced by a single turbine-generator. There is a main feedwater system (MFW) which provides water to the steam generators. During startup and shutdown, a two-train auxiliary feedwater system (AFW) is employed at up to 5% of full power. The auxiliary feedwater system is not provided with power from diesel generators in the event of a loss of offsite power. Feedwater under these circumstances is provided by a 3-train emergency feedwater system (EFW). Each train of EFW includes a water tank with a capacity of 500 m^3, which is sufficient for 8-10 hours of operation.[12]

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[11] This water volume is small in comparison with western PWRs. For example, US PWRs tend to have RWST with water volumes in the range from about 950-1,900 m^3.

[12] Train 1 of EFW can supply either SG-2 or SG-4; with valve realignments, it can supply either SG-1 or SG-3. Train 2 can supply either SG-1 or SG-4. Train 3 can supply either SG-2 or SG-4 (IAEA 1996a: 102).

Table 3.6: VVER-1000 Reactor Operating Experience

Unit	COD	Operating Experience
VVER-1000 Prototypes		
Novovoronezh Unit 5	8102	17: 10
South Ukraine Unit 1	8310	15: 02
Prototype Subtotal		**33: 00**
VVER-1000/338		
Kalinin Unit 1	8506	13: 06
Kalinin Unit 2	8703	11: 09
South Ukraine Unit 2	8504	13: 08
Model 338 Subtotal		**38: 11**
VVER-1000/320		
Balakovo Unit 1	8605	12: 07
Balakovo Unit 2	8801	10: 11
Balakovo Unit 3	8904	09: 08
Balakovo Unit 4	9304	05: 08
Khmelnitsky Unit 1	8808	10: 04
Kozloduy Unit 5	8809	10: 03
Kozloduy Unit 6	9312	05: 00
Rivne Unit 3	8705	11: 07
South Ukraine Unit 3	8912	09: 00
Zaporozhe Unit 1	8504	13: 08
Zaporozhe Unit 2	8510	13: 02
Zaporozhe Unit 3	8701	11: 11
Zaporozhe Unit 4	8801	10: 11
Zaporozhe Unit 5	8910	09: 02
Zaporozhe Unit 6	9510	03: 02
Model 320 Subtotal		**147: 00**
VVER-1000 Grand Total		**218: 11**

Residual heat removal occurs in two stages after the reactor is shut down. The first stage employs the reactor secondary systems, with steam bypassing the turbine to the condenser via the BRU-K valves. If the condenser is unavailable (e.g., loss of offsite power), the steam is released to the environment through atmospheric dump valves (BRU-A).[13] Once hot shutdown is achieved, the steam can be either dumped to the atmosphere (BRU-A) or sent to the technological condensers (cooled by essential service water or ESW system) via the BRU-TK valves. The second stage, with the reactor in cold shutdown, involves the use of heat exchangers (also cooled by the ESW system) in the low pressure injection system.

[13] In the Rivne Unit 3 design, the BRU-A valves cannot be used in a station blackout accident after the batteries are depleted (apparently the valves fail closed) (RiskAudit 1994: 11/115). It is possible that this same design feature is present generically for the VVER-1000/320 units, including Khmelnitsky Unit 2 and Rivne Unit 4.

The essential service water systems for K2 and R4 are different. In the case of K2, the ESW system is an spray pond-based system with makeup possible from a nearby river and also from onsite wells (Mouchel 1998a: 4.9-4.15). The system is common to Units 1 and 2.

At Rivne, the system is a semi-open system serving all four units. There are three trains in the system, which uses spray pools as an ultimate heat sink. There is a two-train makeup system common to the three spray ponds. The intake at the river is 3 km from the plant, and these pumps are not backed by diesel generators (RiskAudit 1994: 11/106. Among the components served by the ESW system are the following (RiskAudit 1994: 2/15):

- The standby generator sets (emergency diesel generators).
- The motors and bearings of the containment spray pumps.
- The motors and bearings of the high-pressure safety injection pumps.
- The motors and bearings of the high-pressure boration pumps.
- The motors and bearings of the low-pressure safety injection pumps.
- The motors and bearings of the auxiliary feedwater pumps.
- The motors, bearings, and oil coolers of the primary circuit makeup water system.
- The safety injection heat exchangers.
- The heat exchangers of the RCP intermediate cooling system.
- The heat exchangers of the spent fuel pool cooling system.

A non-safety service water system is also used, and cools the following systems (among others) (RiskAudit 1994: 11/109):

- The SG drain cooling heat exchangers.
- The drain cooling heat exchangers.
- The post-hydrogen combustion installation cooling heat exchangers.
- The sample-taking cooling heat exchangers.
- The coolant pump motor oil coolers.
- The gas analyzer coolers.
- And room air conditioning systems.

The spent fuel pool is located inside the containment in the VVER-1000/320 design. The pool is in a shaft near the reactor shaft. The pool is divided into three compartments which are interconnected. The first compartment and half of the second compartment are occupied by spent fuel. The other half of the second compartment contains fresh fuel. The third compartment is held in reserve for a full core discharge in the event of an incident (RiskAudit 1994: 2/17). The spent fuel pool cooling system (SFPCS) is a three-train system cooled by the raw water system, with circulation pumps backed up by the diesel generators. In the event of a containment isolation signal, the SFPCS is isolated. After about 7 hours, the containment spray system is used for makeup to the spent fuel pool (RiskAudit 1994: 11/104).

The main AC power system for safety systems is a three-train system. Emergency power is provided by diesel generators. Each diesel is supplied with fuel from a day tank with a capacity of 10 m^3; makeup to the day tank is from a 100 m^3 storage tank, however the makeup pumps are not powered from the diesels. Starting of the diesels is from an air system, with sufficient stored air for four start cycles before the air bottles are depleted (RiskAudit 1994: 5/26; 8/50).

The DC power system is backed by batteries in case of loss of charging capability (such as that resulting from loss of offsite power). The generic VVER-1000/320 design has a battery discharge time of 15-20 minutes. IAEA has recommended a minimum discharge time of 2-3 hours (IAEA 1996a: 24).

The plant seismic design is for a design earthquake of MSK-6, and a maximum credible earthquake of MSK-7 (MOHT 1996: Part 2, Section 1, p. 11/94).[14] In addition, the Rivne plant is designed to withstand a 750 km/h, 10 tonne aircraft impact on the containment (RiskAudit 1994: 7/21), a maximum wind of 37 m/s (RiskAudit 1994: 8/51), and a tornado intensity of 2.75 (RiskAudit 1994: 9/8). Subsonic aircraft are forbidden to fly within 8 km of Rivne, and supersonic aircraft are forbidden to fly closer than 30 km. The closest airport is 300 km away, and there are no air travel routes close to the plant (RiskAudit 1994: 9/10).

3.3.6.2 Unique Design Aspects of K2/R4

Based on documentation reviewed for this project, the only unique design aspects of K2 and R4 compared to the generic VVER-1000/320 design involves the essential service water (ESW) system. This system is different at each VVER-1000 nuclear station. It can be stated, however, that for both K2 and R4 the ESW systems are not seismically qualified and the systems are shared by all units onsite (two units at K2, and four units at R4).[15] Both of these factors make the units at these sites subject to potential multi-unit, concurrent core damage accidents.

3.3.6.3 Chernobyl 1/3 Safety Design Status

RBMK is a Russian acronym for Reaktor Bolshoi Moschhnoski Kanalki (high-power, channel-type reactor). The RBMK design contains a core consisting of vertical pressure tubes containing low-enriched uranium fuel which is cooled by boiling light water and moderated by graphite. The reactor is refueled online.

Three generations of RBMK technology are identified. First-generation RBMKs include Chernobyl Units 1 and 2 (designed to OPB-73). Second-generation RBMKs include Chernobyl Unit 3 (designed to OPB-82). The only operating third-generation RBMK is Smolensk Unit 3 (designed to OPB-88). No containment is provided in the RBMK design; second- and third-generation RBMKs incorporate a partial confinement (accident localization system, or ALS) and pressure suppression system. All RBMK units incorporate an overpressure relief system, which is capable of relieving the steam pressure resulting from simultaneous rupture of 4-9 pressure tubes in the core. First-generation RBMKs have only a limited emergency core cooling system (ECCS), capable of responding to break sizes up to 300 mm equivalent diameter. Second- and third-generation RBMK emergency core cooling systems have full primary break size coverage.

Worldwide RBMK operating experience at the end of 1998 will be almost 267 reactor-years as indicated in Table 3.7.

The RBMK core is constructed from graphite bricks through which run 1661 pressure tubes containing uranium fuel rods in bundles. There are additional channels consisting of control and protection channels as well as reflector cooling channels. The graphite moderator is contained in a cylindrical steel vessel which supports the graphite and acts as a container for the helium-nitrogen gas coolant for the graphite. The pressure tubes are 88 mm in diameter and 4 mm thick. The tubes are zirconium tubes with stainless steel connections to the drum separator and feed headers. Each pressure tube contains two 3.5-meter long fuel assemblies, one above the other, containing 18 fuel rods.

Each of the pressure tubes is connected to one of four steam separators which are located in chambers in the upper part of the reactor building. The steam from the steam separators goes directly to the turbines in a direct cycle arrangement.

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[14] IAEA has recommended that VVER-1000/320 units be re-evaluated in accordance with current international practice for a minimum horizontal acceleration of 0.1g with an appropriate design response spectra (IAEA 1996A: 104).

[15] The EIAs briefly describe the ESW systems (Mouchel 1998a: 4.18, 4.19; Mouchel 1998b: 4.17), but fail to note their lack of seismic qualification or the impact of this on the risks posed by operation of the NPPs at Khmelnitsky or Rivne.

Eight main coolant pumps (two in standby) take suction from the steam separators via a header and provide coolant flow to the pressure tubes via a system of distribution headers. The primary circuit consists of two parallel loops which are independent on the water side but joined on the steam side of the circuit. A separate cooling loop serves the control rod and instrumentation channels.

An annular water tank provides biological shielding. There is also an annular space filled with sand as well as upper and lower shields filled with sand, the latter of which acts as supports for the pressure tubes, control rod assemblies, etc.

Each RBMK is expected to undergo a major reconstruction (including retubing and reboring of graphite channels) at about the mid-point of its operating life (15-20 years). Safety improvements are also implemented in an attempt to bring the reactor into compliance with OPB-88 regulatory standards (AEA 1994: 15).

Table 3.7: RBMK Reactor Operating Experience

Unit	COD	Operating Experience
First-Generation RBMK		
Chernobyl Unit 1	7805	20: 07
Chernobyl Unit 2	7905	19: 07
Kursk Unit 1	7710	21: 02
Kursk Unit 2	7908	19: 04
Sosnovy Bor Unit 1	7411	24: 01
Sosnovy Bor Unit 2	7602	22: 10
First-Generation RBMK Subtotal	**126: 07**	
Second-Generation RBMK		
Chernobyl Unit 3	8206	12: 03
Chernobyl Unit 4	8404	02: 00
Ignalina Unit 1	8505	13: 07
Ignalina Unit 2	8708	11: 04
Kursk Unit 3	8403	14: 09
Kursk Unit 4	8602	12: 10
Smolensk Unit 1	8309	15: 03
Smolensk Unit 2	8507	13: 05
Sosnovy Bor Unit 3	8006	18: 06
Sosnovy Bor Unit 4	8108	17: 04
Second-Generation RBMK Subtotal	**131: 03**	
Third-Generation RBMK		
Smolensk Unit 3	9001	08: 11
Third-Generation RBMK Subtotal	**08: 11**	
RBMK Grand Total		**266: 09**

The original RBMK designs raised several safety concerns (Birkhofer 1996: 457):

- Design features of core and shutdown systems related to reactivity control.
- Core cooling systems for first generation RBMK.
- Confinement/containment.
- Possibility of intolerable consequences due to multiple pressure tube rupture.
- Protection against hazards such as fires and floods.
- Quality of equipment and documentation.
- Conduct of operation and operating experience feedback.

The Chernobyl Unit 4 accident in 1986 was caused by the following main factors (Birkhofer 1996: 456):

- Positive void reactivity coefficient; and
- Deficiencies in the core protection system design that resulted in positive reactivity insertion under the conditions in which the reactor had been placed before the accident.

There is broad agreement in the technical community that the original RBMK design had severe deficiencies in the core and shutdown systems. However, between 1987 and 1991, safety upgrades were performed for all RBMK units addressing the most serious problems in this area. Subsequent improvements addressed other issues (Birkhofer 1996: 458).

First generation RBMKs were designed in the 1960s. Several original design features, including the lack of containment, fall short of current safety requirements. The original design features are characterized by large safety deficiencies. The second and third generation RBMKs have improved safety systems and protection against various hazards. The improvements are basically in line with international safety objectives, but there are deficiencies compared with current standards, in particular regarding containment. A partial containment concept has been implemented, but this design provides less complete protection and incorporates less conservatism than other current reactor designs (Birkhofer 1996: 458-459).[16]

Various reviewers have concluded that the RBMKs can be made safe. For example:

- Adolph Birkhofer (Gesellschaft für Anlagen und Reaktorsitherheit mbH, Garching, Germany), in presenting the summary and conclusions of the 1996 IAEA international forum **One Decade After Chernobyl: Nuclear Safety Aspects**, stated, "*From a fundamental point of view there is no reason why a graphite moderated light water cooled pressure tube reactor could not be safe. … Following the backfitting that was directly related to the causes of the Chernobyl accident, the second and third generation RBMKs basically meet most of the defence in depth objectives applied to modern nuclear power plants. Certainly deficiencies remain, e.g. with regard to the limitations of partial confinement and the control and protection system. Improvements to the core protection system should be considered in relation to improved core design providing a decreased void reactivity coefficient. Questions such as the remaining possibilities of reactivity accidents need further attention. No fundamental problems have been identified so far in solving these issues. However, it must be recognized that the limitations of the partial confinement will require additional investigations and efforts for accident prevention. … Altogether it can be stated that the analyses performed so far have shown that, from a technical point of view, the known safety deficiencies of second and third generation RBMKs could be overcome in a way broadly consistent with the defence in depth concept.*"
- GRS, ISPN, and RRC Kurchatov Institute, in a jointly issued report, stated (GRS 1996: 1-2), *"The events which led to the accident in Unit 4 of the Chernobyl Nuclear Power Plant on April 26, 1986 have essentially been clarified during the past ten years. Although there are*

[16] It is to be noted that there are some power reactors other than RBMKs which operate without a full containment (e.g., VVER-440/230, VVER-440/213, CANDU, AGR, MAGNOX). In addition, CANDU reactors (horizontal pressure tube reactors, light water cooled, heavy water moderated) have a positive void coefficient (Adamov 1996: 470).

still some gaps of knowledge relating to details of some phenomena involved in the accident, the knowledge acquired in the meantime is highly sufficient to identify the causes and to take effective measures to prevent a repetition of such an event. … Soon after the accident the Soviet Union initiated measures to remove the deficiencies of the reactor physical design and the shutdown system. The upgradings served the purpose of reducing the high positive void effect, removing the positive shutdown effect and speeding up the shutdown process. These backfitting measures were carried out in all plants in a similar way. The worst deficiencies of the nuclear design have thus been removed. A repetition of the explosion-like accident seems to be hardly possible today."

- AEA Technologies, in an RBMK safety review performed by 170 scientists and engineers from eleven countries, stated (AEA 1994: 8), "*The review has taken careful note of the changes which have been implemented since the 1986 Chernobyl accident. Considerable progress has been made in reducing the void effect and eliminating the positive scram effect which contributed significantly to the accident. It is very unlikely that a reactivity event of this magnitude could occur again. However, risks for accidents with severe environmental consequences remain.*"[17]
- The Nuclear Energy Institute in the United States (a nuclear industry organization with over 300 members in the US and elsewhere, including reactor suppliers, architect/engineer firms, construction firms, labor unions, etc.) has stated in its 1997 "Source Book" on Soviet-designed NPPs (NEI 1997: 3), "*The corrections and modifications made to all of the RBMKs since the Chernobyl accident are generally considered to be adequate to preclude the type of nuclear excursion – a sudden, rapid rise in power level – that occurred at Chernobyl Unit 4 in April 1986.*"

IAEA has prepared a safety issue categorization report for RBMK units (IAEA 1996c) using a methodology similar to that used for identification of VVER-1000 issues (IAEA 1996a). The issues were categorized as High, Medium, or Low as follows (IAEA 1996c: 32):

- **High** – Issues that reflect insufficient defense-in-depth and have a major impact on plant safety. Short term actions have to be initiated to improve safety as applicable to each specific NPP, until the issue is fully resolved.
- **Medium** – Issues that reflect insufficient defense-in-depth and have a significant impact on plant safety. Short term actions might be necessary to improve safety as applicable to each specific NPP, until the issue is fully resolved.
- **Low** – Issues that reflect insufficient defense-in-depth and have a small impact on plant safety. Actions are desirable to improve defense-in-depth, if applicable and effective from a cost benefit point of view.

A total of 58 issues were identified, of which 13 were operational safety issues and not ranked using the above three categories. Of the remaining 45 issues, 19 were ranked as High, 24 were ranked as Medium, and 2 were ranked as Low. These issues are identified in Table 3.8.

[17] This review involved over 170 scientists and engineers from eleven countries (United Kingdom, Germany, Italy, France, Spain, Russia, Lithuania, Ukraine, Canada, Sweden, and Finland) (AEA 1994: 2, 95-100). The study was focused on the Smolensk Unit 3 and Ignalina Unit 2 RBMKs, both third-generation RBMK designs (AEA 1994: 5); Chernobyl Units 1 and 2 are first-generation RBMKs, while Chernobyl Unit 3 is a second-generation RBMK.

Table 3.8: RBMK Safety Issues Identification by IAEA

Issue Number	Issue Rank	Issue Title and Description
C1	High	**Core design and core design methods** – Development of three-dimensional reactor core models including thermohydraulic feedback and coupling between the 3-D core codes and thermohydraulic modeling of the main coolant circuit.
C2	High	**Core design void reactivity coefficient of the primary and CPS circuit** – Modification of control rod design to reduce CPS void effect; verification and validation of calculation and measurement of void reactivity coefficient of primary circuit; and study of burnable poison (homogeneous solution) to reduce void effect instead of additional adsorbers (heterogeneous) in order to reduce local effects.
C3	Medium	**Spatial power control and protection** – Implementation of 12-zone automatic control/local emergency protection system; independent analyses to confirm designers calculations; implementation of rod insertion limits to limit reactivity worth of individual rods under erroneous rod withdrawal conditions.
C4	High	**Operational reactivity margin (ORM)** – Automation of shutdown actions when ORM value falls below safety limits; implement measures to reduce the contribution of ORM to void effect; implement measures to achieve homogeneous spatial distribution of ORM in the core to prevent high positive local void effects.
C5	High	**Additional shutdown system** – Implementation of fully independent and diverse shutdown system (maintaining reactor subcritical in the event of a CPS LOCA).
C6	High	**Subcriticality margins in first generation RBMKs** – Measures to achieve requirement of 2% subcriticality assuming the most reactive stuck rod for first generation RBMKs (e.g., Chernobyl Units 1 & 2).
I&C1	High	**Diversity and segregation of I&C systems** – Review of diversity and segregation of I&C systems against IAEA recommendations and international best practice; physically segregate electronic equipment for 24 fast action scram rods or implement a separate train of electronics; separation of cables between the core and electronics for flux instruments to form at least three segregated groups; bring ECCS and other support systems to standard of segregation adopted for Smolensk Unit 3; and examination of diversity of means to generate input to safety and control systems.
I&C2	High	**Initiation of ECCS and other safety systems** – Implementation of ECCS initiation equipment to level of Smolensk Unit 3; perform review of other systems important to safety and bring up to standard of Smolensk Unit 3 ECCS initiation equipment.
I&C3	Medium	**I&C system maintenance and periodic testing** – Audit of maintenance and test procedures; extension of scope of periodic detector testing to include discriminator curve measurements checks on detectors; bring QA measures for completion and recording of maintenance activities to IAEA standards; improve scope of equipment failure data base to allow trend monitoring of equipment performance and early identification of aging problems.
I&C4	Medium	**Reliability of I&C systems** – Perform reliability analysis (fault tree and failure modes and effects analysis) of I&C systems, and comparison of analysis with failure data to ensure fidelity of results and that the performance of the equipment in service is as good as claimed in the safety documents.
I&C5	Medium	**Replacement of NPP main computer** – Replacement of existing computer with a modern distributed system with dedicated units for functions important to safety; implementation of additional screens and displays to improve the operator interface.

Issue Number	Issue Rank	Issue Title and Description *(cont.)*
I&C6	Medium	**I&C equipment upgrades** – Installation of modernized versions of I&C systems as soon as possible; replacement of static logic with dynamic logic to reduce failure rates; replacement of slow self powered detectors with prompt detectors for core monitoring and protection; check of environmental qualification of all new equipment prior to installation.
I&C7	Medium	**Operator support** – Improvement of man-machine interface, including safety parameter display system and upgrade of station computers.
P1	High	**Fulfillment of inspection requirements** – Following of examination requirements for reactor pressure boundary integrity and establishment of effective verification mechanisms; implementation of limits on plant operation based on outstanding inspection work; implementation of authorization for further plant operation based on criteria agreed to with the nuclear regulatory authority.
P2	High	**In-service inspection** – Implementation of predictive ISI approach of following defects development and using fracture mechanics; establishment and maintenance of comprehensive ISI documentation; optimization of ISI with respect to inspection locations, frequency, and techniques; use of modern equipment for repeatable measurements of sub-critical defect sizes (consideration of computerized ISI data acquisition, handing, and storage systems).
P3	High	**Break of critical components** – Application of leak-before-break (LBB) the primary coolant circuit components inside and outside ALS; validation of leak detection system.
P4	High	**Fuel channel and tract integrity**— Use of NDE and post-irradiation destructive testing of fuel channels; use of automatic inspection equipment to enable repeatable testing; reduction of number of in-line components which on failure can result in water flow blockage; use of more reliable components; and analysis of possible backfitting of bypasses.
P5	Low	**Special channel integrity** – Implementation of leakage monitoring system for special channels, with high leakage or low water volume resulting in automatic reactor shutdown; periodic removal of special channel in high flux area for analysis of material properties (including hydrogen intake); optimization of ISI programme for special channels.
P6	Medium	**Fuel handling** – Automation of the final refueling machine positioning and of the remaining functions of fuel replacement up to closing of plug; improved sealing of fuel channel plug to eliminate leakages of steam to the reactor hall; analysis of primary coolant circuit system to consider configurations with fuelling machine attached to the circuit.
P7	Medium	**Seismic and aging assessment** – Perform additional seismic analysis of primary pressure boundary piping, components, and component supports; perform additional work on aging of metals and concrete to determine changes in properties and their effects on structural behavior.
AA1	Medium	**Scope and methodology for accident analysis** – Development of RBMK models using state-of-the-art codes; perform additional accident analysis including additional cases, sensitivity studies, and partial power initial conditions; perform best estimate analyses to develop symptom-oriented EOPs and to justify proposed additional trip signals.
AA2	High	**LOCA analysis** – Establishment of rationale for DBA definition, including conservative assumptions and RBMK-specific acceptance criteria; perform additional LOCA analyses with state-of-the-art validated codes to complete the spectrum of cases and to perform sensitivity studies to assess uncertainties.
AA3	High	**Cavity overpressure protection** – Analyses of pressure tube failures in accident scenarios (DBA and BDBA); increase of cavity overpressure protection system capability; evaluation of tradeoffs between protection of reactor cavity and radiological consequences of releases.

Issue Number	Issue Rank	Issue Title and Description *(cont.)*
AA4	Medium	**Steam line break analysis** – Reanalysis of steam line break accidents with state-of-the-art codes with coupled neutronics and thermohydraulic models; analysis of additional cases, including partial initial power conditions and different break locations (inside and outside drum separator compartments); perform best estimate analysis to evaluate existing and proposed ECCS initiation signals.
AA5	Medium	**Pipe whip** analysis – Perform systematic pipe whip analysis with special emphasis on steam and feedwater lines; if damage by pipe whip cannot be excluded, analyze sequences with consequential breaks.
AA6	Medium	**Loss of power** – Perform loss of power analyses on plant-specific basis; perform station blackout analysis to estimate times available for operator actions and to develop accident management strategies.
AA7	Medium	**Radiological consequence analysis** – Independent evaluation of methodology used for consequence analysis of pressure tube rupture; comparison with state-of-the-art methods and codes used in western countries.
AA8	Medium	**Performance and utilization of PSA** – Perform plant-specific PSA to identify plant weaknesses, assessment and prioritization of plant modifications, and input for accident management procedures; development of list of generic RBMK-specific initiating events; development of RBMK-specific reliability data base; review of PSA by independent experts before use.
AA9	High	**Anticipated transient without scram (ATWS)** – Perform comprehensive ATWS analyses.
AA10	Medium	**External hazards** – Perform comprehensive external hazards analysis (airplane crash, flooding, explosions, etc.); perform comprehensive seismic analyses.
S1	High	**ECCS – capability and performance** – Upgrade first generation RBMK ECCS; improvement of primary coolant circuit component design and reliability.
S2	Medium	**Long-term cooling and water make-up** – Improve water makeup system reliability for long-term cooling; improve residual heat removal system reliability in cold shutdown conditions
S3	High	**ECCS reliability improvement** – Review ECCS layout with respect to physical separation requirements; install, where appropriate, an additional diverse emergency feedwater system; improve ECCS equipment redundancies for first-generation RBMKs.
S4	High	**Reactor trip and ECCS actuation signals** – ECCS automatic startup on total loss of feedwater should be investigated; if necessary, implement reactor trip and ECCS initiation on low flow in many channels; use of diverse actuation signals for ECCS.
S5	Medium	**Interfacing system LOCA** – A study should be performed of interfacing LOCAs in the framework of a PSA.
S6	High	**Adequacy of confinement function** – Installation of steam line isolation valves; improvement of ALS leaktightness; upgrade of confinement capability of first-generation RBMKs; investigation of upgrade of upper rooms (reactor hall and steam separator rooms) with respect to leaktightness and confinement function; replacement of main steam safety valves with better quality equipment.
S7	Medium	**Reliability of ultimate heat sink** – Investigate and improve ESW to avoid single failures that produce total loss of heat sink; evaluate automatic isolation of non-essential components; review of ESW system operation to determine if the system is adequately monitored to alert operators to perform actions to avoid complete loss of the system.

Issue Number	Issue Rank	Issue Title and Description *(cont.)*
S8	Medium	**Reliability of electrical system** – Update electrical system to current standards; increase battery depletion time up to one hour.
S9	Medium	**Electrical equipment qualification** – Investigation of seismic and environmental qualification of electrical equipment.
S10	Medium	**Diesel generator reliability** – Consider increase in undervoltage set point for diesels for automatic startup; improvement of diesel generator; evaluate improvements to diesel generator cooling system.
FP1	High	**Passive fire protection** – Review RBMK fire statistical data to identify fire hazards; avoid fire loading from floor plastic, flammable cables, housekeeping, etc.; confirm the venting of gaseous flammables; build fire walls and fire doors/ locks to improve fire separation between redundant trains, control cabinets in the control room, control room and adjacent electric rooms, turbine hall and intermediate building, main transformers and turbine hall, and diesel generators and fuel oil tanks; install fire insulation on power cables, load bearing steel structures, and pipe and cable penetrations; install fire resistant ventilation in control room and electrical rooms.
FP2	Medium	**Automatic fire detection** – Upgrade existing fire detection system by advanced, reliable equipment and increase the coverage.
FP3	Medium	**Manual fire suppression capability** – Improvements for fire fighting, including protective clothing and up-to-date equipment such as nozzles, fire hoses, and portable equipment; replace fire hoses and provide adjustable fog/straight stream nozzles; provide dry stand pipes to the roof of the turbine hall with permanent hoses; provide portable extinguishers corresponding to western practice; add, repair, and maintain emergency lighting.
FP4	Medium	**Automatic fire suppression capability** – Installation of fixed automatic suppression systems for the main turbine generator bearings, the underside of the turbine hall main deck, and the underside of the turbine hall roof; evaluate the coverage and quality of existing fixed suppression systems in cable tunnels and rooms.
FP5	Low	**Fire water supply** – Automate fire protection water pump startup sequencing to replace manual system.
OS1	N/A	**Organization and staffing** – Review organizational structure of NPP management; establish nuclear safety committee; carry out periodic independent review of NPP management (peer review); implement responsibility and accountability at the lowest levels.
OS2	N/A	**Quality assurance** – Complete the development of QA for operation, including operational management in normal and accident conditions, maintenance and repair, and root cause analysis; independent assessment of QA programme effectiveness; all supplied equipment for reconstruction and modernization must be manufactured and repaired in strict compliance with QA programme requirements.
OS3	N/A	**Safety culture** – Establish relationship between personnel based on trust and openness; management should encourage personnel to improve their qualification, self evaluation, and contribution to improvement of safety; self-critical attitudes should be encouraged and developed in all levels of the organization; management should publish its safety culture policy to all state, develop a specific safety culture document, and educate staff on safety culture.

Issue Number	Issue Rank	Issue Title and Description *(cont.)*
OS4	N/A	**Management of documents** – Operational documentation should be kept in good condition and be available for use in the control room and by field personnel; documentation should be regularly checked and necessary changes introduced; state-of-the-art computer techniques should be used for documentation storage and modification; copies of basic operational and design documentation must be securely stored in a separate location taking into account fire protection, flood, etc.
OS5	N/A	**Material condition** – Special attention should be given to housekeeping; equipment must be labeled correctly and lighting must be adequate; equipment must be clean and there must be good access to operate and maintain equipment; leaks from equipment must be minimized.
OS6	N/A	**Training programmes and materials** – Outside the formal programme, seminars should be arranged to actively promote a strong nuclear safety culture in staff at all levels; continuous training should be introduced; training programmes, facilities, and materials should be brought up to the level of international practice.
OS7	N/A	**Operating procedures for normal operation** – Enhancement of normal operational procedures may be achieved by QA programmes in operation; guideline development for writing operating procedures will ensure necessary quality and completeness of procedures.
OS8	N/A	**Emergency operating procedures** – International cooperation should be used in developing symptom-based emergency operating procedures and in training personnel to use them; EOPs should include accident management guidelines.
OS9	N/A	**Experience feedback and event investigation** – ASSET methods used to analyze the root cause of events should be used; action plans must be developed to resolve the issues identified by this process; special programmes should be introduced to carry out corrective measures and check the completion of the same measures at other power plants; the methodology must be used to recognize events which may have occurred if the scenario had developed differently.
OS10	N/A/	**Maintenance programme** – Maintenance programmes should be updated to include all the elements of fully effective preventive, predictive, and breakdown maintenance programmes; special attention should be paid to feedback of experience.
OS11	N/A	**Modification control** – A procedure for temporary and permanent modifications should be established; effective control of implementation of modifications should be established to ensure that modifications are consistent with overall plant safety.
OS12	N/A	**Surveillance test programme** – The period of surveillance testing must be established based on reliability data of the component taking into account operating experience; surveillance tests should be carried out so that the intended function of the system is confirmed; personnel should be provided with detailed instructions and acceptance criteria for tests that verify important safety parameters and functions of systems and trains.
OS13	N/A	**Radiation protection programmes** – ALARA programme should be in place; remote controlled ISI equipment should be used in high dose areas.

3.4 Comments on the Safety Chapter of the EIA and Energoatom Project Presentation

3.4.1 Risk Comparison Between Chernobyl 1/3 and K2/R4

The K2 and R4 EIAs both claim that the K2 and R4 plants, following the implementation of the modernization and upgrade programmes, will be less risky than continued operation of Chernobyl Units 1 and 3 after completion of an upgrading and safety programme for those units (Mouchel 1998a: 0.7; Mouchel 1998b: 0.8). While this **may** be true, it is certainly the case that neither EIA **demonstrates** it to be true.

Considerable additional information, which should be contained in the EIAs, would be needed in order to **demonstrate** that Khmelnitsky Unit 2 and Rivne Unit 4 will be less risky than continued operation of Chernobyl Units 1 and 3. Among the information required for this purpose is the following:

- A **definitive** and **detailed** listing of the upgrades which will be implemented at K2 and R4, both before and after startup. This is very important because Mouchel based its comparisons on **upgraded** VVER-1000/320 (Mouchel 1998a: 0.7),[18] and **upgraded** RBMK-1000 designs, without specifying in detail what the upgrades included in either case.
- A **definitive** and **detailed** listing of the upgrades which Mouchel considered for implementation at Chernobyl Units 1 (or 2) and 3 (note that Units 1 and 2 are first-generation RBMK, while Unit 3 is second-generation RBMK, and thus the upgrades will be different for the two units). This is very important for the reason stated immediately above.
- A detailed treatment of probabilistic safety assessment (PSA) and severe accident study results for VVER-1000/320 units and RBMK units, including a discussion of how these results apply specifically to K2, R4, and Chernobyl Units 1 and 3. This is absolutely necessary since Mouchel bases its expressed preference for completion of K2 and R4 on the basis that there is a lower frequency of core damage and a lower likelihood of a catastrophic release of radioactivity for K2 and R4 compared with the RBMK units at Chernobyl.
- A detailed deterministic comparison of the upgraded K2 and R4 units with the upgraded Chernobyl units. This is important because Mouchel bases its expressed preference for completion of K2 and R4 on the basis that their normal operational releases of radioactivity will be lower than for the RBMK units at Chernobyl.

Without such information, the reader of the K2/R4 EIAs is left with a bald conclusion by the authors of the studies that completion of K2/R4 results in a risk reduction compared to continued operation of upgraded RBMKs at Chernobyl. **No** analyses are presented in support of this conclusion, and **no** documents containing such an analysis are cited in support of this conclusion. The conclusion, therefore, is unsupported and **cannot** be sustained on the basis of the EIAs as they were issued in draft form for public comment.

As discussed first in Section 3.3.3.1, above, there is PSA information in the public domain on both the VVER-1000/320 and RBMK designs. This information, while evaluating plants other than K2, R4, and Chernobyl, is nonetheless of more relevance than the unsupported speculation in the current draft EIAs, and serves to call into question the basis for the conclusion

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[18] Specifically, Mouchel states (Mouchel 1998a: 0.7; Mouchel 1998b: 0.7; underlining emphasis added), "*Given that the* [K2/R4] *project has been proposed as an alternative to continuing to operate the Chernobyl NPP, an initial comparison has been provided for the 'base case alternative'. In this comparison it is assumed that operation of two of the four Chernobyl RBMK reactors is continued **following the completion of an appropriate upgrading and safety programme**.*"

stated in the EIAs that upgraded VVER-1000/320 units will **necessarily** present a lower risk than continued operation of upgraded RBMK units.[19]

Moreover, it must be recognized that there are no residents within 30 kilometers of Chernobyl. Should another accident occur there, there would be no early fatality potential except among the operating staff. The 30-km radius around the plant is **already** abandoned, and the environmental impacts for this area would occur against the background of an existing severe accident-contaminated environment. In contrast, should a large release accident occur at either the Khmelnitsky or Rivne sites, there are substantial populations residing within 30 km of those units, and the environment would not be significantly contaminated from any previous accidents or incidents at these units. The EIAs cannot pretend that the area around Chernobyl is uncontaminated from the 1986 accident at Unit 4 and continues to be occupied by a population which has since been relocated.

3.4.2 Risk Comparison Between K2/R4 and Recently Re-Licensed EU NPPs

The EIAs for K2 and R4 conclude that (Mouchel 1998b: 11.1):

> *The proposed project is based on a reactor type which is already tried and tested in Ukraine. The proposed modernization programme takes into account all major safety issues and international requirements since it has been demonstrated that, if implemented, the safety of the plant would be comparable to that achieved in the European Union for NPPs of similar vintage recently re-approved by national safety authorities.*

This statement, while superficially reassuring, is so heavily qualified and so non-specific as to be nearly useless for characterizing the level of safety which is purported to be achieved by the upgrade program. The following statements can be made about this conclusion:

- The VVER-1000/320 reactor type is "*tried and tested in Ukraine*" to the extent that the reactor actually works (it produces power as intended). But the statement does not indicate in what aspects the design is considered to be "*tried and tested*" – e.g., is this a statement about its reliability as a power producer, its safety, its constructability, or some other aspect of the design? Regardless, the worldwide VVER-1000 operating experience is limited to about 219 reactor-years as indicated in Table 3.6. If the operating experience is limited strictly to VVER-1000 Model 320 units, then the operating experience totals only 147 reactor-years. In comparison, worldwide RBMK operating experience is bout 267 reactor-years (split among 126 reactor-years for first-generation plants like Chernobyl Units 1 and 2; 131 reactor-years for second-generation plants like Chernobyl Unit 3; and 9 reactor-years for third-generation plants). Whether the VVER-1000/320 reactor type is "*tried and tested in Ukraine*" or elsewhere is unimportant to whether this design should be used as a replacement for the Chernobyl RBMK design. The VVER-1000/320 is no more tried and tested than the RBMK or, for that matter, than the VVER-400/230 or VVER-440/213 designs, which have accumulated about 317 reactor-years and 230 reactor-years of experience, respectively. On the other hand, the Westinghouse AP-600 is untested, but potentially would be a much less risky choice (from the severe accident risk standpoint) than a VVER-1000/320. The portion of the statement above about the VVER-1000/320 design being "*tried and tested in Ukraine*" is nearly irrelevant and largely meaningless in the context of the EIAs.
- The statement's conclusion that the modernization programme takes into account all major safety issues and international requirements is **predicated** on the safety of the plant being demonstrated to be comparable to EU NPPs of similar vintage which have been recently

[19] Since it is known that essentially all of the VVER-1000/320 and RBMK units have PSA studies in progress or have completed such studies, there should be an abundance of PSA and severe accident study information available, but perhaps not publicly available. We assume that Energoatom, Mouchel, EBRD, or some other entity associated with the project proposal, project design, or project construction could gain access to this information and provide an appropriate analysis of it in the EIAs and discuss how it pertains to the likely risk profiles of the K2, R4, and Chernobyl plants. Of course, the proposed upgrades in all cases would have to be taken into account.

re-approved for operation by nuclear regulatory authorities. However, such a demonstration is not actually **made** in the EIA or the documents which are cited therein. Rather, comparability is **asserted**, not **demonstrated** by a comparative risk assessment or by a deterministic comparison of the K2/R4 designs with specific "*international requirements*".

- It has not been **demonstrated** that "*all major safety issues*" have been taken into account in the modernization programme. The modernization programme is **entirely** deterministic in nature. Even then, there is not even a complete listing in the EIAs of the components of the modernization programme. Without such a listing and a comparison between the studies cited as the basis for identifying safety issues (e.g., IAEA-EBP-WWER-05, RiskAudit, etc.), there is only the **assertion** that these issues have been addressed by the modernization programme. Whether in fact "*all major safety issues*" have been addressed is left for the reader of the EIAs to determine. Moreover, since the modernization programme has little or no probabilistic basis, it cannot be said to have addressed all major safety issues.
- No PSA results for K2 or R4 are cited, and no such studies are planned to be completed until **after** commissioning. No PSA results for other VVER-1000/320 units are cited. Thus, there is no basis for asserting that severe accident-related safety issues have been addressed by the modernization programme.
- The phrase "*similar vintage*" is not defined in the EIAs. What does this mean? Does "*similar vintage*" refer to a plant designed in the 1970s (the vintage of the VVER-1000/320 design dates to the original design of Zaporozhe Unit 1 in 1978) (Mouchel 1998a: 8.1)? Or a plant designed in the 1980s, which is when the sister units (Khmelnitsky Unit 1 and Rivne Unit 3) came into operation? Or to a plant designed in the late 1990s and early 2000s, which is when the K2/R4 modernization programme will be implemented if the project is approved?
- No specific EU NPP is cited for comparison purposes. No specific nuclear regulatory authority's regulations for re-approval are cited either, nor are the IAEA periodic safety review requirements cited. Thus, the basis for re-approval of an unspecified design, and the comparison of that unspecified design with K2/R4, is completely lacking. The reader does not even know if Mouchel compared the level of safety of K2 and R4 (whatever that level is) with a PWR, a BWR, an AGR, a MAGNOX reactor, or some other design. Indeed, since Finland is part of the EU, it could even be the case that the safety level of K2 and R4 was compared with the Loviisa plant, which is a VVER-440 design with a mixture of VVER and western safety equipment and an ice condenser containment. In short, the reader is **completely uninformed** as to the basis for comparison.
- The safety basis for the comparison is not cited. Is the comparison based on deterministic comparisons of regulatory compliance? If so, with which country's regulations and with what vintage of regulations (current, 1970s, 1980s, 1990s, etc.)? Is the comparison with IAEA NUSS standards (and, if so, with which standards and which vintage of the standards)? Is the comparison based on risk (core damage frequency, large release frequency, consequences per reactor-year, etc.)? If so, what basis was used to estimate these parameters for K2, R4, and the unnamed EU reactors since no PSAs exist for K2 or R4 (such studies are planned for **after** commissioning).

In short, the statement cited above at the beginning of this subsection is so non-specific and so heavily qualified that it is meaningless. Far from serving as an "*internationally acceptable benchmark for safety levels of nuclear power units with VVER 1000 type reactors*" (as asserted by EBRD; see EBRD 1998), the safety levels of K2 and R4 are indeterminate, perhaps better than the basic VVER-1000/320 design but by how much is unknown. How the upgraded design compares with western or international safety standards, or with other NPP designs, is also unknown due to inadequate information presented in the EIAs. Indeed, the EIAs do not even contain a comparison between the K2/R4 designs and Ukrainian national safety standards and regulatory criteria, much less a comparison with any consistent set of western safety criteria (i.e., IAEA NUSS standards, German KTA criteria, USNRC regulations, etc.).

3.4.3 K2/R4 Beyond Design Basis Accident (BDBA) Analysis in the EIAs

From the standpoint of deterministic reactor design, there are two classes of accidents. The first class is referred to as design basis accidents (DBAs), which are events for which the reactor design must be able to respond and bring the reactor to safe shutdown. The second class is beyond design basis accidents (BDBAs). The reactor is not specifically designed to be able to handle these accidents without adverse consequences, but due to conservatism in design (safety margins) the reactor may be able to safely respond to some of these events, especially when operator response actions are considered.

In safety analysis reports, typically both DBAs and BDBAs are analyzed. The purpose in analyzing BDBAs is to ascertain how much margin might be available in the design and what the consequences of BDBAs might be. If a BDBA can be successfully managed, there is very little public consequence. However, there is a subclass of BDBAs referred to as "*severe accidents*" which involve severe core damage and core melting. If such an accident is accompanied by failure of the containment to isolate, structural failure of the containment due to accident phenomenology (e.g., hydrogen burn), or bypass of the containment (e.g., interfacing LOCA), the consequences to the public can be very significant. The most common and disciplined way to study BDBAs is to perform a probabilistic safety assessment (PSA).

The K2 and R4 EIAs purport to analyze the consequences of a BDBA, which they refer to as "*the most representative*" such BDBA. The EIAs acknowledge than a full scope BDBA study for a VVER-1000/320 has not yet been performed (Mouchel 1998b: 8.15). It follows perforce that such a study has not yet been completed for either K2 or R4.

The EIAs go on to state (Mouchel 1998b: 8.15):

> *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 EIAs do not point out that while the selected accident is beyond the **original** design basis of the plant, it is widely acknowledged that the accident should now be considered to be a design basis accident. The EIAs also fail to point out that the accident, as analyzed, does **not** result in severe core damage and is thus not a **severe accident** (such as the Three Mile Island or Chernobyl accidents). The EIAs note that there are other BDBAs (such as anticipated transients without scram or ATWS; total loss of feedwater; and the total loss of onsite and offsite electrical power, or station blackout), however these BDBAs are not analyzed. At least two of the identified accidents (loss of all feedwater and station blackout) will proceed to core damage if they are not recovered in time (loss of all feedwater can be recovered by timely implementation of bleed and feed cooling; station blackout can be recovered by recovery of either offsite power or onsite power before core damage occurs). The exclusion of such severe accidents in favor of one not involving severe core damage is inconsistent with the entire purpose of analyzing BDBAs. One cannot analyze a BDBA which does not result in severe core damage and state or imply that the analysis is "*representative*" of all BDBAs, especially those which **do** result in severe core damage.

The accident actually analyzed has the following aspects (Mouchel 1998b: 8.15):

- Primary to secondary leakage with an equivalent diameter of 100 mm, corresponding to a steam generator header failure.
- Opening and failure to close of an atmospheric dump valve on the damaged steam generator, permitting release of primary coolant to the atmosphere (estimated at 600 tonnes of the mixture of primary and secondary coolant).
- Staff actions beginning at 10 minutes are assumed to cause in a fast cooldown of the reactor, resulting in subcooling of the primary coolant by 40 minutes after the initiating event.

- No additional cladding failures occur.
- The fission product content in the primary coolant is determined by the **design basis** cladding failure rate (1% of the gas gap releases and 0.1% due to direct contact of the fuel with coolant, as well as spike release of radionuclides from such fuel elements).

The release fractions for this accident, expressed in the fraction of the entire core inventory, are as follows:

Kr-85m, 1.5×10^{-5}	I-132, 8.3×10^{-6}	Cs-134, 2.0×10^{-7}
Kr-87, 2.4×10^{-5}	I-133, 4.4×10^{-6}	Cs-137, 4.6×10^{-6}
Kr-88, 2.2×10^{-5}	I-134, 3.6×10^{-6}	La-140, 8.4×10^{-9}
Xe-133, 1.5×10^{-5}	I-135, 3.5×10^{-6}	Ce-144, 1.6×10^{-8}
Xe-135. 2.0×10^{-5}	Sr-90, 1.1×10^{-8}	
I-131, 4.8×10^{-6}	Ru-106, 3.9×10^{-9}	

The **largest** of these release fractions is less than one-hundredth of one percent of the core inventory of the isotope (Kr-87). Clearly, given that there are PWR accident sequences – even in western PWRs – which are capable of releasing substantial core inventories (USNRC 1998a), the choice of the accident selected as a "*representative*" BDBA in the K2/R4 EIAs is open to serious question. The table below compares the release fractions in the K2/R4 EIA BDBA with the release fraction of severe accidents at the Surry reactor from the NUREG-1150 study.[20]

Accident Sequence	Kr & Xe	I	Cs	Sr	Ru	La	Ce
K2/R4 BDBA, SG collector leak, no severe core damage	2.4×10^{-5}	8.3×10^{-6}	4.6×10^{-6}	1.1×10^{-8}	3.9×10^{-9}	8.4×10^{-9}	1.6×10^{-8}
Station blackout, steam explosion (RSUR-1)	1.0	0.35	0.31	0.06	0.006	0.006	0.01
Station blackout, containment leak (RSUR-2)	1.0	0.06	0.03	0.003	0.001	$<10^{-10}$	$<10^{-10}$
Interfacing LOCA (RSUR-4)	1.0	0.12	0.12	0.025	$<10^{-10}$	0.0003	0.004

The accident analyzed by the K2/R4 EIAs is cited by the IAEA as a **design basis accident** (DBA), **not** a BDBA (IAEA 1996a: 55). The rupture of the SG internal manifold is recognized as a DBA for Temelin (Holan et al., 1994: 17), and this accident included consideration of release of primary coolant through a stuck-open SG atmospheric relief valve (Burnett & Tauche 1994: 14). Although the accident was not **analyzed** as a DBA in the pre-1990-era safety analysis reports, severe cracking of collectors has been observed at operating VVER-1000/320 units (IAEA 1996a: 51).

In order for the accident to be beyond design basis, the equivalent diameter needs to be greater than 100 mm, which would lead to a bypass of the containment and the loss of core cooling in the long term due to depletion of the borated water available for high and low pressure injection (IAEA 1996a: 55). Another way for the accident to progress to BDBA is to as-

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[20] The combined frequency of the three cited Surry PWR accidents is about 4×10^{-6} per year; the CDF for Surry for internal and external events is about 1.7×10^{-4} per year. Although this implies that only one in about 40 severe accidents results in containment failure or bypass, this is a highly plant-specific result and depends on the relative likelihoods of various types of accidents which could be quite different from plant to plant. Especially important in this regard is the fraction of accidents resulting in containment bypass.

sume an operator error, which would allow the ECCS water inventory to be discharged to the secondary side of the plant. This would ultimately result in severe core damage and containment bypass. As the accident was **actually** analyzed by the EIAs, no severe core damage occurred and the reactor was assumed to be brought to safe shutdown by operator action.

3.4.4 Spent Fuel Management Risks

The K2/R4 EIAs do not address risks arising from spent fuel management. While it is true that VVER-1000/320 plants have their spent fuel pools located inside containment, spent fuel pool accidents occurring during plant shutdown could occur under conditions in which containment integrity is not fully implemented. Furthermore, it is well known that severe spent fuel pool accidents can result in the production of very large quantities of hydrogen (as a result of the classic metal-water reaction between steam and hot spent fuel, very similar to what goes on in a reactor core in a severe accident), the implications of which for containment integrity of a VVER-1000/320 unit have not yet to date been considered. This issue is potentially important since the radiocesium releases from spent fuel pool severe accidents are much larger than for reactor core accidents, and it is the cesium releases that determine long-term accident consequences.

The potential for spent fuel pool severe accidents at RBMK units is largely an unknown potential due to lack of analysis. Any comparison of the risks of completion of K2 and R4 versus continued operation of Chernobyl Units 1 and 3 should include spent fuel management risks.

Severe spent fuel pool accidents have been subjected to a detailed evaluation for US PWRs. This evaluation has found that under some circumstances, a self-sustaining zirconium cladding fire can occur, which is an exothermic reaction resulting in melting of the spent fuel and the release of radioactivity, including cesium isotopes. Since the radiocesium inventory of the fuel in the spent fuel pool can be quite large compared with the reactor core inventory, the consequences of a severe spent fuel pool accident could be quite large (BNL 1987; USNRC 1989).

In the VVER-1000 context, since spent fuel pool cooling is interrupted in the event of containment isolation, and since containment isolation would eventually occur for nearly all severe reactor core accidents, it is important that the consequences of severe spent fuel pool accidents be well understood for VVER-1000 units. If a severe reactor core accident can result in a consequential severe spent fuel pool accident at a VVER-1000, the overall consequences of the accident could be very substantial with a radiocesium release potentially well in excess of that which was experienced in the Chernobyl accident.

The modernization programme does not address the subject of severe spent fuel pool accidents at all.

3.4.5 Potential Chernobyl 1/3 Replacement Options Other than K2/R4

The K2 and R4 EIAs do not consider any options for replacing the lost capacity of Chernobyl Units 1 and 3 **other** than completion of K2 and R4. This is completely inconsistent with the fundamental purpose of environmental impact analysis.

The scope of an EIA is driven by the proposed action and the range of reasonable alternatives to that action. The range of reasonable alternatives usually includes a “no action” alternative (essentially a continuation of current conditions) as well as other alternatives. The “no action” alternative provides a current environmental baseline against which the impacts of the proposed action and reasonable alternatives can be compared.

The purpose of identifying reasonable alternatives is to examine whether there are alternatives to the proposed action that will avoid or minimize any adverse effects of the proposed action on the environment.[21] Indeed, USDOE guidance on preparation of EIA documents states that reasonable alternatives should be addressed even if they are outside the agency's jurisdiction, **even if they conflict with lawfully established requirements** (USDOE 1993: 10).

Concerning specifically safety issues, EIAs should not just look at design basis events – that is, events that are within the scope of occurrences for which the facility has been designed. EIAs should also look beyond the design basis to see if there may be events of such large consequences that they need to be considered (USDOE 1993: 28).

There are a range of options to replace Chernobyl capacity which are credible, feasible, and within the cost range cited by project proponents for K2 and R4. In short, these options include the following:

- Replacement of Chernobyl Units 1 and 3 by a combined-cycle combustion turbine unit (2,300 MWe capacity, estimated cost of approximately 2.0 billion USD).[22]
- Modernization and upgrade of the existing eleven operating VVER-1000/320 units in Ukraine (estimated to cost 1.7-1.8 billion USD), to both recoup lost Chernobyl capacity by reliability and online performance improvements and to improve plant safety.
- Improvements in energy efficiency in the Ukrainian economy.
- Improvements and refurbishment of non-nuclear generating units.
- Some combinations of the above options.
- Last, but not least the question has to be raised whether the replacement of the Chernobyl capacity is necessary or not.[23]

Instead of considering these possibilities, the EIAs simply reject them outright and focus **only** on the negotiated (between G-7/EU and the government of Ukraine) option of completing K2 and R4 or maintaining the base case of continued operation of Chernobyl. The EIA should not be scoped to consider only alternatives which are politically acceptable to current governments; rather, all reasonable options should be evaluated on an equal footing. The purpose of an EIA is **not** to justify a political choice by artificially constraining the alternatives; rather, the purpose of an EIA is to **inform** the choice of actions.

[21] In the United States, for instance, the regulations of the Council on Environmental Quality (which was established by the original EIA law, the National Environmental Policy Act, or NEPA) hold that the comparative analysis of alternatives is the heart of an EIS. CEQ guidance states that reasonable alternatives include those that are practical or feasible from a common sense, technical, and economic standpoint (USDOE 1993: 9).

[22] Such an option was in fact proposed by the Ukrainian government in 1995, but rejected by the G-7 countries. Whatever the merits of the G-7's position (and these merits are not publicly documented so far as we have been able to determine), there is not reason for not discussiing this possible option in the context of the EIAs.

[23] The experts of the Sussex group came to the result: *"Electricity demand has been so reduced by the highly depressed economic situation that there is a large capacity surplus which is likely to last until at least 2010. Installing further surplus generating capacity would use up limited borrowing authority for a purpose not needed and make it more difficult to achieve the efficiency objectives behind the Government's market-based reforms throughout the energy sector."* (Sussex 1997).

3.5 K2/R4 Modernization and Upgrade: Technical Issue Identification

Sections 3.5.1 through 3.5.13 discuss safety issues identified as a result of review of IAEA documentation and the K2/R4 modernization programme documentation.

3.5.1 General Issues

3.5.1.1 Conservation and Requalification

According to Riskaudit Report No. 120 (RiskAudit 1997a) the construction of Khmelnitsky 2 and Rivne 4 was stopped in 1990 and since then no further construction work has been carried out. The main buildings already exist, some equipment and components are installed, some are stored on site and some parts have not yet been delivered by the manufacturer. The quality of the existing construction and the installed equipment has been checked by inspections carried out by the original Ukrainian or Russian suppliers, under Western engineering supervision with the following main results (RiskAudit 1997a):

- Mechanical equipment: Some repairs or limited replacements are necessary.
- Electrical components: The status is similar except the fact that aging of cables and cable penetrations is of special concern.
- Civil structures: The containment of Khmelnitsky Unit 2 has the problem of pre-stressing tension losses. The prestressing cables of the cylindrical part of the containment have to be replaced.
- Large parts of the I&C have to be replaced because of damage and vintage design.
- A specific Metallurgical Quality inspection programme in order to evaluate the quality of the mechanical components: According to Riskaudit Report No. 120 (RiskAudit 1997), "*The document inspection confirmed that the traceability of the technical documentation is almost complete, with some exceptions, which can be easily corrected. The inspection and surveillance actions were complementary to those performed during the construction. They confirmed that the equipment complies with the rules & norms applicable in Ukraine.*"

According to the project presentation (Energoatom 1998), besides completion to the original design and modernization (safety upgrading, availability upgradings, etc.), repair and replacement of equipment is a separate point of main effort in the completion programme for Khmelnitsky 2 and Rivne 4. See Table 3.4 for a detailed list of planned measures on the field of repair and replacement.

One of the main open problems will be the reconstitution of a complete manufacturing and construction documentation regarding status and quality of the documentation Riskaudit stated (RiskAudit 1997a): *"The documentation is present at various stages of completeness. As the construction stopped at a given moment and as some equipment is still at the manufacturers' premises, the documentation is not always available. Some documentation has still to be found, but it seems that the latter does not concern important items. If corrected, situation can be satisfactory."*

According to Kopchinsky et al. (Kopchinsky 1996) the documentation of the construction work is very poor.

The manufacturers' guarantees for already delivered equipment have expired. At the time of construction of K2 and R4 the Russian manufacturers' guarantees commonly ended after a few years.

Regarding the conservation status of K2/R4 Riskaudit concludes (RiskAudit 1997a): "*The preservative measures to keep the installed equipment free from environmental degradation during the interrupted erection phase were not found to be totally satisfactory. Surface deterioration (oxidation and rust traces, paint deterioration,..) on components and their supports are concerned by curative measures and have to be considered during the completion phase, or even earlier.*"

Riskaudit recommends a faster replacement than planned in the MP (RiskAudit 1997a). Furthermore an improvement of preservation conditions, repair of surface deterioration on the equipment and their supports, as well as random replacement of the most seriously damaged supporting systems, the reconstitution of a complete manufacturing and construction documentation, additional focused and optimized inspections will be considered for the dissimilar welds. Prior to commissioning, a complete and systematic inspection programme has to be developed and implemented together with the commissioning programme.

3.5.1.2 Qualification of Equipment

Qualification of equipment is a generic safety issue of rank III according to IAEA ranking (IAEA 1996a):

> *In accordance with NUSS 50-C-D, Section 12 the qualification of equipment important to safety is required to demonstrate their ability to fulfill their intended functions. This qualification requirement applies to normal operating conditions, to accident conditions and to internal and external events. In addition, according to international practice, it should be possible for the plant operators and the regulatory body to examine the associated qualification reports. A major concern with respect to WWER-1000 nuclear power plants, as shown by safety reviews, is that this practice of qualification of equipment is either lacking or not evident.*

Generic deficiencies identified by IAEA (IAEA 1996a):

- Qualification of electrical and I&C equipment, including cable connections, for LOCA conditions: Neither the specifications concerning the test procedures nor the test reports are available at the nuclear power plants. The cable connections, especially inside the containment of WWER-1000 nuclear power plants, are not able to withstand extreme environmental conditions and consequently, they have a high failure potential under LOCA conditions. The same has to be checked for the valves located inside the containment to operate under LOCA situation.
- No seismic qualification of systems important to safety and safety support systems: Ventilation systems, service water pumps, fire water supply pumps and indication and recording instrumentation are not qualified with respect to seismic loads. Therefore their functional capability on demand in the case of an earthquake would be questionable.

Measures proposed by IAEA (IAEA1996a):

Reconstruct the cable connections inside the containment to ensure their operability under the expected post LOCA conditions.

Analyze the available qualification documentation concerning safety-important equipment.

Qualify the ventilation systems, cable penetrations, fire doors, and fire alarm facilities, etc.

Conclusions and Recommendations by Riskaudit:

Concerning qualification of equipment Riskaudit concluded (RiskAudit 1997a):

> *“The proposed approach was totally not satisfactory and a complementary recommendation has been given by Riskaudit. Basically, the main concern is related to the qualification "proof" which can not for Riskaudit be limited to passport documentation. Documents such as test reports or calculation reports should be also available for review on the file or complementary works would be necessary”.*

According to experts judgement in Attachment 1 of the Riskaudit Report (RiskAudit 1997a) the issue qualification of equipment requires *"measures which may lead to problems difficult to be solved during implementation or studies which may lead to further requirements".* (see also Attachment 1 and 2 of this report)

For the purpose of Accidental Equipment Qualification (EQ) Riskaudit (RiskAudit 1998) recommended a specific program for already installed equipment which has to be divided into 3 groups:

- Group 1: Equipment for which documentation to support EQ status is available.
  (Qualified equipment)
- Group 2: Equipment for which documentation on testing and/or analysis is available, but components do not meet service conditions. (Additional information will be requested from manufacturer. Depending on the outcome, the equipment will be moved to Group 1or 3.)
- Group 3: Equipment for which documentation to support EQ status is not available.
  (Testing or replacement)

Specific equipment which has to be qualified and/or tested according to Riskaudit (Riskaudit 1994b):

- Qualification of SG valves
- Qualification of BRU-A valves
- Qualification of pressurizer valves
- Qualification of purification system valves
- Testability of safety equipment: Riskaudit recommended to perform for all safety systems a complete review of the test program and of periodical tests, the tests should include the support systems such as of diesel generators, I&C, ventilation systems and cooling systems.

3.5.1.3 TMI Requirements (NUREG-0737)

The “Post TMI requirements” (USNRC 1980) comprise a list of 36 items which resulted from the lesson of the accident in Three Mile Island (TMI) NPP in 1979. The TMI accident initiated a set of important safety improvements for all NPPs. The accident had a strong impact on areas such as safety analyses, management practices, safety systems and safety culture, improvements of calculational codes, personnel training, investigation of severe accidents, extension of design base accidents, etc.

The following table 3.10 comprises 36 requirement items, 7 are fully and 9 are partly (i.e. not completely related to the original content of the requirement) addressed in the Ukrainian Modernization Programme. Not all important technical items are addressed in the Ukrainian MP and at present it is unclear which of them are already included in the original design level of the Ukrainian WWER-1000/320 reactors. However important technical items which are explicitly addressed in the MP are planned to be implemented after start-up (see Table 3.10 below). The TMI requirements are generally ranked as class II issues in the IAEA Issue Book for the WWER-1000/V-320 reactors (IAEA, 1996a). Logistic items are generally not explicitly addressed in the Ukrainian modernization programme and it is unclear if they already have been implemented for all operating WWER-1000 reactors.

For US NPPs the implementation of all TMI requirements before start-up is a necessary precondition for obtaining an operating license (USNRC 1980).

3.5.2 Reactor Core Issues

3.5.2.1 Control Rod Insertion Reliability/Fuel Assembly Deformation

The IAEA has ranked this generic issue with rank III (IAEA 1996a):

> *Starting from the end of 1992, an increased drop time of control rods exceeding the maximum design value of 4 seconds has been observed at operating WWER-1000/320 units of Zaporozhe, South Ukraine, Rovno, Khmelnitsky and Balakovo NPPs. The increase in the rod drop time occurs typically at the end of the insertion. In a few cases, a control rod has remained stuck in a position near the bottom 1/3 of the core. Most of the problems have occurred during the third year of operating an assembly in the reactor.*
>
> *There are many factors which could result in the increased drop time. The safety concerns are related to the structural deformation of fuel assemblies affecting the reliable insertion of control rods and leading to the water gap change which will cause a higher local power density. In some accidents with fast transients, the negative reactivity insertion rate needed to shutdown the reactor will be the dominant factor to assure nuclear safety.*
>
> *Investigations of root causes are being made in Russia and the Ukraine (0KB Gidropress, RRC Kurchatov Institute, NIIAR (Scientific Research Institute of Nuclear Reactors, Dimitrovgrad) and other institutes). Root causes have not finally been identified, and the accident analysis for this event needs to be reviewed and extended.*

Measures proposed by IAEA (IAEA 1996a)

- Compensatory and interim measures have been taken to justify continuing operation in the short term:
  - reduction of coolant flow if the CPS time exceeds 4 s by changing from full operation of cooling pumps to operation with three or two reactor coolant pumps at correspondingly reduced power,
  - control rod drop times are measured at least once every 3 months.
  - in order to minimize the potential rod insertion problems, fuel assemblies which have been used for 2 years are not inserted into the control rod locations, but are replaced by new fuel assemblies with nearly the same physical characteristics;
  - tests on stands for verification of free control rod movement before loading of fuel assemblies into the core. The deviations of lifting and lowering forces from normal values should not exceed ± 3 kg. The central instrument thimbles are measured by means of a specially designed calibre; and
  - the position of protective tube unit was readjusted and moved upward to reduce the excess axial load exerted on the fuel assemblies and to alleviate the deformation of guide tubes.
- Investigations of root causes.
- A new design of the control rod, with approximately 30% greater weight to shorten the drop time, and a new fuel assembly design with a modified top nozzle and softer springs are being tested in WWER-1000 reactors.

Table 3.10: TMI Requirements for Operating Reactors (1/2)

NUREG-0737	ITEM	Addressed in Ukrainian Modernization Programme (number of the measure)	Implementation B=before, A=after start-up
I.A.1.1	Shift technical advisor	not explicitly addressed	
I.A.1.2	Shift supervisor responsibilities	not explicitly addressed	
I.A.1.3	Shift manning	not explicitly addressed	
I.A.2.1	Immediate upgrading of RO & SRO training and qualification	not explicitly addressed	
I.A.2.3	Administration of training programs	not explicitly addressed	
I.A.3.1	Revise scope and criteria for licensing exams	not explicitly addressed	
I.C.1	Short-term accident and procedure reviews	partly, not short term: 30211	30211: A,B
I.C.2	Shift & relief turnover procedures	not explicitly addressed	
I.C.3	Shift supervisor responsibility	not explicitly addressed	
I.C.4	Control room access	partly, 35111	35111: B
I.C.5	Feedback of operating experience	yes, explicitly addressed, 31111	31111: A
I.C.6	Verify correct performance of oper. activities	not explicitly addressed	
I.D.1	Control room design reviews	yes, explicitly addressed, 14331, 28411, 28511	14331: A,B 28411: A;B 28511: A
I.D.2	Plant-safety-parameter display console	yes, explicitly addressed, 28411	28411: A,B
II.B.1	Reactor coolant system vents	not explicitly addressed	
II.B.2	Plant shielding	not explicitly addressed	
II.B.3	Post accident sampling	partly, 33211, 33212	33212: A 33211: B
II.B.4	Training for mitigating core damage	not explicitly addressed	
II.D.1	Relief & safety valve test requirements	yes, explicitly addressed 11011	11011: A,B
II.D.3	Valve position indication	not, proposed in the IAEA Report on K2 and R4 (IAEA 1997), (IAEA 1995d)	
II.E.1.1	Auxiliary feedwater system evaluation	partly, 17321, 12221,12211	12211: B 12221: A 17321: B
II.E.1.2	AFW system initiation & flow	not	
II.E.3.1	Emergency power for pressurizer heaters	not	
II.E.4.1	Dedicated hydrogen penetrations	partly, 16131, 16211	16131: A 16211: A,B
II.E.4.2	Containment isolation dependability	not	

NUREG-0737	ITEM	Addressed in Ukrainian Modernization Programme (number of the measure)	Implementation B=before, A=after start-up
II.F.1	Accident-monitoring:	14411,14251	14411: A 14251: A
	• Noble gas monitor	yes, explicitly addressed, 33212, 33211	33212: A 33211: B
	• Post accident sampling	Yes, explicitly addressed 33212, 33211	
	• Containment high range radiation monitors	yes, explicitly addressed 33212, 33211	
	• Containment high range pressure monitor	not explicitly addressed	
	• Containment high range H_2 concentration	yes, explicitly addressed 16211	16211: A,B
II.F.2	Instrumentation for detection of inadequate core cooling	yes, explicitly mentioned, 14251	14251: A
II.G.1	Power supplies for press. relief valves, block valves and level indicators	not, explicitly addressed	
II.K.1	IE Bulletins	not explicitly addressed	
II.K.3	Final recommendation B&O task force	not explicitly addressed	
III.A.1.1	Emergency preparedness short term	partly, 14411, 33231	14411: A 33231: A,B
III.A.1.2	Upgrade emergency support facilities (Technical support center for MCR)	yes, explicitly addressed, 28511	28511: A
III.A.2	Emergency Preparedness	partly, 33231	33231: A,B
III.D.1.1	Primary coolant outside containment	partly, 12411, 16111, 13611, 16121, 19311	12411: B 13611: A,B 16111: B 16121: B 19311: B
III.D.3.3	Improved In-Plant Iodine Instrumentation Under Accident Conditions	partly, 33211, 33212	33211: B 33212: A
III.D.3.4	Control Room Habitability Requirements	not explicitly addressed	

Measures proposed by the Modernization Program (KIEP 1996)

Measure No. 11221: Estimate loads onto supporting frame of fuel assembly, including:

- modification and readjustment of the protective tube unit to reduce the axial force exerted on the fuel assemblies as a compensatory measure;
- determination of the extent of distortion of the guide tubes in order to estimate the power density increase due to the change in gaps between the adjacent fuel assemblies; and
- assessment of the effect of fuel assembly deformation on the control rod drop time under condition of the most severe design basis accident combined with safe shutdown earthquake.

This issue is planned to be studied in relation to the effect of additional mechanical loads on the control rods.

Measure No. 11222: Introduce "heavy weight" control rod of fuel assembly:

- use of heavier control rods to help increase the drop time.

Conclusions and Recommendations by IAEA (IAEA 1997)

Positive results are shown by modified control rods with drilled drive bars. The use of hafnium and other materials has not been properly justified and is not economic, although control rods with hafnium absorbers have already been purchased for Unit 2.

Another possible future improvement is to increase the weight of the absorber cluster up to 27 kg and the drive extension shaft up to 15-18 kg, in order to shorten the drop time. If so, the control rod drive mechanism has also to be replaced with a new one which can handle the heavier absorber clusters.

Some calculations have been made in the RRC Kurchatov Institute, considering the effect of the water gap changes among fuel rods on the power distribution. The calculation was done for a specific whole fuel cycle of Rivne NPP Unit 3. Based on the calculations, it was decided to limit the maximum power to 95% of nominal value after the refueling, considering the penalty of the above effect. The accuracy of the calculation results is not known.

Unreliable insertion of control rods reduces the reactor shutdown capability. During the normal power operation, the safety concern is that the deformation of fuel assemblies leading to the water gaps change will cause a higher local power density, and it has to be ensured that the design limit is not exceeded.

In a reactivity initiated event where positive reactivity is inserted into the reactor core, a delay of the control rod insertion or a control rod jamming may not sufficiently depress the peak power level and may deteriorate the fuel cooling. It could also lead to a transient of high peak primary pressure.

- It is recommended that the operational performance of control rod insertion be carefully followed and that the supplier and user group investigations of the root cause leading to the permanent bowing of the fuel assemblies, which increases the friction between rodlets and guide tubes, be supported.
- The maximum permissible deformation of fuel assembly should be established in order to ensure that the increase in power density is within the design limit.
- Additional analyses of the mechanical stability of the degraded fuel assemblies in the LOCA and/or seismic conditions need to be performed.
- Based on the results of root cause analyses, the improvement of fuel assemblies should be considered in the future.

3.5.2.2 Power Density Control System/Xenon Oscillations

This issue has IAEA rank II (IAEA 1996a). General design criteria require that the plant control systems and monitoring systems be designed to detect and suppress power oscillations due to xenon oscillations. In contrast to western NPPs where Xenon oscillations in the reactor power are automatically suppressed, in WWER reactors this has to be done by the operators following special procedures (KIEP 1996). The current core control strategy prevents xenon oscillations in base-load operation and for infrequent power changes. However, if the plant is used in the load follow mode, the control strategy needs to be improved in order to protect and suppress xenon oscillations (IAEA 1996a).

Measures proposed by IAEA (IAEA 1996a)

Due to the concern regarding the existing in-core instrumentation, there is a proposal to replace the in-core system and to develop a new control strategy.

Recommendations by IAEA (IAEA 1996a)

- The Russian plans to replace the existing in-core system and to use a new control strategy properly address the issue.
- In conjunction with the I&C reconstruction programme, the design of automatic protection for power distribution and DNB should be considered (e.g., the difference between the upper and lower ion chamber signals could be combined with primary system pressure and temperature signals to provide protection based on hot channel factor and peak load power generation).

Conclusions and Recommendations by Riskaudit

As reported in (RiskAudit 1998) an automatic control of Xenon oscillations and power distribution is being developed. The implementation is planned for after start-up because the system must be carefully examined. Regarding Xenon and power control, the proposed measures will permit to solve the safety issue. This item is fully addressed in the modernization programme.

According to experts judgement in Attachment 1 of the Riskaudit Report (Riskaudit 1997a) the issue qualification of equipment requires *"measures which may lead to problems difficult to be solved during implementation or studies which may lead to further requirements"* (see also Attachment 1 and 2 of this report).

3.5.3 Component Integrity Issues

3.5.3.1 Reactor Pressure Vessel Embrittlement

3.5.3.1.1 Available information about the issue

The issue of RPV embrittlement of the VVER 100/320 reactor is addressed in a number of documents (IAEA 1996a; IAEA 1996d; IAEA 1997; KIEP 1996; RiskAudit 1994; RiskAudit 1997)

Several specific measures are addressed in the modernization programme mentioned above. They generally coincide with the measures described and proposed in the other documents referenced above. The mentioned measures deal with the RPV embrittlement, its assessment, monitoring and mitigation. They comprise the following actions:

- Improvement of the accuracy on the assessment of the fluence build-up in critical zones of the reactor vessel;
- Preheating of ECCS water of passive and active systems to reduce stress loads in the RPV wall in case of injection;
- Modernization of the radiation control of surveillance specimens in the downcomer of the reactor vessel to enhance representativeness of test results of surveillance specimens;
- Development and introducing of a new programme of surveillance-specimens of archive vessel material; and
- Development and implementation of the standard procedure for determination of current residual life of safe operation of the reactor vessel based on actual reactor state and operating conditions, data provided by the radiation load monitoring system, results of related experiments and results of tests on surveillance specimens.

Additionally to the above mentioned measures and actions refueling of the reactor core with low neutron leakage is foreseen to improve fuel usage and allow reduction of the neutron fluence to the RPV wall.

The issue of the RPV embrittlement is part of the RPV integrity topic. It is categorized under category III of the IV categories of the defence in depth concept.

The measures are generally requested on basis of national requirements (PNAE G-/-008-89, OPB-88) and international recommendations (IAEA, RiskAudit). The date of realization of the indicated measures and actions are primarily foreseen before startup. The actions concerning the lifetime assessment are intended to be performed after startup.

3.5.3.2 Steam Generator (SG) Collector Integrity, SG Tube Rupture, and Non-Destructive Testing (NDT)

Maintenance of primary system integrity is an important safety function, and this function becomes a critically important one where breach of the primary coolant system boundary leads to loss of primary coolant outside containment. The steam generator collector is just such a primary coolant boundary, failure of which releases primary coolant at high pressure into the secondary side of the plant and, if the atmospheric dump valves (BRU-A) or main steam safety valves lift, out of the containment. Thus, the failure of the SG collector leads to an interfacing LOCA and containment bypass, which **simultaneously** defeats the functioning of the containment **and**, unless terminated by timely operator action, results in loss of primary coolant inventory outside containment and subsequent core damage.

Paradoxically, the IAEA has found the steam generator collector in the VVER-1000/320 design to be "*the weakest element of the reactor coolant boundary*", citing problems with manufacturing technology, environmentally assisted cracking, inadequate secondary water chemistry (chloride intrusion),[24] and the supply of cold water form the emergency feedwater system producing unacceptable stresses in degraded sections of the steam generator as contributing causes (IAEA 1996a: 17).

Each VVER-1000/320 unit has four steam generators (SGs). Each SG has two cylindrical collectors (hot leg and cold leg collectors) which form part of the boundary between the primary and secondary circuit. The SG tubes are attached to these collectors. There are no gate valves in the primary circuit to isolate the SG in case of collector failure (IAEA 1996a: 51).[25] A primary-to-secondary leak due to steam generator collector failure would quickly overfill the steam generator and the main steam line. The main steam line has not been demonstrated to be qualified for this hot water load (IAEA 1996a: 19), and could fail which would guarantee loss of primary coolant into the atmosphere through the turbine building exhaust (which is unfiltered).

During the resulting primary pressure transient, the high pressure injection system (a part of the emergency core cooling system) would actuate, and ECCS water would also be lost out of the break to the secondary side of the plant. This ECCS water would not collect in the containment sump, and would be unavailable for recirculation to the primary coolant system. In addition, ECCS operation keeps the primary system pressure high, counteracting actions needed to depressurize the reactor coolant system and stop primary and ECCS water flow out of the break to the secondary side of the plant (IAEA 1996a: 58).

There are two safety valves and a single BRU-A valve in each main steam line. Neither the BRU-A nor the safety valves in the original design were qualified for water or water-steam mixtures (IAEA 1996a: 65). As noted above, the main steam line is not qualified for water loads (IAEA 1996a: 19). Either a steam line break before the isolation valves outside containment or failure of the BRU-A or a main steam safety valve to close would lead to containment

[24] Chloride intrusion occurs due to failure of the condenser tubes. The K2 and R4 designs retain condensers with tubing containing copper. This design is widely recognized as vulnerable to failure, promoting chloride intrusion events. The K2/R4 upgrade programs do not include replacement of the condensers with a titanium tubing design; such an upgrade is included in the Temelin upgrade program. IAEA recommended consideration of replacement of copper containing alloys in the secondary circuit (IAEA 1996a: 52), but this was done for K2 and R4.

[25] The VVER-400/213 and VVER-440/230 designs incorporate these loop isolation valves. However, the valves were deleted from the VVER-1000/320 design, thus eliminating a key means of avoiding core damage from the SG collector failure initiating event. This design change for the VVER-1000/320 results in an increased contribution of SG collector failure to CDF compared with earlier VVER designs.

bypass and the loss of long-term core cooling due to loss of primary water to the environment. Insufficient emergency operating procedures (EOPs) would further complicate the response to such scenarios (the procedures are event-oriented, rather than symptom-oriented as recommended following the TMI and Chernobyl accidents).

These scenarios are beyond design basis accident (BDBA) scenarios which the IAEA states have not been analyzed in sufficient detail so far (IAEA 1996a: 19). IAEA recommends performing further accident analyses to identify those scenarios in which SG collector failure could lead to severe consequences (IAEA 1996a: 52). Notwithstanding this, Mouchel has limited its BDBA analysis to an SG collector failure event which is fully mitigated (see Sections 3.3.3.1 and 3.4.3, above). Mouchel also did not consider SG collector failure with an effective diameter of greater than 100 mm, which the IAEA has identified as a BDBA for the VVER-1000/320 design (IAEA 1996a: 55).[26]

Severe cracking in steam generator collectors has actually been observed in service. Nonetheless, SG collector failure is not considered by the plant designer to be a design basis accident (DBA) scenario. PSA studies indicate that loss of SG integrity due to this scenario could contribute significantly to the CDF for VVER-1000/320 plants (IAEA 1996a: 30). Indeed, the Temelin PSA, which analyzes the most thoroughly upgraded VVER-1000/320 plant, identifies SG collector failure with long term cooling failure and containment bypass as by far and away the dominant contributor to core damage frequency, contributing a CDF of 4.6×10^{-5} per reactor-year out of the total CDF of 7.9×10^{-5} per reactor-year from internal events. Thus, for Temelin, SG collector failure is responsible for over 58% of the internal events CDF (IAEA 1996b: 113).

Two key factors indicate that the CDF contribution from such events could be higher at K2 and R4 than at Temelin. **First**, since the Temelin plant is implementing symptom-oriented EOPs before startup, while K2 and R4 are **no**t, the core damage frequency contribution per reactor-year for K2 and R4 will be even higher than Temelin since it is clear that operator error (failure to depressurize the primary system and stop the break flow before exhausting the containment sump inventory available to the emergency core cooling system) dominates the scenario frequency.[27, 28] Moreover, not only does this scenario bypass containment, it also results in guaranteed failure of containment sprays (due to lack of containment sump inventory), which eliminates containment pressure mitigation and containment source term mitigation ca-

[26] In the documents which IRR has reviewed for this report, there are no descriptions of the technical basis for concluding that SG collector failures will be limited to a 100 mm effective diameter. A RiskAudit report (RiskAudut 1997b: 13) refers to a statement to this effect by the Russian organization Gidopress, but no specific bibliographic citation is provided. The technical basis for the 100 mm effective diameter should be reviewed and its validity independently confirmed before this value is relied upon.

It is noted that this is the same effective diameter as asserted for the VVER-440/213, but the SGs for the VVER-1000 are different and the component sizes are considerably larger. Given the obvious safety and risk significance of SG collector failures, blind acceptance of the 100 mm effective diameter value is discouraged by IRR.

Finally, it is also noted that the 100 mm effective diameter discussed in the IAEA report on VVER-1000 safety issues applies to the SG collector head lift-up, **not** to the SG collector failure (IAEA 1996a: 116). SG collector head lift-up is a **separate** and **distinct** failure mode from SG collector failure, which involves cracks in the tube sheet, not lifting of the collector cover. The SG collector tubesheet failure involves containment bypass, where as collector head lift-up is simply a primary LOCA (albeit in a unique location). **It has yet to be demonstrated to IRR's knowledge that the 100 mm effective diameter value applies to SG collector tubesheet failure.**

[27] IAEA has stated that the development of emergency operating procedures (EOPs) to mitigate the consequences of this event are "*an essential element to ensure the safety function cooling the fuel*". IAEA **specifically** recommended that these EOPs be **symptom-oriented** (IAEA 1996a: 58). However, symptom-oriented EOPs will not be implemented at K2 and R4 before commissioning.

[28] It should be noted that the procedure being developed for K2/R4 for dealing with primary to secondary leakage such as this (upgrade measure 12411) requires the operators to cooldown by opening the BRU-A (atmospheric dump valves) of the non-affected SGs (KIEP 1996: 51/316). If the operators misdiagnose the unaffected SGs and mistakenly open the BRU-A of the affected, a containment bypass sequence ensues. The procedure is clearly event-oriented as it requires the operators to know that primary to secondary leakage exists.

pability for this severe accident.[29] **Second**, the K2 and R4 condensers retain copper-based tubing, which is much more likely to promote chloride intrusion events due to failure and thus compromise the SG collectors than in plants (such as Temelin) which replace the condensers with titanium tubing. Chloride intrusion events are associated with accelerated degradation of SG components, including SG collectors. The existing monitoring of secondary water chemistry is not adequate to prevent violation of design limits for safe operation. The presence of copper-based tubing in the condensers could lead to a higher SG collector failure initiating event frequency than for Temelin, which implemented titanium condenser tubing, and this higher initiating event frequency will translate directly into a higher CDF from SG collector failure events.

In addition to SG collector failure, SG tube failure could result in similar scenarios (IAEA 1996a: 54), although the effective size of the primary-to-secondary leakage is smaller, providing more time for operator recovery. The Temelin PSA identified SG tube failure with operator failure to depressurize in time (leading to loss of primary coolant and containment sump inventory out of containment, resulting in long-term cooling failure and failure of containment sprays) as the second-leading cause of core damage, just behind SG collector failure. This SG tube rupture scenario – resulting in containment bypass – was estimated to contribute a CDF of 1.8×10^{-5} per reactor-year, or almost 23% of the total CDF due to internal events (IAEA 1996b: 113).

It should be noted that the IAEA recommended improvements to SG tube NDE methods and the development of justified tube plugging criteria. Current practice is to detect tube degradation by use of a camera inside the SG collector to observe bubbles. However, this method does not detect degradation until leakage has **actually begun** (i.e., until there is **already** a through-wall crack) (IAEA 1996a: 54). This method of detecting SG tube degradation increases the risk that one or more SG tubes will fail in the event of a main steamline rupture accident (due to the high differential pressure across the tubes, which "see" full primary pressure on one side and atmospheric pressure on the other side). If the tubes are already cracked nearly through-wall, they are much more likely to rupture in the event of a main steam line failure. This would transfer a main steam line rupture accident in the containment into a containment bypass accident (via the consequential SG tube rupture).

Another accident related to SG collectors involves the lifting of the SG collector cover. RiskAudit has identified that extended ISI of the SG cover studs and threaded holes are an essential condition for avoidance of this accident initiator. However, RiskAudit noted that the modernization programme does not address this issue, nor has it been confirmed that extended ISI can sufficiently reduce the frequency of SG cover lift-up (RiskAudit 1997b: 3-13). This accident initiator, however, is not a containment bypass event; rather, it involves a leakage of primary coolant inside the containment (a LOCA).

The IAEA has identified a number of safety issues related to SG collector and SG tube integrity:

- CI4, Steam Generator Collector Integrity, Category III.
- CI5, Steam Generator Tube Integrity, Category II.
- S2, Mitigation of a Steam Generator Primary Collector Break, Category II.
- S9, Steam Generator Safety and Relief Valves' Qualification for Water Flow, Category III.
- AA7, Steam Generator Collector Rupture Analysis, Category II.

The K2/R4 modernization programme identified the following measures as related to these issues (KIEP 1996: App. 5; RiskAudit 1997a: App. B):

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[29] When the reactor pressure vessel fails in this accident due to interaction with molten core debris in the lower vessel head, the containment pressure will suddenly rise, resulting in containment isolation. At this time, the spent fuel pool cooling system will be isolated. Since the containment sprays will be failed due to the nature fo the reactor core accident, a spent fuel pool severe accident could occur (lack of spent fuel pool cooling and failure of the makeup system, namely the containment sprays). If the containment were to fail due to accident phenomenology concurrently with or subsequent to vessel failure, the reactor core accident source term could be supplemented by the large radiocesium release resulting from a spent fuel pool severe accident. See further discussion in Section 3.4.4.

CI4, Steam Generator Collector Integrity

- 12411 (organizational and technical measures for accident management for 100 mm equivalent diameter primary to secondary leak; **before** startup); these measures are **event-oriented procedures**, not symptom-oriented procedures as recommended following the TMI-2 and Chernobyl accidents;
- 19112, (analysis of selected accidents using modern codes, based on a list of initiating events selected via measure 19111; **before** startup); list of events to be analyzed includes "*leakages from primary coolant system to secondary one*";
- 19311 (analysis of accidents that have not been reviewed in the TOB report; **before** startup); list of events to be analyzed includes "*leakages from the 1st circuit into the 2nd beginning from one or several SG tubes rupture up to the head separation*" and "*maximum SG header rupture*";
- 22111 (modernization of SG blowdown; **before** startup); not applicable to SG collector failure or SG tube rupture;
- 26121 (measure deleted; already implemented);
- 26131 (improved and automatic secondary chemical water treatment; **before** startup); relatively insignificant since copper-tubing is retained in the condenser, permitting chloride intrusion;
- 26132 (continuous automatic monitoring of primary system parameters; **after** startup, compensatory measures unavailable); not only after startup, but unrelated to root causes of SG collector failure and SG tube rupture, which have little or nothing to do with primary water chemistry;
- 31311 (missing from Rev. 2 of modernization programme);
- 32212 (radiation monitoring of primary system, steam generator, and stack; **after** startup, compensatory measures unavailable);

CI5, Steam Generator Tube Integrity

- 12441 (determination of SG tube plugging criteria for eddy current inspection; **after** startup, no compensatory measures identified);
- 33212 (See CI4, above);
- 33311 (measure missing from Rev. 2 of modernization programme);

S2, Mitigation of a Steam Generator Collector Primary Break

- 12411 (See CI4, above);
- 12421 (measure missing from Rev. 2 of modernization programme);
- 19112 (See CI4, above);
- 19311 (See CI4, above);
- 22111 (See CI4, above);
- 30211 (development of symptom-oriented EOPs; **after** startup); existing draft event-oriented EOPs will be completed before startup, but these do not comply with post-TMI and Chernobyl EOP guidance;

S9, Steam Generator Safety and Relief Valves' Qualification for Water Flow

- 11011 (equipment qualification, including BRU-A; **after** startup, compensatory measures unavailable);
- 12411 (See CI4, above);
- 13321 (replacement of main steam safety valves; **before** startup); could result in lower conditional probability of an MSSV sticking open, but cannot eliminate this possibility;

AA7, Steam Generator Collector Rupture Analysis

- 19112 (See CI4, above);
- 19311 (See CI4, above).

Based on the identified measures and their status, it is clear that SG tube rupture will be a higher frequency accident initiator than would otherwise be necessary since at startup reliance will continue to be placed on an NDE method which requires the existence of a through-wall crack before degradation can be detected (i.e., measure 12441 will not be implemented until after startup). In addition, since the copper-based condenser tubing is retained in the design instead of being replaced as at Temelin, an enhanced potential exists for SG tube degradation due to secondary side corrosion.

Concerning the more serious SG collector failure accident initiator as well as SG tube rupture, the EOPs for managing this accident will be event-oriented at startup. This increases the frequency of core damage from both initiators compared with Temelin, where the SG collector failure initiator alone already contributes more than 5×10^{-5} per year to CDF and SG tube rupture already contributes almost 2×10^{-5} per year to CDF (the increase is due to the higher likelihood of operator error; K2/R4 will have event-oriented EOPs whereas Temelin has Westinghouse-developed symptom-oriented EOPs). Replacement of the MSSVs will help lower the conditional probability of one of these valves sticking open following the initiating event, however the BRU-A valves will not be environmentally qualified for water and two-phase flow until after startup, and the BRU-A valves will be the first valves challenged in the event of an SG collector break or SG tube rupture (indeed, one of the purposes of the BRU-A is to reduce the rate of challenges to the MSSVs). On balance, it is unclear whether replacement of the MSSVs without also qualifying the BRU-A valve for water and two-phase flow results in any significant reduction in the likelihood of a containment bypass core damage accident resulting from an SG collector break (this is because the MSSVs would only be challenged if the BRU-A valve fails to open; if the BRU-A valve opens and fails to close, the MSSVs are never challenged and whether or not they can handle two-phase or water flow is irrelevant).

Non-destructive testing (NDT) for primary cooling system components is carried out using the defect-reject approach rather than the defect-follow approach. The latter approach is capable of timely detection of degradation. The existing procedures are not adequate for NDT of SG collectors and tubing. The IAEA identified this issue as a Category III Safety Issue (IAEA 1996a: 49). Thus, IRR examined the K2/R4 modernization programme for evidence that issue was being treated. Appendix 5 to the modernization programme lists the measures which address this safety issue, as does Appendix B of RiskAudit Report No. 120 (RiskAudit 1997a). None of the measures cited adequately address this issue, as indicated below:

- 12221 (leak-before-break application to primary system components; **after** startup, no compensatory measures identified); not applicable to SG collectors since causes of SG collector failure include environmentally-assisted cracking for which LBB is inapplicable, as well as secondary chloride intrusion due the presence of copper tubing in the condensers (the condensers are not being modified to eliminate this problem);
- 28111 (diagnostic system; **after** startup, no compensatory measures identified);
- 28112 (computerized network for diagnosis and monitoring; **after** startup, no compensatory measures identified);

- 28113 (vibration diagnosis system; **before** startup for Khmelnitsky, **after** startup for Rivne); may assist in diagnosis depending on system design and implementation, but such details are unavailable, and the system will not be implemented at startup for Rivne 4 in any event;
- 28114 (loose parts monitoring system; **before** startup for Khmelnitsky, **after** startup for Rivne); may assist in diagnosis depending on system design and implementation, but such details are unavailable, and the system will not be implemented at startup for Rivne 4 in any event;
- 28115 (noise diagnosis system for SG headers; **after** startup, compensated by installation of "*appliances for control over SGs blowing water activity and increasing of sampling number and samples analysis*"); the cited compensatory measure does not assist in timely diagnosis of which SG collector has failed;
- 28116 (primary coolant leakage detection system for leak-before-break; **before** startup); may assist in diagnosis depending on system design and implementation, but such details are unavailable;
- 28117 (residual fatigue lifetime diagnosis system; **before** startup); not applicable since the SG collector failure mode is not related to fatigue but to corrosion;
- 28118 (RCP vibration monitoring and diagnosis system; **after** startup, no compensatory measures identified); not applicable to SG collector failure;
- 28119 (mode diagnosis system; **after** startup, no compensatory measures identified);
- 28121 (in-core noise diagnostic system; **before** startup); not applicable to SG collector failure;
- 28122 (diagnostic system for back pressure of check valves; **after** startup, no compensatory measures identified); not applicable to SG collector failure;
- 28123 (diagnostic system for air-operated valves; **after** startup, no compensatory measures identified); not applicable to SG collector failure;
- 28124 (remote television system for components inside containment; **after** startup, no compensatory measures identified); not applicable to SG collector failure;
- 34111 (internal monitoring of RPV metal; **before** startup); not applicable to SG collector failure.

In short, the SG collector-related NDT will apparently remain in the defect-reject mode, which has been found by operating experience to be inadequate and which is a deviation from current standards (Russian OPB-88 and IAEA NUSS). The NDT regime is unaffected, and thus the frequency of occurrence of SG collector failures is unchanged. A few of the measures **may** assist with diagnosis of the accident, but this is unclear since no design details are available for measures 28113, 28114, and 28116. More importantly, the EOP for this event will be an event-oriented EOP at startup, so operator error in choice of the correct EOP or in execution of the correctly-chosen EOP will remain as dominant contributors to core damage resulting from this initiating event.

The modernization programme thus leaves, at the time of startup, two IAEA Category III Safety Issues (CI4 and S9) inadequately resolved as well as two IAEA Category II Safety Issues (CI5 and S2). The IAEA Category II Safety Issue AA7 on analysis of SG collector ruptures appears to be adequately addressed by the modernization programme.

3.5.3.3 Reactor Coolant Pump (RCP) Seals

The VVER-1000 reactor coolant pumps (RCPs) are equipped with seals which prevent the passage of primary coolant along the pump shaft, resulting in a loss of coolant accident (LOCA). Injection of seal cooling water is necessary to maintain the integrity of the RCP seals. This injection function is carried out by the makeup system, which is isolated in the case of a safety injection signal (RiskAudit 1997b: 8-20). (It should be noted that due to the signals used to actuate safety injection, such a signal would inevitably be generated at some point in all severe accident sequences.)

The makeup system function of RCP seal injection is backed up by a self-contained cooling system which is designed to cool the lower RCP bearing and which includes an emergency supply pump. However, even this system is not maintained in the case of a containment isolation signal (RiskAudit 1997b: 8-20). IAEA categorized the RCP seal cooling system as a Category II Safety Issue (IAEA 1996a: 59).

RiskAudit reports (RiskAudit 1997b: 8-20): "*Loss of both systems* [makeup pumps and the self-contained RCP seal cooling system] *or loss of the cooling system ZUP* [intermediate cooling system] *results in unacceptable consequences within a small delay.*" The project proponents have argued that a 24-hour RCP seal test proves that there will be no adverse consequences (RiskAudit 1997b: 8-22). However, the tests were performed with either seals that were new or had been used for only 400 hours (a little less than 17 days). Considering that the reactors would be run as baseload units, and that it is unlikely that RCP seals would be changed on other than a major outage (i.e., refueling outage) basis, the pump seals could be operated for 7000-8000 hours during any given operational cycle. Using new seals or seals aged only 400 hours is thus seen to be completely unrealistic and optimistic. Whether the seals would have survived if properly aged is thus an open question. RiskAudit questioned whether the test was representative (RiskAudit 1997b: 8-22, 8-23).

RiskAudit recommended that tests be performed using representative conditions, and noted that the seals could be vulnerable to failure in loss of heat sink, station blackout, or containment isolation conditions. The NPPs replied that during commissioning tests would be performed to confirm the resistance of the seals to loss of cooling. RiskAudit accepted this position (RiskAudit 1997b: 8-23). However, unless the commissioning tests are performed using aged seals, the results will **still** not be representative of conditions in the field faced by the pump seals, and there is no guarantee that the seals will be aged in the proposed commissioning tests.

3.5.4 Systems Issues

3.5.4.1 ECCS Sump Screen Blocking

The issue is addressed in a number of documents (IAEA 1995d; IAEA 1996a; IAEA 1997; KIEP 1996; RiskAudit 1997a).

The issue is addressed in the modernization programme. A detailed description of the problem is provided. It is mentioned, that actions have to be selected and substantiated to exclude common mode failure of the safety system trains due to clogging of ECCS suction lines.

Two requirements are listed which are intended to be performed:

- Analysis of insulation material behaviour under LOCA conditions
  - The analysis is referred to the "available material of mineral filaments". It comprises the determination of the thermal insulation washing out rate, of the possible fraction dimensions and of the fragments transfer into a sump-tank.
  - Specific solutions for the task are referred to without giving evidence where these solutions were implemented.
  - Reference is made that the "analysis is being developed together with western company". No name of this company is given.
  - The analysis is dated to be realized before start up.
- Implementation of selected technical solution to ensure residual heat removal under LOCA
  - Both requirements of measures are international recommendations and specific requirements of the National Safety Authority based on the ???89 document.
  - The related measures require for both tasks theoretical and experimental studies before implementation of the safety improvement.

3.5.4.2 Main Steam Safety Valves (MSSV) and Atmospheric Dump Valves (BRU-A) Qualification for Two-Phase and Water Flow

On each main steam line there are two main steam safety valves (MSSVs) and a relief valve (BRU-A, an atmospheric dump valve). None of these valves is currently qualified for water flow or water-steam mixture flow. Failure of the valves to re-close in the event of a steam generator tube rupture or an SG collector failure results in containment bypass since there are no loop isolation valves in the VVER-1000/320 design (unlike the VVER-400/230 and VVER-440/213 designs). Neither the BRU-A valve nor the MSSVs are isolable as they are located before the main steam isolation valve. The IAEA has identified the qualification of the BRU-A and MSSVs for water and two-phase flow as a Category III issue (IAEA 1996a: 65). If the atmospheric dump valves and the MSSVs are not qualified for two-phase and water relief, this increases the likelihood of containment bypass accidents which can result in core damage and large releases of radioactivity.

The modernization programme for K2 and R4 envisions replacement of the MSSVs before startup. However, the modernization programme technical description appears to foresee a difficulty in accomplishing this, and describes a process by which one year before startup an application would be made to the nuclear regulatory authority to delay this replacement. In any event, neither the original MSSVs nor the BRU-A valves will be environmentally qualified before startup (KIEP 1996: 12/316). (See additional discussion of this issue in Section 3.5.3.2.)

3.5.4.3 SG Feedwater Capacity

The steam generator emergency feedwater (EFW) system capacity is limited to 8-10 hours of flow (KIEP 1996a: 63/316). IAEA identified the capacity of the EFW system as a Category I issue, but assumed the water supply was adequate for 24 hours (IAEA 1996a: 68); whether this judgment would remain as Category I with only an 8-10 hour supply is questionable. Even with the assumed 24-hour capacity, however, IAEA recommended the development of procedures to provide makeup to the EFW tanks from other sources (demineralized water tanks, pure condensate tanks, turbine tanks, ESW system, and fire engine tank trucks).

The K2/R4 upgrade programme includes this measure (Item 13311), but does not foresee its implementation until **after** startup (KIEP 1996: 63-64/316).

3.5.4.4 ECCS Sump Capacity

The ECCS sump in the VVER-1000/320 design integrates the function of the containment sump and the refueling water storage tank. All safety systems requiring borated water flow (high pressure injection, low pressure injection, and containment sprays) always draw suction from the containment sump; there is no need for switchover from an “injection” to a “recirculation” mode as in western PWRs. Rather, the systems always operate in a “recirculation” mode in the VVER-1000/320 design.

However, there is a point of comparison where the VVER-1000/320 design suffers in comparison regarding ECCS design. This has to do with the ECCS sump capacity. The capacity of the containment sump water volume is 630 m^3 in the VVER-1000/320 design (RiskAudit 1994: 11/73). In comparison, US PWRs have ECCS water volumes ranging from 950-1900 m^3. What this means is that VVER-1000/320 have less time – and thus a greater propensity for human error – in dealing with accidents where primary coolant is lost outside containment (e.g., SG tube rupture, SG collector failure, and interfacing LOCA) compared with western PWR operators. In addition, since symptom-oriented EOPs will not be implemented at commissioning, the human error rates will be even higher in comparison since all western PWRs employ symptom-oriented EOPs instead of the event-oriented EOPs that are known to have a higher operator error potential (the potential of which was demonstrated in the TMI and Chernobyl accidents).

3.5.5 Instrumentation and Control Issues

3.5.5.1 Reactor Vessel Head Leak Monitoring System

In the VVER-1000/320 design, the control rod drive mechanisms (CRDMs), instrumentation, etc., are attached to the reactor vessel head penetrations through bolted joints (flanges). Each joint is sealed by two parallel sealing rings. The existing leak detection system is based on collection of the leakage water between these two sealing rings. The leak detection system is not tested or inspected periodically. There is a humidity monitoring system in the upper reactor block, but it is not sensitive enough to detect leaks in the bolted joints. IAEA has identified this issue as a Category III safety issue. (It should be noted that the reactor vessel head of the Khmelnitsky Unit 1 had to be replaced due to corrosion damage associated with leaks in the upper reactor block.) (IAEA 1996a: 80) In the worst case condition associated with this safety issue, a control rod ejection could occur (IAEA 1996a: 80).

IAEA recommended replacement of the humidity detectors to improve the reliability of leak detection for the reactor vessel head. In addition, IAEA recommended development and implementation of inspection and testing methods to ensure proper operation of the existing leak detection system. IAEA also recommended consideration should be given to upgrading the reactor vessel head leakage monitoring system to provide for timely detection of leaks (IAEA 1996a: 80).

IAEA has identified the reactor vessel head leak monitoring system issue as a Category III Safety Issue (IAEA 1996a: 80). The modernization programme includes measure 28116 (KIEP 1996: 239-240/316; RiskAudit 1997a: App. B), which is a generic primary coolant circuit leakage detection system to be implemented before startup. The description of the measure does not even mention detection of reactor vessel head leaks specifically. Thus, it cannot be concluded that this issue is adequately addressed in the upgrade programme. This leaves an IAEA Category III Safety Issue unresolved.

3.5.5.2 Instrumentation & Control Replacement

The poor reliability of the original I&C system of the WWER-1000 reactors is a generic problem (IAEA 1996a): "*The I&C equipment of WWER-1000 units are based on a technology that is known to present reliability problems. The failure modes found, including relay contact oxidation and low insulation resistance of wiring and terminals, are typical of the technology. Operational experience has shown that the I&C failure rate is relatively high and can cause power reduction." "Without major efforts in maintenance, the I&C reliability may have a serious impact on safety. As the equipment becomes older, the amount of maintenance required to keep an acceptable status of I&C reliability will increase remarkably.*"

Recommendation by IAEA (IAEA 1996a)

Replace safety and safety related I&C systems with up to date technology which features high reliability, self monitoring, testability and fail-safe-design.

3.5.6 Electrical Power Supply Issues

3.5.6.1 Emergency Battery Discharge Time

In the event of a loss of offsite power and failure of the diesel generators to start, the station batteries provide for continued ability to monitor the status of the plant and allow for operation of critical plant components (such as the BRU-A valves, which require DC power; failure of DC power causes these valves to fail closed).

The VVER-1000/320 design has three redundant batteries to provide DC power to vital loads in each of the safety trains. The as-designed discharge time for the batteries is 15-30 minutes

(IAEA 1996a: 24, 90). Modern safety requirements provide for extended discharge times, and 30 minutes is inconsistent with these requirements. The batteries are critical to maintain vital I&C systems and illumination in the control room. Extending the battery discharge time allows more time for recovery of offsite power or diesel generators, and provides the operators with larger time periods to decide on further actions. IAEA has identified this issue as a Category III safety issue for the VVER-1000/320 design (IAEA 1996a: 90). The reactor cannot be controlled in a station blackout accident after the batteries discharge (IAEA 1996a: 123).

The IAEA has inconsistently recommended a minimum battery discharge time of one hour (IAEA 1996a: 90) and 2-3 hours (IAEA 1996a: 24). The K2/R4 upgrade programme includes a measure to increase the battery discharge time to at least one hour (measure 15121); this item is scheduled to be implemented before startup (KIEP 1996: 103/316).

There is no analysis presented to demonstrate that a one-hour battery capacity is adequate. A probabilistic safety analysis is needed in order to evaluate whether one-hour is sufficient. A one-hour capacity is certainly better than a 30-minute capacity – the key question, however, is: How much difference in risk is provided by this extra capacity, and could risk be lowered on a cost-effective basis (considering that the batteries will be replaced in any event) by a greater capacity (perhaps of the order of 2-3 hours or more)? The modernization programme leaves this issue unaddressed, and simply opts for a one-hour discharge time. Without an analysis which considers the offsite power system reliability at the specific sites of K2 and R4 (including consideration of the experience with recovery of offsite power), there is no guarantee that merely increasing the battery discharge time from 30 minutes to one hour results in any reduction in risk.

The basis for accepting a one-hour battery discharge time seems to be that this will support critical loads for **design basis accidents** (RiskAudit 1997b: 4-6). This is most assuredly **not** the issue. Rather, the issue is how much battery time is needed under severe accident conditions (i.e., station blackout conditions) to stabilize the plant and maximize the amount of time the plant can ride out the blackout without suffering core damage. If the batteries are exhausted after one hour, the BRU-A valves will fail closed. Thus the operators lose control over secondary cooldown since the only means of heat removal is through the MSSVs which cannot be manually controlled.[30]

3.5.6.2 Replacement of 6 kV Switchgear

Safety systems are provided with power directly or indirectly from the 6 kV switchgear. Operational experience has shown that the original equipment is unreliable. RiskAudit's review of Rivne Unit 3 recommended replacement of the 6 kV switchgear (RiskAudit 1994: 5/29). The K2/R4 upgrade programme includes replacement of the 6 kV switchgear, but only **after** startup. In addition, the upgrade programme mischaracterizes this item as an "*availability improvement*" (KIEP 1996: 109/316). Clearly the unavailability of a train of 6 kV power is a **safety** issue as it eliminates one of three trains of safety equipment across the board. A compensatory measure is available, consisting of additional checking of auxiliary contacts when installing wheel-out cartridges into the working position, and replacement of dampers after 20 drive operations. No evaluation is provided of the efficacy of these compensatory measures (indeed, these measures are not identified as being related to the problem at hand causing the unreliability of the 6 kV switchgear). Even RiskAudit suggests replacing the 6 kV switchgear with new components at least for safety-related components (RiskAudit 1997b: 4-14).

[30] The IAEA has clearly recognized this (IAEA 1996a: 90): "*The international trend goes towards an extension of the battery discharge time in order to better cope with accident management and station blackout requirements. In case of a station blackout event, the battery is the ultimate energy source of the unit. A higher battery capacity maintains vital I&C systems in operation and illuminates the main control room. This would enable monitoring of essential plant parameters and safety significant motor operated valves would remain maneuverable. Therefore, the reactor can be controlled and can be kept in a safe condition by performing accident management actions (e.g. bleed). The extended battery discharge time leads to larger time margins for operators to decide on further actions.*"

3.5.7 Containment Issues

3.5.7.1 Containment Structure and Containment Bypass Accidents

The VVER-1000/320 design includes a prestressed concrete containment which is similar in **concept** (but not in **execution**) to containments used in some western PWRs. However, there is a unique difference in the layout of the VVER-1000/320 containment.

The bottom of the VVER-1000/320 containment is not placed on the reactor building basemat. In most western PWRs, in the event of a core damage accident, core debris will collect in the space under the reactor vessel. Penetration of the bottom of the containment leads core debris into the space under the reactor building, where the debris is isolated from the atmosphere by the fact that the bottom of the containment rests on the ground. Even if a release pathway is created, there is filtration of the release by dirt, etc., and mostly what is released is noble gases and small fractions of other materials. This release pathway was identified in the WASH-1400 study ("Reactor Safety Study") in 1975 as one associated with relatively small source term release fractions for cesium and iodine.

In the case of the VVER-1000/320 design, however, the bottom of the containment is elevated above grade, and there are three levels of non-containment spaces below the containment. (These spaces include the main and emergency control rooms, both of which would have to be evacuated in the event of containment melt-through due to lethal radiation doses which would occur if the operators remained.) Penetration of core debris into these spaces would result in containment bypass since the spaces below the containment are not designed to be pressure-retaining, nor are they lined to prevent radioactive release to the environment. (Indeed, if the HVAC system remains in operation, it would actually **promote** a higher release by "pumping" the airborne radioactivity to the environment. The filtration system in the HVAC system would fail since it is designed for non-severe accident source terms, pressure conditions, and thermal loads on the filters. Although the release point would be elevated, at the plant stack, this would still be a considerably worse outcome than would be experienced in most western PWRs, which do not have this containment bypass pathway present.

There are at least seven other containment bypass pathways associated with the VVER-1000/320 design:

- Steam generator tube rupture and release via the BRU-A (atmospheric dump valve) or main steam safety valves, none of which are isolable and none of which (at least in the original design) are designed for passing water or two-phase flow (which increases their chances of sticking open) (IAEA 1996a: 19). Steam generator tube rupture is classified by the IAEA as a Category II Safety Issue (IAEA 1996a: 54).[31]
- Steam generator collector failure and release via the same pathway (IAEA 1996a: 19, 30, 51, 55, 58, 116). SG collector failure is itself classified by the IAEA as a Category III Safety Issue (IAEA 1996a: 51).
- Interfacing LOCA involving failure of isolation between the reactor coolant system and the low pressure injection system piping, with failure of the piping outside containment (the "Event V" sequence from WASH-1400) (IAEA 1996b: 113).
- Rupture of the heat exchanger of the cooling circuit of the reactor coolant pumps, leading to a two-phase flow discharge to the intermediate closed cooling circuit at a high flow rate. The integrity of the system cannot be ensured under such conditions, and rupture of the

[31] It should be noted that the upgrade programme does not foresee implementation of SG tube plugging criteria based on eddy-current testing before startup. Although eddy-current testing is performed, the information obtained is not used since there are no criteria for plugging tubes based on wear. Tubes are only plugged when they are detected to be actually leaking (that is, once a through-wall crack has actually developed) (KIEP 1996: 55/316). This method of detecting and plugging defective SG tubes increases the likelihood of an SG tube failure, and thus the frequency of SG tube rupture accident sequences which bypass containment.

closed cooling circuit could occur outside containment, resulting in containment bypass (IAEA 1996a: 92).[32]

- Rupture of the primary circuit letdown flow after-cooler would result in containment bypass, resulting in loss of primary water outside containment (IAEA 1996a: 92).[33]
- Failure of low pressure injection system residual heat removal system heat exchanger tubing, bypassing the containment with a release pathway into the essential service water system (such a release would be limited to noble gases and nonsoluble, volatile species) (IAEA 1996a: 21, 63, 165). This vulnerability is itself classified by IAEA as a Category II Safety Issue (IAEA 1996a: 63).[34]
- Failure of any one of three suction lines between the ECCS water storage tank (in the containment sump) and the containment isolation valve in the ECCS lines would lead to containment bypass (IAEA 1996a: 22, 62, 163). This vulnerability is itself classified by IAEA as a Category II Safety Issue (IAEA 1996a: 62).

All seven of the containment bypass events described above would be associated with guaranteed failure of the containment spray system due to loss of water inventory outside containment. Thus, the spray system would be unavailable to mitigate the airborne source term within the containment in the event of a severe accident associated with the bypass events.

IAEA recommended studying the possibility of installing rupture membranes to ensure that any ruptures in the intermediate closed cooling circuit of the reactor coolant pumps or the letdown after-cooler occur inside the containment, thus permitting recirculation of spilled primary coolant (IAEA 1996a: 92). In so doing, however, a risk comparison study would be needed to ensure that the core damage frequency is not increased by inadvertent LOCA events due to failure of the rupture membranes (i.e., a risk tradeoff study would be needed to ensure that risk is not increased by installation of the rupture membranes).

IAEA also recommended that all lines going through the containment should be checked and the possibility of isolating them from the primary circuit to avoid containment bypass should be analyzed (IAEA 1996a: 92).

Containment bypass is categorized by IAEA as a Category II Safety Issue (IAEA 1996a: 92). The modernization programme includes two items to address this issue (KIEP 1996: App. 5; RiskAudit 1997a: App. B):

- 13611 (improved leak tightness of the ECCS heat exchangers; **after** startup, except for analysis of boron dilution implications; compensatory measures limited to control over service water and primary coolant under cooldown mode flow rate and temperatures, and control over service water radioactivity); the compensatory measures do not address the issue, but rather only detect the occurrence of the failure which this issue addresses, thus the compensatory measures do not affect the likelihood of the failure on the potential accuracy with which the operators can diagnose it;
- 16111 (implementation of design change to prevent loss of integrity of independent RCP cooling circuit and let down aftercooler heat exchanger; **before** startup); this item mounts protective devices inside containment to ensure that failures occur inside containment rather than outside containment, so that the spilled coolant can be recirculated by the ECCS.

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[32] This issue may be resolved by the upgrade programme, which includes a measure (Item 16111) to mount protective devices inside containment to ensure leakage in containment instead of bypassing containment (KIEP 1996: 113/316).

[33] This issue may be resolved by the upgrade programme, which includes a measure (Item 16111) to mount protective devices inside containment to ensure leakage in containment instead of bypassing containment (KIEP 1996: 113/316).

[34] A K2/R4 upgrade programme measure (Item 13611) is defined to improve the leaktightness of the ECCS heat exchangers, but no hardware modifications are envisioned before startup (KIEP 1996: 69/316). A measure to detect radioactivity in the ESW water has already been implemented (KIEP 1996: 70/316).

Of the eight containment bypass mechanisms cited above (containment melt-through into above-ground, non-pressure-retaining rooms; SG tube rupture; SG collector failure; interfacing LOCA via low pressure ECCS piping outside containment; RCP heat exchanger failure; letdown after-cooler heat exchanger failure; ECCS heat exchanger tube failure into ESW system; and failure of ECCS suction lines), only two are addressed by the measures above before startup.

3.5.7.2 Containment Ultimate Strength

It is typical western safety practice (e.g., USNRC IPE programme) for a containment ultimate strength analysis to be performed. This is done in order that the margin to failure for loads in excess of design is understood, along with the likely failure mode.

The modernization programme proposed by the project sponsors lacks such an analysis. RiskAudit has proposed that the analysis be included in the programme (RiskAudit 1997b: 17).

Without such an analysis, there is no adequate understanding of the reliability of the containment for beyond design basis accidents. Such an understanding is fundamental to understanding the risk from BDBAs, as well as properly structuring accident management programmes.

3.5.7.3 Pneumatic Containment Isolation Valves

A compressed air system is provided to supply pneumatically controlled containment isolation valves. The system consists of three independent trains, each including a compressor and a buffer tank which allows the pneumatic valves to maintain their position for seven hours or to move five times (RiskAudit 1997b: 8-113).

RiskAudit states that the presence of the independent trains, interconnected by isolable links, each with a buffer tank, enables compliance with the single failure criterion until depletion of the air reserves (RiskAudit 1997b: 8-113).

However, the specific valves and their behavior under loss of air conditions needs to be explored. RiskAudit did this for Rivne Unit 3, and it is inexplicable that they did not understand the implications of the system design for Rivne Unit 4 and Khmelnitsky Unit 2. Among the pneumatic containment isolation valves are the isolation valves for the LP ECCS system. The valves fail **open** on loss of air. Given loss of air, the only thing separating the primary coolant system from an interfacing LOCA (due to failure of the LP ECCS piping/valves/packing outside containment) is either one or two check valves (one train has a single check valve, the other two trains have two check valves). There are three such lines penetrating containment, one for each train of LP ECCS (RiskAudit 1994: 11/21, 11/22). (The HP ECCS system and the SFPCS also contain pneumatic containment isolation valves.) (RiskAudit 1997b: 8-114; RiskAudit 1994: 11/22)

During operating conditions in which the air system is failed and the pneumatic containment isolation valves are failed open, failure of the check valves in the LP ECCS line could lead to overpressurization of the LP ECCS system outside containment. If the LP ECCS system piping fails due to this overpressurization, the primary coolant system will blow down outside containment. Even if the remaining trains of ECCS operate, the injected water will also spill outside containment via the broken LP ECCS line. Once the primary system blowdown and ECCS water are depleted, the core will melt and radioactivity will be transported to the LP ECCS break point (at least until the reactor pressure vessel fails due to interaction of core debris with the lower vessel head). Such accidents are called “interfacing LOCAs” and have been recognized as large release accidents since they were first studied in the 1975 Reactor Safety Study (WASH-1400).

It is not clear whether there are other pneumatic containment isolation valves in the VVER-1000/320 design. However, given the presence of such valves in the HP ECCS and LP ECCS systems, this seems likely.

3.5.8 Internal Hazards Issues

3.5.8.1 Fire Prevention

There are typically 0.3 fire per year at NPPs (based on US experience), and there have been three serious fires identified by the USNRC as "near misses" for core damage (Browns Ferry, 1975; Vandellos, 1989; Narora, 1993). The defense-in-depth concept of nuclear safety requires attention to fire prevention, fire detection, fire suppression, and fire mitigation (USNRC 1998c). The IAEA has identified fire prevention (Internal Hazards 2) as a Category III Safety Issue. In particular, IAEA identified weaknesses associated with lack of qualified fire doors in fire barriers, location of redundant cable trains too close to one another, lack of qualified penetrations, and lack of fire resistance of cables (IAEA 1996a: 94).

Cable spreading rooms are recognized generically as a potential fire vulnerability point due to proximity of redundant cable trains. In the VVER-1000/320 design in particular due to a lack of passive fire protection. Specifically, the cable spreading rooms under the main and emergency control rooms contain substantial quantities of safety system control cables which penetrate the ceiling of the cable spreading rooms into the control rooms above. A fire in one of these areas could potentially lead to loss of control over all three trains of safety systems (IAEA 1996a: 94). Finally, the IAEA stated that it was not clear whether a fire affecting cables in one control room would affect the functioning of the remaining control room (IAEA 1996a: 94).

In addition, it has been found that there is a common mode failure potential for all 6 kV main distribution boards to fail simultaneously in a fire because they are not separated by fire barriers (IAEA 1996a: 26). The 6 kV distribution boards are located in an electrical distribution building which is attached to the turbine hall. On the same floor are additional switchgears for 0.4 kV and DC power systems. According to the Ukrainian and Russian requirements and specifications, these areas of the floor are **not** considered to be a fire protection zone (IAEA 1996a: 95).

IAEA has identified a number of fire protection-related safety issues. These issues and the modernization programme elements which attempt to address them are identified in summary form as follows:

IH1, Systematic Fire Hazards Analysis (Category II)

- 17111 (systematic fire hazards analysis; **before** startup); however, the methodology to be used is not identified, nor from the description in the modernization is it clear that the purpose and methods typically used in systematic fire hazards analysis are even understood;[35] IAEA guidance on this subject is available (IAEA 1992; IAEA 1994c);

IH2, Fire Prevention (Category III)

- 17121 (replacement of combustible petroleum oil with noncombustible lubricating fluids in the RCP lubrication system; **after** startup, with compensatory measures consisting of an RCP fire detection system and a fire extinguishing system);[36]

[35] The description in the modernization states, in its entirety (KIEP 1996: 121.316): "*At a preliminary stage the methodical base is being created, grounded on IAEA requirements to implement manual and computer calculations of NPP premises fire danger quantitative parameters and, as consequence, of needed barriers resistance. On a base of calculation results the permissible limits of fire sections and zones, necessity and type of passive protection, and active protection means are being determined. To inspect thte technology of fuel material restriction and fire sources, elimination in all rooooms with safety system equipment inside. Before plant commissioning (if necessary) the qualitative analysis or quantitative one should be performed.*" These statements are so confusing that they call into question whether the project management really understands what is required for a fire hazard analysis, and – as a result – whether they really understand the cost and schedule implications of the remedial measures which may be required.

[36] RiskAudit has recommended that this action be "*urgently performed, if possible before start-up*" (RiskAudit 1997b: 7-23).

- 17131 (replacement of existing input switching devices of RTZO type switchboards for 0.4 kV power system; **before** startup);
- 17132 (coating of safety system cables with fire resistant coating; **before** startup);
- 17151 (replacement of fire resistant doors in rooms containing safety system trains; **before** startup);
- 29111 (improvement of resistance of turbine hall steel structures by applying fire resistant compound; **before** startup);
- 29112 (implement means to automatically dump hydrogen from the generator housing outside the turbine building in the event of a fire alarm actuation; **after** startup, with compensatory measures unavailable).

IH3, Fire Detection and Extinguishing (Category II)

- 17141 (implement improved fire protection actuation system; **before** startup);
- 29131 (implementation of gas fire extinguishing system for NPP rooms containing safety control systems and monitoring and control systems; **before** startup); however the specific system to be implemented remains to be selected, thus its efficacy cannot be determined.

IH4, Mitigation of Fire Effects (Category II)

- 17112 (analysis of possibility of maintaining reactor in safe shutdown and maintaining long-term subcriticality in the event of a fire in the cable spreading rooms under the MCR and ECR or a fire in the 6 kV switchgear; **before** startup);
- 17161 (installation of fire protection valves in air conduits; **before** startup);
- 29121 (implementation of smoke prevention system for rooms and corridors used for personnel evacuation of reactor building; listed as both **before** and **after** startup without any specification of what is to be done in these phases).

The IAEA evaluation of VVER-1000/320 safety issues singled out fire protection as a key issue for these plants (IAEA 1996a: 26):

> *Fire protection is considered to be an especially important topic, since operating experience with nuclear power plants worldwide has shown that the possibility of fires cannot be fully excluded and the risk of a fire leading to a major event is not sufficiently low. According to the NUSS requirements, an adequate degree of fire protection should be achieved by a defence in depth concept in the design. A key element of this concept is the performance of a systematic fire hazards analysis prior to initial loading of reactor fuel and updating this analysis during operation. This would enable the determination of the required fire resistance of the fire compartment boundaries and requirements of the fire extinguishing systems and other features necessary to fulfill the fire protection requirements.*
>
> *A systematic fire hazards analysis, as discussed here, has not been performed so far for any of the WWER-1000 nuclear power plants. As a consequence, the defence in depth concept of fire protection is lacking. This includes identified weaknesses in passive fire protection in general and in the cable spreading room in particular. A further safety concern in conjunction with fire protection is the possibility that all the 6 kV main distribution boards can simultaneously fail in the event of a fire, since they are not separated by fire barriers.*

Although a complete fire hazard analysis report, in conformance with western safety practice, has apparently not been performed for a VVER-1000/320 plant, a limited analysis of the Zaporozhe NPP by Burns and Roe Company from the US provides little cause for optimism. This study found the following problems (Burns & Roe 1992):

- Housekeeping was found to be a problem, including problems with debris, wood, oil, rags, and other debris present in cable trays, cable runs, behind cabinets, etc.

- Smoking permitted in various areas of the plant (recommended to be restricted or eliminated altogether).
- Plastic floor coverings.
- Inadequate emergency lighting.
- Lack of fire barriers (e.g., between the transformer area and the auxiliary electrical switchgear building).
- Lack of deluge sprinkler system for station transformers.
- Inadequate detector and sprinkler coverage for diesel generators.
- Overloaded cable trays.
- Inadequate fire suppression capability for the turbine hall.
- Lack of covers on rear panels in control rooms.
- Lack of fire dampers in ventilation systems.
- Insufficient smoke detector capability.
- Lack of fire protection gear for the unit operators.
- Cracks in penetration seals.

Considerable actions are intended to be implemented before startup. However, given the significance of fires as potential source of severe accidents and the lack of previous detailed fire hazard analyses, it will only be in the **details** of implementation that it can be seen whether the measures planned for implementation are sufficient. These details are lacking at the current time. It should also be noted that US NPPs have been required to perform detailed fire hazards analyses as part of the measures growing out of the adoption of new USNRC fire protection regulations following a fire at the Browns Ferry NPP in 1975. Notwithstanding the very prescriptive USNRC fire regulatory requirements and the performance of fire hazards analyses, fire PSA studies performed on US NPPs have frequently found fires to be an important contributor to CDF and risk. This indicates the importance of using fire PSA to confirm the adequacy of the fire hazard analysis for safety purposes.

3.5.8.2 Pipeline Break Impacts Inside the Reactor Building and Turbine Building

This issue concerns both primary coolant circuit and secondary circuit piping failures in either the reactor building or the turbine building. Assessments of primary pipe whip restraints for the Temelin, Stendal, and Kozloduy 5 and 6 plants have found that the restraints are not adequate to perform their safety function. IAEA recommended that the adequacy of the restraints be reassessed on a plant specific basis or that as a compensatory measure the leak-before-break concept be applied (IAEA 1996a: 50). Neither of these alternatives is being proposed for implementation at K2 or R4 before startup (see below). For secondary piping, no mention is made in the modernization programme of erosion-corrosion which is well-known from PWR NPP experience to be the dominant cause of failure of secondary piping. Thus, there is very little basis for confidence in the adequacy of the measures adopted for pipe breaks in the modernization program at the time of proposed K2/R4 startup. The situation potentially improves only for primary piping at some point after startup. However, until erosion-corrosion is addressed, there is no basis for confidence in the adequacy of integrity of secondary piping.

The IAEA safety issues and the related modernization programme measures identified for this issue are as follows (IAEA 1996a; KIEP 1996: App. 5; RiskAudit 1997a: App. B):

CI3, Primary pipe whip restraints (Category II)

- 12211 (provide rigid embedding of steam and feedwater piping at the containment boundary; **before** startup); done to transmit dynamic loads to building structure;
- 12221 (implement leak before break; **after** startup, no compensatory measures identified)

- 17321 (analysis of pipeline breaks and impacts in the reactor building, including secondary effects; **before** startup); compensatory measures to be defined depending on outcome of analysis;
- 28116 (primary coolant leak detection system; **before** startup); however, confirmation of applicability of leak-before-break is deferred until **after** startup, thus the usefulness of the leak detection system is in question and the absence of primary pipe whip restraints may cause secondary failures in the event of a pipe failure in the primary system.

CI6, Steam and feedwater piping integrity (Category III)

- 12211 (See CI3, above)
- 12221 (See CI3, above)
- 17321 (See CI3, above)
- 22412 (missing from Rev. 2 of upgrade programme)
- 26131 (improved and automatic secondary chemical water treatment; **before** startup); relatively insignificant since copper-tubing is retained in the condenser, permitting chloride intrusion; ineffective without implementation of erosion-corrosion control and monitoring programme;
- 26132 (continuous automatic monitoring of primary system parameters; **after** startup, compensatory measures unavailable); ineffective without implementation of erosion-corrosion control and monitoring programme.

IH7, Protection against the dynamic effects of steam and feedwater line breaks (Category II)

- 17321 (See CI3, above).

AA5, Main steam line break analysis (Category I)

- 19112 (analysis of selected accidents using modern codes: **before** startup); main steam line break not specifically mentioned.

It is not clear how measure 17321 will be implemented. It is well known that there are potential common mode failures resulting from steam and feedwater piping arrangements at the 28.8 meter deck in the turbine building. The IAEA has noted that the EFW lines supplying SGs 1 and 4 run close to a main steam line and a main feedwater line, and in case of rupture of either line the EFW supply to these SGs could be lost. IAEA also noted that there is a lack of physical separation of steam and feedwater piping as well as limited pipe whip restraints outside containment. The rupture of a steam line could lead to damage to other lines (feedwater lines). (A historical steam line failure, due to erosion-corrosion, at the Millstone NPP in the US resulting in the failure of a secondary nearby stream line.)[37]

In addition to the main steam and feedwater lines, the MOHT consortium identified a flooding problem associated with a large diameter ESW pipeline in the reactor building. Such a large diameter pipe entails the potential for rapid accumulation of floodwater. MOHT also called for special attention to be given to the possibility of flooding electrical devices, including switchgear and I&C devices important to safety (MOHT 1996: Part 3, p. 130). It is not clear whether this specific vulnerability has been addressed by the modernization programme.

Finally, it should be noted that the primary piping LBB proposal seems to envision only one leak detection system. German requirements are for detection of leakage by **three** leak detection systems, working by diverse physical principles (RiskAudit 1997b: 8).

[37] The Temelin NPP (also a VVER-1000/320) is implementing erosion-corrosion controls using the CHECMATE program, which is the program being used by many utilities in the US.

3.5.9 External Hazards Issues

3.5.9.1 Extreme Weather Conditions, Low Temperatures

According to IAEA (IAEA 1988b; IAEA 1996a) proposed sites are required to be adequately investigated with respect to all the characteristics that could affect safety in relation to design basis natural events. A site specific assessment is the first step to reach a decision regarding a particular event. A systematic site specific assessment of this nature is not evident for WWER-1000 nuclear power plants.

Recommendations by IAEA (IAEA 1996a)

- A site specific assessment should be made with respect to the design basis natural events.
- A probabilistic analysis could be utilized to assess the potential hazard.

Conclusions by IAEA (IAEA 1995d and IAEA 1997)

- The natural phenomena like temperature, snow and wind have been considered in the design. It is assumed that the related severe conditions occur once in 10,000 years.
- Both plants have no special design against a tornado.

Conclusions and Recommendations by Riskaudit (RiskAudit 1997a):

No measure has been proposed originally by the utilities regarding extreme temperature conditions. The situation was found to be satisfactory for hot temperature. A study has been recommended to be performed for extreme low temperature in order to identify needed modifications (heating systems are not classified), engagement is taken in revision 2.

Existing situation is satisfactory regarding external flooding. On tornado aspect, a study is proposed to be performed.

3.5.9.2 Man-Induced External Hazards

There is a wide variety of potential man-induced hazards that can impact NPP safety. The IAEA safety issue review of VVER-1000/320 units mentions specifically only the potential impact of blast and impact loads on NPP buildings other than the reactor building. However, IAEA recommends that screening analyses be performed to identify man-induced hazards which should be evaluated in more detail if the screening analysis cannot exclude possible impacts on safety. The IAEA has identified man-induced external events as a Category II Safety Issue (IAEA 1996a: 106).

The K2/R4 modernization programme has identified the following measures as responding to this issue:

- 18211 (risk assessment of shock wave loads and their impact on plant structures; **before** startup);
- 18311 (analysis of aircraft crash onto reactor building; **before** startup);
- 18321 (risk assessment of impact on MCR/ECR personnel of toxic gases; **before** startup).

In contrast with the IAEA recommendation, the modernization programme does not include an overall assessment of external man-induced hazards. Further, the analysis which is included of aircraft crash is limited to the reactor building. Clearly, the reactor building is important, but not uniquely so for aircraft crash. Aircraft crash into other buildings which are not designed for such impacts is possible, including the ESW building, the turbine building, and the diesel generator building. The ESW building is of particular relevance since the ESW system is shared among multiple units at both the Khmelnitsky and Rivne sites. An aircraft crash that disables the ESW system at either site could result in multi-unit concurrent core damage accidents. In addition, it should be recognized that it is not just direct impact that could have an effect on plant systems. For buildings such as the ESW building, nearby impact and sliding into the building could cause considerable damage, as could a nearby impact and a resulting fire. Guidance in this regard has been provided in a USDOE standard that is freely available (USDOE 1996). Another potential vulnerability that should be explored is aircraft crash into the switchyard, causing loss of offsite power with a resulting fire causing failure of the diesel generators (either due to ignition of spilled fuel or due to choking of the diesels from ingestion of combustion products).

The turbine building structures in the VVER-1000/320 design are not designed to withstand special external impacts such as the design basis earthquake, hurricanes, shock waves, etc. Under such impacts, for which the building structures were not designed, the roof and other turbine hall structures dropping onto steamlines, feedwater lines, etc., is possible. The dynamic forces arising from such impacts could rupture more than one steamline or the safety/protective devices installed on the steamlines (SG safety valves, atmospheric dump valves, fast-acting shutoff valves) (MOHT 1996: Part 2, Section 3, p. 25/316). There is no indication in the modernization programme that the special vulnerability of the turbine building to man-caused external hazards is being adequately addressed.

3.5.9.3 Seismicity/Geology

3.5.9.3.1 Geophysical and geological aspects of safety

Shortcomings of the EIAs for Rivne 4 and Khmelnitsky 2

- The EIAs for the NPP's in the districts of Rivne and Khmelnitsky lack documentation of seismological observations and geophysical, engineering geological and hydrogeological investigations. It should be pointed out that there are no real maps in the EIAs (no scale, no altitudes), no geological maps, no cross sections or usual borehole documentation.
- Previous assessments by Riskaudit recommend exhaustive geophysical studies for the sites of Rivne and Khmelnitsky NPP's but deny their urgency. This fits well into "predictions" of some experts that strong earthquakes of a certain higher intensity may be expected not earlier than in 10,000 years. Examples from other regions proved this assumption to be totally wrong.
- There is no comparison of the seismic hazard and the hazards on groundwater between the sites of K2, R4 and Chernobyl NPP.
- The complex hydrogeological situation (Shestopalov 1970) was neither investigated nor presented sufficiently by Mouchel environmental consultancy. Despite the importance of the different groundwater tables for drinking water supply of big cities there is no information on the situation of several cities e.g. on the interaction of different groundwater levels, protection areas, risks from hazardous wastes (Shestopalova 1998).
- The environmental impacts on groundwater and karst due to pile foundations and grouting as well as the effects of groundwater-contamination and those of aggressive waters and ongoing karst activity on pile foundations (R4) have not been dealt with.
- According to EIA for K2 (Mouchel 1998a: 9.1), construction impacts have not been defined by Energoatom. On the contrary Energotechprojekt performed a quality inspection of civil engineering buildings and structures of K2 and reports on settling and tilting of reactor compartment and turbine generator foundation. From 1986 to 1996 a 52.1 mm mean settling value of the reactor compartment was observed with a maximum rate up to 18mm per year during 1988-1990. Foundation deformations are supposed to result from uneven settling of bases, a strongly heterogeneous, strongly weathered laminated Upper Proterozoic formation consisting mostly of claystones with sandstone intercalations, classified as a "semi-rocky base".

- The Engineering-Geological characteristics of K2 NPP site remain unclear in the EIA and the above mentioned Quality Inspection report: According to the EIA (Mouchel 1998a: 9.3), bedrock is in 5m depth but the weathering zone and geotechnical characteristics are not documented and in chapter 3.2.8 on geology and hydrogeology the geological situation of K2 NPP is not addressed but ground waters in Cretaceous layers are listed.
- Different and contradictory situations are described by Energotechprojekt for K2 NPP site, chapter 5.2.1. Therein fluvioglacial Quaternary deposits varying in thickness from 2-4 meters are underlain by Proterozoic bedrock. Only a few lines later in the same report a thixotropic chalk layer, 5-10 meters in thickness, with a soft plastic consistency above and a fluid plastic consistency below ground water level is mentioned. Neither can be assured that the complete chalk layer was redeposited during the Quaternary nor can a foundation of parts of the buildings on that layer be excluded.
- These contradictions cannot be only the result of insufficient and inaccurate geological documentation but point towards a safety problem. Because the different foundations of buildings may not be seismically qualified a serious seismic hazard cannot be excluded.

3.5.9.3.2 Seismic Hazard and Seismic Engineering

The EIAs for both Khmelnitsky Unit 2 and Rivne Unit 4 report that according to an (unidentified) seismic hazard map of Ukraine, the NPPs are located in a zone where the design earthquake (DE, 100-year return period) is MSK-5 (0.02g PGA), and the design basis earthquake (DBE, 10,000-year return period) is MSK-6 (0.05g PGA) (Mouchel 1998a: 3.6; Mouchel 1998b: 3.6). With the source of this information unidentified, the reader is unable to confirm the validity of these estimates. The EIAs should contain a specific citation to the seismic hazard map, along with the identification of its date of publication in order that the reader may ascertain the credibility of the estimates.[38]

There is no discussion in the EIAs of the use of probabilistic seismic hazard assessments (PSHAs). PSHAs are now widely used to estimate seismic hazards at specific sites, and their use should be encourage here for a facility, such as a nuclear power plant, whose safety is potentially significantly affected by seismic events. The fact that the asserted seismic hazard (as cited above) is so steep (100-year return period at 0.02g, 10,000-year return period at 0.05g) (see Figure 3.9.3-1) suggests that the seismic hazard map is unreasonably optimistic. If this is the case, the design may be biased toward too low a PGA value, making the plant vulnerable to seismic events.

DBE levels set under USNRC regulations for eastern US plants indicate that there may be problems with seismic hazard estimates made in the 1970s and 1980s compared with modern PSHA results. The eastern US is a generally aseismic-area; it is large, mid-plate region with no large faults (such as the San Andreas Fault in California). NPPs in the eastern US are generally associated with DBEs of 0.10g-0.17g PGA. It has been found that these DBE levels are generally associated with earthquakes with return period of 1,000 years rather than 10,000 years (LLNL 1988: 16-18). Prior to this assessment, the DBEs for the eastern US NPPs were asserted to have been conservatively established. This is now called into question. The

[38] RiskAudit has indicated that the existing seismic hazard analysis dates from the 1970-1979 time period (RiskAudit 1997b: 7-40). RiskAudit noted several departures from international practice in the study (RiskAudit 1997b: 7-40, 7-41), and considering the vintage this is not surprising as seismic hazard analysis procedures have substantially matured since the 1970s. A key deficiency is the correlation between macroseismic intensity and PGA. The uncertainties in such a process are large and it is unclear whether the study adequately accounted for this factor. Indeed, it has been recognized in the 1980s and 1990s as a result of earthquake experience that capable tectonic sources are not always exposed at th ground surface. This has been demonstrated by the buried (blind) reverse causative faults of the 1983 Coalinga, 1988 Whittier Narrows, 1989 Loma Prieta, and 1994 Northridge earthquakes (USNRC 1997: App. D). If this can occur in California along the San Andreas Fault, one of the most heavily studied and heavily instrumented faults on the planet, it can surely occur elsewhere.

USNRC's new seismic design regulations are based on the use of PSHA.[39] In addition, once the design has been established, the NRC requires (via Generic Letter 88-20, Supplement 4) that a search for severe accident vulnerabilities from earthquakes be conducted. The vulnerability analysis can take the form of a seismic PSA (which necessarily uses the result of a PSHA) or a seismic margin analysis (SMA), which uses a review level earthquake (RLE) which is about twice the design basis earthquake for the purpose of estimating seismic margin. Guidance for performing SMAs has been issued by the Electric Power Research Institute (EPRI) (EPRI 1991), as modified by USNRC Generic Letter 88-20, Supplement 4 (USNRC 1991b).[40] IAEA found that the seismic design for VVER-1000/320 units was not in accordance with current international practice, and recommended that the seismic margin of the structures, systems, and components be checked. This was cited as a Category II Safety Issue by IAEA (IAEA 1996a: 104).

It is known, as indicated in Section 3.5.1.2, that the ESW pumps for K2 and R4 are **not** seismically qualified (IAEA 1996a: 26, 39). Similarly, it is known that the ESW makeup pumps (which provide makeup water to the ESW system) are not seismically qualified. Without seismic qualification nor a seismic margin assessment of these pumps, it must be assumed that they will fail in a design basis earthquake. If the ESW pumps fail, the Khmelnitsky site will experience a dual-unit concurrent severe accident since the ESW system is shared between Units 1 and 2. Similarly, the Rivne site will experience a four-unit concurrent severe accident since the ESW system at that site is shared between Units 1 and 2 (VVER-440/213 design) and Units 3 and 4 (VVER-1000/320 design).

It is also noted that the turbine building is not seismically qualified, and that collapse of this building could lead to multiple steam line/feed line failures as well as unavailability of steam line isolation. (Concurrent failures of SG tubes under these conditions would lead to containment bypass.) The batteries in the original VVER-1000/320 design were also not seismically protected (IAEA 1996a: 90), but this may have been corrected as part of the upgrade programme (this needs to be confirmed by the EIAs). Finally, neither the ventilation systems nor the fire protection system water pumps are seismically qualified (IAEA 1996a: 26, 39). Any fires occurring after a DBE or more severe earthquake would have to be fought using manual methods since the automatic suppression system would be unavailable.

Finally, it is noted that the ventilation system which provides HVAC functions for the main control room (MCR) and emergency control room (ECR) is not seismically qualified (specifically, the air ducts are not so qualified) (RiskAudit 1997b: 8-110). Thus, following an earthquake all components in the MCR and ECR which require cooling could be damaged. Correction of this matter is not included in the modernization programme, and RiskAudit concluded that the air ducts "*have to be qualified to cope with seismic requirements*" (RiskAudit 1997b: 8-111).

Given this backdrop, considerably more discussion of the seismic hazard in the vicinity of the Khmelnitsky and Rivne sites is required in order to establish comparability with western safety standards. It is noted that Ukraine authorities opposed further seismic analyses beyond ensuring that its plants were designed to the IAEA minimum of 0.1g (IAEA 1996a: 188). The EIAs should, as a minimum, ensure (through the Environmental Action Plan or EAP) that proper seismic analyses are required to be performed for K2 and R4.

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[39] The USNRC has also defined an SMA methodology. Guidance is contained in three USNRC reports: (a) NUREG/CR-4334, **An Approach to the Quantification of Seismic Margins in Nuclear Power Plants**, Lawrence Livermore National Laboratory, August 1985; (b) NUREG/CR-4482, **Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margins Reviews of Nuclear Power Plants**, Lawrence Livermore National Laboratory, March 1986; and (c) NUREG/CR-5076, **An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR Plant Systems and Functions to Seismic Margins**, Lawrence Livermore National Laboratory, May 1988. Additional guidance is also contained in NUREG-1407 (USNRC 1991a).

[40] A K2/R4 upgrade programme measure (Item 13611) is defined to improve the leaktightness of the ECCS heat exchangers, but no hardware modifications are envisioned before startup (KIEP 1996: 69/316). A measure to detect radioactivity in the ESW water has already been implemented (KIEP 1996: 70/316).

A modern seismic hazard analysis of the Khmelnitsky site has been performed. The study shows a PGA value of 0.18g at a return frequency of 10,000 years (Gelder & Varpasuo 1998). This is considerably higher than the EIA value of 0.05g at 10,000 years return period, and well in excess of the 0.1g PGA value used in the assessment of structures, systems, and components in the modernization programme. Rather than 0.1g representing a “conservative” value, it considerably underestimates the seismic hazard. Since the hazard a Khmelnitsky for return periods of less than 1,000 years is dominated by the Vrancea region, it is expected by IRR that similar results would be obtained for the Rivne site. Western safety practice in seismic is to confirm the adequacy of design by performing either a seismic PSA or a seismic margin analysis. Given the disparity between the plant design and the currently-assessed seismic hazard, it is essential for one or the other of these studies to be performed for K2 and R4. If a seismic margin analysis is performed, it should use a Review Level Earthquake (RLE) of at least 0.3g, as recommended by EPRI and the USNRC.

3.5.10 Accident Analysis Issues

3.5.10.1 Probabilistic Safety Assessment (PSA)

It is important to distinguish between deterministic safety analysis and probabilistic safety assessment (PSA). The **deterministic** approach is based on leaktight barriers and defense-in-depth. It attempts to ensure that an NPP is adequately designed to respond to design basis accidents. It has to be recognized, however, that there are some types of accident initiating events and some types of equipment failures which can cut across multiple levels of defense-in-depth. In addition, the design basis for NPPs is not without limit. The **probabilistic** approach involves an assessment of the residual risks which remain after the deterministic design approach is applied at a particular NPP. The probabilistic approach attempts to treat all aspects of NPP safety in a common context, allowing an understanding of the relative importance of various initiating events, hardware failure modes, and operator actions to core damage frequency (CDF). (Of course it is true that not **all** aspects of NPP design and operation can be modeled in a PSA, e.g., management influence on “safety culture”.)

IAEA states (IAEA 1996a: 119; underlining emphasis added), “PSA is an important tool which evaluates all the different aspects (technical and human) in the assessment of plant safety. PSA may be used to rank the importance of the different aspects of the plant in terms of nuclear safety. **Specifically, PSA results are an important base for the assessment of the measures directed to upgrading the safety**.” Similarly, the IAEA INSAG has stated (INSAG 1992: 18; underlining emphasis added), “… it can be said that PSA has matured to a point where there is now a broad basis of understanding. There is agreement that it constitutes a valuable additional basis for safety assessment and for decisions on safety improvements. **In particular, the methodology has proven to be a most valuable tool for identification of plant weaknesses.**” IAEA recommends that a Level 1 PSA (core damage frequency assessment) be performed as a minimum for all VVER-1000/320 NPPs (IAEA 1996a: 119).

There are no plans to perform PSAs for K2 and R4 until after commissioning. Since the IAEA safety issue report is nearly entirely **deterministic** (only one issue mentions PSA as a partial justification), and since the K2/R4 modernization programme similarly mentions PSA as a partial justification for only one item, this is a potentially crucial matter for the success or failure of the upgrade programme to reduce risk. **Without a PSA, there is no assurance that the modernization programme has not missed one or more important core damage frequency contributors which could nonetheless contribute to a high CDF despite implementation of the modernization programme.** Indeed, this seems likely since no PSA to date has identified I&C issues as dominant contributor to CDF and since by the project sponsor’s own admission the K2/R4 modernization programme is largely directed to I&C issues

(Mouchel 1998a: 0.3; Mouchel 1998b: 0.3).[41] The proposal by the project proponents for performing the PSA after startup seems to be driven by the assumption, contained in the modernization programme, that such an analysis would require 48 months for completion. This is an excessively pessimistic schedule. A PSA programme for Rivne Unit 3 foresaw completion in 24 months (RiskAudit 1994b: 9-36). The USNRC PSA Procedures Guide indicates that a representative PSA schedule is 22 months for a Level 1 PSA and 24 months for a Level III PSA (USNRC 1983: 2-23, 2-24). Clearly, it should be eminently possible to perform a PSA within 24 months, leaving 12 months available in the 36-month schedule established by the project proponents to implement any necessary remedies indicated by the PSA results.[42]

3.5.10.2 Rapid Reactivity Increase/Control Rod Ejection

The effect of rapid reactivity increase caused by ejection of control rods was addressed by IRR (IRR 1997):

The rapid reactivity and power increase in WWER-1000s during operation could be caused mainly by control rod ejection. Sixty-one neutron absorbers (control rods), can be moved up and down in the core by individual drive mechanisms. The WWER-1000 control rod drive mechanisms are installed in the housings, which are connected to the reactor upper unit and thus to the primary circuit. In the case of sudden rupture of the particular housing, large pres-

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[41] RiskAudit (RiskAudit 1997b: 23, for example) has taken the position that a PSA performed before operation of an NPP "*can only give some highlights on some sequences but can not be considered as really specific*". IRR states plainly that this is nonsensical, and RiskAudit is the only technical organization in the world which holds to this belief for which there is no technical basis whatsoever. A PSA is indeed best performed **precisely** at this time in order to confirm the risk adequacy of the design. (Indeed, the seminal WASH-1400 PSA was performed with very little plant-specific data; the study was issued in the fall of 1975, and the Surry plant had come on line in December 1972, while the Peach Bottom unit came online in July 1974.) PSA was originated as a **design tool**, not as a solely post hoc analysis which RiskAudit advocates as the only real PSA. It is true that there is no plant-specific data available prior to startup, but it is also true that **any** PSA relies on considerable non-plant-specific data, and this makes such analyses no less valid. The PSA performed prior to operation should, of course, be continuously updated (living PSA) to reflect plant-specific data as it becomes available. But it is **absurd** to suggest, as RiskAudit has done, that a PSA performed before operation has no validity. The primary value of the PSA is the engineering analysis embedded in the fault and event trees and systems relationships among one another. It is precisely to capture and update plant experience data that it is widely recommended that PSAs be maintained as living documents (i.e., that they be updated periodically, such as after each refueling outage, to capture design changes and to update the plant-specific data).

There are numerous PSAs which have been performed on PWR NPPs prior to operation (e.g., Sizewell, Seabrook, Mochovce, Temelin, Millstone Unit 3, Watts Bar, Comanche Peak, South Texas, etc.), and it is to be noted that the USNRC regulations **require** the performance of PSAs for all new NPPs and standard NPP designs. Most of the US PWRs cited in the previous sentence were subjected to detailed peer reviews sponsored by the USNRC, and **not one of those reviews cited the lack of commerial operating experience as a par to either performing the analyses or using their results in assessing the safety of the plants involved**. In addition, the INSAG specifically identifies the use of PSA for plants with no operating history, referring to such a PSA as an *a priori* analysis (INSAG 1992: 4). RiskAudit's position on PSAs performed before plant operation is **technically indefensible** and without merit, and it should be ignored by all parties.

[42] IRR does not regard the 36-month project completion schedule as reasonable. The modernization programme documentation (KIEP 1996) makes it abundantly clear that there are many key design desicions yet to be made. Clearly there will be a period of detailed engineering which must be accomplished before construction is restarted. US experience over the past 20 years indicates that project completion even **with** an established design (i.e., **without** major design modifications) and without economic constraints allows a completion rate of about 1% complete per month. In addition, for projects which experienced a construction hiatus, it generally takes 3-6 months to re-establish full construction. Thus, with a project 80% complete and construction suspended with an established design, it would be expected that completion would require 23-26 months. Given design modifications as contemplated by the project proponents, as well as refurbishment and requalification activities, as well as the need for engineering of design changes to precede construction, it seems to IRR to be excessively optimistic to suppose that 36 months will be adequate to permit project completion from the time funds are authorized. A period of 48-60 months seems much more plausible, considering all of these factors. Thus, there is all the more imperative to perform PSAs now in order to understand more precisely the risks posed by the **existing** design and how those risks might be ameliorated if necessary, since there will most likely be sufficient time to perform the PSA, have it peer reviewed, identify needed design changes, and implement those changes before commissioning.

sure differences will eject the corresponding control rod and the drive shaft to the fully withdrawn position within 0.1-0.2 s. In the event that the control rod was initially inserted in the core, the consequences of this failure are rapid increase of the neutron flux and thermal power, with an adverse power distribution in the core.

The effect depends mainly on which of the 61 control rods will be ejected, its initial position, and the initial neutron flux distribution. The transient could be mitigated by the negative reactivity feedback and terminated by actuation of the scram system. However, in some cases (effective control rod fully inserted in the core) the rapid control rod ejection could result in fuel melting, damage to the fuel rod cladding, and damage to the primary circuit boundary (IAEA 1995a). After such an accident, the possibilities to cool the core could be significantly affected. Sudden rupture of a control rod drive mechanism housing will also perforate the reactor upper unit, leading to loss of coolant accident.

The consequences of such an accident are restricted by the limited reactivity efficiency of a single neutron absorber and by the presence of control rod insertion limits, which vary as a function of power level. However, the control rod insertion limits for WWER-1000s are assured not by technical means, but merely by administrative measures. The reactor is operated with control rod group No. 10 partially inserted in the core and, as a rule, well above the insertion limits; nevertheless it could be also operated with control rods near or below the insertion limits after unplanned changes of the power level or during load follow operation (IAEA 1995c). In the beginning of fuel cycles, or after scrams, the reactor could be operated by control rod group No. 9, while group No 10 is fully inserted.

Conclusion by (IAEA 1995a)

Code calculations for Temelin NPP were performed with the result that conservative analysis indicate that the fuel and clad limits for the WWER-1000 reactor are not exceeded. No danger for sudden fuel dispersal into the coolant and no danger of further consequential damage to the RCS has be expected.

Comments by IRR:

The following aspects are not included in the control rod ejection analysis for Temelin:

- According to the original Russian design, during physical start-up of WWER-1000s the control rod groups Nos. 10 and 9 are fully inserted in the core in order to decrease the boric acid concentration in the primary coolant and to obtain a negative temperature reactivity coefficient. The criticality and power level of the core are controlled by group No. 8.
- For Temelin the worst case assumed is that the rods are positioned at the insertion limits. However the reactor operator is fully responsible for remaining within these limits and they could be easily violated during some transients.

3.5.11 Spent Fuel and Radioactive Waste Management Issues

3.5.11.1 Spent Fuel Storage

In the VVER-1000/320 design, the spent fuel pool is located inside the containment. This has advantages over designs where the spent fuel pool is located outside containment, since in the event of a severe spent fuel pool accident radioactive releases would be contained in the VVER-1000/320 design.

There are certain problems with the design of the spent fuel pool cooling system (SFPCS), and the location of the spent fuel pool inside the containment. First, the SFPCS design includes normally-open pneumatic valves. If the pneumatic system fails, the valves fail closed, interrupting spent fuel pool cooling. The containment spray system must be used (manually actuated by the operators) in order to maintain the spent fuel pool inventory under boiling conditions (MOHT 1996: Part 3, p. 79). It is not clear to what extent the operability of plant systems located inside the containment has been evaluated under boiling spent fuel pool conditions and/or containment spray conditions. (The SFPCS is a three-train system with heat exchangers cooled by the raw water system. It is unclear from this RiskAudit report reference whether the “raw water system” refers to the essential service water system at Rivne or some other water system; if the latter, it is not clear whether the system is provided with backup power if offsite power is lost.) (RiskAudit 1994: 2/17)

In the event of a loss of spent fuel pool cooling, there are only seven hours available to restore cooling before the pool water temperature reaches the maximum allowable limit (RiskAudit 1994: 11/104).

Lack of a spent fuel cask drop accident analysis in the VVER-1000/320 design has been identified by IAEA as a Category I Safety Issue (IAEA 1996a: 121). In addition, the need for interlocks to prevent transport of heavy loads by the polar crane over the spent fuel pool has been identified by IAEA as a Category II Safety Issue (IAEA 1996a: 103).

3.5.12 Operating Procedures Issues

One of the key post-TMI accident requirements issued by the USNRC in NUREG-0737 was the requirement for plant operators to implement **symptom-oriented** emergency operating procedures (EOPs). Prior to the accident, EOPs were event-oriented, and such procedures were recognized in the aftermath of the accident as more prone to error than symptom-oriented procedures. This insight was reinforced by the Chernobyl accident experience. The generic weakness of event-oriented EOPs is that the operating personnel first have to identify (diagnose) the event, then select the correct procedure to control the event, and finally correctly implement the procedure. In any event, even if the event were correctly diagnosed and the right procedure were selected for implementation, the event-oriented EOPs were “one way” procedures oriented toward success, which fails to take into account unforeseen events and mistakes by operators along the way (IAEA 1996a: 126).

The existing EOPs at Rivne Unit 3 are instructive as to quality and usability in an accident; and absent an effort to shift to symptom-oriented EOPs, the Rivne Unit 3 EOPs can be taken as exemplary of the situation which will likely exist at K2 and R4 at startup. The Rivne Unit 3 EOPs are event-oriented – that is, the operators must directly recognize what accident they are dealing with and select the appropriate procedure. Moreover, the procedures are not intended to be used during an accident; rather, they are supposed to be known by heart. There is no system for diagnosing the accident; the qualification of the operators (graduate engineer equivalency) and the training of the operators is relied upon for this purpose. There is no description in the procedure of how the diagnosis of the accident is to be made, nor is there a description of the systematic verification of the proper operation of the safety systems (RiskAudit 1994: 12/19).

The IAEA has recommended implementation of symptom-oriented EOPs for VVER-1000/320 NPPs (IAEA 1996a: 126).

Symptom-based EOPs have been developed for the Balakovo and Zaporozhe VVER-1000/320 units (IAEA 1996a: 126), as well as for the Temelin units. Thus, there is no reason why these existing symptom-oriented EOPs could not be used as the basis for creating plant-specific EOPs for K2 and R4. However, the modernization programme for K2 and R4 envisions use of event-oriented EOPs at startup. Symptom-oriented EOPs will not be developed and implemented until after startup as part of upgrade programme Measure 30211 (KIEP 1996: 278-279/316).

One specific procedure deserves mention, that being implementation of feed and bleed cooling. Loss of all feedwater is well recognized as a potentially dominant contributor to CDF for

PWRs. Feed and bleed can be used to reduce the risk of this accident type, and it involves the opening of the pressurizer PORV and the use of high pressure injection to remove core heat. It appears that K2/R4 will startup (proposed) without a bleed and feed procedure implemented (RiskAudit 1997b: 8-128, 8-129). It must be observed that one would not encounter feed and bleed except under conditions of inadequate heat removal in which without feed and bleed the core would melt in any event. If the PORV fails while relieving water flow, it will fail **open**. This is not a significant problem since the valve needs to be open to perform feed and bleed in any event. It would represent a design basis accident (small LOCA) for which reason the high pressure ECCS system is specifically provided. There is no reason for not implementing a feed and bleed procedure, at least as a last resort to forestall core damage. The plan to commission K2 and R4 without a feed and bleed procedure could result in a high CDF contribution (of the order of 10^{-4} per year).[43]

3.5.13 Logistics and Infrastructure

General economic crisis

According to Kopchinsky (Kopchinsky 1997): *"The economic situation in Ukraine remains extremely complicated."* H. Boss from the Vienna Institute for International Economic Studies (Boss 1998) summarized: *"It is ironic that Russia's "international" financial crisis struck when Ukraine's recorded economy had finally achieved stagnation and the first glimmers of recorded growth, after eight years of decline."* Boss also stated that the IMF expects zero growth for 1998 and has cut his forecast for 1999 from 3% to 1%.

Financial situation in the power generation industry

According to Kopchinsky (Kopchinsky 1997):

> "*Payment crisis in the branch appears to be the major cause of the crucial situation in the electric power industry of Ukraine....The creditors debt of the Ministry of Energy subordinate power plants (according to statistics as of 01.06.1997) reaches 4,4. bln grivnas (2,3 bin US dollars). The constituents of this debt are: gas: 1628 mln grivnas, electric power of NPPs: 1 583 mln grivnas, coal: 598 mln grivnas." "The worst fact is that the creditors debt is 1 bln grivnas higher than the debits one, which actually points out a financial paralysis of power industry. The consumers' debt as of 01.07.97 reaches 3,4 bln grivnas. Main debtors are industrial enterprises, public utilities and budget-financed organisations." "Among the causes of the crucial financial situation in the branch are also an advancing growth of organic fuel prices and an overrun of the generated power first cost above the selling tariffs. The government cannot decide to settle this obvious situation, being afraid of social problems and inflation.*"

[43] This is unquestionably the case because loss of offsite power as well as loss of main and auxiliary feedwater places a demand on the EFW system. The EFW system must be manually placed into action by operator action to open the admission valves to the SGs. Failure of the operators to perform this action will dominate the EFW system failure probability. In the Dukovany PSA (VVER-440/213), the human error probability for this identical action was estimated at 5.0×10^{-3} per demand (SAIC 1994: Section 7.3). Based on VVER-1000 operating history, the initiating event frequency for loss of feedwater is 1.8×10^{-1} per year (IAEA 1994d: 37); for loss of offsite power it is 6.7×10^{-2} per year (IAEA 1994d: 63). The conditional failure probability of AFW (2-train system, motor-operated pumps) is of the order of 3.6×10^{-3} per demand. Without feed and bleed, there are two sequences which can be defined: (a) loss of feedwater, failure of AFW, failure of EFW; and (b) loss of offsite power, failure of EFW. The frequencies of these sequences are: (a) 1.8×10^{-1} per year times 3.6×10^{-3} per demand times 5.0×10^{-3} per demand; and (b) 6.7×10^{-2} per year times 5.0×10^{-3} per demand; the frequencies are 3.2×10^{-6} per year and 3.4×10^{-4} per demand, without considering recovery. Even a very poorly reliable feed and bleed procedure would reduce these frequencies by a factor of 30 or more (rule-based procedure, with hesitancy and low success likelihood, 30 minutes or more to implement; SAIC 1994: Table 7.1-5).

Spare Parts

Kopchinsky (Kopchinsky 1997):

> "*The situation at nuclear power plants is now extremely complicated. A considerable part of the equipment has worked out its useful life (valves, control system, elements of the control and protection system, etc.). A lack of finance for purchasing necessary spare parts and replaceable equipment, insufficiency of machine-building industry (more than 60% of spare parts supplies come from Russia) result in a decrease of planned maintenance of nuclear units.*"
>
> "*NPPs' reconstruction and modernisation are going very slowly for the same reason. In 1996, about 400 mln grivnas were allocated for these purposes (about 210 mln US dollars). However, only 30% of the allocated sums were used." "Reconstruction of the operating Ukrainian power plants is planned. However, those plans are implemented very slowly because of lack of finance.*"

Unpaid salaries

Within the last years the personnel of Ukrainian NPPs held several protest meetings or protest demonstrations to obtain their salaries being unpaid for some month. As reported several times the non-payment has a negative impact on the motivation of the Ukrainian workers in the nuclear industry.

Similar problems exist in other sectors of electricity generation:

Kopchinsky (Kopchinsky 1997):

> *"Economic recession and deficiency of organic fuel have resulted in inefficient operation of thermal units. The utilisation factor of their installed power decreased 2,2 times compared to 1990. It concerned to a greatest extent the most efficient gas-and-oil-burning power units of 300 and 800 MW unit power."*

An increasing number of power lines is reaching end of lifetime.

3.5.14 Additional Safety Issues

Two additional safety issues have been identified which are treated below. These issues are:

- Complete loss of heat sink (loss of ESW), Section 3.5.14.1.
- Attempted application of leak-before-break (LBB) to secondary piping, Section 3.5.14.2.

3.5.14.1 Complete Loss of Heat Sink (Loss of ESW)

The ESW systems at both K2 and R4 serve as the ultimate heat sink for the plants. As noted previously, the ESW systems are shared systems at both plants. That is, the same ESW system at Khmelnitsky serves both Units 1 and Unit 2, and the same ESW system at Rivne serves Units 1, 2, 3, and 4. A failure of ESW at either site could result in multi-unit concurrent core melt accidents; in addition, the spent fuel stored in the containment in the spent fuel pool would be vulnerable to a severe accident caused by boiloff of inventory. Such an accident would generate considerable additional hydrogen and fission products which would be released into the containment. No makeup to the spent fuel pool would be possible since all makeup source pumps would be failed by the loss of ESW.

The importance of the ESW system is illustrated by a listing of the equipment served by the ESW system at Rivne Unit 3; all of this equipment would fail unless a failure of ESW is recovered in time (RiskAudit 1994: 2/15):

- The emergency diesel generators.
- The motors and bearings of the containment spray pumps.
- The motors and bearings of the high-pressure safety injection pumps.
- The motors and bearings of the high-pressure emergency boration pumps.
- The motors and bearings of the low-pressure safety injection pumps.
- The motors and bearings of the auxiliary feedwater (AFW) pumps.
- The motors, bearings, and oil coolers of the primary coolant system makeup pumps.
- The ECCS heat exchangers.
- The heat exchangers of the RCP cooling system.
- The heat exchangers of the spent fuel pool cooling system.

Also as previously noted, the ESW system (both the main pumps and the pumps of the ESW makeup system which can pump makeup water from the river) is not seismically qualified. Given the shared nature of the service water systems at Khmelnitsky Unit 2 and Rivne Unit 4, failure of service water makeup without recovery would eventually result in concurrent core damage accidents at multiple units (2 units at Khmelnitsky, 4 units at Rivne).

The IAEA has ranked complete loss of heat sink as a Category II Safety Issue. IAEA recommended that the scenario be analyzed to have a clear understanding of the consequences of the loss of ESW and to develop compensatory measures to cope with this BDBA scenario (IAEA 1996a: 124).

The modernization programme for K2/R4 has identified the following measure to respond to this issue:

- 19211 (analysis of NPP behavior for beyond-design-basis accidents; **before** startup only for complete loss of feedwater and station blackout; **after** startup for all other accidents; no compensatory actions identified); loss of heat sink is not specifically mentioned.

As loss of heat sink is not explicitly mentioned in the list of BDBAs for which calculations will be done, it cannot be concluded that the modernization programme is responsive to this issue. Moreover, even if it is included, the calculations will not be done until **after** startup. Finally, no measures are proposed to seismically qualify the ESW system, to separate the system among the units to avoid concurrent accidents, or to otherwise enhance the capability of the plant to avoid core damage given a loss of heat sink.

3.5.14.2 Attempted Application of Leak-Before-Break (LBB) to Secondary Piping

The K2/R4 modernization program envisions application of leak-before-break (LBB) for steam lines and feedwater pipelines (KIEP 1996: Section 3, p. 28/316). The MOHT consortium also recommended implementation of leak-before-break (LBB) for secondary as well as primary piping (MOHT 1996: Part 2, Section 3, p. 27/316).

It is well-recognized that LBB is **not** applicable to secondary piping which is vulnerable to erosion-corrosion-induced pipe wall thinning. This is amply demonstrated by a lengthy history of steam and feedwater line ruptures in PWR NPPs. If indeed leaks did occur before these breaks, the time interval between leak and break was so short that nothing could be done to prevent the break. In a number of these incidents (including the 1986 feedwater line break at the Surry plant in the US) resulted in fatalities for workers who were in the vicinity of the pipes when they failed. Were LBB applicable to steam and feedwater piping, these fatalities would not have occurred. Whatever its merits for primary piping, LBB is not applicable to secondary piping.

IAEA guidance on LBB is quite explicit that LBB is inappropriate for secondary piping. IAEA states that it is “not a recognized practice to apply the LBB concept to secondary and small diameter primary piping” (IAEA 1994a: 8). The reasons for this are clear – LBB applies **only** to piping where there is no significant corrosion (IAEA 1994a: 9,11). Implementation of LBB to piping requires a corrosion damage analysis, which requires that none of the materials is "*susceptible to intergranular corrosion cracking and the corrosion wear is negligible*" (IAEA 1994a: 12). Secondary piping (steam and feedwater) are significantly subject to erosion-corrosion damage and other forms of corrosion, and have a long history of pipe ruptures to indicate that LBB is inapplicable to secondary piping. Indeed, at least one recent event in the United States (which occurred at the Millstone plant) involved pipe whip which ruptured a redundant, nearby steam line. Erosion-corrosion occurs in both single-phase and two-phase secondary piping. Indeed, it has been reported that feedwater lines at Rivne Unit 3 have already experienced erosion-corrosion-related pipe thinning (RiskAudit 1994: 4/28). Accordingly, it is inappropriate to attempt to apply LBB to secondary piping.

3.6 K2/R4 Modernization and Upgrade: Technical Issue Evaluation

3.6.1 Issue Evaluation Criteria

In selecting issues for treatment in this report, expert judgment was used. This expert judgment was applied in the light of experience in risk analysis and safety evaluation, and based on previous work on VVER-1000/320 and RBMK reactors in general (and on Khmelnitsky Unit 2, Rivne Unit 4, and Chernobyl in particular).

In evaluating the selected issues to identify those which are considered to be the most significant from the standpoint of safety, the following criteria and considerations were taken into account:

- IAEA Safety Issue Categorization in IAEA-EBP-WWER-05. IAEA categorized VVER-1000/320 safety issues into four categories (IAEA 1996a: 10-11):

  Category I – Issue reflects a departure from recognized international practices. It may be appropriate to address them as part of actions to resolve other higher priority issues.

  Category II – Issue is of safety concern. Defense-in-depth is degraded. Action is required to resolve the issue. (Defense-in-depth is considered to be "*degraded*" if one or more levels of protection are affected, and a safety function is impaired for design basis accidents or is questionable for beyond design basis accidents.)

  Category III – Issue is of high safety concern. Defense-in-depth is insufficient. Immediate corrective action is necessary. Interim measures might also be necessary. (Defense-in-depth is considered to be "*insufficient*" of one or more levels of protection are seriously affected, and a safety function is questionable for design basis accidents or disabled for beyond design basis accidents.)

  Category IV – Issue is of the highest safety concern. Defense-in-depth is unacceptable. Immediate action is required to overcome the issue. Compensatory measures have to be established until the safety problem is resolved. (Defense-in-depth is considered to be "*unacceptable*" if one or more levels of protection are lost, and a safety function is disabled for scenarios within the design basis.)

  The issue categorizations are based on an evaluation of the potential degradation of defense-in-depth. There are five levels of defense-in-depth (IAEA 1996a: 10):

  Level 1: Conservative design; quality assurance; safety culture.

  Level 2: Control of abnormal operation and detection of failures.

  Level 3: Safety systems and protection systems.

  Level 4: Accident management including confinement protection.

  Level 5: Offsite emergency response.

  Defense-in-depth is implemented to prevent damage to the plant and to mitigate the consequences of damage if it nonetheless occurs. Impairment of defense-in-depth was judged based on degradation of barriers and an evaluation of the performance of the main safety functions of controlling power, cooling the fuel, and confining radioactive material. Plant conditions with a relatively high frequency of occurrence should have small consequences, and plant conditions resulting in plant damage with large radioactive releases should be low in frequency. Operational safety issues need to be evaluated on a plant-specific basis, and were therefore identified without being placed in a category (IAEA 1996a: 10-11).

- The issue was considered to be significant if it is associated with a credible (i.e., with an estimated frequency of occurrence of $1.0×10^{-6}$ per year or greater) potential for a common-mode, common-unit accident – that is, concurrent core damage accidents in two or more units at the same site due to a common cause initiator (e.g., an external man-made or natural phenomena hazard that can affect more than one unit simultaneously; or the failure of a shared support system that can both initiate a transient or LOCA **and** cause the failure of systems needed to respond to the initiating event).
- PSA results for VVER-1000/320 reactors (or other VVER reactors or other PWRs, provided that design differences are taken into consideration) are considered. If an individual sequence (or a group of related sequences sharing a common initiator or functional failure) is identified with a CDF contribution greater than or equal to $5.0×10^{-5}$ per year, the issue related to the sequence is considered to have high safety significance. (The value of $5.0×10^{-5}$ per year is half the INSAG safety target of $1.0×10^{-4}$ per year for CDF.) If an individual sequence (or a group of related sequences sharing a common initiator or functional failure) is identified with a CDF contribution between $1.0×10^{-6}$ and $5.0×10^{-5}$ per year, the issue related to the sequence is considered to have safety significance
- PSA results are also considered in the context of large release frequency (LRF) (i.e., sequences which result in early containment failure or containment bypass). If an individual sequence (or a group of related sequences sharing a common initiator or functional failure) is identified with an LRF contribution greater than or equal to $5.0×10^{-6}$ per year, the issue related to the sequence is considered to have high safety significance. (The value of $5.0×10^{-6}$ per year is half the INSAG safety target of $1.0×10^{-5}$ per year for LRF.) If an individual sequence (or a group of related sequences sharing a common initiator or functional failure) is identified with a CDF contribution between $5.0×10^{-7}$ and $5.0×10^{-6}$ per year, the issue related to the sequence is considered to have safety significance.

The IAEA Category rankings for safety issues are noted throughout this report where applicable. This ranking was not used in isolation, however, since the rankings are **deterministic** in nature, and there is no assurance that issues with the same ranking have a comparable effect on **risk**.

3.6.2 Issue Evaluation

This section evaluates the issues identified in previous sections of the report. Each issue is briefly identified, the source of the issue is specified, and the issue is evaluated against the evaluation criteria set forth above in Section 3.6.1.

3.6.2.1 Comparability of K2/R4 “Safety Level” With Western NPPs

Mouchel claims that the modernization programme will result in K2 and R4 being completed to an internationally accepted level. Mouchel claims that the safety level for K2 and R4 will be similar to those of similarly aged and recently re-licensed western NPPs. Finally, Mouchel claims that the CDF for K2 and R4 will be close to the value for recently re-approved PWRs and significantly lower than the corresponding values for the RBMK.

Based on IRR’s review, we believe that none of these claims is correct. In contrast, see the conclusions below in Section 3.6.2.12. IRR believes, based on substantial evidence, that the K2/R4 designs will exceed the INSAG CDF target due solely to SG collector failures and SG tube ruptures, and will also exceed the INSAG large release frequency target by a large margin (a factor of ten or more) also due solely to SG collector failures and SG tube ruptures. IRR is aware of no western PWR with a large release frequency in excess of 10^{-4} per year, so there can be no pretense of western safety comparability for K2 and R4. (These conclusions are also valid for the existing eleven operating VVER-1000/320 units in Ukraine.)

The K2 and R4 core damage frequency will be high, but probably comparable to western PWRs. The large release frequency, in contrast, will be far in excess of western PWR values, exceeding the IAEA INSAG large release frequency safety target by a large margin. The CDF and large release frequency values for K2 and R4 will be comparable to and perhaps in excess of those for Chernobyl Unit 3. (A comparison with Chernobyl Unit 1 is not possible due to lack of PSA results for a first-generation RBMK. In any event, operations at Chernobyl Units 1 and 2 have already been ended by Ukraine.)

3.6.2.2 Upgraded VVER-1000/320 Risk Level vs. Upgraded RBMK Risk Level

Mouchel has claimed that normal operational levels of radioactivity release are less in a VVER-1000/320 than in an RBMK. IRR has found this to be true, but irrelevant since the actual doses at 3 km meet regulatory requirements. Moreover, since there are no offsite members of the public within 30 km of Chernobyl, the actual dose to the most exposed individual (MEI) is **lower** at Chernobyl than at either Khmelnitsky or Rivne.

Mouchel has also claimed that the upgraded designs of Khmelnitsky Unit 2 and Rivne Unit 4 pose a lower level of risk of a catastrophic accident than would upgraded Chernobyl units. IRR has found that Mouchel has failed to demonstrate such an advantage. There is no risk assessment presented in the EIAs. There is no risk assessment of either type of unit (VVER-1000/320 or RBMK) cited in the EIAs, nor referenced in the EIAs. The available information (on Temelin and Ignalina) for internal accident initiators suggests that there is reason to question whether in fact K2 and R4 pose a lower level of risk of catastrophic accidents. The problem lies with VVER-1000/320 accidents which bypass containment, thus negating the design advantage of the VVER-1000/320 over the RBMK (the former has a containment while the latter lacks one).

Mouchel has not identified the specific upgrades which it considered in its assessment, either for the K2/R4 units or for the Chernobyl units. Mouchel has also not compared the design or risk characteristics of the K2/R4 units with other VVER-1000/320 units which have performed PSAs. Finally, Mouchel has not compared the Chernobyl units' design or risk characteristics with the Ignalina plant for which a PSA has been performed. Thus, Mouchel has little basis for judging the **risk** of catastrophic accidents at either type of facility.

3.6.2.3 Beyond Design Basis Accident (BDBA) Analysis

Mouchel analyzed what it purported to be "*the most representative*" beyond design basis accident. In fact, Mouchel did not justify its selection. The accident actually analyzed is a small variation on a design basis accident, and the additional failure included is not significant to the outcome of the accident due to the assumption of timely, effective operator action. This is an optimistic assumption, given the lack of effective, symptom-oriented emergency operating procedures at either K2 or R4. The IAEA has identified the specific scenario analyzed as one that should be considered to be **within** the design basis for the VVER-1000/320 design, so Mouchel's characterization of the scenario as a BDBA is questionable in any event. The accident actually analyzed is **not** associated with severe core damage, whereas with a small variation it could be not only a severe accident but one which bypasses containment. Moreover, there are a host of BDBAs which clearly involve core damage, some of which are also containment bypass accidents.

The IAEA has identified numerous potential BDBAs for the VVER-1000/320 design, **all** of which are more serious than the accident selected for analysis by Mouchel. RiskAudit also identified a number of BDBAs, as have a number of PSA and severe accident studies. In short, Mouchel failed to acknowledge most of the other BDBA possibilities, and fail to justify its choice as "*most representative*". Mouchel's choice is demonstrably optimistic. The release fractions estimated for the selected BDBA are so small that the conclusion of no offsite protective actions being required is pre-ordained and not at all surprising. However, given other equally or even more reasonable choices of a BDBA, it is likely that this outcome would not be sustained.

3.6.2.4 Project Options Other Than K2/R4

One of the basic purposes of an EIA of **any** kind is to evaluate **alternatives** to the proposed project. One alternative that is a **given** is the no action alternative. In this case, the no action alternative is continued operation of one or more units at Chernobyl. In the EIAs, however, **no other alternative is evaluated**. This is completely inconsistent with the purpose of an EIA.

Clearly, there are other possibilities for recouping lost capacity from shutdown of Chernobyl. And in fact IRR has identified another possibility which not only accomplishes this financial objective but makes a much more significant contribution to overall risk levels in Ukraine by improving the safety and efficiency of **all** operating VVER-1000/320 units in Ukraine for the same money as is being proposed to complete only two additional units and leaving the existing eleven units untouched.[44]

Other alternatives clearly exist. Mouchel should technically justify its exclusion of other alternatives, and not just buy the "party line" without regard to whether reasonable alternatives exist. To do otherwise is to do the lending institutions **and** the public a serious disservice.

3.6.2.5 Conservation and Requalification

This issue has been addressed by IRR because **compliance with international acceptable practice** in conservation and requalification such as the NRC procedures (USNRC 1986a,b) developed for a longer the construction halt of US plants (Limerick and Watts Bar NPP) **is questionable for K2/R4**.

The issue is **not addressed in the MP**, but included in a repair and replacement program for K2/R4. The safety relevance of this issue is demonstrated by the fact that a reconstitution of a complete manufacturing and construction documentation is yet pending, which has also a negative impact on the next issue.

Conclusions and Recommendations by IRR:

No detailed information about conservation procedures applied to K2/R4 and their compliance or noncompliance with international acceptable standards (e.g. NRC procedures developed for the construction halt of Limerick and Watts Bar NPP (USNRC 1986a, b)) has been provided until now. However, the type of measures in the list of repairs and replacements indicates that conservation was insufficient and confirms the statement of IRR (IRR 1997), that at least during 1990-1993 conservation was marginal, or not accomplished at all, because of the Ukrainian moratorium of completing nuclear projects. The economic crisis in Ukraine, lack of money in the nuclear industry and high inflation rate during this period exacerbated the problem.

The quality of a requalification program is highly dependent on the availability of manufacturing and construction documentation. The reconstitution of a complete manufacturing and construction documentation is a pending, not yet solved issue.

Because of the reported restricted funds in the Ukrainian nuclear industry, it is questionable whether a repair and replacement program, which represents an important contributor to the completion costs, will be fully implemented. This could have a very negative effect on plant reliability and safety.

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[44] The Science Policy Research Unit (SPRU) at the University of Sussex also identified this alternative in their least cost analysis in 1997 (Sussex 1997).

3.6.2.6 Qualification of Equipment

Qualification of equipment (EQ) important to safety is required to demonstrate the ability of the equipment to fulfill their intended functions. This qualification requirement applies to normal operating conditions, to accident conditions and to internal and external events. The EQ issue is a generic safety issue for all WWER-1000 with **IAEA rank III** (IAEA 1996a).

According to IAEA the issue was identified from safety reviews and represents a **deviation from international practice (IAEA 1988a)**. Insufficient or lacking qualification of equipment important to safety would seriously affect levels 1 to 3 of protection of defence in depth and the safety functions (controlling power, cooling the fuel, and confining radioactive material) would be questionable within the class of design base accidents. EQ is of special importance concerning the guarantee of safety function of equipment under extreme environmental or seismic conditions.

Measure 11011 addresses this issue. The implementation is planned partly before and partly after start-up.

The approach in the MP to qualification of equipment is the following: (Riskaudit 1994b)

- For the already installed equipment it is enough to present the available special documents (passport, or technical conditions report or documentation) with the corresponding requirements.
- If these back-up documentation does not exist for equipment already installed then additional requirement for qualification is required.
- New equipment has to be supplied with the proper documentation.

Conclusions and Recommendations by IRR:

- It has also been stated by Riskaudit (Riskaudit 1994b) this approach is insufficient and existing equipment has to be demonstrated to be qualified for accidental situations as a new one.
- Until now no complete programme for equipment qualification has been set up by the project organizations (Riskaudit 1994b). Already existing parts of this programme are not planned to be implemented before start-up. This is a deviation from international acceptable practice. For example these reactors could not be licensed according to US Nuclear Regulatory Commission (NRC)-regulations because equipment qualification is a precondition for licensing.
- Unknown difficulties can be expected due to the unsatisfactory conservation procedures. Qualification of equipment is highly dependent on the status of documentation being available (see above).
- Recommendations on the qualification against seismic loads are included in the seismicity issue (see chapter 3.5.9.3 in this report)
- The process of qualification must be strictly observed by an independent licensing authority.

3.6.2.7 Post-TMI Requirements

The “Post TMI requirements” (USNRC 1980) comprise a list of 36 items which resulted from the lesson of the accident in Three Mile Island (TMI) NPP in 1979. The TMI accident initiated a set of important safety improvements for all NPPs. The accident had a strong impact on areas such as safety analyses, management practices, safety systems and safety culture, improvements of calculational codes, personnel training, investigation of severe accidents, extension of design base accidents, etc.

They are generally ranked as **IAEA rank II** issues (IAEA, 1996a). For US NPPs the implementation of all TMI requirements before start-up is a necessary precondition for obtaining an operating license (USNRC 1980). It is unclear which of the TMI requirements are already implemented in the original design of the Ukrainian WWER-1000/320. TMI requirements exist which are explicitly addressed in the modernization program for K2/R4 which are planned to be implemented after start-up. This fact represents a **deviation from international practice.**

Conclusions and Recommendations by IRR

It is highly recommended that the as yet not implemented or in the MP unaddressed TMI requirements are included in the MP for K2/R4. All TMI requirements have to be implemented before start-up.

3.6.2.8 Control Rod Insertion/Fuel Assembly Deformation

The IAEA ranks this generic issue with category III. The issue was identified from **operational experience**. It represents a **deviation from NUSS and the Russian standard PBJa-89** (PBJa 1989).

According to IAEA this issue indicates a weakness of the core design, i.e. level 1 of protection of defence in depth is seriously affected. In the case of fast transients, an effective reactivity control may not be possible within the required time period (level 2 of protection). In the case of a LOCA or a seismic event, a control rod insertion may not be possible at all due to deformation and, in addition, the flow channels may also be blocked (level 3 of protection). The safety function controlling the power by shutting down the reactor and maintaining safe shutdown conditions may be questioned. The fuel cladding may be damaged.

Compensatory and interim measures have been established as described below to ensure the design limit for control rod insertion time. However, the experience to verify the design modifications by normal operation is not sufficient and the root cause is not fully established.

Measure 11221: "Estimate loads onto supporting frame of fuel assembly" and measure 11222: "Introduce "heavy weight" control rod of fuel assembly" are included in the MP and will be implemented **before start-up** (see Attachment 2 and 3). Up to now it is unclear to what extent the proposed measures will solve the safety issue.

Conclusions and Recommendations by IRR

At the moment the situation is unclear. Contrary to the IAEA assessment (IAEA 1997) Atomaudit claims that the reasons for control rod jamming have been clarified and measures to eliminate them have been developed and implemented in all working units (Atomaudit, 1997).

Additional root cause analysis and extensive operational performance feedback of control rod insertion is recommended.[45]

3.6.2.9 Power Density Control System/Xenon Oscillations

The IAEA ranks this issue in **category II** with the following justification (IAEA 1996a): The core control strategy for load follow mode of operation does not adequately suppress xenon oscillations, thus affecting levels 1 and 2 of protection. The safety function is impaired for scenarios within the DB envelope because the xenon oscillations can cause the local linear power density to exceed the design limit.

Measures (11211 and 11212, see Attachment 2 and 3) proposed in the MP will be implemented **after start-up**.

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[45] RiskAudit has recommended that the K2/R4 modernization programme include a study of the need to install an automatic emergency boration system to counter the potential for mechanical blockage of the control rods (RiskAudit 1994b: 8-131). The USNRC adopted formal ATWS regulations (10 CFR 50.62) in 1984 following an ATWS event at the Salem NPP. The regulation required NPPs in the US to implement equipment, diverse from the scram system, that would automatically initiate actions required to mitigate ATWS events. PWRs were required to have a diverse scram system and a system to automatically trip the turbine and initiate EFW. BWRs receiving construction permits after 26 July 1984 are required to have an automatically initiated emergency boration system (referred to as the standby liquid control system).

Conclusions and Recommendations by IRR:

Adequate measures are still in the testing phase. Neither the MP nor the Riskaudit documents provide more specified information about the automatic power control system planned to be implemented.

The automatic control of Xenon oscillations and power distribution should be implemented before start-up.

3.6.2.10 Reactor Pressure Vessel Embrittlement

This issue is ranked in **category III**. According to (IAEA 1996a) the issue was identified as a **deviation from applicable standards** (OPB-88, PBJa 1989) and from **operational experience**. Unexpected degradation due to improper material specification has been observed which affects level 1 of protection. Adequate monitoring to detect the degradation is not provided affecting level 2 of protection. This may lead to increase the frequency of a vessel failure. This is a BDBA scenario, which would result in an inability to cool the fuel and confine the radioactive material and consequently in loss of all barriers with unacceptable consequences.

The RPV walls represent one part of the third barrier of the radioactivity confining structures. Maintaining the integrity of this barrier under all operating and emergency conditions during the lifetime of the reactor is mandatory. The phenomenon of irradiation embrittlement of the RPV wall is well known for all PWRs. However it is of stronger influence with the VVERs due to the smaller gap of the downcomer and generally unfavorable material conditions (relative high Ni in the belt line weld of the VVER 1000/320 reactors).

All the measures summarized above contribute to a better assessment and mitigation of the RPV embrittlement. Based on the available material however it cannot be assessed whether critical RPV embrittlement will be reached within the planned lifetime of the reactor or not. No results of a Pressurized Thermal Shock (PTS) analysis for the RPV are provided in the available documents.

The number and type of planned measures and actions indicate, that the problem of RPV embrittlement is not only serious for the first generation of VVERs but is still of strong influence for the last generation. Based on the documents review it appears that the usage of data of the surveillance-specimens irradiation embrittlement for predicting reliable RPV wall embrittlement is still an approach with relatively high margins of uncertainty.

Conclusions and recommendations

The RPV embrittlement is generic for all VVER reactors. However, this should not be an excuse but a stronger enhancement to overcome and manage the situation thoroughly.

The proposed measures and actions are generally favorable. However they do not satisfactorily add to the clarification whether the usage of the improved data of the surveillance-specimens reduce the uncertainty in predicting RPV irradiation embrittlement.

An experimental verification of the irradiation embrittlement of the RPV wall is still open. This verification should prove that the measurement of the irradiation embrittlement of the surveillance-specimens provides results which can be applied reliably for the determination of the irradiation embrittlement status of the RPV wall and the core belt line weld.

A comparative assessment is recommended to be performed assessing predicted RPV embrittlement based on embrittlement data of the surveillance-specimens in comparison with measurements of the embrittlement of the actual RPV material. RPVs of reactors which are already taken out of operation are predestinated for such measurements (e.g. VVERs in Greifswald).

It is recommended that experiments of the above mentioned type should become the basis for accurate positioning of surveillance specimens and for the reliable use of their irradiation and embrittlement data for predicting the status of the RPV material in the region of the irradiation influence of the reactor core.

It is recommended that licensing of VVERs like K2/R4 should be based on the demonstration of satisfying test results from above mentioned type of experiments.

3.6.2.11 SG Collector Integrity, SG Tube Rupture, and Non-Destructive Testing (NDT)

Despite the fact that the IAEA has identified the SG collector as the weakest element of the reactor coolant boundary in the VVER-1000/320 design, and despite the fact that this weakness occurs in an interfacing location where failure results in containment bypass (requiring complex operator actions to avoid core damage, using outdated event-oriented EOPs), no substantial improvements have been proposed prior to startup of K2 and R4. Compared with Temelin, which is being upgraded more substantially than K2 and R4 and where SG collector failure is a dominant contributor to CDF and large release frequency (containment bypass accident) with a frequency greater than 5×10^{-5} per year, K2 and R4 should see an even higher CDF contribution from this event because the BRU-A valves are not being environmentally qualified for water and two-phase flow before startup, and because the EOPs are not being upgraded to symptom-oriented procedures before startup. SG collector failure is almost certainly a dominant contributor to risk (both CDF and large release frequency) for K2 and R4, but very little of substance is being proposed to be done prior to startup to ameliorate this risk contribution.

Similarly, SG tube rupture leading to core damage is also largely unaffected by the modernization programme because the dominant contribution to CDF and large release frequency from this event is due to operator error, the likelihood of which will be unchanged due to the lack of implementation of symptom-oriented EOPs before startup. In addition, the SG tube rupture initiating event frequency will remain high because SG tube plugging criteria based on eddy current testing will not be implemented at startup, leading to continued reliance on a procedure which has been shown to yield a high initiating event frequency. This is combined with a failure to replace the condensers with titanium tubing, resulting in greater likelihood of SG tube failure due to environmentally-assisted corrosion resulting from secondary water chemistry upsets.

Finally, the NDT procedures will remain in a defect-reject mode, which leads to a higher frequency of the SG collector initiator than would be the case if a defect-follow mode were to be employed.[46]

At Temelin, SG collector failure and SG tube rupture combine to produce a core damage frequency and large release frequency contribution of 6.4×10^{-5} per year (IAEA 1996b: 113). The same contributions will be higher at K2 and R4 due to higher initiating event frequencies and higher conditional probabilities of operator error following the initiators (which is the dominant cause of core damage from these initiators even **with** symptom-oriented EOPs). IRR concludes that considering these factors, it is plausible that the CDF and large release frequency for K2 and R4 **due solely to SG collector failure and SG tube rupture** may exceed the INSAG CDF safety target of 10^{-4} per year and may exceed the INSAG large release safety target of 10^{-5} per year by a factor of ten or more. This is unacceptable and is strong evidence of a lack of comparability of K2/R4 risk levels with western PWRs.

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[46] It should be noted that the Temelin SG collectors will be made of carbond steel with a lower contact of sulfur and phosphorous using secondary metallurgy which improves the mechanical features of the material. In addition, in attaching the SG tubes a technology was used which reduces stresses considerably. Finally, the collector header cover design was modified to reduce the consequences of header lifting due to connecting bolt failure, resulting in an equivalent diameter for the leak of 40 mm vs. 100 mm for K2/R4 (IAEA 1996a: 153). These improvements have also not apparently been adopted as part of the K2/R4 upgrade programme.

3.6.2.12 ECCS Sump Screen Blocking

The thermal insulation used inside the containment and the total area of the screen above the sump/ECCS water storage tank form a combination that raises high safety concerns regarding the possibility of maintaining ECCS circulation after a medium or large LOCA. **Operational experience** based on recent events in Sweden and in the USA have demonstrated that even a relatively small amount of similar fibers can efficiently block a large screen area. Tests in Zaporozhe came to the same result.

IAEA has ranked this issue in **category III.** This issue was identified on the basis of **international operating experience**. The insufficient design of thermal insulation of equipment and pipelines inside the containment affects level 1 of protection and can, under LOCA conditions, lead to a common mode failure by clogging the sump screens and/or the ECCS heat exchangers. The high risk of losing ECCS recirculation seriously affects level 3 of protection of defence in depth. In this situation the function is thus questionable (or disabled in extreme situations) for scenarios **within the DB envelope**.

The reliable functioning of the ECCS in case of request is essential to manage LOCA consequences. Lessons learned from other NPPs force to take measures to avoid sump screen clogging at the ECC pump inlets of the suction lines under all LOCA conditions.

The measures planned for K2/R4 comprise two important areas of interest. The investigation of the behaviour of the insulation material during a LOCA is one essential task to be performed. The results of this investigation strongly determine the technicyolution to avoid sump screen clogging.

However, it was recognized that due to the uncertainty of the LOCA boundary conditions an installation for each individual sump should be provided for backflushing the sump screen in case of increased pressure drop over the screen using nitrogen from a low-pressure compressed nitrogen system.(Information from the Co-operation Forum of the VVER Regulators, 4,th Meeting, August 97, Tervakoskio, Finland; "Improvements in LOVIISA EFW and ECC Safety Functions", Juhani Hyvärinen, Radiation and Nuclear Safety Authority (STUK)).

The Loviisa example demonstrates that considerable improvements can be made to the normal sump screen solution to avoid clogging in any case by providing a backflushing solution of the sump screens. There is no indication in the MP documents, that this essential provision for cleaning the screens in case of high pressure drop due to blocking insulation is foreseen.

Conclusions and Recommendations by IRR

The proposed and indicated measures for avoiding ECCS sump screen clogging concerning the analysis of insulation material behaviour under LOCA conditions and the implementation of a selected technical solution to ensure residual heat removal under LOCA appear to be appropriate but not sufficient.

A reliable technical solution for backflushing the sump screens should be foreseen.

The pressure drop across the sump screens should be automatically measured and taken as input for an automatic start of backflushing the sump screens if necessary.

3.6.2.13 Main Steam Safety Valves (MSSV) and Atmospheric Dump Valves (BRU-A) Qualification for Two-Phase and Water Flow

The MSSVs **may** be replaced before startup. The BRU-A valves will **not** be environmentally qualified for water relief and two-phase flow before startup. Since it is the BRU-A valves that are the first to be challenged by an overpressure event (this is by design, to reduce the rate of challenge to the MSSVs, which perform a safety function), it is not at all clear that the likelihood of a stuck-open secondary valve has been affected significantly by the modernization programme at the time of startup. The importance of this issue is emphasized by the discussion in Section 4.2.12, above.

3.6.2.14 SG Feedwater Capacity

The existing design capacity of the SG emergency feedwater system will remain unchanged at startup. Procedures may be defined to enhance this capacity, but the procedures will be event-oriented, not symptom-oriented and thus will suffer from an unnecessarily high operator error rate. Use of the EFW system may still continued to be limited to 8-10 hours. This could be a limiting factor for external events which result in long-term loss of offsite power.

3.6.2.15 ECCS Sump Capacity

The ECCS sump capacity of the VVER-1000/320 design is limited compared with western PWRs (630 m^3 vs. a range of 950-1900 m^3 for western PWRs). This results, in combination with the lack of use of symptom-oriented EOPs, in a higher human error rate for events in which primary coolant is lost outside containment, because this limits the amount of time the operators have in which to depressurize the reactor coolant system and stop the loss of primary coolant before the ECCS sump inventory is exhausted. See the discussion in Section 3.6.2.12 above which places this issue into further perspective.

3.6.2.16 Reactor Vessel Head Leak Monitoring System

The IAEA Category III issue of a reactor vessel heat leak monitoring system is not explicitly addressed in the modernization programme. Rather, the modernization programme references a generic primary system leak detection system, without specifically mentioning the special issue of reactor vessel head leaks and their implications as possible contributors to control rod ejection accidents.

3.6.2.17 Instrumentation & Control Replacement

From **operating experience** it is known that the I&C equipment of WWER-1000 units is based on a technology that is known to present reliability problems.

The issue I&C reliability has **IAEA rank II** which was justified by IAEA (IAEA 1996a):

> *The issue affects the design provisions and may have a direct or indirect impact on deviations from normal operation (level I of protection), on bringing back the installation to normal operating conditions (level 2 of protection) and on the capability of engineered design features to prevent the evolution of deviations into more severe accidents (level 3 of protection). One or more safety functions can be impaired due to the insufficient reliability of the I&C system. The issue may cause initiating events during normal operation and can aggravate the abnormal conditions.*

Conclusions and Recommendations by IRR

Replacement of large parts of the I&C because of damage and vintage design is included in the repair and replacement program of K2/R4 (see Table 3.4). In contrast to other completion projects, e.g. the Temelin, the I&C replacement planned for K2/R4 has not been specified.

Temelin NPP has demonstrated that the substitution of the I&C system could have a great impact on the modernization project. In the Temelin project the original instrumentation and control system is exchanged by a Westinghouse distributed digital system. The merging of two technologies at an advanced completion level is one of the major technical problems for time delays and cost overruns in the Temelin completion project.

In the latest report on the status of the Temelin project the responsible minister K. Kühnl concluded (Kühnl 1998):

Each additional equipment or more modern equipment (with other parameters) must be incorporated in the power plant design, including a new design of instrumentation and control system provided by Westinghouse. A major part of the delay in the date of construction completion is therefore attributed to design work resulting from the changes described above. One of the direct causes of delay of final date is the delay of cabling (design and physical cable pulling) which is impacted by a majority of changes and where it is most complicated to incorporate the changes. Delay in cable pulling (stop of pulling for the reason of completion of the design change or its slow progress for the reason of complicated design) is directly reflected in the date of power plant completion (cabling is on so-called critical path of the construction schedule).

3.6.2.18 Emergency Battery Discharge Time

The modernization programme includes upgrade of the battery capacity from 30 minutes to 60 minutes. However, there is no demonstration that this is sufficient, and the IAEA report referenced by the measure is inconsistent, recommending minimum discharge times ranging from one to three hours. Since the batteries will be replaced in any case, it should be ascertained whether 60 minutes is adequate or not.

3.6.2.19 Replacement of 6 kV Switchgear

Although acknowledged historically as a reliability issue, replacement of the 6 kV switchgear is deferred until after startup. The RiskAudit report on Rivne Unit 3 noted that there have been, on average, 2 failures per year of these breakers (RiskAudit 1994: 5/28). Unreliability of 6 kV switchgear results in common-mode unavailability of an entire train of all safety systems, and is therefore an important safety issue for all initiating events.

3.6.2.20 Containment Structure and Containment Bypass Accidents

There are three containment bypass mechanisms identified by IAEA in Safety Issue Cont1 (Category II Issue). Only two of these three mechanisms are addressed by the modernization programme before startup. For the remaining issue, the compensatory measures do not affect the likelihood of the bypass mechanism; rather, the measures only enhance the detection of the failure **after** it has occurred. The other containment bypass mechanisms are not adequately addressed by the modernization programme, or are not addressed at all. See Section 3.6.2.12 for additional discussion of two such mechanisms (SG collector failure and SG tube rupture).

In addition, the modernization programme lacks a measure to analyze the ultimate strength of the containment. Such an analysis is needed to support proper analyses of BDBAs as well as to structure an accident management programme.

3.6.2.21 Fire Prevention

Considerable actions are planned to be implemented before startup in the fire prevention/ protection area. However, it is well recognized that the adequacy of fire prevention and fire protection measures is dependent on fine details of analysis and implementation. Few details are provided in the modernization programme. Indeed, a methodology for performing the fire hazards analysis is not even identified, despite the availability of IAEA methods guides in this regard.

Implementation of prescriptive regulatory requirements for fire protection in the US has nonetheless led to a wide range of resulting CDF results from fire PSAs. It is clear that a fire PSA is needed to ensure that fire prevention and fire protection measures are adequate.

3.6.2.22 Pipeline Break Impacts Inside the Reactor Building andTurbine Building

A number of measures are included in the modernization programme to address these issues. However, the key measure (implementation of leak-before-break) for primary piping is not going to be implemented at startup. Further, the measures adopted for secondary piping are of limited use without implementation of controls on erosion-corrosion problems, and the modernization programme does not even mention erosion-corrosion despite the fact that it has been identified as a problem in Rivne Unit 3 (RiskAudit 1994: 4/28) and other VVER-1000 units. In addition, it is not clear how the modernization programme has addressed the issue of large diameter ESW pipeline failure in the reactor building (identified as a concern by the MOHT report).

3.6.2.23 Extreme Weather Conditions, Low Temperatures

In accordance with NUSS (IAEA 1988b), proposed sites are required to be adequately investigated with respect to all the characteristics that could affect safety in relation to design basis natural events.

This issue has IAEA rank I (IAEA 1996a). The lack of an adequate investigation of the nuclear power plant site with WWER-1000 reactors with respect to natural events is a **deviation from current international practice**.

According to the IAEA Safety Series on treatment of external hazards in PSA (IAEA 1995c) international experience indicates that external hazards can significantly contribute to plant risk. Therefore such hazards should be included in a PSA.

This issue is partly addressed in the Ukrainian Modernization Programme (KIEP 1996). (Measures: 18212 will be implemented before and 18221 partly after start-up)

Conclusions and Recommendations by IRR

This issue might gain significant safety relevance, e.g. for emergency cooling water tanks exposed to low external temperatures. The related heating system must be able to cope with all low temperature situations.

The sites of K2/R4 have to be assessed with respect to the natural phenomena prior to start-up and the site specific aspects should be included in the MP (IAEA 1995a and IAEA 1997).

Performing a PSA including external weather hazards is highly recommended. Protective measures against hazards of tornadoes have to be implemented.

3.6.2.24 Man-Induced External Hazards

The modernization programme limits its consideration of man-induced external hazards to aircraft crash onto the reactor building, the impact of shock loads on plant structures from explosions, and the possible intake of toxic gases and their impact on MCR/ECR personnel. In contrast, the IAEA recommended a global analysis of man-induced external hazards using screening techniques to identify those hazards requiring more detailed analysis. In addition, limiting the aircraft crash analysis to the reactor building has been shown to be a weakness since impacts into the ESW building could result in multi-unit concurrent core damage accidents. Aircraft impacts into the switchyard or turbine building could also be important to risk.

3.6.2.25 Seismicity/Geology

Based on the disparity between the design basis earthquake level (0.05g PGA) and the assessed seismic hazard for K2 (0.17g PGA) at the same return frequency, IRR concludes that a more substantial effort is required. The Vrancea region dominates the seismic hazard at return periods below 1,000 years, it seems likely that the Rivne site will experience a similar seismic hazard level. There are already **known** seismic vulnerabilities in that the ESW system is not seismically qualified, nor are the ventilation system or the fire protection water system. The ESW system vulnerability is particular serious since failure of the ESW system following an earthquake would result in multi-unit concurrent core melt accidents (four units simultaneously at Rivne, two units simultaneously at Khmelnitsky).

Additional combined neotectonic and seismicity studies as well as geological investigations of ongoing karst activity are recommended. Seismic PSA or seismic margin analysis (with an RLE of not less than 0.3g) should be performed for both K2 and R4, and any seismic vulnerabilities should be rectified before startup. These analyses should be performed as soon as possible in order that the costs associated with the upgrades are known.

3.6.2.26 Probabilistic Safety Assessment (PSA)

A full-scope PSA has already been performed for the Temelin plant. PSAs of varying depth have also been performed for Kozloduy Units 5 and 6, the Balakovo plant, and the Novovoronezh plant. Thus, there is no inherent reason why a PSA could not be performed now for K2 and R4. The modernization programme includes a PSA but only after startup. This is inconsistent with the basic and original purpose of PSA which is as a design aid. PSAs are recognized by IAEA and others as a basic ingredient in formulating a safety improvement programme that adequately addresses risk. The current modernization programme for K2 and R4 is nearly completely deterministic.

3.6.2.27 Rapid Reactivity Increase/Control Rod Ejection

According to IAEA (IAEA 1996a) accident analyses are needed in the licensing of plants to demonstrate meeting of the minimum requirements of safety systems. However, these analyses have not been reviewed with respect to their completeness. The issue was identified from **safety reviews**.

The lack of a complete set of full scope safety analyses affects level 3 of protection and makes it impossible to understand if the plant can safely cope with accidents with different probabilities of occurrence. Consequently, all safety functions can be impaired and may not be performed as demanded.

The IAEA (IAEA 1995c) has developed guidelines for accident analysis of WWER power plants. Control rod ejection accidents are included in these guidelines.

This issue is **not addressed** in the MP.

Conclusions and Recommendations by IRR

A complete set of control rod ejection analyses must be accomplished before start-up of K2 and R4. Additionally it is recommended to consider control rod ejection initiating events also in the PSA. This is not included in the Modernization Program (KIEP 1996)

In the Modernization Programme plans to develop fuel of new design with burnable neutron absorbers (measure 11212) after start-up are mentioned but not specified. New patterns for the initial fuel loading of K2/R4 are only partly addressed in the Modernization Program. Finally, the process of commissioning these units will follow the traditional approach along with the above-mentioned safety deficiencies. Regarding the potential severity of a control rod ejection accident it is highly recommended to implement both measures **before start-up**.

3.6.2.28 Spent Fuel Storage

Spent fuel storage risks are not addressed in the EIAs. In order to make risk comparisons between K2/R4 and Chernobyl, spent fuel storage risks must be known.

3.6.2.29 Operating Procedure Issues

One of the key post-**TMI-2 accident requirements issued by the USNRC in NUREG-0737** was the requirement for plant operators to implement **symptom-oriented** emergency operating procedures (EOPs). Prior to the accident, EOPs were event-oriented, and such procedures were recognized in the aftermath of the accident as more prone to error than symptom-oriented procedures. This insight was reinforced by the Chernobyl accident experience. The generic weakness of event-oriented EOPs is that the operating personnel first have to identify (diagnose) the event, then select the correct procedure to control the event, and finally correctly implement the procedure. In any event, even if the event were correctly diagnosed and the right procedure were selected for implementation, the event-oriented EOPs were "one way" procedures oriented toward success, which fails to take into account unforeseen events and mistakes by operators along the way (IAEA 1996a: 126). Therefore the IAEA has recommended implementation of symptom-oriented EOPs for VVER-1000/320 NPPs (IAEA 1996a: 126).

Symptom-based EOPs have been developed for the Balakovo and Zaporozhe VVER-1000/320 units (IAEA 1996a: 126), as well as for the Temelin units. Thus, there is no reason why these existing symptom-oriented EOPs could not be used as the basis for creating plant-specific EOPs for K2 and R4. However, the modernization programme for K2 and R4 envisions use of event-oriented EOPs at startup. Symptom-oriented EOPs will not be developed and implemented until **after startup** as part of upgrade programme Measure 30211 (KIEP 1996: 278-279/316).

3.6.2.30 Logistics and Infrastructure

A number of issues concern the infrastructural and logistic preconditions of NPPs. Besides technical, these issues include economic, political and societal aspects and, if at all, usually cannot be resolved by a set of relatively simple measures. They **could have a considerable influence on the nuclear safety**.

The importance of infrastructural and logistic preconditions for the safety of NPPs operation was stressed by the Kemeny Commission as a conclusion of their investigation of the TMI accident (Kemeny 1979) and the Steinberg (Shteynberg) Commission, in re-investigating the causes of the Chernobyl accident before the political change in the former Soviet Union. An interdisciplinary approach in this area is mandatory.

> (Kemeny 1979): *"When we say that the basic problems are people-related, we do not mean to limit this term to shortcomings of individual human beings-although those do exist. We mean more generally that the revealed problems with the "system" that manufactures, operates, and regulates nuclear power plants. There are structural problems in the various organizations, there are deficiencies in various processes, and there is a lack of communication among key individuals and groups."*

> (Shteynberg 1991): *"The system of legal, economic and sociopolitical correlations that existed prior to the accident and still exists in the field of nuclear power has no legal basis, and did not and does not meet the requirements of ensuring the safe utilization of nuclear power in the USSR."*

This issue is **not addressed** in the MP.

Conclusions by IRR

Compared to the IRR report (IRR 1997) the situation in the Ukrainian power generating industry has not improved. Therefore it is questionable that two new nuclear units could be completed in an environment where great problems exist to keep the existing power generating units in operation. Furthermore it is questionable that a sound repair and replacement program and a safety upgrading program to reach an "acceptable" level of safety could be implemented. One possible scenario could be that measures in the MP which are planned to be implemented after start-up in order will be shifted into a better future.

Additionally after the disintegration of the USSR the Ukrainian nuclear industry was confronted with a drastic loss of nuclear infrastructure (main designer and construction organizations, scientific institutions, facilities necessary for the fuel cycle, independent nuclear regulatory body, qualified personnel, etc.). For a detailed evaluation see (IRR 1997). Concerning the crucial situation it seems questionable that large improvements in creating a nuclear infrastructure necessary for the operation of NPPs have been achieved.

In summary, due to the lack of financial and industrial resources, the Ukrainian supporting infrastructure is not favorable for the further development of nuclear power.

3.6.2.31 Additional Safety Issues

Two additional issues were identified: loss of heat sink and application of leak-before-break (LBB) to secondary piping. The modernization programme does not adequately respond to the loss of heat sink issue. Failure of ESW (ultimate heat sink) would result in multi-unit concurrent severe accidents with potentially severe offsite consequences. Depending on the configuration of the spent fuel pool at the time, the spent fuel pools could also undergo a severe accident which would result in considerable additional hydrogen and fission products being released into the containment.

LBB is not applicable to secondary piping because this piping is susceptible to cracking due to corrosion. Lack of an erosion-corrosion programme for K2 and R4 only serves to re-emphasize the inapplicability of LBB to secondary piping.

3.6.3 Most Significant Issues Not Adequately Addressed in the K2/R4 Modernization and Upgrade Programme

Based on the analysis and discussion presented above, IRR has identified the following issues as the most safety significant issues which are not adequately addressed in the K2/R4 modernization and upgrade programme:

- **Steam generator collector failure** – important due to high CDF contribution, high containment bypass frequency (large release frequency), continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), failure to implement an automatic safety system response to the initiating event, failure to replace the steam generators, lack of symptom-oriented EOPs, limited ECCS water inventory, and lack of adequate compensatory measures at the time of startup.
- **Steam generator tube rupture** – important due to high CDF contribution, high containment bypass frequency (large release frequency), inadequate NDE, continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), lack of symptom-oriented EOPs, limited ECCS water inventory, and lack of adequate compensatory measures at the time of startup.
- **Qualification of BRU-A for water and two-phase flow** – important due to containment bypass implications in the event of a steam generator collector failure or a steam generator tube rupture.
- **Probabilistic safety assessment (PSA)** – important because the modernization programme is almost completely deterministic, because the upgrade programme ignores a number of

recommendations from MOHT based on PSA results for VVER-1000/320 reactors, and because the PSAs for K2 and R4 are not scheduled to be completed until **after** startup.

- **Emergency operating procedures (EOPs)** – important because the existing procedures and the ones which will be in place at startup are event-oriented procedures instead of symptom-oriented EOPs as recommended following the TMI-2 and Chernobyl accidents, because of the high CDF contribution of human errors with event-oriented EOPs, and because of the importance of human actions in mitigating containment bypass accidents which dominate CDF for the VVER-1000/320 design.
- **Fire prevention/fire protection** – important due to lack of previous fire hazards analysis, the lack of fire PSA for VVER-1000/320 (except Temelin, for which the results are not publicly available), and the lack of coverage of fire in K2/R4 PSA until **after** startup.
- **Seismicity** – important due to low PGA level for design (0.05g) compared with seismic hazard at 10,000 year return interval (0.17g), lack of seismic qualification of ESW system (multi-unit concurrent accident risk), lack of seismic qualification of ventilation and fire protection water pumps, and lack of seismic PSA/seismic margin analysis until **after** startup.
- **Geology** – Monitoring of karst phenomena and karst water, consequences of possible accidents for groundwater safety areas (emergency preparedness)and impact of karst activity on pile foundations of R4 **not addressed** in the MP for K2 and R4. Necessary paleoseismic and seismotectonic studies **not included** in the MP for K2/R4. Foundation deformations (settling and tilting of K2 reactor compartment) resulting from uneven settling of strongly weathered Proterozoic rock are not mentioned in the EIA. Foundations on strongly weathered "semi-rocky" bedrock can be a safety issue in combination with seismicity – even more if the thixotropic soft/fluid plastic chalk layer is involved.
- **Loss of heat sink** – important due to multi-unit concurrent accident potential, dependent failure potential, implications for spent fuel pool severe accidents, possible high CDF contribution from loss of ESW, and lack of improvements in the modernization programme.
- **Conservation/requalification and qualification of equipment** – the quality of the requalification program and equipment qualification addressed in the MP is questionable because of the poor quality of conservation measures during the construction halt and the non-availability of large parts of the manufacturing and construction documentation. Furthermore a complete programme to qualify equipment under extreme environmental conditions and seismicity is still pending for K2/R4. Already existing parts of this programme are not planned to be implemented before start-up. This is a deviation from international acceptable practice. For example these reactors could not be licensed according to US Nuclear Regulatory Commission (NRC)-regulations because equipment qualification is a precondition for licensing.
- **TMI requirements** – Their implementation is a precondition for obtaining an operating license for US plants. Not all TMI issues are addressed in the MP. Some are planned to be implemented **after** start-up.
- **Reactor Core** – it is unclear if the measures to solve control rod jamming addressed in the MP deal with the root causes of this issue. Further studies and operational feedback is necessary. Automatic control of Xenon oscillations and power distribution is not specified in the MP and will be implemented **after** start-up.
- **DBA and BDBA** – a more comprehensive spectrum of accidents (including reactivity accidents) should be analyzed than proposed in the MP before start-up.
- **ECCS sump screen clogging** – proposed measures for avoiding concerning the analysis of insulation material behaviour under LOCA conditions and the implementation of a selected technical solution to ensure residual heat removal under LOCA appear to be appropriate but **not sufficient**.
- **Logistic and infrastructural preconditions** – due to the lack of financial and industrial resources, the Ukrainian supporting infrastructure is not favorable for the further development of nuclear power. This issue is **not addressed** in the MP.

3.7 Conclusions and Recommendations

3.7.1 Summary Perspective

3.7.1.1 Project Sponsor and EIA Claims for K2/R4 Safety and Risk Levels

The project sponsor (Energoatom) and the EIAs (prepared by Mouchel) make a number of claims for K2/R4 safety and risk levels. These claims are:

1. The project will complete Khmelnitsky Unit 2 and Rivne Unit 4 to an "*internationally acceptable safety level*" (Energoatom).
2. After completion the two units will have a safety level similar to that of similarly aged but recently re-licensed, western plants (Energoatom).
3. The project would allow the safety of the plant to be comparable to that achieved in the European Union for NPPs recently re-approved by national safety authorities (Mouchel).
4. The routine discharges of radioactivity from two RBMK units operating at Chernobyl would significantly exceed those arising from operation of two VVER-1000 units (Mouchel).
5. The RBMK reactor is inherently less safe than is the VVER reactor. The no-change option therefore would result in an increased risk of a catastrophic accident leading to widespread contamination (Mouchel).
6. The core damage frequency for the upgraded K2/R4 units will be close to the value for recently re-approved PWRs and significantly lower than the corresponding value for the RBMK (Mouchel).
7. PWRs and VVERs have a strong leaktight containment, are stable reactors, and physically cannot generate an explosive Chernobyl-type accident. Therefore, the overall safety level of an RBMK can never be equivalent to that of a VVER 1000 (Mouchel).
8. Continued operation of Chernobyl Unit 3 has many implications for the final entombment of Unit 4 (Mouchel).
9. The EIAs have analyzed the "*most representative*" beyond design basis accident (BDBA), and the lower intervention level for implementation of counter-measures would not be reached at the boundary of the 3 km zone using worst-case dispersion conditions (Mouchel).

Of these claims, only Number 7 is partially substantiated in the documentation reviewed. Claim Number 7 is substantiated to the extent that the VVER-1000/320 design does in fact include a prestressed concrete containment with a steel liner which is designed to withstand the pressure and temperature loading of the design basis accident. However, as to the portion of the claim that the overall level of safety of the RBMK can never approach that of the VVER-1000/320, this is not substantiated. The reason for this is the high percentage of VVER-1000/320 core damage frequency which is comprised of scenarios which **bypass** the containment, thus negating its value as a risk reduction mechanism. **None of the other claims above are substantiated in the EIAs.**

- **Risk Comparison Between K2/R4 and Chernobyl 1/3**: The EIAs claim that the level of risk posed by completion of the upgraded K2/R4 units will be lower than that posed by upgrade Chernobyl units. However, no risk estimates are presented for any of these units (not for K2/R4, or for Chernobyl Units 1, 2, or 3), nor are any risk estimates are cited by the EIAs. There is no listing, even in summary fashion, of the specific upgrades which were considered in drawing the conclusion that K2/R4 poses less risk than continued operation of the Chernobyl units. In addition, the EIAs fail to acknowledge, in making risk comparisons, that there are no public residents within 30 km of Chernobyl, while within 30 km of K2 and R4 there are 250,700 and 134,680 persons, respectively.
- **Risk Comparison Between K2/R4 and Recently Re-Licensed EU NPPs**: The EIAs claim that the upgraded K2/R4 NPPs would have a safety level comparable to recently re-licensed

EU NPPs of similar vintage. However, the EIAs do not define "*similar vintage*", nor do the EIAs identify any particular EU nuclear regulatory authority nor any particular NPP in making this claim. The claim is unsubstantiated and is not based on a detailed comparison of either risk profiles or of conformance with any specific nuclear regulatory authority's safety criteria (including the IAEA NUSS standards).

- **EIA Evaluation of Beyond Design Basis Accidents (BDBAs)**: The EIAs contain an analysis of what is claimed to be "*the most representative*" beyond design basis accident (BDBA) for the K2/R4 units. In fact, the accident actually evaluated is one that the IAEA has identified as one which should be considered to be within the design basis, not the design basis, not beyond it. The selected accident does not result in severe core damage, but as analyzed is fully mitigated by installed safety systems and timely operator action. A whole host of other BDBAs are known for VVER-1000/320 units, many of which do result in severe core damage, and some of which (including those identified by PSA studies as being the most likely) result in containment bypass as well.
- **Chernobyl Replacement Options Other than K2/R4 Completion**: The EIAs do not evaluate any other Chernobyl replacement options other than K2/R4 completion. There are a number of readily identifiable and reasonable options which should have been addressed in the EIAs, including use of the same amount of money required for K2/R4 completion to modernize and upgrade Ukraine's existing eleven operating VVER-1000/320 units. Such an option would make up for the lost capacity arising from final closure of Chernobyl, would significantly reduce overall risks of a catastrophic nuclear accident in Ukraine, and would result in more of the money being expended in Ukraine and Russia instead of being spent outside these countries in western Europe and the United States. The latter benefit would assist the economies of Ukraine and Russia by providing an infusion of hard currency to an extent not envisioned in the proposal to complete K2 and R4.
- In fact, the RBMK-1000 design is **not *inherently* unstable**. The instability which led in part to the Chernobyl Unit 4 is **not** inherent, and has indeed been corrected in all operating RBMK units as a result of the Chernobyl accident. A number of international organizations [47] have reviewed the upgraded RBMK design and have concluded that a repeat of the Chernobyl Unit 4 accident is not credible. (IRR agrees that the VVER-1000/320 design is inherently stable with respect to reactivity-insertion accidents.)
- It is true that the RBMK-1000 design **lacks a containment**. Chernobyl Units 1 and 2 lack a pressure suppression system and has only limited confinement capabilities. Chernobyl Unit 3 has a partial confinement and a pressure suppression system, but this does not constitute a containment in the conventional sense. The VVER-1000/320 design includes a prestressed concrete containment. However, this does **not** mean that the risks posed by operating a VVER-1000/320 reactor are **necessarily** lower than operating an RBMK-1000 reactor. What matters is the **frequency of a large radioactivity release**.

  The EIAs neither contain nor reference analyses of the frequency of large release accidents for VVER-1000/320 or RBMK-1000 units generally or for K2/R4 or Chernobyl specifically. For an RBMK-1000 unit, the frequency of a large release is expected to be essentially the core damage frequency (CDF) – that is, most if not all core damage accidents will progress to a large release due to the absence of containment. For the VVER-1000/320 design the consideration of nor only the core damage frequency but also the relative contribution of containment bypass sequences to the core damage frequency. While PSA results are not available for K2 or R4, Level 1 PSA results for the Temelin units (which will be more significantly upgraded than either Khmelnitsky Unit 2 or Rivne Unit 4) indicate that the fraction contribution of containment bypass events is very high and that the overall CDF is near the INSAG safety target. The net result **may** be that, for internal events, the frequency of a large release of radioactivity is actually be higher for a VVER-1000/320 than for an RBMK reactor (at least for a second-generation RBMK; it is not clear that this is true for a first-

[47] Specifically, GRS (Germany), IPSN (France), RRC Kurchatov Institute (Russia), AEA Technologies (United Kingdom), and the Nuclear Energy Institute (United States).

generation RBMK, such as Chernobyl Units 1 and 2). No inferences are yet possible for external events due to the lack of publicly-available PSA results for external events for either VVER-1000/320 or RBMK reactors. Although these results are **not** plant-specific to K2, R4, or Chernobyl, they are suggestive of the proposition that it is not **obviously** the case that an upgraded VVER-1000/320 reactor **necessarily** poses lower risk than an upgraded RBMK-1000 reactor. **Unless it can be demonstrated to be the case that K2 and R4 pose a lower risk of a catastrophic accidents than one or more of the Chernobyl reactors, the project proponents have failed to prove a risk advantage for completion of K2 and R4.**

- **Plant-specific PSAs** are planned for Khmelnitsky Unit 2 and Rivne Unit 4, but the studies will not be completed until **after commissioning**. Although the MOHT organizations have identified PSA-related and severe accident-related upgrades for the VVER-1000/320 units, these upgrades are **not** included in the Khmelnitsky Unit 2 or Rivne Unit 4 modernization programmes. Thus, the upgrade programmes are essentially entirely deterministic in nature, and any actual risk reduction which occurs will be a matter of serendipity. It cannot be concluded that the risk-dominant sequences have been addressed by the Khmelnitsky Unit 2 and Rivne Unit 4 modernization programmes; rather, to the contrary, it must be conceded by the project proponents that there is a possibility that risk-dominant sequences have been missed. The K2 and R4 modernization programmes, as currently proposed, are not so comprehensive and in-depth that it can be claimed that risk has been reduced across the board, regardless of the accident initiator being considered. Certainly, it **cannot** be claimed that the level of safety of Khmelnitsky Unit 2 and Rivne Unit 4 is the same as western European PWRs – not even the same as older western European PWR units, which have undergone continuous safety upgrade programmes, complete with PSA analyses and risk optimization upgrades to emergency procedures and plant hardware based on the PSA study outcomes.

3.7.1.2 Most Significant Safety Issues Not Adequately Addressed in the K2/R4 Modernization and Upgrade Program

Based on review of additional project specific documentation (most of them not made available in the frame of the PPP) analysis and discussion by the Institute of Risk Research (IRR), IRR has identified the following issues as the most safety significant issues which are not adequately addressed in the K2/R4 modernization and upgrade program:

- **Steam generator collector failure** – important due to high Core Damage Frequency (CDF) contribution, high containment bypass frequency (large release frequency), continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), failure to implement an automatic safety system response to the initiating event, failure to replace the steam generators, lack of symptom-oriented Emergency Operating Procedures (EOPs), limited Emergency Core Cooling System (ECCS) water inventory, and lack of adequate compensatory measures at the time of startup.
- **Steam generator tube rupture** – important due to high CDF contribution, high containment bypass frequency (large release frequency), inadequate NDE, continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), lack of symptom-oriented EOPs, limited ECCS water inventory, and lack of adequate compensatory measures at the time of startup.
- **Qualification of atmospheric dump valve (BRU-A) for water and two-phase flow** – important due to containment bypass implications in the event of a steam generator collector failure or a steam generator tube rupture.
- **Probabilistic safety assessment (PSA)** – important because the modernization program is almost completely deterministic, because the upgrade program ignores a number of

recommendations from MOHT[48] based on PSA results for VVER-1000/320 reactors, and because the PSAs for K2 and R4 are not scheduled to be completed until after startup.

- **Emergency operating procedures (EOPs)** – important because the existing procedures and the ones which will be in place at startup are event-oriented procedures instead of symptom-oriented EOPs as recommended following the TMI-2 and Chernobyl accidents, because of the high CDF contribution of human errors with event-oriented EOPs, and because of the importance of human actions in mitigating containment bypass accidents which dominate CDF for the VVER-1000/320 design.
- **Fire prevention/fire protection –** important due to lack of previous fire hazards analysis, the lack of fire PSA for VVER-1000/320 (except Temelin, for which the results are not publicly available), and the lack of coverage of fire in K2/R4 PSA until after startup.
- **Seismicity** – important due to low Peak Ground Acceleration (PGA) level for design (0.05g) compared with seismic hazard at 10,000 year return interval (0.17g), lack of seismic qualification of Essential Service Water (ESW) system (multi-unit concurrent accident risk), lack of seismic qualification of ventilation and fire protection water pumps, and lack of seismic PSA/seismic margin analysis until after startup.
- **Geology** – Monitoring of karst phenomena and karst water, consequences of possible accidents for groundwater safety areas (emergency preparedness) and impact of karst activity on pile foundations of R4 not addressed in the MP for K2 and R4. Necessary paleoseismic and seismotectonic studies not included in the MP for K2/R4.
- **Loss of heat sink** – important due to multi-unit concurrent accident potential, dependent failure potential, implications for spent fuel pool severe accidents, possible high CDF contribution from loss of ESW, and lack of improvements in the modernization program.
- **Conservation/requalification and qualification of equipment** – the quality of the requalification program and equipment qualification addressed in the MP is questionable because of the poor quality of conservation measures during the construction halt and the non-availability of large parts of the manufacturing and construction documentation. Furthermore a complete program to qualify equipment under extreme environmental conditions and seismicity is still pending is not planned to be implemented before start-up. This is a deviation from international acceptable practice (e.g. equipment qualification is a precondition for licensing in US plants).
- **Three Mile Island (TMI) requirements** – Their implementation is a precondition for obtaining an operating license for US plants. Not all TMI issues are addressed in the MP. Some are planned to be implemented after start-up.
- **Reactor Core** – it is unclear if the measures to solve control rod jamming addressed in the MP deal with the root causes of this issue. Further studies and operational feedback is necessary. Automatic control of Xenon oscillations and power distribution is not specified in the MP and will be implemented after start-up.
- **Design Base Accident (DBA) and Beyond Design Base Accident (BDBA)** – a more comprehensive spectrum of accidents (including reactivity accidents) should be analyzed than proposed in the MP before start-up.
- **ECCS sump screen clogging** – proposed measures for avoiding concerning the analysis of insulation material behavior under Loss Of Coolant Accident (LOCA) conditions and the implementation of a selected technical solution to ensure residual heat removal under LOCA appear to be appropriate but not sufficient.
- **Logistic and infrastructural preconditions** – due to the lack of financial and industrial resources, the Ukrainian supporting infrastructure is not favorable for the further development of nuclear power. This issue is not addressed in the MP.

[48] MOHT is an association of the following organizations: Atomenergoproject, OKB Gidopress, Kurchatov Institute, VNIIAES, Zarubejatomenergostroy, Rosenergoatom, et al.

- **Presence of fail-open pneumatic containment isolation valves** – the designs use fail-open pneumatic valves as containment isolation valves, which is at variance with western safety criteria and which result in a greatly increased risk of an interfacing LOCA given failure of the pneumatic system.
- **Additional Not Adequately Addressed Safety Issues:**
  - Reactor coolant pump (RCP) seal failures
  - Emergency battery discharge time
  - Replacement of 6 kV switchgear
  - Reactor pressure vessel embrittlement
  - Reactor vessel head leak monitoring system
  - Replacement of I&C
  - Containment structure and containment bypass accidents
  - Containment ultimate capacity
  - Man-induced hazards
  - Extreme weather conditions
  - Spent fuel storage
  - Leak before break application to secondary piping
  - Pipeline break impact inside reactor building

3.7.2 Conclusions

Based on the above analysis and evaluation, IRR has reached the following conclusions:

- There is no basis for Mouchel to assert **comparability of the K2/R4 safety level with western NPPs**. No PSA is available for K2/R4 for comparison purposes. No comparison of K2/R4 with any specific set of western nuclear safety standards is presented. No comparison of K2/R4 with the IAEA NUSS criteria is presented. In contrast, there is significant evidence (in the form of PSA results from VVER-1000/320 units) and the lack of treatment of relevant issues in the K2/R4 modernization programme to believe that the CDF for K2/R4 will exceed the INSAG CDF safety target of 10^{-4} per year and that K2/R4 will exceed the INSAG large release frequency target of 10^{-5} per year by more than a factor of ten. These results will occur due to SG collector failure and SG tube rupture **alone**, not accounting for any of the other various sources of severe accident risk. It cannot be concluded, therefore, that the K2/R4 upgraded designs, as set forth by the project proponents, will meet western safety standards (e.g., INSAG-3).
- There is no basis for Mouchel to assert **risk superiority for K2/R4 over Chernobyl Unit 3**. The doses from normal operational releases are **lower** for Chernobyl due to the 30-km exclusion radius. The large release frequency for K2/R4 cannot be shown on the basis of current information to be less than for Chernobyl Unit 3 and may be higher based on the PSA results from Temelin (VVER-1000/320) and Ignalina (RBMK).
- The **beyond-design-basis-accident (BDBA)** contained in the EIAs is flawed and misleading. The selected accident is identified by IAEA as an accident **within** the design basis, not **beyond** it. There are a host of BDBAs which are easily identified for VVER-1000/320 units which result in severe accidents (and some of which result in severe accidents with containment bypass) which were ignored by the EIAs without explanation. There is no basis for concluding that the selected accident is “the most representative” of the BDBAs. It is only representative of a group of accidents which have little public or environmental impact because they are assumed to be fully mitigated by plant safety features and/or operator recovery actions. Moreover, the BDBA analysis lacks an assessment of containment ultimate capacity, which is also needed for the accident management programme.

- The EIAs fail to examine any **project alternatives** except replacement of Chernobyl by K2 and R4.
- From the risk reduction point of view one **readily identifiable and reasonable nuclear option** is to fund a program of PSAs, reliability upgrades, and safety upgrades for the existing eleven operating VVER-1000/320 units in Ukraine. If the 1.725 billion USD **currently** estimated as necessary for completion and commissioning K2 and R4 (the estimate will almost **certainly** rise) could be instead used to improve safety at **already operating** VVER-1000/320 units in Ukraine, this would provide a safety improvement budget of almost 157 million USD per unit.[49] In the process, the safety of eleven NPPs already operating in Ukraine would be improved,[50] instead of just two **additional** units as the K2/R4 proponents would have it. Further, unlike the proponents proposal, which would have 730 million USD spent on activities by western firms, much of the money in the alternative plan discussed here would be spent indigenously in Ukraine or in Russia, which would benefit both economies with an infusion of hard currently and which would achieve greater total safety improvements overall.[51]
- **Significant safety issues** have been identified by IRR which are **not adequately addressed in the K2/R4 Modernization and Upgrade Program**. A comprehensive treatment of these issues is a precondition to reach the minimum acceptable safety level, formulated by IAEA in the INSAG-3 targets for core damage frequency and frequency of large releases, which is one of the EBRD guidelines for funding.

3.7.3 Recommendations

Based on the above analysis and evaluation, IRR makes the following recommendations:

- Before making a decision to fund the completion of K2/R4, EBRD must in compliance with its own guidelines (acceptable safety level) require that Energoatom demonstrates how significant safety issues not adequately addressed in the K2/R4 Modernization and Upgrade Program will be resolved before start-up. Without solving these significant safety issues K2/R4 will not reach a minimum acceptable safety level formulated by IAEA in INSAG 3.
- Completion of K2 and R4 on basis of the present MP should not be funded by EBRD.
- Instead, it is recommended to assess possible alternatives on the basis of safety and risk comparisons.

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[49] MOHT has stated that the most important measures related to VVER-1000/320 availability and safety can be implemented for 238 million USD for two units for plants in operation (MOHT 1996: 30), or 119 million USD per unit. Similarly, a panel of high-level advisors on nuclear safety to the European Union has indicated that the cost of safety upgrading for VVER-1000/320 units is in the range of 100-150 million ECU (Contzen 1998, Section 2.1.3), about 112-168 million USD (at the current exchange rate).

[50] A recent paper co-authored by the former chairman of the Ukraine nuclear regulatory authority states (Budnitz & Steinberg 1998: 5), "*The problem of maintaining and upgrading the safety of the operating NPPs remains the main issue, despite an external appearance of health and the high share of electricity produced by the NPPs. … When NPPs in operation in Ukraine consider various proposals for safety upgrading, they are forced to make guesses, because of a lack of detailed and comprehensive analyses of their safety, which would take into account their operational practices, siting, and human factors. The absence of such analyses was typical in the former USSR, and was one of the underlying causes of the Chernobyl accident.*"

[51] It must be noted that Ukrainian officials have repeated stated that K2 and R4 will be completed irrespective of EBRD action on the loan application. Perhaps the most recent example of such statements was carried in a 23 September 1998 Xinhua report, citing statements by Ukraine President Leonid Kuchma. This story can be found on the WWW at the following URL: http: //news.poweronline.com/wires/19980924-27876445.html. Since K2 and R4 will be completed by Ukraine in any case, EBRD could have a far greater impact on overall nuclear safety in Ukraine by funding upgrades to the existing eleven operating VVER-1000/320 units.

- From the risk reduction point of view one readily identifiable and reasonable nuclear option is to fund a program of PSAs, reliability upgrades, and safety upgrades for the existing eleven operating VVER-1000/320 units in Ukraine. The resulting improved plant capacity factors will allow replacement of capacity which would be lost by closing the remaining Chernobyl reactor (Unit 3) and by foregoing restart of either Unit 1 or 2 at Chernobyl. This programme would have the same cost as the proposed completion of K2 and R4, but would result significant overall reduction in the risk of catastrophic nuclear accidents for Ukraine.

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3.9 Abbreviations

AC .... Alternating Current
AGR.............. Advanced Gas-Cooled Reactor
AFW .... Auxiliary Feedwater
ALS............... Accident Localization System
ANL .... Argonne National Laboratory
ANS .... American Nuclear Society
ATWS........... Anticipated Transients Without Scram

BDBA............ Beyond Design Base Accident
BNL .... Brookhaven National Laboratory
BRU-A .... Russian acronym for atmospheric dump valve
BRU-K .... Russian acronym for turbine bypass valve
BRU-TK .... Russian acronym for valve that opens to permit flow into technological condensers

C................... Celsius
CANDU......... Canadian Deuterium Uranium
CDF .... Core Damage Frequency
Ce................. Cerium
CEQ.............. Council on Environmental Quality
CIEMAT........ CENTRO de Investigaciones Energéticas Medioambientales y Tecnológicas
CPS .... Control and Protection System
CRDM........... Control Rod Drive Mechanism
Cs................. Cesium
CSS .... Containment Spray System

DBA .... Design Base Accident
DC .... Direct Current
DGH .... Distribution Group Header

EAP .... Environmental Action Plan
EBRD............ European Bank for Reconstruction and Development
EBP .... Extrabudgetary Programme
EC .... European Commission
ECCS............ Emergency Core Cooling System
ECR.............. Emergency Control Room
EdF............... Electricitié de France
EFW .... Emergency Feedwater
EIA................ Environmental Impact Assessment
EMI .... Electromagnetic Interference
ENEA............ Ente per le Nuove tecnologie, l'Energia e l'Ambiente
EOP.............. Emergency Operating Procedure
ESW .... Essential Service Water
EU .... European Union
EURATOM.... European Atomic Energy Community

g .... gravity (force of gravity)
G7................. Group of Seven
GAN.............. GOSATOMNADZOR (Russian nuclear regulatory authority)
GRS.............. Gesellschaft für Anlagen- und Reaktorsicherheit mbH (German acronym)

h .... hour
HPI .... High Pressure Injection
HVAC............ Heating, Ventilation, and Air Conditioning

I .... Iodine
I&C .... Instrumentation and Control
IAEA .... International Atomic Energy Agency
IEEE .... Institute of Electrical and Electronics Engineers
INSAG .... International Nuclear Safety Advisory Group
IPE................ Individual Plant Examination
IPEEE........... Individual Plant Examination of External Events
IPSN .... Institut de Protection et de Sûreté Nucléaire
IRR .... Institute of Risk Research
ISI................. Inservice Inspection
ISO .... International Standards Organization

K2................. Khmelnitsky Nuclear Power Plant, Unit 2
KIEP .... Kievenergoproyekt
km .... kilometers
Kr.................. Krypton
kV .... kilovolts

La .... Lanthanide
LBB............... Leak-Before-Break
LLNL............. Lawrence Livermore National Laboratory
LOCA............ Loss of Coolant Accident
LPI .... Low Pressure Injection
LPS............... Low Power and Shutdown
LRF............... Large Release Frequency

m .... meters
MAGNOX...... magnesium clad oxide fuel
(type of gas-cooled reactor and fuel originated in the United Kingdom)
MCR .... Main Control Room
MFW............. Main Feedwater
mm .... millimeters
MOHT........... Russian design and engineering organization
(consisting of consortium of Atomenergoproject, OKB Gidopress, Kurchatov Institute, VNIIAES, Zarubejatomenergostroy, Rosenergoatom, et al.)
MP .... Modernization Program for K2/R4
MSK.............. Medvedev-Sponheurer-Karnik
MSSV .... Main Steam Safety Valve
MWt.............. Megawatts Thermal

NDE.............. Nondestructive Examination
NDT .... Nondestructive Testing
NEI .... Nuclear Energy Institute
NEPA............ National Environmental Policy Act
Ni.................. Nickel
NPP .... Nuclear Power Plant
NUSS............ Nuclear Safety Standards (IAEA)

OECD .... Organization for Economic Cooperation and Development
OPB.............. Obschiye polojeniya obespecheniya bezopasnosti atomnikh stansi (General Safety Regulations)
ORM .... Operating Reactivity Margin

PGA.............. Peak Ground Acceleration
PSA .... Probabilistic Safety Analysis
PSHA............ Probabilistic Seismic Hazard Analysis
PSR .... Periodic Safety Review
PWR .... Pressurized Water Reactor

QA .... Quality Assurance

R4................. Rivne Nuclear Power Plant, Unit 4
RBMK .... Reaktor Bolshoi Moshchhnoski Kanalkni (High-Power Channel Type Reactor)
RCP.............. Reactor Coolant Pump
RDIPE........... Research and Development Institute of Power Engineering
RLE .... Review Level Earthquake
RPV .... Reactor Pressure Vessel
RRC.............. Russian Research Centre (Kurchatov Institute)
Ru................. Ruthenium
RWST........... Refueling Water Storage Tank

s .... second
SFPCS.......... Spent Fuel Pool Cooling System
SG .... Steam Generator
SKI................ Statens Karnkraft Inspection (Swedish Nuclear Power Inspectorate)
SMA.............. Seismic Margin Analysis
Sr.................. Strontium
STUK............ Säteilyturvakeskus (Finnish acronym for Finland's Radiation and Nuclear Safety Authority)

TACIS........... Technical Assistance to the Commonwealth of Independent States
TMI .... Three Mile Island NPP
TOB.............. Teknichiskoye Obosnovaniye Bezopasnostni (Russian for "Technical Substantiation of Safety", equivalent to Safety Analysis Report)
TSC .... Technical Support Centre

US .... United States
USA.............. United States of America
USD.............. United States dollars
USDOE......... United States Department of Energy
USNRC......... United States Nuclear Regulatory Commission

VNIIAES .... Vserossiski Nayuchno-issleyedovatel'ski i proyektni institut energyeticheskikh technologi (All Russia Scientific and Project Institute of Energy Technology)
VVER............ Vodo-Vodyannoy Energeticheskiy Reactor (water-cooled, water-moderated, reactor)

WWER.......... See VVER

Xe................. Xenon

ATTACHMENT 1: Safety Issues Difficult To Solve

Extracted from Riskaudit Report No. 120:
"Evaluation of the Modernisation Programmes Rivne 4 and Khmelnitsky 2 units",
Appendix 1:

Date of implementation (proposal of Riskaudit):
b: before the start-up
a: after the start-up

P: Measures which may lead to problems difficult to be solved during implementation or studies which may lead to further requirements are identified with the symbol (P). It has to be pointed out that this assessment is purely based on personnel judgement and not technically argumented. This personnel judgement (not engaging Riskaudit responsibility) has been added by the authors of the report following a strong requirement of the Austrian member of the Phare-Tacis expert group.

Issue No.	Item in R4/K2 MP	Item	Date of implementation	Riskaudit Recommendation
Core 3 (I)	11211	New control strategy (Xenon oscillation and power distribution)	a	Riskaudit recomm. automatic system when available (P)
Components 13 (S)	12411	Development and implementation of measures to control leakage primary/secondary circuit DN 100	b	pay attention to ISI documents (P)
Components 22 (H)	12211	Providing "rigid embedding" of steam and feedwater pipelines at 28,8m level	b	(P)
Hazards 1	17321	Analysis to determine the extent of pipeline breaks impact inside the reactor building	b	(P)
Hazards 3 (C)	12211	Rigid support of steam and feed water lines	b	(P)
Hazards 21 (S)	18221	Assessment of the risk of "average minimal temperature" and "extreme cold condition"	b(a)	implementation in 2 steps (P)
Sysems 1	11011	Develop materials on equipment qualification	a.b	recomm. not to limit to passport (P) •

ATTACHMENT 2: Comparison of IAEA Safety Issues with Measures in the K2/R4 Modernisation Program

Safety issues for VVER-1000 comparison IAEA (Issue-Book) – Rivne 4 and Khmelnitsky 2 Modernisation Programmes (Extracted from Riskaudit Report No. 120)

IAEA-Code	Issue Name	Measures in MP Rivne 4/Kh 2
	General	
G1	Classification of components	11011
G2	Qualification of equipment	11011
G3	Reliability analysis of safety class 1 and 2	19411
	Core	
RC1	Prevention of inadvertent boron dilution	19121, 13111, 13611
RC2	Control rods insertion reliability/Fuel assembly deformation	11221, 11222, 14281
RC3	Subcriticalily monitoring during reactor shutdown conditions	14211, 14221, 11111
	Low leakage strategy	25111
	Components Integrity	
C11	RPV embrittlement and its monitoring	12311, 12341, 12351, 12352, 12321, 12331, 12361, 25111
C12	Non-destructive testing	12221, 21114, 34111, 12441
C13	Primary pipe whip restraints	12221, 28116, 17321, 12211
C14	Steam-generator collectors integrity	22111, 26131, 26132, 26121, 26132, 12411
C15	Steam generator tube integrity	12441, 33212, 31311
C16	Steam and feedwater piping integrity	12211, 17321, 12221, 26131, 26132, 22412
	Systems	
S1	Primary circuit cold overpressure protection	12111
S2	Mitigation of SG primary collector break	19112, 19311, 30211, 12411, 12421, 22111
S3	MCP coolant pump seal cooling system	21111, 21115, 24411
S4	Pressurizer safety and relief valves qualification for water flow	11011, 13411
S5	ECCS sump screen blocking	13211, 13213
S6	Emergency core cooling system sump-tank and suction lines integrity	16121
S7	ECCS heat exchanger integrity	13611
S8	Power operated valves on the ECCS injection lines	Already solved
S9	Qualification of SG safety and discharge valves for operation with water	11011, 13321, 12411
S10	SG safety valves performance at low pressure	13321 (Need not obvious. Residual risk: available PSA show sequence <10-7/y).

IAEA-Code	Issue Name	Measures in MP Rivne 4/Kh 2
S11	SG level control valves	22441
S12	Emergency feedwater makeup procedures	13311
S13	Cold emergency feedwater supply to SG	24411
S14	Ventilation system of control rooms	measures already implemented
S15	Hydrogen removal system	16131, 16211
	Instrumentation and Control	
IC1	I&C reliability	14321, 14421, 14231, 14421, 23111, 14241, 14331, 15111, 14111
IC2	Safety system actuation system	14231, 14331
IC3	Automatic reactor protection for power distribution and DNB	14211
IC4	Control rooms	14331, 28511, 28411
IC5	Control and monitoring power distributions in load follow mode	14261, 14221, 11211
IC6	Monitoring of mechanical equipment status	28111, 26112, 28113, 12221 28114, 28115, 28116, 28117, 28118, 28119, 28121-28124
IC7	Primary circuit diagnostic system	28111, 28112, 28113, 12221 28114, 28115, 28116, 26132 28117, 28118, 28119, 28121-28124
IC8	Reactor vessel head leak monitoring system	28116
IC9	Accident monitoring instrumentation	14251, 14411, 16211
IC10	Technical support centre	28511
IC11	Water chemistry control and monitoring equipment (primary and secondary)	26131, 26132
	Electrical power supply	
EP1	Off site power supply via stand-by transformers	24311, 24441
EP2	Diesel-generator reliability	15131, 15132, 24411
EP3	Protection signals for emergency Diesel generators	15132
ERA	Power supply for accidents and events control	24411, 15131, 15132
EPS	Emergency battery discharge time	15121
EPS	Ground faults in DC circuits	24122
Containmen	t and building structures	
C1	Containment bypass	16111, 13611
—	Structural aspects	27211, 27212, 27214, 32231
	Internal Hazards	
IH1	Systematic fire safety analysis	17111
IH2	Fire prevention	17121, 17131, 17132, 17151, 29111, 29112
IH3	Fire detection and extinguishing	17141, 29131

IAEA-Code	Issue Name	Measures in MP Rivne 4/Kh 2
IH4	Mitigation of fires effects	29121, 17161, 17112
IH5	Systematic flooding analysis	17211
IH6	Protection against flood for emergency electric power distribution boards	17211
IH7	Protection from dynamic effects due to ruptures of steam and feedwater pipelines	17321, 26211, 26212, 12211, 273111
IH8	Polar crane interlocking	already introduced
IH9	Missiles hazards	17311
	External hazards	
EH1	Seismic design	18111
EH2	Analysis of natural environmental conditions of NPP site	18212, 18221
EH3	Man induced external events	18311, 18321, 18211
	Accident Analysis	
AA1	Scope and methodology of AA	19111, 19112
AA2	QA of plant data used in AA	19112, 19121, 19311
AA3	Computer code and plant model validation	19112
AA4	Availability of accident analyses results for supporting plant operation	30211, 19311, 19112
AA5	Main steamline break analysis	19112, 19111
AA6	Overcooling transients related to pressurised thermal-shocks	19311
AA7	Analysis of SG collector rupture accidents	19112, 19311
AA8	Accidents at low power and shutdown operation conditions	19111, 19311
AA9	Severe accidents	19211, 16131, 16211
AA10	Probabilistic Safety Assessment	19411
AA11	Accidents connected with boron dilution	19121, 13111, 19311
AA12	Accidents connected with drop of spent fuel container	already done
AA13	ATWS-type accidents	19311, 19211
AA14	Total loss of electrical power	19211
AA15	Total loss of heat sink	19211
-–	Loss of feedwater	13411, 19211
	Operational Safety	
OP1	Normal operation procedures	30111
OP2	Emergency operating procedures	30211, 30111
OP3	Limits and conditions	32112, 32111, 30111
M1	Need for safety culture improvements	Included in different measures
M2	Exchange of operational experience	31111
M3	Quality Assurance Program	31211
M4	Management of documentation keeping	30111, 31111

IAEA-Code	Issue Name	Measures in MP Rivne 4/Kh 2
P01	Philosophy of procedures application	30111, 30211
P02	Program for conduct of inspections and tests	32111
P03	Communication system	Already solved
Trail	Program for conduct of inspections and tests	32112
EM 1	Emergency centre	28511
	Personnel protection and radiation safety	
RP1	Radiation protection and monitoring	33211, 33212, 33231

ATTACHMENT 3: Riskaudit Evaluation of the Modernization Program, Appendix 1

Riskaudit Report No. 120:
"Evaluation of the Modernisation Programmes Rivne 4 and Khmelnitsky 2 units", Appendix 1:

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
		Core						
K1	11111	Monitoring of subcriticality during sutdown	x	x			b	
K2	14211	Improvement of neutron flux control measurement	x	x			b	
K3 (I)	11211	New control strategy (Xenon oscillation and power distribu-tion)	x	x			a	Riskaudit recomm. automatic system when available (P)
K4	11212	Study of new control strategy	x	x			a	
K5 (I)	25111	Implementation of refueling strategy	x	x			b	improvement of nuclear codes is necessary
K6	25131	Use of improved engineering margin factors	x	x			a	
K7	11221	Measures to improve drop time of reactor shut down rods and fuel bending		x	x		b	consideration of Gidropress list is recommended
K8	11222	Introduce “heavy weight" control rods		x	x		b	
K9	14281	Replacement of CPS drives	x	x			b	•
K10	20111	Monitoring of fuel rods leak tightness (new system as part of refuelling machine)	x	x			a	
K11	30141	Implementation of methodology to determine the correspondence between damaged fuel operational limit and primary coolant activity by reference isotopes.		x			a	Comments for further steps

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
K12	20121	Develop equipment for completing fuel assembly placing procedures in loss of power		x	x		a	
K13	20131	Sufficient storage capability to ensure emergency reloading			x	x	a	Comments for further steps
K14	33111	To develop equipment for transportation of the spent CPS AR dusters from the reactor and for their burial at the NPP site (withcompacting)	x				a	'.
K15	33112	To develop equipment for transportation of the spent CPS AR clusters from the reactor and for their burial at the NPP site (without compacting)	x				a	
		Components						
C1 (S)	12331	Heating up to 20° C (ECCS active part)			X		b	
C2 (S)	12321	Heating up to 55° C (ECCS passive part)			x		b	
C3	12311	Standard system of reactor vessel of radiation load monitoring	x	x			b	
C4	12361	Verification of residual life of reactor vessel	x				a	
C5	12351	Develop and introduce new programme of surveillance specimens		x			b	change to safely improvement
C6	12341	Replace irradiation specimen from above the core in the water gap		x			a partly b	Before implementation calculation of characteristics of neutron/radiation field
C7	12352	Develop a system for monitoring of radiation load to determine remaining life time	x				a	
C8 (K)	25111	Optimisation of fuel loads (fuel strategy)	x				b	quantify benefit of shielding
C9	12221	Develop and introduce facilities and systems to implement “leak before break" concept	x	x			a	linking to leak detection system necessary
C10 (H)	12211	Rigid support of steam and feedwater lines at the outlet of the reactor building			X	x	b	
C11 (H)	26211	Recalculation of strength of piping essential to safety; implementation of measures	x				b	
C12	26212	Measures to Increase strength of piping if necessary (link to 26211)	x				b	
C13 (S)	12411	Development and implementation of measures to control leakage primary/secondary circuit DN 100	x	x	x	x	b	pay attention to ISI documents (P)
C14 (0)	28111	Implement a full diagnostic system		x			a	
C15 (I)	28113	Implement a vibration diagnostic system		x			b (K2) a (R4)	
C16 (I)	28116	Implement a primary circuit leakage detection system		x			b	specific conditions for leak detection systems

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
C17 (I)	28117	Implement a residual fatigue lifetime diagnostic system		x			b	
C18 (S)	12421	Develop and introduce a SG leakage control system		x	(X)		b	proposal is deleted, but should stay in Mod-ernisation Programmes
C19 (I)	26131	Implementation of secondary coolant parameter automatic control system fornormal conditions		x			b	collector status has to be provided "b"
C20 (S)	22111	Modernisation steam generator blowdown system	x				b	
C21	12441	Develop and implement a criterion for preventive plugging of SG-tubes		x			a	
C22 (H)	12211	Providing "rigid embedding" of steam and feedwater pipelines at 28,8m level					b	(P)
C23 (H)	17321	Analyses to determine extent of pipeline breaks					b	.
C24	34111	In-service Inspection of RPV by TV orultrasonic inspection		x			b	
C25	21114	Procedure for determination of defects in MCP-195M	x	x			a	
C26	12371	Introduction of an equipment set to manufacture and anneal high quality gaskets for the main joint	x				b	
C27	12381	Reconstruction of the upper head sealing assemblies	x				–	issue deleted; already implemented, demon-stration necessary
C28	26132	Implementation of primary coolant parameter automatic control system for normal conditions		x			a	
C29	12391	Strength calculation of the air duct weld ofreactor top head	x				a	
C30	12431	Strength calculation of the reactor vesselhead	x				b	
C31	21211	Strength analysis of make up nozzle thermal shield	x				a	
C32	26111	Chemical water treatment with higher inventory of alkaline metals	x				–	issue deleted; already implemented, demonstration necessary
C33	26121	Programme to determine inventory of alkaline metals	x				–	issue deleted; already implemented, demon-stration necessary
C34	31361	Develop evaluation criteria for metal state	x				–	issue deleted; will be implemented in branch programme, demon-stration necessary
C35	34241	Install tools for maintenance of upper unit nozzles					–	issue deleted; already implemented, demon-stration necessary
C36	33221	Install displacement indicators for piping					a	

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
		Electrical Supply						
E1	15111	Replacement of ail inverters for the emergency power supply			x		b	some comments for further steps
E2	15121	Increase of battery discharge time			x	x	b	
E3	15131+ (24111)	Analysis of additional sources of energy for safety systems				x	b	some comments for further steps
E4	15132	Improvement of emergency DG reliability			x	x	b	
E5	15211	Replace 6 kV switches			x		a	
E6	24211	Procedures to assess residual lifetime ofcables	x	X			a	some comments for further steps
E7	24221	Fit additional self contained emergency lighting fixtures			x	x	a	
E8	17131	Replacement of input switching devices of RTZO type switchboards			x		b	
E9	15221	Replacement of cable penetration	x					issue deleted; already implemented, demon-stration necessary
E10	24421	High voltage transformers bushings replacement	x				b	some comments for further steps
E11	24311	Analysis of external power grid	x				b	
E12	24131 (24121)	Computerized monitoring turbine generator stator windings	x	x			a	
E13	24122	Computerized monitoring 6 kV motor statorwindings	x	x			a	
E14	24111	Implement a multi-channel system "Regina"		x			b	some comments for further steps
E15	24441	Install stand-by transformers		x	x		b	in R4 already implemented
E16	31351	Programme for replacement of electrical wiring		x			b	
		I&C						
11 (C)	11211	Upgrading reactor power control system to improve Xe and power distribution	x	x			a	
12	14271	Electromagnetic Interference (EMI) immunity			x	x	b	
13	-	Improvement of unit control computer (UCTF)	x	x			–	in 14331 included, rec. to replace UCTF-U of the 2. generation
14 (K)	14211	Replacement of neutron flux monitoringsystem	x	x			b	'1
15	14221	Implementation of reactivity measurement	x	x			b	some comments for further steps
16	14231	Separate impulse lines for primary circuit pressure measurement			X		b	
17	14421	Replacement sensors, transducers andsecondary instruments		x			b, a	compensatory measures before start-upfor Hz measurement

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
18	14251	Monitoring the gas volume under the reactor cover (post accident monitoring system)			x	x	b	5-point transducer after start-up implemented
19	23111	Modernisation of monitoring generator process parameters	x	X			a	
110	14111	Redesign temperature monitoring racks for protective tube units	x	X			b	
111	14261	Replacement Computer and software (Hindukush, SM-2M)	x	x			a	
112	14321	Improvement of the turbine regulating system		x			a	
113	14241	Improvement water level measurement in SG	x	x			b	
114	28511	Develop Technical Support Centre				x	b (R4) a (K2)	
115	28124	Implementation television for closed premises	*x*				b	
116	14331	Replace power unit control system (Titan 2)	*x*	x			b (R4) a (K2)	software must be produced to appropriate standards
117	14411	Implement data storage (black box)			x		a	
118	28411	Implement a system displaying safety parameters (SPDS)			x	x	b (R4) a (K2)	software will require appropriate validation and verification
119 (C)	28111	Introduction of a full diagnosis systems		x			a	
120	28112	Introduction of a computerized network for diagnosis		x			a	
121 (C)	28113	Implementation vibration diagnosis system		x			b (K2) a (R4)	
122	28114	Implementation loose pans diagnosis system		x			b (K2) a (R4)	
123	26115	Implementation noise diagnosis system for SG headers		x			a	
124 (C)	28116	Implementation primary circuit coolant leakage diagnosis system		x			b	
125 (C)	28117	Implementation residual fatigue lifetime diagnosis system		x			b	
126	28118	Implementation MCP vibration monitoring diagnosis system		x			a	
127	28119	Implementation mode diagnosis system		x			a	
128	28121	Implementation In-core noise diagnosis system		x			b	
129	28122	Implementation back pressure valve diagnosis system		x			a	
130	28123	Implementation air operated valves diagnosis system		x			a	

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
		Containment						
B1	27211	Analysis of building structure (especially penetrations)			x		a	
B2	27212	Analysis of adequacy structure incl. Diagnostics		x	x		a	
B3	27213	Procedure of the containment state assessment during operation		x			a	
B4	27214	Prepare calculated groundings ofcontainment reliability	x		x	x	a, b	Implementation in 2 stages
B5	32241	Improvement of containment stale monitoring		x			b	
B6	32231	Develop proposals on diagnosis of forces in fitting cables		x			b	
B7	32251	Implement equipment for containment vacuum test		x			b	
		Hazards						
H1	17321	Analysis to determine the extent of pipeline breaks impact inside the reactor building			x		b	(P)
H2	17311	Develop of criteria for shut-off valves protection against internal missiles			x		a	
H3 (C)	12211	Rigid support of steam and feed water lines			x	x	b	(P)
H4	17211	Complete analysis of internal flooding in Reactor compartment and Machine halt rooms	x	x	x		b	
H5	17111	Performance of a systematic fire hazard analysis	x	x	x		b	modification (recommendations for analysis, implementation depends on safety importance)
H6	17132	Coat the cable bundles with fire resistant coating	x	x			b	
H7	17112	Analysis of situation (fire) in cable compartment under MCR and ECR			x	x	b	
H8	17121	Replace combustible petroleum oil inlubrication system	x	x			a	
H9 (E)	17131	Replacement of input switching devices of RTZO type switchboards		x			b	
H10	17141	Development and implementation of fire extinguishing system special for NPP	x		x		b	
H11	17151	Replace fire resistant doors		x	x		b	
H12	17161	Install fire protection valves in air conduits			X		b	
H13	29111	Improve fire resistance rate of turbine hall roof	x	x			b	
H14	29112	Implement automatic Hydrogen dumping from generator housing	x				a	
H15	29121	Implement smoke prevention system for personnel evacuation			x	X	b, a	

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
H16	29131	Furnishing the compartment containing electronic equipment with gas fire fighting means	x		x		b	
H17	18311	Analysis of possibility of air craft crash				x	b	
H18	18321	Analysis of risk impact on MCR (ECR) personnel of toxic gases			x	x	b	
H19	18211	Analysis of risk of shock wave loads		x	x		b	presentation of meth-odology for next step
H20	18111	Additional instrumental seismic instru-mentation and geophysical studies		x	x		b, a	
H21 (S)	18221	Assessment of the risk of "average minimal temperature" and "extreme cold condition"	x	x	x	x	b (a)	implementation in 2 steps (P)
H22	18212	Analysis of risk of tornado loads		x	x		b	
		Systems						
S1	11011	Develop materials on equipment qualification			x	x	a, b	recomm. not to limit to passport (P) •
S2	12111	Replacement of non-qualified valves and implementation of technical and administrative measures to prevent overpressure events	x	x			b	
S3 (C)	12321	Heating of safety injection water tanks			x		b	
S4 (C)	12331	Heating of the sump water			x		b	
S5 (C)	12411	Organisational engineering measures for management of accidents Involving primary to secondary coolant leak up to D, nom. 100mm.	x	x	x	x	b	
S6	13111	Implementation of devices to measure Boron 10 concentration,	x	x			b	
S7	13211	Analysis of insulation material behaviour under LOCA conditions.			x		b	
38	13213	Ensure residual heat removal under LOCA (replacement of insulation)			x		b	other solutions possible if demonstration of aptitude is given
89	13311	Increase the volume of steam generator make up water,				x	a	
S10	13321	Replacement of steam generator safety valves.			x		b	
S11	13411	Updating of pressurizer pulse safety device to implement "Feed and Bleed" procedure				x	b	
S12	13611	Implementation of lightness diagnosis system for ECCS exchangers.		x			b, a	
S13 (I)	14251	Steam detector under vessel head.			x	x	b	
814	16111	Take measures to prevent radioactive release outside the containment building (MCP heat exchanger)	x		x	x	b	recommendation to study systematic all possibilities of containment bypass
S15 (A)	19311	Carry out the analysis of initialing events not taken into consideration in Technical Report on Safety Substantiation (TOB)			X		b	List should be presented la before SAR

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
S16	16121	ECCS suction pipes- prevention of leakage (bypass of containment)			x	x	b	check of vibration measurement for fur-ther steps
S17	16131	To perform analysis and calculations of hydrogen accumulation Inside the reactor plant and its release to the outside for BDBA.				x	a	
S18	16211	Take measures to prevent explosive hydrogen concentration,				x	b, a	analysis before start-up, equipment after start-up
S19	30131	Hydrogen removal from the reactor plant primary circuit equipment in the process of the cool-down and "cold" shutdown and analysis of hydrogen safety	x	x			b	
S20 (A)	19111	Prepare a list of Design Basis Accidents and define a list of initiating events.			x		b	proposals of Riskaudit, measures if necessary
S21 (A)	19112	Carry out analysis of selected accidents using modem codes			x		b	
S22 (A)	19121	Analysis of reactivity accidents			x		b partly, a	
S23	19211	Identification of beyond design basis accidents to be analysed, Performance of related analysis.				x	b partly, a	proposals of Riskaudit\ at least compensatory measures if necessary
S24	19311	Carry out the analysis of initiating events not taken Into consideration In Technical Report on Safety Substantiation (TOB)			x		b	If necessary at least compensatory measurements have to be implemented
S25	21111	Modernise thermal barriers to improve operational reliability and safety of GZN-195M	x				b	
S26 (C)	21114	Develop a procedure for the determination of allowable defects in body components GZN-195M		x			a	
S27	21115	Develop documentation and carry out auxiliary systems reconstruction to increase the time of interruption in supply of blocking water to sealing of GZN-195M			x		b	modifications can be necessary
S28	22111	Upgrading of steam generator blowdown system	x	x			b	
S29	22441	Retroit balanced (disk) steam generator feed control valves.		x			a	
S30 (H)	18221	Carry out an analysis on possibility of ensuring the normal air conditions inside the rooms of safety system at lower ambient temperature	x	x	x	x	b, a	
S31	22351	Replacement of the air-conditioners.					b	
S32	24411	Installation of an additional Diesel generator set	x	x			b	list of components to be backed up for further steps
S33	13521	Installation of seated valves 1600 diameter		x			b	only in R4 necessary

N°	Item in R4/K2 Mod. Progr.	Item	Concerned Level of Defence in Depth				Imple-menta-tion-date	Riskaudit Recommendation
			1	2	3	4		
		Accident Analysis						
A1 (S)	19111	Prepare a list of Design Basis Accidents and define a list of initiating events.			x		b	1a before SAR justification of list
A2 (S)	19121	Analysis of reactivity accidents			x		b, a 2 steps	
A3 (S)	19311	Carry out the analysis of initiating events not taken into consideration in Technical Report on Safety Substantiation (TOB)			x		b	recomm. additional cases
A4 (S)	19112	Carry out analysis of selected accidents using modern codes			x		b	
A5 (S)	19211	Identification of beyond the design basis accidents to be analysed.				X	b.a	recomm. additional cases
A6	19411	Carry out level 1 and 2 probabilistic safely analysis			x	x	b, a 2 steps	
		Operational Safety						
01	31211	Develop General NPPs Quality Assurance programs.	x	x	x		b	modifications (e.g. independent quality review body)
02	32111	Improve operation procedure for safety related reactor systems			x		b	
03	32112	Improve verification and testing procedure of safety-related reactor system			XX		b, a	
04	30111	Improve technical Instructions and normal operation procedures on reactor equipment and systems	x	x			b	
05	30112	Improvement of maintenance and repair procedures for reactor equipment and procedures	x	x			a	
06	31111	Develop an information system "Computer-aided history of NPP equipment operation"	x				a	
07	30121	Include the list of works involving a nuclear hazard into the regulatory documents.		x	x		b	
08	30211	Elaboration of accidental procedures			x	x	a, b	
0/1	NPP-progr	Improvement of the organisational structure and management	x	x	x	x	b	
0/2	NPP progr.	Personnel training programme	x	x	x	x	b	
0/3	NPP progr.	Emergency planning				x	b	
		Radiation Protection						
R1	33211	Enhance the function of the existing radiation protection	x	x	x	x	b	
R2	33212	Replace of the radiation monitoring system AKRB-03	x	x	x		a	
R3	33231	Development and implementation of an automatic radiation monitoring system			x	x	b, a	
R4	Branch progr.	Development and implementation of an automatic environmental radiation monitoring system	x	x	x	x	b	

ATTACHMENT 4:
K2/R4 Documentaion Requested by the IRR in the Frame of the PPP

The following list of documents was send to Energoatom Kiev without any response:

K2/R4 Documentation Requested by the IRR in the Frame of the PPP

(Institute of Risk Research, 30. Sept. 1998)

Specified Safety Relevant Reports (ranked according to priority)

1. Plant Quality Status of Rivne 4 and Khmelnitsky 2 Units, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 11, 11/97.
2. PSA Documents on WWER-1000 and RBMK reactors.
3. Modernization Programme for the Ukrainian Power Plants with VVER-1000 (B-320), Revision 2, 31.10.1996, Kiev Institute ENERGOPROJECT.
   Part 1: Generic Basic Program
   Part 2: Program for the Modernization of K2
   Part 3: Program for the Modernization of R4.
4. Evaluation of the Modernization Programmes (Rivne 4 and Khmelnitsky 2) Revision 2, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 10.2A, 5/11/1997.
5. Khmelnitsky NPP – Data for environmental Impact Assessment – UKK000IR, Kyivenergoproekt 1996.
6. Rivne NPP – Data for environmental Impact Assessment – UKK000IR, Kyivenergoproekt 1996.
7. Review of the seismological information available at the Kyivenergoproekt Institute on the Khmelnitsky NPP and Rivne NPP sites, which forms the basis for taking decisions on the seismicity of the sites, Kyivenergoproekt 1996. (Reference 3.3 in Rivne EIA).
8. Khmelnitsky NPP – data for environmental impact assessment, Kyivenergoproekt, UKKE00001, 1996. (Reference 1.3 and 6.19 in Khmelnitsky EIA).
9. Information for updating of EIA for KNPP (Parts 1 and 2), SSEC CSER, November 1997. (Reference 1.5 and 6.20 in Khmelnitsky EIA).
10. Khmelnitsky NPP – data for environmental impact assessment, Kyivenergoproekt, UKKE00001K, 1996. (Reference 7.6 in Khmelnitsky EIA).
11. Khmelnitsky NPP – data for environmental impact assessment, Kyivenergoproekt, UKK0001R, 1996. (Reference 8.6 in Khmelnitsky EIA; Reference 8.6 in Rivne EIA).
12. Information for updating of EIA for Rivne NPP, SSEC CSER, November 1997. (Reference 10.8 in Khmelnitsky EIA).
13. Rivne NPP – data for environmental impact assessment, Kyivenergoproekt, UKKE00001R, 1996. (Reference 1.3 and 6.10 in Rivne EIA).
14. Information for updating of EIA for Rivne NPP (Parts 1 and 2), SSEC CSER, November 1999. (Reference 1.5, 6.11, and 10.8 in Rivne EIA).
15. Rivne NPP – data for environmental impact assessment, Kyivenergoproekt, 1996. (Reference 3.1 in Rivne EIA).

16. Rivne NPP – data for environmental impact assessment, Kyivenergoproekt, UKKE0001R, 1996. (Reference 5.34 in Rivne EIA).
17. Modernization Programme for the Ukrainian Power Plants with VVER-1000 (B-320), Revision 1, Consortium KIEP-MOHT-ENAG, 6/1996.
18. Evaluation of the Modernization Programmes (Rivne 4 and Khmelnitsky 2) Revision 1, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 10.2, 15/11/1996.
19. General Safety Objectives, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 7, 23/4/1996.
20. Approach to review of the upgrading programmes Rivne 4/Khmelnitsky 2, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 7a.
21. Riskaudit remark on licensing procedure Rivne 4/Khmelnitsky 2, Riskaudit, TACIS/U/TSO/VVER/02, Report N° 6.
22. Nuclear Safety Expert for Modernization of Kalinin NPP Unit 3, Riskaudit Report N° 78, Fonenay-aux-Roses/Berlin 1997.
23. Nuclear Safety Expert for Modernization of Kozloduy 5&6, Final Report GRS, IPSN, Fonenay-aux-Roses/Berlin 1997.

Safety Relevant Background Information (ranked according to priority)

1. All reports which serve as the basis for the conclusion in the Khmelnitsky EIA (page 0.7) and the Rivne EIA (page 0.8) stated as follows: "*It is also noted that the RBMK reactor is inherently less safe than is the VVER reactor.*" All documents which contain such a comparison, or which serve as the basis for such a comparison, and which serve as the basis for this conclusion should be provided.
2. At page 8.3 of the Rivne EIA, in Section 8.2, it is stated that the promoters of the modernisation programme used INSAG-3 objectives and principles to "*achieve an updated plant which will comply with the current Ukranian rules and which will reach a safety level in line with western safety objectives and practices for both aspects of design and operational safety*". Please provide all documents which address how the design of Rivne 4 meets the INSAG-3 technical safety objectives of a frequency of severe core damage below about 10^{-4} per plant operating year and frequency of large off-site releases at least a factor of 10 below this level (see INSAG-3, page 9, §25). Provide copies of all probabilistic analyses which demonstrate Rivne 4 conformance to either of these INSAG-3 technical safety objectives.
3. Provide all documents which form the basis for concluding that the accident described in pages 8.15-8.18 of the Rivne 4 EIA constitutes a "beyond design basis accident". Note that this accident is an SG collector leak of 100 mm equivalent diameter with failure of the turbine stop valves to close an doperator recovery beginning at 10 minutes, with no additional cladding failures, and no fuel damage exceeding 1% of gas gap releases and 0,1% due to direct contact of fuel with collant, and the spike release of radionuclides from such fuel elements. (Note that IAEA-EBP-WWER-05, page 55, identifies such an accident as a **design basis accident**, and that the same document states that the **beyond design basis accident** would require additionally a failure of long-term core cooling. This accident, involving failure of long-term core cooling, would result in severe core damage and containment bypass.)
4. Provide the reports which identify and describe the "*appropriate upgrading and safety programme*" for the Chernobyl NPP as identified in the first paragraph on page 0.8 of the Rivne EIA. Identify all measures contained in that programme, their basis, their technical nature, their costs, and their schedule for implementation.
5. Safety and Reliability Documents on Zaporozhe Unit 6.

6. Information about severe accidents investigations for K2/R4 and Chernobyl
7. All reports which document a probabilistic safety assessment (PSA) for a VVER-1000/320 NPP or an RBMK NPP should be provided. Such documents would be required in order for the EIAs to state that the no-change option would result in "*an increased* ***risk*** *of a catastrophic accident leading to widespread contamination*" (Rivne EIA, page 0.8; emphasis added). Also, define "*catastrophic accident leaidng to widespread contamination*" as that phrase is used in the cited portion of the Rivne EIA.
8. Provide the report(s) which document the "*detailed safety evaluation*" of the Rivne 4 project as described as follows and which form the basis for the conclusion in the last sentence of the quotation (Rivne EIA, p. 0.6): "*A detailed safety evaluation of the project has been completed. The partner companies in the organisation that undertook this study act as independent technical safety advisors to nuclear regulatory agencies in Germany and France. The conclusion of the study was that the project would allow safety of the plant to be comparable to that achieved in the European Union for NPPs recently re-approved by national safety authorities.*"
9. Specifically identify which European Union NPPs the authors of the Rivne EIA had in mind when making the statement contained in the last sentence of the quotation above.
10. Define "*safety evaluation*" as that phrase is used in the quotation above.
11. Certification Documents on the RPV materials and surveillance specimen.

Seismic and Geological Reports (ranked according to priority)

1. RIVNE NPP, Kuznetsovsk settlement. 1-st phase of construction. Report on the geological surveying in the scale 1: 5000. Lvov, ATEP,1984. 98 p. (in Russian).
2. SHECHTMAN et al. Rivne NPP, extension (2-nd phase of construction). The project. V.1 (in 3 books). Report on engineering-geological conditions of the construction region. Kiev, KoATEP, 1984. (in Russian).
3. SHECHTMAN et al. Rivne NPP, extension (2-nd phase of construction). The project. V.2 (in 7 books). Report on engineering-geological conditions of the operating site and adjacent hydrotechnic constructions. Kiev, KoATEP, 1984. (in Russian).
4. SHECHTMAN et al. Rivne NPP, extension (2-nd phase of construction). The project. V.3 (in 3 books). Report on engineering-geological conditions of the objects of housing construction. Kiev, KoATEP, 1984. (in Russian).
5. PALIENKO V.P.: Neogeodynamika etc, Kiev, Nauk. Dumka, 116c,1992.
6. Report on the exploration of groundwater for industrial and potable water supply of the 2-nd phase of Rivne NPP construction with calculation of reserves, by the state of October 1, 1983. Preliminary and detailed exploration. Riga's ATEP.- Riga, 1984. 287 p. (in Russian).
7. Results of search and exploration of groundwater for water supply of West-Ukrainian NPP, conducted in 1969-1971. Lvov Geological Expedition of Kiev Geological-Exploration Trust. – Lvov, 1971. – 189 p. (in Russian).
8. SHESTOPALOV V.M., RYBIN V.F. Report on the prospective assessment of groundwater reserves of Volyn-Podolian artesian basin. 1973-1977. Lvov, 1977.-520 p. (in Russian).
9. SHECHTMAN et al. Rivne NPP, extension (2-nd phase of construction). The project. V.3 (in 3 books). Addition to the volume 2 (in 5 books) Report on stationary observations of groundwater regime. Kiev, KoATEP, 1984. (in Russian).
10. LOMAYEV. Geology of Volyn-Podolian karst. – Kyiv, "Naukova Dumka" Publisher. 215 p. (in Russian).

11. SHESTOPALOV V.M., GOUDZENKO, V.V. et al. Studying migration of radionuclides in the hydrosphere of regions subjected to radioactive contamination after Chernobyl disaster and elaborating recommendations on industrial and potable water supply. Report of the Institute of Geological Sciences, NAS of Ukraine. Kiev, 1990. – 219 p. (in Russian).
12. Report of the Team of Geoenvironmental studies on the results of works conducted in 1984-1986. Kiev, 1988. (in Russian).
13. DBN A2.2-1-95. The structure and contents of the materials for the environmental impact assessment in the course of designing and constructing enterprises, buildings and structures. The design outline. Derzhkomatom of Ukraine and the Ukrainian Ministry of Environmental safety. 1. July 1995.
14. Requirements for siting of nuclear power plants NPP, Moscow, 1987.
15. Guidelines for conducting the state Ecological Examination. Approved by the order of the ministry of environmental protection and nuclear safety of July 71995, N55.
16. The procedure of submitting documentation for the state ecological examination, approved by the resolution of the cabinet of ministers of Ukraine of October 31,1995.
17. Rivne NPP – Data for Environmental impact assessment – UKKE0000IR – Kyivenergoprojekt 1996.
18. Khmelnitsky NPP – Data for Environmental Impact Assessment – UKK000IR – Kyivenergoprojekt 1996.
19. 15 years of Rivne NPP operation- Rivne NPP 1995.
20. The law of Ukraine “On Environmental Protection” adopted 25 June 1991.
21. The Law of Ukraine “ On Ecological examination” February 9, 1995.
22. The Law of Ukraine “On Radioactive Waste treatment” 30.June 1995.N256.
23. Shestopalov V.M. Chalk karst of Volyn and hydrogeological conditions of its formation. In: Physical Geography and Geomorphology. No. 4, 1970. Kiev. (in Ukrainian).

ATTACHMENT 5: Short Comments on Riskaudit Report No 136:

In June 1997 the Institute of Risk Research published a report on safety relevant issues and measures for K2/R4 NPPs (IRR 1997). In May 1998 the Riskaudit Report No 136 commented the IRR report (Riskaudit 1998). In the present IRR Report additional safety relevant issues which are not adequately addressed in the Modernization Program (Rev. 2) were found and included in this table. As an attachment IRR has produced a comparative table in which comments on safety relevant issues of all three publications are included.

Riskaudit presented in the Riskaudit report No. 120 and No 136 the overall conclusion that, to the extent that all 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 International Atomic Energy Agency (IAEA) documents. These include, in particular the IAEA Safety Series No 75 – INSAG-3, and the Nuclear Safety Standards (NUSS) Codes of Practice.
- 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.

This overall conclusion was not substantiated by Riskaudit and the present report of IRR contradicts this conclusion.

Important Safety-Related Issues (institution which addressed the issue)	**IRR Comments June 1997 (IRR 1997)**	**Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)**	**IRR Comments Oct. 1998**
Area – Logistics and Infrastructure			
Economic Situation in the Ukrainian Energy Sector (IRR97)	The economic situation in Ukraine is characterized by a deep crisis. No domestic funds are available for modernization projects in the energy system.	Not technical issue – Has already been improved and will continue to be improved*.	Compared to the IRR report (IRR 1997) the situation in the Ukrainian power generating industry has not improved. Due to the lack of financial and industrial resources, the Ukrainian supporting infrastructure is not favorable for the further development of nuclear power.
Nuclear Infrastructure (IRR97)	After the disintegration of the USSR, an unsatisfactory situation exists in Ukraine.	Not technical issue – The modernization project is planned to be conducted in close co-operation with nuclear countries. Ukraine is not isolated and not as weak as mentioned. No issue*	After the disintegration of the USSR the Ukrainian nuclear industry was confronted with a drastic loss of nuclear infrastructure. Because of the unfavorable situation in the Ukrainian power generating industry it is questionable that large improvements in creating a nuclear infrastructure necessary for the operation of NPPs have been achieved.
Spare Parts (IRR97)	The lack of spare parts is a problem which exists for the whole Ukrainian nuclear industry.	Not technical issue. The utility r e-organization (pre- condition for financing) will permit to solve this issue financially. No more issue*.	A lack of funds for purchasing necessary spare parts and replaceable equipment, insufficiency of machine-building industry (more than 60% of spare parts supplies come from Russia) result in a decrease of planned maintenance of nuclear units.

* Riskaudit has no specific technical competence on these issues.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Fresh Fuel (IRR97)	The lack of fresh fuel is a problem which exists for the whole Ukrainian nuclear industry.	Not technical issue. There is absolutely no such problem in Ukraine. All NPPs are regularly re-loaded. Not safety issue*.	The situation has improved. At present there is no problem with fresh fuel.
Safety Culture (IRR97)	The safety culture is generally insufficiently developed in Ukraine, especially on responsible levels of management.	Safety culture cannot be improved by a „single" measure. Has been (and continues to be) improved.	Two indicators for an insufficiently developed safety culture: • The MP has not adequately addressed significant safety issues. • Lack of transparency – lack of requested documentation been made available in the PPP
Area – General			
Preservation and Mothballing (IRR97)	This issue is not yet sufficiently investigated. Strong indications exist for minimal or missing conservation/mothballing of equipment and components, which might result in large cost overruns.	Demonstration of existing quality has been provided. Needed corrections are identified and are in the way to be solved.	The quality of the requalification program and equipment qualification addressed in the MP is questionable because of the poor quality of conservation measures during the construction halt and the non-availability of large parts of the manufacturing and construction documentation.
Qualification of Equipment (IAEA, Riskaudit)	This task is still pending. Implementation has not yet been satisfactorily demonstrated.	Modernization measures are planned. Safety issue will be solved	A complete program to qualify equipment under extreme environmental conditions and seismicity is still pending is not planned to be implemented before start-up. This is a deviation from international acceptable practice (e.g. equipment qualification is a precondition for licensing in US plants).
TMI Requirements (IRR98)			Their implementation is a precondition for obtaining an operating license for US plants. Not all TMI issues are addressed in the MP. Some are planned to be implemented after start-up.
Area – Core			
Control Rod Insertion Reliability/Fuel Assembly Deformation (IAEA)	This is a generic problem for WWER-1000/V-320s. It remains unresolved.*	Modernization measures are planned. Safety issue will be solved	At present it is unclear if the measures to solve control rod jamming addressed in the MP deal with the root causes of this issue. Further studies and operational feedback is necessary.
Power Density Control System (Riskaudit)	This is a TMI requirement which must be fulfilled.	Modernization measures are planned. Safety issue will be solved	Measures are still in the testing phase. Automatic control of Xenon oscillations and power distribution is not specified in the MP and will be implemented after start-up.
Xe-Oscillations (Riskaudit)	This is a generic issue, which is not yet resolved.	Modernization measures are planned. Safety issue will be solved	Measures are still in the testing phase. Automatic control of Xenon oscillations and power distribution is not specified in the MP and will be implemented after start-up.

* for Atomaudit´s comments see issue 3.3.1.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Area – Component Integrity			
RPV Embrittlement and its Monitoring (IAEA)	This problem is generic for all WWERs. Only limited solutions appear possible. Generally there is insufficient space for inspection of the RPV walls from the outer side on the level of the critical (highly irradiated) weld.	Modernization measures are planned including re-location of surveillance specimen containers, low leakage, fluence measurement, … Problem is not linked with inspection from outside. Safety issue will be solved.	The proposed measures and actions are generally favorable. However they do not satisfactorily add to the clarification whether the usage of the improved data of the surveillance-specimens reduce the uncertainty in predicting RPV irradiation embrittlement. A comparative assessment is recommended to be performed assessing predicted RPV embrittlement based on embrittlement data of the surveillance-specimens in comparison with measurements of the embrittlement of the actual RPV material. RPVs of reactors which are already taken out of operation are predestinated for such measurements (e.g. VVERs in Greifswald).
Non-Destructive Testing (IAEA)	See above.	Modernization measures are planned. Safety issue will be solved	Non-destructive testing (NDT) for primary cooling system components is carried out using the defect-reject approach rather than the defect-follow approach. The latter approach is capable of timely detection of degradation. The existing procedures are not adequate for NDT of SG collectors and tubing.
Steam Generator Collector Integrity (IAEA)	This situation is insufficiently taken into account in the original WWER-1000/V-320 design. A design solution is still pending.	Modernization measures are planned (prevention and mitigation). Safety issue will be solved	This issue is important due to high Core Damage Frequency (CDF) contribution, high containment bypass frequency (large release frequency), continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), failure to implement an automatic safety system response to the initiating event, failure to replace the steam generators, lack of symptom-oriented Emergency Operating Procedures (EOPs), limited Emergency Core Cooling System (ECCS) water inventory, and lack of adequate compensatory measures at the time of startup.
Steam Generator Tube Rupture (IRR98)			This issue is important due to high CDF contribution, high containment bypass frequency (large release frequency), inadequate NDE, continued use of copper-based tubing in the condenser (which contributes to secondary side chemical attack), lack of symptom-oriented EOPs, limited ECCS water inventory, and lack of adequate compensatory measures at the time of startup.
Steam and Feedwater Piping Integrity (IAEA, Riskaudit)	The integrity is impaired for all WWER-1000/V-320 reactors. Basic acceptable solutions are needed. Related measures might become cost intensive.	Modernization measures are planned. Safety issue will be solved	LBB is not applicable to secondary piping because this piping is susceptible to cracking due to corrosion. Lack of an erosion-corrosion program for K2 and R4 only serves to re-emphasize the inapplicability of LBB to secondary piping.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Area – Systems			
ECCS Sump Screen Blockage (IAEA, Riskaudit)	A solution for this problem is generally possible.	Modernization measures are planned. Safety issue will be solved	The proposed measures for avoiding concerning the analysis of insulation material behavior under Loss Of Coolant Accident (LOCA) conditions and the implementation of a selected technical solution to ensure residual heat removal under LOCA appear to be appropriate but not sufficient. A reliable technical solution for backflushing the sump screens should be foreseen.
Steam Generator Safety and Relief Valves (IAEA, Riskaudit)	This safety issue is generic. A satisfactory solution is generally possible.	Modernization measures are planned. Safety issue will be solved	Qualification of atmospheric dump valve (BRU-A) for water and two-phase flow important due to containment bypass implications in the event of a steam generator collector failure or a steam generator tube rupture.
Loss of heat sink (IRR98)			Important due to multi-unit concurrent accident potential, dependent failure potential, implications for spent fuel pool severe accidents, possible high CDF contribution from loss of ESW, and lack of improvements in the modernization program.
Reactor coolant pump (RCP) seal failures (IRR98)			Both means of ensuring reactor coolant pump (RCP) seal integrity are isolated on safety injection and containment isolation signals, raising the possibility of RCP seal LOCA. In addition, the seals are vulnerable on loss of offsite power and loss of service water conditions. Tests have been performed on prolonged loss of seal cooling, but the tests are not representative of actual plant conditions with aged pump seals.
ECCS Sump Capacity (IRR98)			The ECCS sump capacity of the VVER-1000/320 design is limited compared with western PWRs (630 m^3 vs. a range of 950-1900 m^3 for western PWRs). This results, in combination with the lack of use of symptom-oriented EOPs, in a higher human error rate for events in which primary coolant is lost outside containment, because this limits the amount of time the operators have in which to depressurize the reactor coolant system and stop the loss of primary coolant before the ECCS sump inventory is exhausted.
Area – Instrumentation & Control			
Reactor Vessel Head Leak Monitoring System (IAEA)	This safety issue is generic for all WWER-1000/V-320 reactors. An adequate solution seems possible.	Modernization measures are planned. Prevention + monitoring of primary leak. Safety issue will be solved	The IAEA Category III issue of a reactor vessel heat leak monitoring system is not explicitly addressed in the modernization program. Rather, the modernization program references a generic primary system leak detection system, without specifically mentioning the special issue of reactor vessel head leaks and their implications as possible contributors to control rod ejection accidents.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Replacement of I&C (IRR98)			Replacement of large parts of the I&C because of damage and vintage design is included in the repair and replacement program of K2/R4 (see Table 1.5-2). In contrast to other completion projects, e.g. the Temelin, the I&C replacement planned for K2/R4 has not been specified. Temelin NPP has demonstrated that the substitution of the I&C system could have a great impact on the modernization project. In the Temelin project the original instrumentation and control system is exchanged by a Westinghouse distributed digital system. The merging of two technologies at an advanced completion level is one of the major technical problems for time delays and cost overruns in the Temelin completion project.
Area – Electrical Power			
Emergency Battery Discharge Time (IAEA)	Reliable solutions for this issue can be found.	Modernization measures are planned. Safety issue will be solved	The modernization program includes upgrade of the battery capacity from 30 minutes to 60 minutes. However, there is no demonstration that this is sufficient, and the IAEA report referenced by the measure is inconsistent, recommending minimum discharge times ranging from one to three hours.
Residual Life Time of Cables (Riskaudit)	This issue is not yet assessed. Corresponding measures might become cost intensive.	Modernization measures are planned. Safety issue will be solved	The issue is addressed in the MP.
Replacement of 6 kV switchgear (IRR98)			Although acknowledged historically as a reliability issue, replacement of the 6 kV switchgear is deferred until after startup. The RiskAudit report on Rivne Unit 3 noted that there have been, on average, 2 failures per year of these breakers (RiskAudit 1994: 5/28). Unreliability of 6 kV switchgear results in common-mode unavailability of an entire train of all safety systems, and is therefore an important safety issue for all initiating events.
Area – Containment			
Containment Bypass (IAEA)	A satisfactory solution is limited due to the specific steam generator design used and its potential to fail.	Modernization measures are planned. Safety issue will be solved	There are three containment bypass mechanisms identified by IAEA in Safety Issue Cont1 (Category II Issue). Only two of these three mechanisms are addressed by the modernization program before startup. For the remaining issue, the compensatory measures do not affect the likelihood of the bypass mechanism; rather, the measures only enhance the detection of the failure **after** it has occurred. The other containment bypass mechanisms are not adequately addressed by the modernization program, or are not addressed at all.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Containment Structure (IRR97)	Any deficiencies of the containment have thoroughly to be assessed and corrected.	Modernization measures are planned. Safety issue will be solved	In the case of the VVER-1000/320 design, the bottom of the containment is elevated above grade, and there are three levels of non-containment spaces below the containment. (These spaces include the main and emergency control rooms, both of which would have to be evacuated in the event of containment melt-through due to lethal radiation doses which would occur if the operators remained.) Penetration of core debris into these spaces would result in containment bypass since the spaces below the containment are not designed to be pressure-retaining, nor are they lined to prevent radioactive release to the environment. Indeed, if the HVAC system remains in operation, it would actually **promote** a higher release by "pumping" the airborne radioactivity to the environment. The filtration system in the HVAC system would fail since it is designed for non-severe accident source terms, pressure conditions, and thermal loads on the filters. Although the release point would be elevated, at the plant stack, this would still be a considerably worse outcome than would be experienced in most western PWRs, which do not have this containment bypass pathway present.
Presence of pneumatic containment isolation valves (IRR98)			The designs use fail-open pneumatic valves as containment isolation valves, which is at variance with western safety criteria and which result in a greatly increased risk of an interfacing LOCA given failure of the pneumatic system.
Containment Ultimate Capacity (IRR98)			The modernization program lacks a measure to analyze the ultimate strength of the containment. Such an analysis is needed to support proper analyses of BDBAs as well as to structure an accident management program.
Area – Internal Hazards			
Fire Prevention (IAEA)	The fire hazards potential and its prevention have not yet been sufficiently addressed in the modernization program. A PSA is necessary to take effective measures.	Modernization measures are planned. Safety issue will be solved	Important due to lack of previous fire hazards analysis, the lack of fire PSA for VVER-1000/320 (except Temelin, for which the results are not publicly available), and the lack of coverage of fire in K2/R4 PSA until after startup.

Important Safety-Related Issues (institution which addressed the issue)	**IRR Comments June 1997 (IRR 1997)**	**Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)**	**IRR Comments Oct. 1998**
Pipeline Breaks Impact Inside the Reactor Building (Riskaudit)	Sufficient and reliable measures are still open.	Modernization measures are planned. Safety issue will be solved	A number of measures are included in the modernization program to address these issues. However, the key measure (implementation of leak-before-break) for primary piping is not going to be implemented at startup. Further, the measures adopted for secondary piping are of limited use without implementation of controls on erosion-corrosion problems, and the modernization program does not even mention erosion-corrosion despite the fact that it has been identified as a problem in Rivne Unit 3 (RiskAudit 1994: 4/28) and other VVER-1000 units. In addition, it is not clear how the modernization program has addressed the issue of large diameter ESW pipeline failure in the reactor building (identified as a concern by the MOHT report).
High Energy Pipes Ruptures (Riskaudit)	This is a safety issue applicable to all WWER-1000/V-320 reactors. Basic solutions to safely separate high energy pipes are still needed. Appropriate measures are potentially cost intensive.	Modernization measures are planned. Safety issue will be solved	LBB is not applicable to secondary piping because this piping is susceptible to cracking due to corrosion. Lack of an erosion-corrosion program for K2 and R4 only serves to re-emphasize the inapplicability of LBB to secondary piping.
Area – External Hazards			
Extreme Weather Conditions: Low Temperature (Riskaudit)	Assessing this issue will require performing a review of the design basis.	Modernization measures are planned. Safety issue will be solved	The sites of K2/R4 have to be assessed with respect to the natural phenomena prior to start-up and the site specific aspects should be included in the MP (IAEA 1995a and IAEA 1997). Performing a PSA including external weather hazards is highly recommended. Protective measures against hazards of tornadoes have to be implemented.
Man-induced external hazards and seismicity (IRR97)	This issue must be assessed in site-specific investigations, which have not yet been performed.	Modernization measures are planned. Safety issue will be solved	The modernization program limits its consideration of man-induced external hazards to aircraft crash onto the reactor building, the impact of shock loads on plant structures from explosions, and the possible intake of toxic gases and their impact on MCR/ECR personnel. In contrast, the IAEA recommended a global analysis of man-induced external hazards using screening techniques to identify those hazards requiring more detailed analysis. In addition, limiting the aircraft crash analysis to the reactor building has been shown to be a weakness since impacts into the ESW building could result in multi-unit concurrent core damage accidents. Aircraft impacts into the switchyard or turbine building could also be important to risk.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Seismicity and Geology (IRR98)			The issue seismicity is important due to low Peak Ground Acceleration (PGA) level for design (0.05g) compared with seismic hazard at 10,000 year return interval (0.17g), lack of seismic qualification of Essential Service Water (ESW) system (multi-unit concurrent accident risk), lack of seismic qualification of ventilation and fire protection water pumps, and lack of seismic PSA/seismic margin analysis until after startup. Geology: Monitoring of karst phenomena and karst water, consequences of possible accidents for groundwater safety areas (emergency preparedness) and impact of karst activity on pile foundations of R4 not addressed in the MP for K2 and R4. Necessary paleoseismic and seismotectonic studies not included in the MP for K2/R4.
Area – Accident Analysis			
Plant-specific PSA (IRR)	The proposed modernization program for K2/R4 is not based on plant-specific PSA results. Thus the possibility exists that measures are taken with unknown level of impact on plant safety.	Modernization measures are planned. Safety issue will be solved	Important because the modernization program is almost completely deterministic, because the upgrade program ignores a number of recommendations from MOHT[52] based on PSA results for VVER-1000/320 reactors, and because the PSAs for K2 and R4 are not scheduled to be completed until after startup. This is inconsistent with the basic and original purpose of PSA which is as a design aid. PSAs are recognized by IAEA and others as a basic ingredient in formulating a safety improvement program that adequately addresses risk
Rapid Reactivity Increase (IRR)	A complete set of rod ejection analyses has to be accomplished for the start-up phase of operation of K2 and R4, taking into consideration the potential severity of this type of accident.	Modernization measures are planned. Safety issue will be solved	A complete set of control rod ejection analyses must be accomplished before start-up of K2 and R4. Additionally it is recommended to consider control rod ejection initiating events also in the PSA. In the Modernization Program plans to develop fuel of new design with burnable neutron absorbers (measure 11212) after start-up are mentioned but not specified.
DBA and BDBA (IRR98)			A more comprehensive spectrum of accidents (including reactivity accidents) should be analyzed than proposed in the MP before start-up.
Area – Operation			
Symptom Oriented Emergency Operating Procedures (IRR98)			This issue is important because the existing procedures and the ones which will be in place at startup are event-oriented procedures instead of symptom-oriented EOPs as recommended following the TMI-2 and Chernobyl accidents, because of the high CDF contribution of human errors with event-oriented EOPs, and because of the importance of human actions in mitigating containment bypass accidents which dominate CDF for the VVER-1000/320 design.

[52] MOHT is an association of the following organizations: Atomenergoproject, OKB Gidopress, Kurchatov Institute, VNIIAES, Zarubejatomenergostroy, Rosenergoatom, et al.

Important Safety-Related Issues (institution which addressed the issue)	IRR Comments June 1997 (IRR 1997)	Riskaudit Comments Riskaudit Report No 136 (Riskaudit 1998)	IRR Comments Oct. 1998
Area – Spent Fuel and Radioactive Waste			
Spent Fuel Storage (IRR)	A critical situation with the spent fuel storage capacity can be expected by the year 2000.	No technical difficulties to deal with the issue.	Depending on the configuration of the spent fuel pool at the time, the spent fuel pools could also undergo a severe accident which would result in considerable additional hydrogen and fission products being released into the containment.
Radioactive Waste Management (IRR)	There is a lack of a proper infrastructure for radioactive waste treatment and management in Ukraine	No technical difficulties to deal with the issue.	The situation has not changed.