NNSA/IAEA VVER Reactor Safety Workshops
May 2002 – April 2003

Executive Summary

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July 18, 2003

This work was performed under the U.S. Department of Energy National Nuclear Security Administration office NA-23 under Contract W-31-109-ENG-38. The work was performed in cooperation with the International Atomic Energy Agency, the Nuclear Regulatory Authority of Slovakia, and the Center for Nuclear Safety in Central and Eastern Europe.

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1. Introduction

Over the past year, the U.S. National Nuclear Security Administration (NNSA) has sponsored four workshops to compare the probabilistic risk assessments (PRAs) of Soviet-designed VVER power plants. The *International Workshop on Safety of First-Generation VVER-440 Nuclear Power Plants* was held on May 20 – 25, 2002, in Piešťany, Slovakia. A short follow-on workshop was held in Bratislava, Slovakia, on November 5-6, 2002, to complete the work begun in May. Piešťany was the location also for the *International Workshop on Safety of Second-Generation VVER-440 Nuclear Power Plants* (September 9-14, 2002) and the *International Workshop on Safety of VVER-1000 Nuclear Power Plants* (April 7-12, 2003). The four workshops were held in cooperation with the International Atomic Energy Agency (IAEA), the Nuclear Regulatory Authority of Slovakia (UJD), the Center for Nuclear Safety in Central and Eastern Europe (CENS), and Argonne National Laboratory (ANL).

The objectives of the workshops were to identify the impact of the improvements on the core damage frequency; the contribution to the PRA results of different assumptions about events that can occur at the plants; and to understand, identify, and prioritize potential improvements in hardware and plant operation of VVER nuclear power plants. These objectives were achieved based on insights gained from recent PRAs completed by the plants and their technical support organizations.

Nine first-generation VVER-440 plants (nominally of the VVER-440/230 design) are currently operating in Armenia, Bulgaria, Russia, and Slovakia. Sixteen VVER-440/213 plants are currently operating in the Czech Republic, Hungary, Russia, Slovakia, and Ukraine. Twenty-three VVER-1000 plants are currently operating in Bulgaria, the Czech Republic, Russia, and Ukraine. Eleven addition plants are in the advanced stages of construction in various parts of the world.

The workshops reviewed the current configuration and safety status of each plant represented, which have all implemented numerous safety upgrades from the original VVER designs. The PRAs were reviewed in order to identify differences among the studies.

1.1 Safety Analysis as the Basis for Risk-Informed Plant Operation

The National Nuclear Security Administration (NNSA) International Nuclear Safety Program (INSP) set out to improve the safety of the Soviet-designed reactors in the former Soviet Union and satellite countries. A fundamental aspect of the program is safety analysis, which is the sole basis for fully understanding the normal and off-normal behavior of the plant. This knowledge enables the designers, operators, and maintainers to develop the optimum set of procedures for operation of the plant under emergency conditions.

The two branches of safety analysis are deterministic analyses, which are used to identify the progression of events following off-normal faults, and probabilistic risk assessment (PRA), which is used to determine the frequency of those sequences whose outcome is undesirable and the relative contribution of plant and personnel failure to these sequences. Deterministic safety analyses start with presumed accident scenarios (pipe breaks, loss of power, etc.) and attempt to quantify the consequences of those accidents. Those consequences (e.g., the maximum fuel cladding temperature during the event) are compared with prescribed acceptance criteria to determine whether the plant can adequately cope with the accident scenario. Deterministic analyses for design basis accidents are a necessary part of Western plant licensing processes. For extreme and unlikely scenarios (i.e., beyond design basis accidents), deterministic analyses are useful for quantifying the potential radioactive release to the environment.

Probabilistic risk assessments, unlike deterministic analyses, do not presume an accident scenario for the plant. Rather, probabilistic analyses use the probabilities of individual component failures and an understanding of how components are interlinked to predict the probability of an initiating event propagating into a severe plant accident. Probabilistic risk assessments are extremely useful in determining important accident scenarios that the plant may encounter and in identifying components and operator actions that are key to those accident sequences. Thus, probabilistic analyses are used to define accident scenarios for simulator training of operators, just as deterministic analyses are used to determine the plant response during those events.

The main objectives of deterministic and probabilistic assessments are

- to reflect the true safety status of the unit;
- to show whether the technical condition of the structures, systems, and components of the power plant ensures the safe operation of the unit;
- to reveal possible deviations from the requirements of the existing rules and regulations and to justify the adequacy and effectiveness of the compensatory measures taken; and
- to determine whether the effects of the operation of the unit on the plant personnel, the public, and the environment exceed the safety limits established by the rules and regulations.

Deterministic and probabilistic assessments are at the root of risk-informed operation and regulation of the nuclear power plant and the establishment of a safety culture within the utility and regulatory bodies. A background to the increasing use of PRA as an integral part of plant design, upgrade, and operation is provided in the next section and the results of the NNSA workshops are highlighted in section 2. The conclusions and recommendations for future activities are in section 3.

1.2 Development of PRA in the U.S. as a Cornerstone of Safety Culture

The gestation period for the use of PRA in the U.S. and other Western countries has been approximately 25 years, but has now reached maturity and is the basis for risk-informed regulation as one of the twin pillars of the licensing process, the other being the deterministic design basis criteria laid down in the regulations. This historical perspective is summarized in NUREG/CR-6813 Section 1.2.1 (*Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making*). It should be emphasized that the deterministic design basis analyses are performed to demonstrate that the plant design can *prevent* core damage from the design basis events and define the envelope of safe operation. Deterministic analyses, however, give no information about the frequency of events that go beyond the design basis and lead to offsite releases of radiation.

In 1975 it was recognized that the rules by which a complex nuclear power plant was designed did not cover all aspects of potential accidents, nor did they address the frequency of occurrence of potential accidents. A methodology using Boolean logic in the form of event trees and fault trees was developed. The appeal of this type of analysis was that it enabled the potential accidents to be evaluated in an inductive manner, paralleling the events that would happen at the plant in the form of the sequence of events given the success and failure of various plant protection functions and systems. Starting from a given initiating fault there could be many paths, some of which would be acceptable and others leading to an unacceptable plant state. These potential paths were developed in the form of an "event tree" for each identified fault or groups of faults that resulted in similar sequences of events. Each branch in the sequence represented either success of a safety function or plant system. It is possible to quantify the probability that failure occurs at the branch by identifying all the ways in which the system (and all its support systems, including the human operator) might fail by developing a logic model known as a fault tree in which each component failure is an event with a given probability. The quantification of this tree gives the probability of failure of the function. More important, the quantification of all such trees in a given sequence gives the conditional probability of the sequence occurring while accounting for commonalities among the different systems and functions in each event tree sequence.

Initially the ability to accurately model the plant was severely limited by computational capacity of existing computers. Now this is no longer an issue, so it is possible to develop risk models that comprehensively model plant performance.

Three levels of PRA are typically performed. A level-1 PRA considers sequences that could lead to core damage. It focuses on plant design and operation, and can include initiating events both internal and external to the plant. Internal events would include hardware failures, operator errors, internal fires and floods, and operational perturbations. External events could include seismic disturbances, flooding, man-made hazards, and weather-related events. Level-1 PRAs have been performed for full-power, low-power, and shutdown conditions. A level-2 PRA looks at the consequences of a core-damage accident and considers the response of radiation confinement measures and the transport of radioactive material into the environment. A level-3

PRA analyzes the dispersion of radioactive material beyond the power plant and the potential environmental and health effects. A probabilistic risk assessment, then, is not a single analysis. There are many aspects of plant operation and accident phases that have to be addressed in a progressive manner. Each phase builds on the work performed in the earlier phases, and in each case new information and a unique approach is required.

The most influential body in ensuring wide acceptance of PRA has been the IAEA, which has taken the lead in publishing a wide range of documents in the performance and applications of PRA.

2. Workshop Findings

The workshops were designed to use the results of probabilistic risk assessments to identify the impact of the improvements on the two components of the core damage frequency: the initiating event frequency and the conditional core damage probability. This study helped to relate assumptions about events that can occur at the plants to the PRA results and to understand, identify, and prioritize potential improvements in hardware and plant operation.

Comprehensive workshop reports have been issued for each class of VVER. These contain a complete list of the implemented and planned modifications for the plants and the results of the current PRAs. The differences in results arising from different assumptions relating to the frequency of initiating events and the subsequent performance of the safety systems are reported on. The summary of the findings in each workshop is presented in the following sections.

Figure 2-1 is a scatter plot of initiating event frequency versus conditional core damage probability for all the VVER plants examined.[1] The initiating event frequency value plotted represents a summation of the frequencies for all full-power internal-event initiators modeled in the PRA. Fire, flooding, and external hazards (e.g., weather) are excluded. The conditional core damage probability value plotted represents a weighted average of probabilities for the individual events modeled in the PRA. The diagonal dashed lines are lines of constant core damage frequency (the product of the initiating event frequency and the conditional core damage probability). Mochovce-1 (VVER-440/213) and Tianwan-1 (VVER-1000) claim to have the lowest internal-event core damage frequencies; Novovoronezh-5, Kalinin-1, and South Ukraine 1 (all VVER-1000) have the highest values.

2.1 VVER-440/230 Reactors

The risk profiles from the three VVER-440/230 PRAs examined at the workshops, in terms of both the dominant initiating events and accident sequences, are quite different (Table 2.1-1). Taking into consideration the differences among the current plant configurations, it appears that the risk profiles are strongly influenced by assumptions that have been made in the PRA modeling about initiating events that have not yet occurred and a misuse of Bayesian statistical methods. These assumptions mask some of the risk differences that arise as the result of the differences in plant modification status. Harmonization of initiating event frequencies would enable the true impact of the changes to the plant systems to be determined and the identification of further beneficial plant modifications.

The reasons for the differences in the core damage frequency (CDF) for each of the plants are discussed in the following subsections.

[1] This plot includes the latest VVER-440 PRA results as provided at an IAEA-led and NNSA-supported follow-up workshop held in Budapest, Hungary, in May 2003. Several of the VVER-440 plants had upgraded their PRAs since the NNSA workshops in 2002. Some plants have improved their safety culture by implementing “living PRA” programs that call for periodic updating of the PRA models to reflect current plant configurations and operations. Results reported elsewhere in this report are based on data presented at the earlier NNSA workshops.

Table 2.1-1. VVER-440/230 Rankings of the Most Significant Initiating Event Contributors to Core Damage Frequency (CDF)

Initiating Event Category	Bohunice-V1 (Slovakia)			Kozloduy 3&4 (Bulgaria)			Novovoronezh-3 (Russia)		
	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%
LOCA > 100 mm	***1***	8.61E-06	37%	7	5.62E-06	7%	8	1.00E-06	3%
LOCA 32-100 mm	***2***	5.04E-06	22%	6	6.24E-06	8%	6	1.79E-06	5%
LOCA < 32 mm	***3***	3.88E-06	17%	***1***	1.90E-05	24%	4	2.33E-06	7%
Pressurizer Safety Valve LOCA	7	7.26E-07	3%	***3***	7.60E-06	10%	***2***	6.91E-06	20%
Steam Generator Tube Rupture	8	6.82E-07	3%	***2***	1.34E-05	17%	14	1.42E-07	0.4%
Steam Generator Collector Rupture	4	9.29E-07	4%	4	7.14E-06	9%	***1***	1.01E-05	29%
Steam System Breaks	10	5.39E-07	2%	5	6.41E-06	8%	***3***	5.32E-06	16%

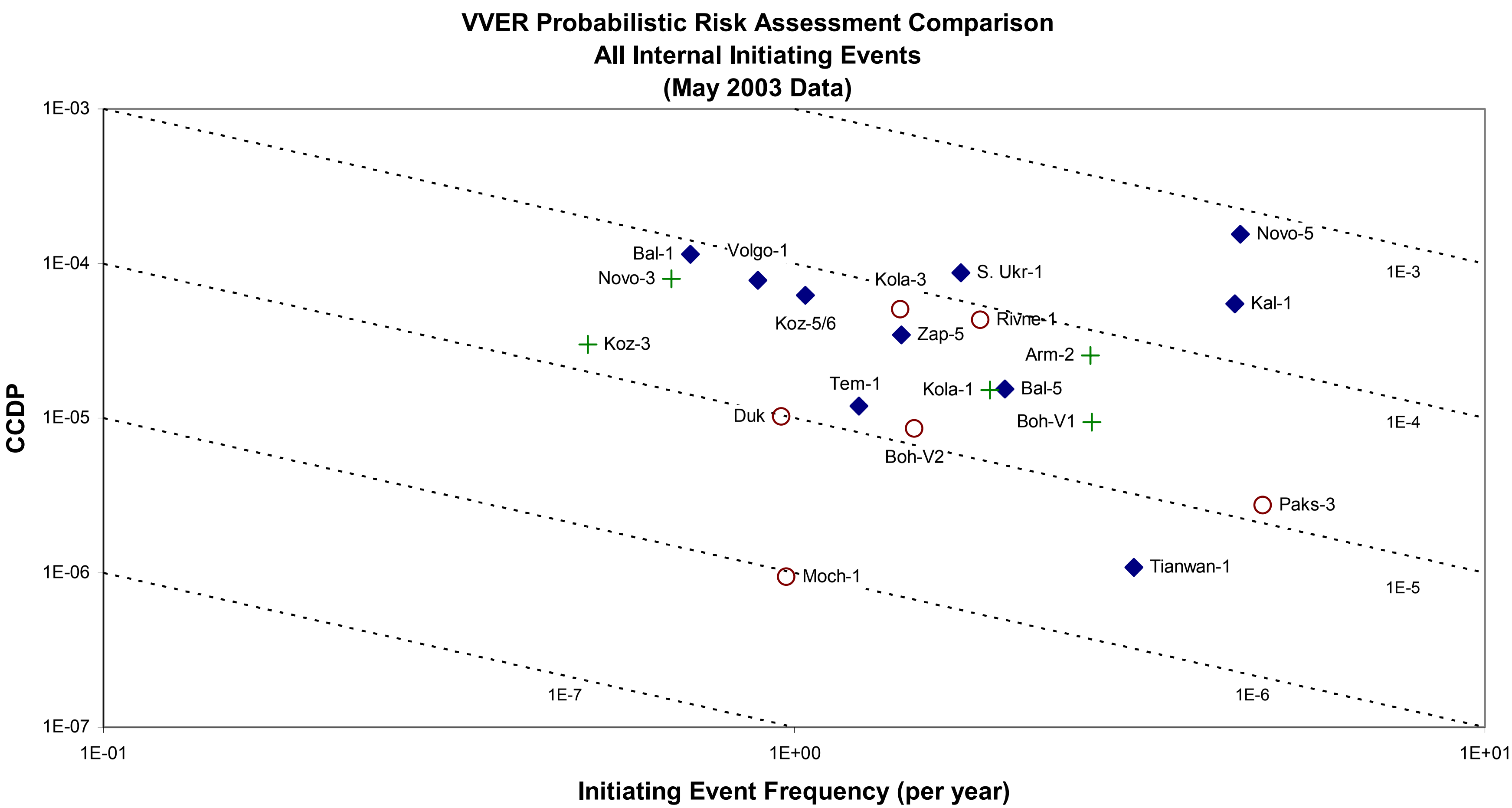

Figure 2-1. Conditional Core Damage Probability vs. Initiating Event Frequency for VVER Reactors

2.1.1 Modeling Assumptions and Initiating Events

Each of the three units used a different approach to deriving the frequency of initiating events for which there are no data (i.e., postulated events that have not yet occurred at any power plant). As a result, there is a particularly wide range for the frequency of large loss of coolant accident (LOCA) and steam generator header rupture events. The latter event has being excluded from two of the PRAs altogether on the basis of low occurrence frequency. In the case of large LOCAs, Bohunice-V1 and Kozloduy-3/4 (which have a low pressure injection system and, hence, the capability of providing sufficient make up in the event of a large LOCA) use a higher initiating event frequency than Novovoronezh-3, in line with the frequency used for many Western PWRs.

This difference is specifically related to assumptions on the performance of leak-before-break (LBB) equipment, which has been fitted at all three plants. In the U.S., the purpose of such equipment is to justify revised (less frequent) in-service inspection programs on the basis that incipient wear or crack growth will not propagate into a large LOCA. The LBB concept has not been used in U.S. PRA studies to justify that a large LOCA cannot occur. The occurrence of leakage rather than pipe rupture indicates that LOCAs occurring in the primary pipe work will fall into the category of small or medium LOCAs for which the plant response is scram and automatic coolant injection. Is the impact of the hydraulic transients resulting from this plant response included in the consideration of the large LOCA frequency? This issue needs to be addressed further.

It is necessary, then, to develop a rational approach for all VVER-440/230 plants for the treatment of all LOCAs in the range from 100-500mm that is based on the performance of equipment installed for the detection and indication of leaks. The same need exists for the treatment of the larger primary-to-secondary leaks. There should be a uniform basis for deriving the frequency of large leaks (leaks greater than a single tube, but less than 100mm equivalent diameter) and header rupture (leaks greater than100mm equivalent diameter).

A second concern relating to LOCAs is that Bohunice-V1 and Novovoronezh-3 have not included very small loss of coolant accidents above the technical specification for leakage but below the break size that will immediately initiate an automatic scram. This size break requires the reactor to be shut down, cooled down, and depressurized (i.e., it requires the use of plant safety systems). This LOCA category contributes 13% to the core damage frequency at Kozloduy. It is standard PRA practice to include such LOCAs. If they are not included in the full-power PRA, then they need to be included in the low-power and shutdown PRAs since all initiators that lead to reactor shutdown must be included.

The other initiating event for which there is wide divergence among the plants is the modeling of steam line breaks. Novovoronezh-3 uses plant experience to justify a very high frequency for a failed-open steam generator relief valve. The other plants use significantly lower initiating event frequencies.

2.1.2 Plant Improvements and Conditional Core Damage Probability

The modification implementation status tables developed at the workshops indicate the area of the PRA impacted by each improvement and include information on whether the plant change has been incorporated in the PRA. Thus it is possible, when comparing the results of the PRAs for each plant, to identify whether differences in results are related to specific plant improvements. This eliminates one of the uncertainties in the PRA result comparison. The impacts of plant improvements are compared for transients and loss of coolant accidents.

Transients

It is clear from the PRA results that the improvements to the systems supporting secondary and decay heat removal have had a major impact on the risk profiles for the various transient initiating events at the plants that have made the modifications. In particular in the Bohunice-V1 and Kozloduy-3/4 PRAs, credit has been taken for primary feed-and-bleed cooling in the event that all steam generator feedwater is lost. Pressurizer relief valves qualified for two-phase flow have been installed at Novovoronezh-3; however, there are no plans to implement primary feed-and-bleed into the emergency operating procedures. Hence the contribution from transient events is higher at this plant. Armenia-2 has installed qualified pressurizer safety valves, but has yet to implement a feed-and-bleed procedure.

In the case of high-energy line breaks, Bohunice-V1 has also made modifications to the emergency feedwater injection lines to ensure physical independence from the other injection mechanisms and consequential effects such as certain steam and feedwater line breaks. Novovoronezh-3 has not yet considered the dynamic effects of high-energy line breaks (e.g., pipe whip and jet blast).

Loss of Coolant Accidents

In the case of LOCAs, all plants have made modifications to the high-pressure emergency core cooling system to improve system reliability. However, the key difference between the units is the replacement of two of the high-pressure pumps by low-pressure pumps. This was included in the initial design at Kozloduy-3/4 and has been included in the PRA for large LOCAs, but not for medium ones. The new system has been installed at Bohunice-V1 and is included in the current PRA. Thus two plants have the capability to mitigate the large-break LOCA. The third plant (Novovoronezh-3) indicates that the risk from such a break is low because the fitting of leak-before-break monitoring equipment ensures that the event frequency is low. Armenia-2 has neither leak-before-break measures nor low-pressure injection pumps to respond to such an event.

For LOCAs in the range 32-100 mm equivalent diameter, there are differences in the PRA success criteria that affect the results. For Bohunice-V1, credit has been taken to depressurize the primary circuit in the event that the high-pressure emergency core cooling system fails, so that low-pressure emergency core cooling may be used. In contrast, no credit has been taken at Kozloduy-3/4 even though this plant also has a low-pressure emergency core cooling system. As noted above, there is no low-pressure emergency core cooling system at Novovoronezh; however, credit has been taken to manually isolate the LOCA using the main loop isolation valves. The efficacy of the main loop isolation valves to isolate a LOCA has been a matter of continuing debate among VVER PRA practitioners. The design differences between Bohunice and Kozloduy on the one hand and Novovoronezh on the other will lead to different core damage frequencies among the plants if a common set of assumptions is used in relation to the actual plant design in each case.

2.2 VVER-440/213 Reactors

As with the VVER-440/230 PRAs, the risk profiles from the six VVER-440/213 PRAs examined at the 2002 workshop, in terms of both the dominant initiating events and accident sequences, are quite different (Table 2.2-1). This is primarily due to a number of assumptions that have been made in the PRA modeling about initiating events that have not yet occurred (i.e., extremely rare events) or a misuse of Bayesian updating in the treatment of these events. These assumptions have a direct impact on the initiating event frequencies. They mask some of the risk differences that arise as the result of the differences in plant modification status.

In the case of Bohunice-V2 the interaction between high-energy pipe failures and emergency feedwater has been eliminated by plant upgrades, but this is not yet reflected in the PRA, which distorts the results.

The harmonization of the initiating event frequencies would enable the true impact of the changes to the plant systems to be determined and the identification of further beneficial plant modifications.

The reasons for the differences in the core damage frequency for each of the plants are discussed in the following subsections.

Table 2.2-1. VVER-440/213 Rankings of the Most Significant Initiating Event Contributors to Core Damage Frequency

Initiating Event Category	Bohunice-V2 (Slovakia)			Dukovany 1-4 (Czech Republic)			Kola-3 (Russia)			Mochovce-1 (Slovakia)			Paks-3 (Hungary)			Rivne-1 (Ukraine)		
	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%
Main Feedwater System Breaks	*1*	2.86E-05	39%															
Steam System Breaks	*2*	2.36E-05	32%	*1*	4.42E-06	31%										6	4.98E-06	6%
Loss of Normal Power	*3*	1.64E-05	22%	*3*	1.73E-06	12%	4	6.93E-06	10%				*1*	5.02E-06	42%	5	6.81E-06	8%
Interfacing LOCA	4	2.44E-06	3%	*2*	4.10E-06	28%				*3*	1.22E-07	13%						
LOCA < 20 mm	5	7.04E-07	1.0%				*1*	1.89E-05	26%	*2*	1.60E-07	18%	*2*	2.14E-06	18%	*2*	1.67E-05	21%
LOCA 20–200 mm	6	4.78E-07	0.7%	5	4.55E-07	3%	*2*	1.64E-05	23%	4	1.21E-07	13%	5	5.89E-07	5%			
Pressurizer Safety Valve LOCA							*3*	1.00E-05	14%	6	5.96E-08	7%	*3*	1.50E-06	13%			
LOCA > 200 mm				4	1.64E-06	11%	5	5.96E-06	8%	*1*	2.77E-07	30%	6	5.48E-07	5%			
Steam Generator Tube Rupture				6	3.98E-07	3%	6	5.68E-06	8%	5	9.82E-08	11%	4	1.12E-06	9%	4	7.16E-06	9%
Steam Generator Collector Rupture																*1*	1.86E-05	23%
General Transients																*3*	1.60E-05	20%
Total Internal-Event CDF:		7.33E-05			1.44E-05			7.21E-05			9.17E-07			1.20E-05			8.06E-05	

2.2.1 Modeling Assumptions and Initiating Events

The differences among the VVER-440/213 plants are discussed in terms of the two categories of events, loss of coolant accidents and transients

Loss of Coolant Accidents

There is a relative degree of consistency in the initiating event frequencies used for large LOCAs, unlike the frequencies used in the PRAs for the VVER-440/230 plants. Interestingly the VVER-440/213 plants all have initiating event frequencies greater by a factor of twenty or more than the lowest value used in the first-generation VVER-440 comparison. At first glance this might imply that the second-generation VVER-440 plants are more likely to have a large LOCA than the earlier plants, but that is not consistent with engineering advances in the second generation. As was pointed out in section 2.1.1 the inconsistent large-LOCA initiating event frequencies for those plants require further examination. The VVER-440/213 results confirm that anomaly.

In the case of the very small LOCAs there is inconsistency among the VVER-440/213 plants beyond that arising from any uncertainty in the data used. It appears that these differences stem from different assumptions concerning this event and not from design differences. These inconsistencies should be addressed, in particular the inclusion of very small loss of coolant accidents above the technical specification for leakage but below the break size that will immediately initiate an automatic scram. This size break does require the reactor to be shut down, cooled down, and depressurized, and hence requires the use of the safety systems.

It is necessary to develop a rational approach for both VVER-440/213 and VVER-440/230 plants for the treatment of all LOCAs over the whole range of break sizes if the PRAs are to be comparable for use in the determination of further design improvements.

Transients

There are four transient groups in which differences in the initiating event frequency needs further explanation.

Primary-to-secondary leakage frequency due to steam generator tube rupture at Rivne and Paks is much greater than at the other plants. It is not clear that there are sufficient design differences to warrant the initiating event frequency difference. Only one of the plants, Rivne, considers an event equivalent to steam generator collector rupture, and it is the highest contributor to core damage. Again this is related to the design of the steam generators, so there needs to be a proper basis for the inclusion or exclusion of this event across the whole class of reactors.

In the case of high-energy line breaks occurring in the steam and feedwater systems inside and outside confinement, there is a factor of 40 difference in the break frequency among the plants, and this initiator is the greatest contributor to core damage in the plant using the highest frequency (Bohunice-V2). This is an initiator that has not occurred at any of the VVER-440/213 units. The design differences are not sufficient to warrant such a wide range of frequencies.

The treatment of loss of offsite (or normal) power at Bohunice-V2 and Dukovany specifically includes failures within the switchyard that will lead to loss of normal power as well as the loss of offsite power. At both these plants the loss of normal power is the third most significant contributor to the CDF. This treatment should be used for all plants. If the higher Bohunice initiating event frequency for loss of power had been used at Kola-3, the core damage frequency for this event would increase significantly.

Although Kola-3 uses an exceptionally low frequency for the loss of main feedwater, this does not affect the overall results for that plant. The remaining transients are not significant contributors to core damage.

2.2.2 Plant Improvements and Conditional Core Damage Probability

The plant improvement comparison is considered for the same categories of events as in the previous section.

Loss of Coolant Accidents

In the case of loss of coolant accidents the design basis for all plants is the large-break LOCA (guillotine rupture of the reactor coolant system loop piping), so there is little difference in the PRA results among the plants, and the contribution to the CDF is never greater than 6E-6/yr. Large LOCAs make the greatest relative contribution at Mochovce-1, where it is 30% of a very low CDF. The largest difference in core damage frequency comes at the other end of the scale, the very small LOCA (0-20mm), as a result of inconsistent modeling of the response to the initiator among the plants.

Transients

In the original design for this VVER class the main steam piping, the main feedwater piping, and auxiliary feedwater piping were collocated immediately outside confinement, which led to a common mode failure in the event of steam line failure. It is clear from the PRA results that the most significant improvement is the relocation of the emergency feedwater piping to an area where it cannot be affected by high-energy line breaks. The impact of this is demonstrated by the difference in the conditional core damage probability for the two plants that have implemented the change and included it in the PRA (Mochovce and Paks) and the two plants that have not implemented the change (Kola and Rivne). When this improvement is combined with the ability to use primary system high-pressure make-up and venting of the primary circuit (feed-and-bleed capability) the contribution to CDF from transients is low (<10% at Mochovce).

The other modifications that together have a major impact are improvements to diesel operation in the event of loss of offsite power and the addition of a mobile diesel generator for use in station blackout conditions. This has proven effective at a number of plants in the U.S.

2.3 VVER-1000 Reactors

In the case of the VVER-1000 design the plants loosely fall into three groups. In the first group are the early plants known as the small-series units, in the second group is the main body of VVER-1000 Version 320 reactors with very similar systems, and in the third group are the modern plants just going into operation. The latter incorporate the majority of modifications proposed for the earlier designs plus extensions to a number of the systems and new automation for some safety functions. The contribution to the total CDF for the top six initiators is shown in Table 2.3-1. The variation among units is a combination of the differences in the modification state and the variation in the initiating frequencies selected by the units.

As there are more similarities among the VVER-1000 plants than there are differences, the spread in CDF (with the exception of Tianwan, the newest plant design) is less than for the VVER-440 plants (Tables 2.1-1 and 2.2-1). Nevertheless, as in the previous plant reviews, variations in the initiating event frequencies and assumptions among plants make it difficult to interpret the exact impact of the modifications. In the case of the VVER-1000 plants this is eased by using a set of reference initiating event frequencies developed at the 2003 workshop to calculate the CDF for each plant. The resulting ranking of the plants enables the impact of the safety system modifications to be determined, as shown in Table 2.3-2. This harmonization of the initiating event frequencies through the use of the reference values shows that the summated effect of the changes between Volgodonsk-1 and Tianwan-1 (including possible lower failure probabilities for equipment) is approximately a factor of 100.

Table 2.3-1. VVER-1000 Rankings of the Most Significant Initiating Event Contributors to Core Damage Frequency

Initiating Event Category	Balakovo-1 (Russia)			Balakovo-5 (Russia)			Kalinin-1 (Russia)			Kozloduy 5 and 6 (Bulgaria)			Novovoronezh-5 (Russia)		
	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%
Loss of Normal Power	*1*	4.22E-05	52%				*3*	3.16E-05	13%	*1*	2.83E-05	37%	4	5.65E-05	8%
Loss of Main Feedwater	*2*	2.39E-05	29%	4	4.14E-06	13%				*2*	1.14E-05	15%			
Steam Generator Collector Rupture	*3*	7.76E-06	10%	*1*	7.49E-06	24%				*3*	1.14E-05	15%			
Steam Generator Tube Rupture	4	3.81E-06	5%	*3*	4.20E-06	14%				4	5.76E-06	8%			
Small LOCA	5	2.48E-06	3%												
General Transients	6	7.53E-07	0.9%	5	4.03E-06	13%	*2*	3.87E-05	16%				*2*	1.16E-04	17%
Very Small LOCA				*2*	7.12E-06	23%									
Large LOCA				6	2.42E-06	8%	*1*	9.58E-05	40%	6	2.80E-06	4%	*3*	1.00E-04	15%
Pressurizer Safety Valve LOCA							4	1.91E-05	8%				5	4.68E-05	7%
Medium LOCA							5	1.80E-05	8%				6	3.20E-05	5%
High-Energy Line Breaks							6	1.09E-05	5%	5	5.41E-06	7%	*1*	2.91E-04	42%
Loss of Service Water															
Total Internal-Event CDF:		8.14E-05			3.11E-05			2.38E-04			7.62E-05			6.87E-04	

Table 2.3-1. VVER-1000 Rankings of the Most Significant Initiating Event Contributors to Core Damage Frequency (continued)

Initiating Event Category	South Ukraine 1 (Ukraine)			Temelin-1 (Czech Republic)			Tianwan-1 (China)			Volgodonsk-1 (Russia)			Zaporizhzhya-5 (Ukraine)		
	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%	Rank	CDF (per year)	%
Loss of Normal Power	*3*	1.60E-05	11%	*3*	2.64E-06	18%	*3*	6.74E-07	20%	*2*	2.08E-05	30%	4	6.71E-06	14%
Loss of Main Feedwater	6	9.50E-06	6%	5	1.73E-06	12%	4	5.29E-07	16%	5	2.30E-06	3%	5	5.72E-06	12%
Steam Generator Collector Rupture	4	1.60E-05	11%	*2*	3.16E-06	21%				4	2.38E-06	3%	*2*	8.57E-06	17%
Steam Generator Tube Rupture										*3*	1.17E-05	17%	6	3.22E-06	6%
Small LOCA				*1*	3.29E-06	22%	*2*	6.77E-07	20%						
General Transients	5	1.20E-05	8%				*1*	8.71E-07	26%	*1*	2.84E-05	41%			
Very Small LOCA	*1*	5.40E-05	35%	6	7.03E-07	5%	5	1.51E-07	5%				*3*	7.89E-06	16%
Large LOCA	*2*	2.00E-05	13%							6	2.22E-06	3%			
Pressurizer Safety Valve LOCA															
Medium LOCA															
High-Energy Line Breaks				4	2.06E-06	14%	6	1.12E-07	3%				*1*	1.30E-05	26%
Loss of Service Water															
Total Internal-Event CDF:		1.52E-04			1.48E-05			3.35E-06			6.92E-05			4.95E-05	

Table 2.3-2. VVER-1000 Rankings of Core Damage Frequency Using a Reference Set of Initiating Event Frequencies

Plant	Base-Case Core Damage Frequency (per year)	Core Damage Frequency Using Reference IEFs (per year)	Comments
Volgodonsk-1	6.92E-05	3.13E-04	88% from loss of feedwater.
Novovoronezh-5 (*Small Series*)	6.87E-04	2.60E-04	50% from loss of normal power and general transients.
Kozloduy-5/6	7.62E-05	1.35E-04	70% from loss of service water and normal power.
South Ukraine 1 (*Small Series*)	1.52E-04	1.03E-04	41% from very small LOCA and 44% from loss of normal power and main feedwater transients.
Kalinin-1 (*Small Series*)	2.38E-04	9.37E-05	50% from loss of normal power and general transients.
Balakovo-1 (*Small Series*)	8.14E-05	8.22E-05	75% from loss of normal power and general transients.
Zaporizhzhya-5	4.95E-05	5.54E-05	
Temelin-1	1.48E-05	2.77E-05	
Balakovo-5	3.11E-05	1.97E-05	
Tianwan-1	3.35E-06	3.18E-06	

2.3.1 Modeling Assumptions and Initiating Events

The differences among the plants are discussed in terms of the two categories of events, loss of coolant accidents and transients.

Loss of Coolant Accidents

Considering the lack of actual events there is relative consistency in the frequency of large and medium LOCAs for all the units, with only factors of 13 and 10 for the range of the two events. It is interesting to note that there is a relatively larger degree of inconsistency in the initiating event frequencies used for the small LOCAs (25), but for the very small LOCA the range is again a factor of 13. LOCAs in these two categories, however, have occurred in light water reactors, including VVERs, so the data need to be collected and used consistently in the PRAs. This is a difference between the PRAs for these plants compared with the PRAs for the first-generation VVER-440 plants that were examined. More interesting still is that the VVER-1000 plants (even Tianwan) all have large-LOCA initiating event frequencies greater by a factor of *fifty* or more than the lowest value seen in the first-generation VVER-440 comparison. At first glance this might imply that the very latest VVER-1000 plants are more likely to have a large LOCA than the earlier plants. That is not consistent with engineering advances in the most recent generation of VVERs. As was pointed out in section 2.1.1 the inconsistent large-LOCA initiating event frequencies for the VVER-440/230 plants requires further examination. The VVER-1000 results confirm that anomaly.

The inconsistency in the treatment of interfacing system LOCAs and the lack of detailed information available at the workshop made it impossible to compare the frequency of the interfacing system LOCAs. A consistent approach should be adopted in all PRAs.

Transients

The first subgroup considered in the transient category is primary-to-secondary leakage due to steam generator tube rupture. It is understood that this event has not occurred in any horizontal steam generator used in VVER plants. Leakage has occurred, but always within the capacity of the normal make-up system. For this event South Ukraine uses a value (4.6E-2 per year) five times greater than the next-highest frequency. Tianwan-1, taking into account plant experience and improved steam generator design, uses a value of 6.6E-4 per year. If these two extreme values are disregarded, the other plants fall within a factor of seven range, which shows close agreement.

The second subgroup consists of the high-energy line breaks occurring in the steam and feedwater systems inside and outside containment. There is a factor of 106 difference in the break frequency among the plants, and this initiator is the greatest contributor to core damage at two of the plants. Failure of steam generator safety and relief valves is a contributor that has occurred at a number of plants. It is considered that the determination of an appropriate frequency for a given unit can be established only when event data have been collected and analyzed. Major pipe failures have not occurred at any of the VVER-1000 plants. The design differences seem insufficient to warrant such a wide range of frequencies.

In the third subgroup, transients in which the primary circuit remains intact, the widest discrepancy is in the treatment of loss of main feedwater and service water. For some reason the value for loss of feedwater used by Volgodonsk-1 is an order of magnitude lower than at Temelin-1, and two orders of magnitude lower than at the other plants. Again, these events have occurred at plants, so the collection of data would enable a generic basis that should be used by all plants.

The ratio in the range of loss of service water frequency is skewed by the approach used by Khmelnytskyy-1 to derive the frequency. The approach that should be adopted is the use of a fault tree to establish the frequency, so taking into account plant-specific features.

2.3.2 Plant Improvements and Conditional Core Damage Probability

The use of a reference set of initiating events to determine the core damage frequencies in Table 2.3-2 eliminates one of the uncertainties in the PRA result comparison. Tianwan-1 essentially embodies all the potential modifications identified to date plus some fundamental changes in design such as the introduction of a fourth train of safety systems and complete train separation on both the safety system and steam/feedwater arrangements. It could be concluded, then, that the net impact of all the changes is an effective reduction factor of 100 in the conditional core damage probability.

It seems anomalous that in applying reference initiating event frequencies Volgodonsk-1 is ranked highest with 80% of the CDF coming from loss of feedwater transients. This is one of the more recent plants and has both emergency and auxiliary feedwater systems. Nevertheless, there is no indication that the modification for feed-and-bleed capability is planned. It is known that there are still conservative assumptions in the Kozloduy-5/6 PRA, which may explain why it is ranked higher than other VVER-1000/320 units.

The impact of modifications in the systems that have most impact on reductions in the conditional core damage probability for the various groups of initiators is discussed in the following sections.

Loss of Coolant Accidents

There are three modifications that affect the differences in the contribution to CDF from loss of coolant accidents. These are the ability to connect the high-pressure pumps to the containment sump, the ability of the pump to operate when primary system pressure is below 40 kg/cm^2, and modifications to the sump strainers to prevent clogging. The completion of these upgrades in the plants that do not have them will reduce the conditional core damage probability for LOCAs to the same level as the more recent plants that have the first two features as part of the original design. The first of these modifications means that either the high-pressure or low-pressure systems can provide decay heat removal in the initial phases of the accident, whereas without it both the high- and low-pressure systems are required for small and very small LOCAs, which have the highest frequency.

Transients

The contribution from transients in which the primary circuit remains intact is determined by the number of feedwater systems available (auxiliary and emergency) and the ability to achieve feed-and-bleed cooling through the high-pressure injection system with venting from the primary circuit.

In the event of loss of normal power, the introduction of additional diesel generators to support the auxiliary feedwater system enhances safety. The addition of a mobile diesel generator for use in station blackout conditions has proven effective at a number of plants in the U.S.

The other modifications that are important from both the initiating event perspective and the capability to cope with primary-to-secondary leakage are those associated with improvements to steam generator valves, enabling them to pass steam/water mixtures without failing open.

3. Conclusions and Recommendations

It is clear from the workshop summaries in section 2 that it is possible to use the PRAs to determine the effectiveness of plant improvements and identify the impact of further improvements. For those plants that have embraced the use of safety analysis over the past decade there has been a consistent reduction in risk. This has been achieved by the performance of PRA and the evaluation of what modifications will most effectively reduce the risk to an acceptable level.

The majority of plants have (quite rightly) concentrated initially on improvements to systems that have a direct impact on the achievement of the primary safety functions of inventory control and decay heat removal for events occurring when the rector is at power. Now that this has been achieved in the majority of plants, the workshop results showed that there are differences in the assumptions made in the frequency of initiating events that are not justified by plant differences. In addition it was shown that there are differences in data sources used because of a lack of data collection for VVER plants and differences in the extent and quality of the human reliability modeling.

Table 3-1, which compares the scope of completed PRA activities at each of the plants, shows that there are still three areas with major potential for impact on risk (based on experience at older plants in the U.S. and at the Temelin VVER-1000 in the Czech Republic) that have not been addressed. These are:

- Fire and internal flooding probabilistic analyses;
- PRA for all low-power and shutdown modes of operation;
- Containment analysis and level-2 PRA.

At many older plants the risk calculated from the first two analyses may exceed those calculated for equipment failures or operator mistakes leading to plant trip. Continuing the safety analyses in these areas are likely to lead to requirements for additional plant modifications.

The success of the NNSA INSP program in ensuring that the relationship between risk and the implementation of plant modifications was amply demonstrated at the four workshops held over the past year. For each reactor design the impact of the plant improvements implemented on the reduction of risk was clearly identified. In addition those plants that had not made certain improvements could see the impact of making such improvements. This is particularly the case for the VVER/440-230 unit in Armenia.

Eight years ago, there were no effective PRAs for any VVER-440/230, VVER-440/213, or VVER-1000 unit. The ability to perform such analysis was limited to a small number of research establishments that understood the theory, but had little or no plant experience. Further, there was no experience within the regulatory bodies of what constituted an adequate PRA or why one should be performed. The NNSA/IAEA workshops in 2002 and 2003 showed that at many plants personnel now understand and, in some cases, are actively performing and maintaining living PRAs in the same way as was started in the U.S. some ten years ago. They have started to develop the necessary safety culture.

The necessity now is to move forward to the next stage where the PRA is considered to be good enough to be used on a day-to-day basis as it is in the U.S. This is a two-stage process. First the PRA must be completed. That is, it must cover all events that can occur, both internal and external to the plant, and the resulting frequency and magnitude of radiological release must be quantified. Second the PRA model must be converted for on-line use by all plant personnel, not just the PRA specialists. This is the current status in a number of countries. Only then can management have confidence that the risk is being adequately managed in the daily operation, inspection, and maintenance of the plant.

The U.S. is the most advanced practitioner in the use of safety analysis methodologies. It was the existence of such analytical tools that enabled the adoption of symptom-based emergency operating instructions (EOIs) in the U.S. and the export of this technology to other countries with light water reactors. It was the U.S. that was the first to use level-1 and level-2 PRAs to extend the emergency procedures to severe accident management guidelines to minimize harm after a core-damage event. The INSP is allowing this core of knowledge to be put to good effect in those countries which have just started out on this process (Ukraine, Armenia, and Russia) in a timescale ¼ that of the time it took to develop and implement these standards in the U.S.

The current status in Ukraine is equivalent to the U.S. approximately 15 years ago; Armenia is still further behind. The status of what has been achieved against what is left to do is shown graphically in the workshop reports (reproduced in part as table 3-1a-c in this summary report).

The recent NNSA sponsored VVER-440 and 1000 workshops to determine the impact of the plant modification status on risk have shown that there are differences in the PRA results that are not the result of plant design or modification status. The same was true in the past among plants in the U.S. This was resolved in the U.S. by the development of PRA standards and PRA certification by specialist groups for each reactor type. This is an essential step in the process towards risk-informed regulation.

Workshops of the type reported on in this document are an essential element of this process. They enable the countries that have advanced in the development of their safety culture (Czech Republic, Slovakia, and Bulgaria) to share their information and experience with all plants and ensure that unresolved issue for each class of reactor are dealt with from a common basis. This not only expedites resolution of such issues, but enables the work to be done at lower cost, ensuring that limited funds are directed to the areas most in need.

Specific areas in which there is still much work to be done are the evaluation of operator performance, the upgrading of procedures, and the development of accident management guidelines. The safety analysis required to support this work is common to all plants and beyond the capacity of individual units.

Table 3-1a. Scope of PRA Studies—VVER-440/230 Reactors

Plant	Year	Level-1										Level-2										Level-3	
		Full Power					Low Power and Shutdown					Full Power					Low Power and Shutdown						Comment
		Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events		
Kozloduy Unit 3 (Bulgaria)	2002	C	C		C		C	C		C													Other events include aircraft crash, severe weather conditions, pipe whip, missiles, jet impingement, and affect of surrounding industry.
Kola (Russia)	2003 (Unit 1)	C																					
	2002 (Unit 2)	C																					
Armenia Unit 2	2003	O																					
Novovoronezh (Russia)	2002 (Unit 4)	O																					
	2001 (Unit 3)	C		O		O	C																
J. Bohunice Unit V1 (Slovakia)	2002	C	C	C	C *	C	C	C	C	C	C *	O	O	O	O	O	O	O	O	O	O		* Simplified analysis of seismic events was performed.

C – analysis is completed
L – "living" PRA is implemented
O – analysis is ongoing

Table 3-1b. Scope of PRA Studies—VVER-440/213 Reactors

Plant	Year	Level-1										Level-2										Level-3	Comment
		Full Power					Low Power and Shutdown					Full Power					Low Power and Shutdown						
		Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events		
Kola Unit 3 (Russia)	2002 (Unit 3) 1999 (Unit 4)	C C	C		C		C	C		C													
Paks Units 1-4 (Hungary)	2002	C L	C L	C L	O		C					O											
Rivne Unit 1 (Ukraine)	2002	C	O	O																			The PRA is currently under in-country peer review.
Dukovany Units 1-4 (Czech Republic)	2002	C L	C L	C L			C L	C L	C L														Limited scope of level-2 PRA.
J. Bohunice V2 Unit 3 (Slovakia)	2003	C L	C	C			C	C	C			C	C	C									
Mochovce Unit 1 (Slovakia)	2002	C	C	C	C	C	O	O	O	O	O												

C – analysis is completed
L – “living” PRA is implemented
O – analysis is ongoing

Table 3-1c. Scope of PRA Studies—VVER-1000 Reactors

Plant	Year of PRA-1	Level-1										Level-2										Level-3	Comment
		Full Power					Low Power and Shutdown					Full Power					Low Power and Shutdown						
		Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events		
Kozloduy-5/6 (Bulgaria)	2003	C	C	C	C		O	O	O			O	O	O									Level 2 PSA is planned to be completed in 2004. LSPSA is planned to be completed in 2005.
Tianwan-1 (China)	2002	C					O																PRA performed for PSAR for plant under design..
Kudankulam (India)	2001	C	C																				Performed at PSAR stage of design development.
Bushehr (Iran)	2003	C	O		O		C					O											To be finished in 2003.
Temelin-1 (Czech Republic)	2002	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L	C L							Level 1and 2 (power) and Level 1 shutdown in safety monitor.
Balakovo-1 (Russia)	1998	C	C				C																
Balakovo-5 (Russia)	2002	C	C				C					C											Performed at PSAR stage of design development.
Kalinin-1 (Russia)	1999	C										O											Internal fire and flooding PSA are performed in limited scope. For seismic study walkdown was performed.

Table 3-1c. Scope of PRA Studies—VVER-1000 Reactors

Plant	Year of PRA-1	Level-1										Level-2										Level-3	Comment
		Full Power					Low Power and Shutdown					Full Power					Low Power and Shutdown						
		Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events	Internal events	Internal fire	Internal flood	Seismic	Other external events		
Novovoronezh-5 (Russia)	1999	C	C	C		C	O	O	O		O	C	C	C		C						C	For seismic study walkdown was performed.
Volgodonsk-1 (Rostov-1) (Russia)		C	O				O																
Khmelnytskyy-1 (Ukraine)		O																					Follow on PRA from Zaporizhzhya Unit 5.
Khmelnytskyy-2 (Ukraine)		O																					
Rivne-4 (Ukraine)		O																					
South Ukraine-1	1999	C	O	O																			
Zaporizhzhya-5 (Ukraine)	2001	C	O	O																			

C – analysis is completed
L – "living" PRA is implemented
O – analysis is ongoing