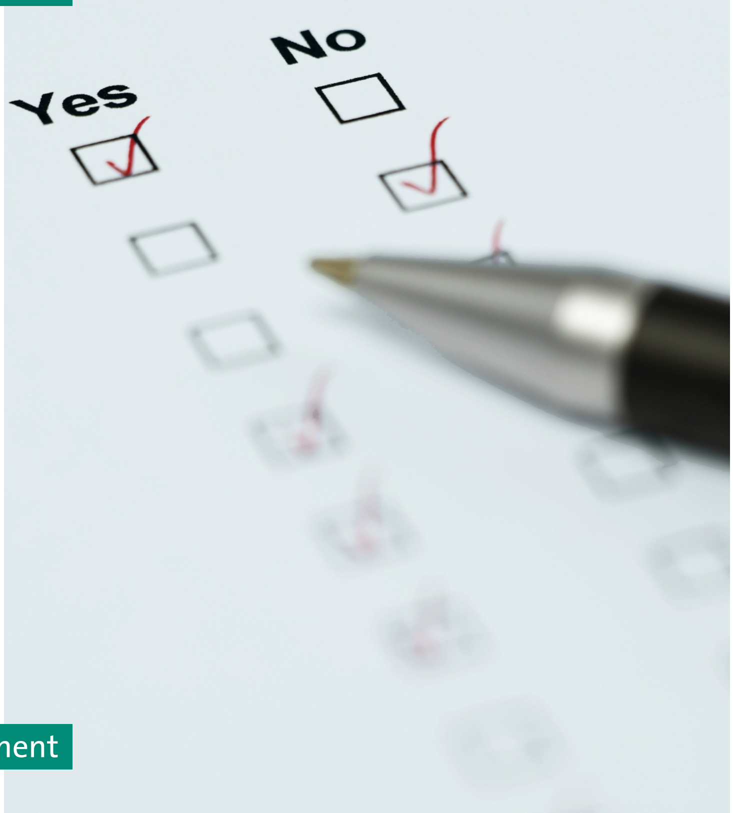
# Construction of a NPP in Belarus




Expert Statement

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ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

ENVIRONMENT AGENCY AUSTRIA umweltbundesamt

# CONSTRUCTION OF A NPP IN BELARUS

## Expert Statement on the Preliminary EIA Report

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# CONTENT

# ZUSAMMENFASSUNG

Die weißrussische Regierung hat sich entschieden, ein Kernkraftwerk mit einer Kapazität von 2.300–2.400 MWe zu errichten. Österreich beteiligt sich am grenzüberschreitenden UVP-Verfahren. Das Umweltbundesamt hat das Österreichische Ökologie-Institut gemeinsam mit Dr. Helmut Hirsch, wissenschaftlicher Konsulent, beauftragt eine Fachstellungnahme zur weißrussischen Umweltverträglichkeitserklärung (UVE) auszuarbeiten.

Die Begutachtung der UVE konzentriert sich auf die Sicherheits- und Risikoanalyse, mit dem Ziel festzustellen, ob die UVE dazu geeignet ist, belastbare Aussagen über die möglichen Auswirkungen grenzüberschreitender Emissionen zu machen.

Aufgrund einer gründlichen Analyse verfügbarer Reaktorblöcke entschied sich Weißrussland für das Projekt AES-2006, einen Druckwasserreaktor (DWR) russischer Bauart der 3. Generation mit einer elektrischen Leistung von 1.200 MWe (VVER-1200/V491 im folgenden  VVER-1200 ). In der UVE werden einige andere Reaktortypen als alternative Optionen mit dem AES-2006 verglichen. Dieser Vergleich verschiedener DWR-Typen zeigte keine signifikanten Unterschiede hinsichtlich der Häufigkeit der Kernschäden (CDF/core damage frequency) und der Häufigkeit großer Freisetzungen (LRF/large release frequency). Für den VVER-1200 wurde in einem kürzlich erschienen Artikel eine LRF von < 1 E-7/a angegeben (NEI 2009), das 10-fache der LRF in der UVE (1 E-8/a). Grundsätzlich ist zu beachten, dass Ergebnisse von PSA (probabilistischen Sicherheitsanalysen) nur als grobe Risiko- Indikatoren zu betrachten sind.

Die UVE gibt an, dass die Erfahrung mit den Komponenten und Sicherheitssystemen in Vorläufer-Reaktoren ein wesentlicher Grund für die Auswahl des VVER-1200 war. Allerdings besteht bisher keinerlei Erfahrung aus dem Betrieb des VVER-1200. Verglichen mit anderen neuen DWR-Typen weist der VVER-1200 keine wesentlich größeren Erfahrungen auf. Die meisten neuen Reaktortypen sind Weiterentwicklungen bestehender Modelle, wie z. B. der französische N4 und deutsche Konvoi DWR Vorgängermodelle für den EPR von Areva sind.

In der UVE werden etliche Grundlagen zum Reaktorkonzept und zu Betriebsparametern angeführt. Anderseits fehlt eine detaillierte Beschreibung der in der UVE aufgezählten Sicherheitssysteme. Daher ist es nicht möglich, sich aus der UVE einen ausreichenden Überblick über Funktion und Zuverlässigkeit dieser Systeme zu verschaffen.

Die Widerstandsfähigkeit des VVER-1200 gegen äußere Einwirkungen (die wesentlich von der Dicke der Wände des Containments abhängt) ist – nach Angaben in der UVE – in mancher Hinsicht geringer als bei modernen Druckwasserreaktoren der Generation II.

Mehrere technische Merkmale werden in der UVE als neu angeführt, die in vielen Reaktoren der Generation II längst vorhanden sind (z. B. zwei voneinander unabhängige Systeme zur Regelung der Reaktivität). Daneben gibt es im Vergleich zu Generation II Reaktoren einige völlig neue Systeme. Das bedeutendste ist die Einrichtung zur Eingrenzung des schmelzenden Reaktorkerns, in der englischsprachigen Literatur als “core catcher” bezeichnet. Allerdings fehlt in der UVE eine Beschreibung dieses core catchers. Wenn dieses System wie geplant funktioniert, hätte es das Potential zur Verringerung der Eintrittswahrscheinlichkeit großer Freisetzungen radioaktiver Stoffe bei schweren Unfällen. Allerdings gibt es keine Garantie, dass dieser Zweck erreicht wird, da vorerst noch eine Reihe ungelöster Probleme zu klären ist (z. B. die Gefahr von Dampfexplosionen).

Die quantitativen probabilistischen Ziele scheinen vom VVER-1200 erfüllt. Allerdings bleibt in der UVE ungeklärt, ob die CDF- und LRF-Werte tatsächlich alle Betriebszustände des Reaktors beinhalten (Volllast, Teillast und Stillstand), und ob alle auslösenden Ereignisse (intern und extern) berücksichtigt sind.

Hinsichtlich der allgemeinen Sicherheitsstandards ist es wichtig festzuhalten, dass der AES 2006 eine Weiterentwicklung des AES-92 ist, der ein European Utility Requirements (EUR) Zertifikat hat. Deshalb ist es plausibel anzunehmen, dass auch der AES-2006 die EUR erfüllt. In der UVE werden hingegen die Sicherheitsmerkmale und das Multi-Barrieren-Konzept so allgemein beschrieben, dass dies auch von vielen Generation II-Reaktoren erfüllt wäre.

Die Informationen zu Unfällen sind in der UVE über viele Kapitel verstreut. Es fehlt eine systematische Analyse der Auslegungs- und darüber hinausgehenden Unfälle (DBA und BDBA[1]). Für schwere Unfälle werden in der UVE mehrere Quellterme angegeben, allerdings ohne Beschreibung der auslösenden Ereignisse und des Unfallablaufs. Für das KKW Novovoronesh II werden in de UVE zwei Unfallszenarien präsentiert, aber ohne Details und ohne Quellenangabe. Es ist unklar ob die Quellterme aus deterministischen oder Wahrscheinlichkeitsanalysen stammen. Dasselbe gilt für das „worst case“ Unfallszenario in Kapitel 5 der UVE zu grenzüberschreitenden Auswirkungen.

Die Schlussfolgerung der UVE, dass keine größeren Quellterme auftreten könnten als die in der UVE dargestellten beschränkten Emissionen, ist nicht ausreichend begründet. Für alle existierenden Reaktoren und auch für die in Bau befindlichen neuen Generation III-Reaktoren können Unfälle mit einer Freisetzungsrate von einigen Prozent (2–20 %) des Cäsium-137 Inventars des Reaktors nicht ausgeschlossen werden. Auch wenn die Eintrittswahrscheinlichkeit für einen Unfall mit großen radioaktiven Emissionen in der PSA sehr klein erscheint, sollten Quellterme schwerer Unfälle in einem grenzüberschreitenden UVP-Verfahren berücksichtigt werden.

Hinsichtlich des Managements radioaktiver Abfälle werden in der UVE lediglich die Volumen fester und flüssiger Abfälle angegeben, Angaben zum radioaktiven Inventar fehlen. Das System zur Behandlung der radioaktiven Abfälle wird dargestellt, aber ohne Details. In der UVE wird weder ein Zwischenlager für den abgebrannten Brennstoff noch Pläne für ein Endlager für radioaktiven Abfall in Weißrussland erwähnt.

Hinzuzufügen ist noch eine allgemeine Bemerkung: Manche Teile des Textes der UVE sind schwer verständlich, wahrscheinlich liegt das an Schwierigkeiten bei der Übersetzung vom (Weiß)Russischen ins Englische. Die Übersetzung mancher Ausdrücke ist irreführend oder unverständlich, zum Beispiel:

“максимальной проектной аварии “ = design basis accident, statt “maximum projected damage” (hier geht es um den Auslegungsstörfall)

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[1] DBA = design basis accident, Auslegungsstörfall; BDBA = beyond design basis accident, auslegungsüberschreitender (schwerer) Unfall

“Предельный аварийный выброс” = limit for severe accident release; die Phrase “maximum emergency emission” ist unklar. Hier stellt sich die Frage, was mit der Übersetzung wirklich gemeint ist: ein Grenzwert oder die größte denkbare Freisetzung, und ob überall im Text auch das gleiche gemeint ist.

# SUMMARY


The government of Belarus decided to construct a nuclear power plant (NPP) with a capacity of 2,300–2,400 MWe. Austria takes part in the transboundary Environmental Impact Assessment (EIA) for the construction of the NPP in Belarus. The Environmental Agency Austria, "Umweltbundesamt" has assigned the Austrian Institute of Ecology, in cooperation with Dr. Helmut Hirsch, scientific consultant, to elaborate an expert statement on the EIA Report presented by Belarus.

This review of the EIA Report is focused mainly on the safety and risk analysis. The goal is to assess if the EIA allows making reliable conclusions about the potential impact of transboundary emissions.

As a result of comprehensive analysis of industrial reactor units, the Russian project NPP-2006, a Generation III Russian Pressurized Water Reactor (PWR) with 1,200 MWe (VVER-1200/V491 further VVER-1200) was chosen for the Belarussian NPP. In the EIA some other types were compared to this reactor as alternative options. The PWR types considered in the report do not display significant differences regarding core damage frequency (CDF) and large release frequency (LRF). For the VVER-1200, a large release frequency of <1 E-7/a is given in a recent article (NEI 2009), one order of magnitude higher than the value in the EIA Report. Results of Probabilistic Safety Assessments (PSA) in any case should only be taken as rough indicators of risk.

Experience with equipment and safety systems in prototype units was according to the EIA Report the main reason to select VVER-1200. However, no operating experience has been gained so far in proper VVER-1200 prototypes. Thus, compared to other PWR types, experience relevant for VVER-1200 is not significantly more comprehensive. For most new reactor types, there are earlier models from which they are developed, for example the French N4 and the German Konvoi PWRs for the Areva EPR.

While a number of basic data concerning reactor design and operational parameters are provided in the EIA Report, there is no detailed description of the safety systems which are mentioned. It is not possible to gain a comprehensive picture of the functioning and reliability of those systems.

The resistance of the VVER-1200 against external impacts (which depends to a considerable extent on the wall thickness of the containment building) as specified in the EIA Report is, in some cases, inferior to that of modern Generation II PWRs.

Several technical features are presented as new in the EIA Report which already are implemented in many currently operating Generation II plants (for example two fully independent reactivity control systems). But there are also some new features compared to Generation II plants. Most notable is the corium localization device usually referred to as "core catcher" in the English literature. There is no description of the core catcher in the EIA Report. If this device is functioning as planned, it would have the potential to reduce the probability of large releases in case of a severe accident. However, there is no guarantee, to date, that it will indeed fulfill its purpose because a number of problems have not been sufficiently clarified so far (for example the hazard of steam explosions).

The quantitative probabilistic targets appear to be fulfilled by the NPP-2006. However, it is not entirely clear from the EIA Report that the CDF and LRF values provided really include all plant states (full power, low power and shutdown) and all initiators (internal and external). Regarding general safety requirements, it is important to note that the NPP-2006 was developed from NPP-92, which is certified by European Utility Requirements (EUR). Thus, it is plausible that NPP-2006 also fulfills the EUR. But the main safety characteristics and the concept of multiple barriers are described only in a very general manner. In this form, they apply to many operating NPPs of Generation II.

Information concerning accidents in the NPP is distributed over different parts of the EIA Report. There is no systematical analysis of design basis (DBA) and beyond design basis accidents (BDBA). Several BDBA source terms are presented, but without description of the initiating events and the progress of the emergency situation. Two Novoworonesh NPP severe accident scenarios are described without details and without reference. Further it is unclear whether the source terms are derived from deterministic or probabilistic assessments. That is also the case for the “worst case” BDBA emission scenario in Chapter 5 “Transboundary impact”.

The conclusion of the Report that no greater source terms than the presented limited releases could occur is not sufficiently substantiated. For all existing reactors and also for the new Generation III reactors now under construction severe accidents with a release in the range of some percent of the radioactive Cesium inventory (2–20%) are not excluded. Even if the frequency of occurrence of accidents with a large release appears very small according to PSA, such severe accident source terms should be considered in the transboundary EIA.

Regarding waste management, only the volumes of solid and liquid radioactive wastes are provided, there is no information on the radioactive inventory.

The radioactive waste handling system is described without details. No interim storage for the spent fuel and no plans for radioactive waste disposal in Belarus are mentioned in the EIA Report.

There is one general remark to make: Some parts of the text are difficult to understand, probably due to translation problems. The translation of some phrases is misleading or incomprehensible: e.g.

“максимальной проектной аварии“ = design basis accident, instead of “maximum projected damage”

“Предельный аварийный выброс” = limit for severe accident release; the phrase “maximum emergency emission” is unclear: It could mean this is the largest conceivable emission or this is the maximum emission permitted.

# 1 INTRODUCTION

The government of Belarus decided to construct a nuclear power plant (NPP) with a capacity of 2,300–2,400 MWe.

As a result of comprehensive analysis of industrial reactor units, the Russian project NPP-2006 of the Generation III VVER was chosen for the Belorussian NPP. The government of Belarus is convinced that this project conforms to modern international nuclear safety and radiation protection requirements: *"The advantage of NPP-2006 project is that in comparison with other projects the main equipment and security systems of NPP are already tested on operating NPPs. The nearest prototype of NPP-2006 project was putted into revenue service in 2007 in China (2 power supply units). 2 units upon the Russian third generation projects are in construction process now in India. The construction of 2 units in Bulgaria and 4 units in Russia is already started."* (REPORT 2009, p. 38)

Another advantage of the Russian offer is that Russia not only delivers the nuclear fuel for the NPP but that the used fuel could be returned to Russia for long-term storage and reprocessing at the territory of Russian Federation.

In the Environmental Impact Assessment (EIA) some other types were compared to NPP-2006 as alternative options (REPORT 2009, p.31, table 5 & 6).

For the construction of the NPP some governmental institutions have been established in the Republic of Belarus:

- The state company "The Directorate of Nuclear Power Plant Construction" for carrying out the customer's functions of fulfilling the complex of preparative, design and survey works on nuclear power plant construction
- The "Department of Nuclear and Radiation Safety" for performance of the state supervision in the field of the nuclear and radiation safety provision in the Ministry of Emergency Situations.
- The Design Scientific-Research Republican Unitary Enterprise "Belnipienergoprom" has been determined as the general designer for coordination of the design and estimate documentation for construction of the NPP.

The site selection procedure started with 74 potential locations in the Republic of Belarus. From these the three most promising locations were selected for detailed evaluation. By the end of 2008 the potential sites at the locations have been selected:

- Mogilyov region: Bykhov location, Krasnaya Polyana site
- Mogilyov region: Shklov-Gorki location, Kukshinovo site
- Grodno region: Ostrovets location, Ostrovets site

In the EIA Report tables of comparison for these sites are presented (REPORT 2009, p.15 ff, table 1–3).

Austria takes part in the transboundary EIA for the construction of a NPP in Belarus. The Environmental Agency Austria, "Umweltbundesamt" has assigned the Austrian Institute of Ecology, in cooperation with Dr. Helmut Hirsch, scientific consultant, to elaborate an expert statement on the EIA Report presented by Belarus.

The expert statement is referring to the EIA Report "Substantiation of investments in construction of the nuclear power plant in the republic of Belarus – Environmental impact assessment" as (REPORT 2009).

The review of this EIA Report is focused mainly on the safety and risk analysis. The goal is to assess if the EIA allows making reliable conclusions about the potential impact of transboundary emissions. For that safety features, equipment and procedures for severe accident management should be explained in detail. Results of Probabilistic Safety Assessment (PSA) should be presented in such detail that the conclusions are transparent and consistent.

# 2 SELECTION OF THE NPP TYPE

## 2.1 Treatment in the EIA Report

Three basic reactor types were under consideration for the Belarus project: PWR, BWR and HWR. The reactor types are briefly and generally described. The PWR was given preference for the following reasons:

- Maximum power density in the core; hence, minimum size per unit of power
- Double-circuit structure providing localization of radioactive equipment, with protective containment
- Minimum dose burden during maintenance

Several NPP types with PWR are considered as alternatives: AP-600, AP-1000, EPR and VVER (variants VVER-1000/V412 and V428, as well as NPP-2006 (VVER-1200/V491). Core damage frequencies (CDF) and large release frequencies (LRF) for AP-600, AP-1000, EPR and NPP-2006 are listed; they are all in the same order of magnitude (CDF from <1 E-7 to <5.8 E-7/a, and LRF from <1 E-8 to <6 E-8/a). It is emphasized that all those NPPs fulfill IAEA guidelines, European Utility Requirements (EUR) as well as national norms.

From this short list, the NPP-2006, a Generation III Russian PWR, has been selected (reported CDF<5.8 E-7/a, LRF <1 E-8/a).

This selection is stated to be the result of a comprehensive industrial analysis. This analysis is not presented, however, in the EIA Report. It is only briefly mentioned that the advantage of the NPP-2006 is that, in comparison to other NPPs, the main equipment and safety systems are already tested in operating NPPs. Two prototype units are in operation in China, and eight more units are under construction in India, Bulgaria and Russia.

## 2.2 Assessment

The selection of PWR as the basic reactor type is not particularly remarkable. More than half of all operable commercial power reactors worldwide belong to this type, as well as more than three quarters of reactors under construction (as of beginning of 2009 (WNIH 2009)).

It can be observed, however, that a high power density – which is given as one of the reasons for selecting the PWR – tends to be disadvantageous from a safety point of view. It implies a high density of decay heat, and hence rapid heating up of the reactor core in case of loss of coolant. On the other hand, the small core size which is a consequence of high power density can be economically advantageous.

The PWR types considered in the report do not display significant differences regarding CDF and LRF; the variation is only a factor of six in both cases. Mostly, the PSA results listed in the EIA Report correspond to other values to be found in the published literature. However, for the VVER-1200, a large release frequency of <1 E-7/a is given in a recent article (NEI 2009), one order of magnitude higher than the value in the EIA Report. For the EPR, a CDF which is higher by a factor of about four to five is given in other sources (UMWELTBUNDESAMT 2008).

PSA results in any case should only be taken as very rough indicators of risk. It is problematic to compare results of different studies where different methodologies might have been applied. Furthermore, all PSA results are beset with considerable uncertainties; and there are factors contributing to NPP hazards which cannot be included in PSAs. (The issue of PSA results is taken up again in the section on project targets, criteria etc.)

Experience with equipment and safety systems in prototype units was according to the EIA Report the main reason to select VVER-1200/V491. However, no operating experience has been gained so far in proper VVER-1200 prototypes. There is operating experience from an earlier model, the VVER-1000/V428 (an advanced version of VVER-1000); two units are in operation since 2007 at Tianwan (China).

Other advanced VVER-1000s are under construction in India (VVER-1000/V392, Kudankulam-1 and -2). Whether two units of VVER-1000/V466 will be built at Belene, Bulgaria, has not been definitely decided yet; the financing of those reactors remains unclear. Three VVER-1200 are under construction in Russia, due to go into operation 2012/13 (WNIH 2009).

Thus, compared to other PWR types, experience relevant for VVER-1200 is not significantly more comprehensive. For most new reactor types, there are earlier models from which they are developed; for example, the French N4 and the German Konvoi PWRs for the Areva EPR.

The Westinghouse-Toshiba AP600 and AP1000 also mentioned in the EIA Report are more innovative reactor types. Experience with earlier types is less applicable in their cases. On the other hand, they rely more on passive safety features than both VVER-1200 and EPR which is generally seen as an advantageous feature.

It is also interesting to note that the AP1000 has NRC certification as well as EUR certification. The EPR is EUR certified, the NRC certification process is ongoing. NPP-2006 (VVER-1200) has neither EUR nor NRC certification. Only the earlier model NPP-92 (VVER-1000/V446) has been EUR certified (KRAEMER 2008).

It was reported in a technical magazine that there are two variants of VVER-1200, V-392M and V 491. Passive safety systems prevail in the former, whereas the latter focuses more on active systems (NEI 2009). There is no discussion in the EIA Report why V-491 was chosen and not V-392M.

## 2.3 Questions

1. Can the reasons for the choice of the reactor type (VVER-1200) be explained in more detail – regarding experience with components and systems, and possible other reasons?
2. What were the reasons for selecting the variant V491 and not V392M, and hence relying more on active, than on passive safety systems?

# 3 DESCRIPTION OF PROJECT

## 3.1 Treatment in the EIA Report

The design service life of the NPP is 50 years. (A service life of 60 years is claimed for the reactor pressure vessel.)

The following differences between NPP-2006 and existing NPP projects with PWRs are claimed:

- Two fully independent reactivity control systems
- Long-time residual heat removal by active and passive safety systems
- Double protective containment

Apart from an active emergency core cooling system, a high-pressure passive boron injection system is described. Also a system of passive heat removal from the steam generators (complementing the active emergency feedwater system which is not explicitly mentioned, but existent at every PWR) via water-cooled heat exchangers and a passive filtration system for the space between the two containment hulls are mentioned.

Also mentioned are an automated process control system, fasteners for protection of equipment and pipelines against dynamic loads and a system for monitoring equipment and pipeline conditions.

Systems for controlling and mitigating in case of a severe accident (BDBA) are referred to, in particular a corium localization device (system for core melt trapping and cooling). A further description of those systems is lacking.

Basic reactor parameters and specifications are listed (for example reactor coolant flow, pressure and temperatures, fuel burn-up and residence time in the reactor). The expected capacity factor is given as about 96%. Also, the main objects of the NPP and the main equipment are briefly listed and the general layout is briefly described.

The reactor building is designed to withstand the following external impacts:

- Airplane crash (not specified further)
- Snow loads (0.6 kPa)
- Earthquakes (intensity 7, ground acceleration 0.25 g)
- Hurricanes, whirlwinds, tornadoes (F3 – F4)
- External explosions (10 kPa, 1 s)
- Floods

In case of a severe accident, emergency evacuation is planned in a zone with a radius of 800 m; this zone lies almost completely within the NPP territory.

The handling of fresh and spent fuel is briefly described. The spent fuel pools in the reactor buildings will have a capacity for ten years' spent fuel arisings, plus the full core (in case of emergency unloading). After three years, the spent fuel can be removed from the pool for reprocessing or long-term storage.

## 3.2 Assessment

While a number of basic data concerning the reactor design and operational parameters are provided in the EIA Report, there is no detailed description of the safety systems which are mentioned. It is not possible to gain a comprehensive picture of the functioning and reliability of those systems.

The capacity factor provided in the report (96%) appears to be very optimistic. An expected capacity factor of up to 90% has been recently reported in a technical article (NEI 2009).

Several technical features are presented as new in the EIA Report which already are implemented in many currently operating Generation II plants.

It is general practice for PWRs and all other reactor types to be equipped with two fully independent reactivity control systems; indeed, this is required by IAEA guidelines (IAEA 2000, 2005).

The passive high-pressure boron injection system mentioned in the EIA Report appears to be equivalent to the passive borated water flooding tanks (hydraulic accumulators) being part of the Emergency Core cooling System (ECCS) in Western PWRs of Generation II. They are set to inject at intermediate pressure (in German PWRs, for example, typically about 27 bar). The injection pressure of the passive system of the VVER-1200 is not provided in the EIA Report.

Long-time residual heat removal by active systems is a feature of all Generation II PWRs; the low-pressure part of the emergency core cooling system generally functions as long-time decay heat removal system. From the EIA Report, it is not apparent that there are any innovations in this respect. Long-time passive heat removal, however, is a new feature – see below.

A double containment is often found in modern Generation II plants. Current French PWRs of the type P4, P'4 and N4 have a double containment with the following wall thicknesses: Inner hull 0.9–1.2 m (cylinder), 0.9–0.95 m (dome); outer hull 0.55 m (cylinder), 0.4 m (dome) (COSTAZ 1983). German Konvoi PWRs have a single concrete shell with thickness of about 2 m, plus a separate steel shell inside.

The wall thickness of the VVER-1200 containment is not mentioned in the EIA Report. For the VVER-1000/V466, the wall thickness of the inner hull is reported as 1.2 m (cylinder), 1.0 m (dome); for the outer hull as between 0.6 and 2.2 m (NEK 2004). If the situation is similar for the VVER-1200, the strength of the containment building of this reactor type is roughly comparable to that of the newer plants with PWR of Generation II.

The resistance of the VVER-1200 against external impacts (which depends to a considerable extent, but not exclusively, on the wall thickness of the containment building) as specified in the EIA Report is, in some cases, inferior to that of modern Generation II PWRs.

In the EIA Report, the airplane crash the building has to withstand is not specified. According to a recent technical article, the aircraft can weigh up to 5.7 tons, at a speed of 100 m/s (NEI 2009). In a report on the VVER-1000/V466, it is stated that the crash resistance can vary, depending on the thickness of the outer hull, from 5 tons and 120 m/s to 20 t and 214 m/s (NEK 2004). The higher load corresponds to the crash of a fast military airplane. Almost all German pressurized water reactors of Generation II (namely, those of construction lines 3 and 4) are designed to withstand such a crash (RSK 2001).

Regarding external explosions, the containment building can withstand a shockwave with 10 kPa overpressure and duration of 1 sec. The requirement in Germany is considerably more demanding: Linear increase of overpressure at building wall to 45 kPa, within 100 ms, then linear decrease to 30 kPa from 100 to 200 ms, then constant overpressure of 30 kPa for further 800 ms (BMI 1976). However, according to the article mentioned above (NEI 2009), the overpressure of the shockwave can be up to 30 kPa for a VVER-1200 (no duration provided).

The assumptions for the maximum design earthquake appear to be plausible for the site, which is located in a seismically rather quiet area. However, it is not explained how they were derived.

There are also some new features compared to Generation II plants.

Most notable is the corium localization device (system for core melt trapping and cooling, usually referred to as “core catcher” in the English technical literature). This device, however, is only mentioned in the EIA Report; no description is provided.

Somewhat more information has been published in other sources concerning the core catcher of the VVER-1000/V466 (Belene type), which can be assumed to be similar to that of the VVER-1200: This device is placed in a concrete shaft below the reactor pressure vessel. It is filled with sacrificial material. The molten reactor core falls into this device after it has penetrated the pressure vessel bottom, and is cooled from above, with water. The water from a building sump and the fuel pool is destined for this task (NEK 2004).

If functioning as planned, this new feature would have the potential to reduce the probability of large releases in case of a severe accident. However, there is no guarantee, to date, that it will indeed fulfill its purpose because the functioning of a core catcher is beset with a number of problems which have not been sufficiently clarified so far; for example (SEVON 2005; CSNI 2007a):

- It is impossible to accurately simulate the interaction between the molten core and concrete.
- There are considerable uncertainties regarding heat transfer between the materials involved.
- Cracks in the concrete of the device can occur.
- Hydrogen formation creates additional complications.

The possibility of steam explosions constitutes another problem. Such explosions, which can damage the containment, can occur when the molten core falls into a pool of water. In this case, the melt can fragment into small particles. In this way, heat transfer to the water is very fast, with abrupt vaporization as a result.

For those steam explosions, it is not possible today to predict the level of potential damage (CSNI 2007b).

The steam explosions constitute a particularly severe problem for the core catcher design selected for the VVER-1000/V466 (which is, as already pointed out, likely to be similar to that of the VVER-1200): It is not guaranteed that the molten core will reach the core catcher all at once, as a whole. If, at first, only a part gets into the concrete shaft, it is likely that this will trigger flooding. Further molten core material then falls into water. This can lead to an explosion.

To avoid this problem, the core catcher of the Areva EPR is constructed in a different manner: At first, the melt is collected completely in the shaft below the reactor, which is to remain absolutely dry. Then, the melt is to flow to the area where it is cooled with water. This transfer is to be initiated passively by melting through of an aluminium plug (TVO 2008). A construction of this kind is complicated and has its own disadvantages – in particular, accurate timing of the accident sequences is required. Also, it does not solve all of the problems of the core catcher mentioned above. But at least the construction selected for the EPR demonstrates that the developers of this reactor type were aware of the steam explosion hazard and attempted to reduce it – in contrast to the VVER-1000/V466 (and, presumably, the VVER-1200). Furthermore, the area for cooling the melt is larger than the shaft below the reactor; the melt can spread more, forming a layer of lower thickness. Chances for successful cooling are thus increased.

The passive system for heat removal from the steam generators appears to be a genuinely new system, compared to Generation II PWRs. However, this system is not described in any detail and hence cannot be assessed further here.

The claimed capability of long-time residual heat removal by passive systems is only mentioned in the EIA Report, and not explained and discussed further. This point also cannot be further assessed here.

## 3.3 Questions

3. The capacity factor given in the EIA Report (about 96%) is very high. What is the basis for this assumption?
4. Can a description of the passive high-pressure boron injection system (design, operating parameters) be provided?
5. What are the wall thicknesses (cylinder and dome) of the double containment building of the VVER-1200?
6. What are the parameters of the maximum aircraft crash (plane mass and speed) the containment building can withstand?
7. Regarding external explosions, the maximum shockwave overpressure the containment building can withstand, according to the EIA Report, appears rather low (10 kPa). On the other hand, a higher value reported in the literature. Which value is correct, what is specified in the regulations in this respect?
8. How were the assumptions for the maximum design earthquake (intensity, ground acceleration) arrived at?
9. Can a detailed description of the corium localization device be provided? How has the functioning of this device been proven (tests, computer simulations)? In particular, how can it be guaranteed that steam explosions can be avoided?
10. Can the passive system for heat removal from the steam generators be described (design, operating parameters)? Which role does this system play in the context of long-time passive residual heat removal, what other systems are there for this purpose? How has their functioning been proven?

# 4 PROJECT TARGETS & DESIGN LIMITS

## 4.1 Treatment in the EIA Report

The quantitative probabilistic project targets are:

- Core damage frequency <E-5 per reactor year (for all initiating events)
- Large release frequency <E-7 per reactor year
- Furthermore, there are the following requirements:
- Active and passive safety systems
- Low susceptibility to human factors
- Low susceptibility to failures of supply and control systems

The targets are to be achieved by increasing the safety level, by improving the nuclear fuel and main equipment, development of improved safety systems (active and passive), reduction of susceptibility to human errors, improvement of reliability of the equipment,and maximum consideration of experience in design and operation of earlier VVERs.

It is also aimed at improving the economic characteristics by reducing capital investment and operating costs.

The design limits for effective radiation doses are specified in accordance with recommendations of ICRP and IAEA, as well as with regulation NRB-99 of the Russian Federation.

For example, the upper limit for the population during normal operation is 0.1 mSv/a. For DBAs with a probability higher than E-4/a the limit is 1 mSv/event; for a probability lower than E-4/a, 5 mSv/event.

Operating and safety limits for the nuclear fuel are also specified (for example, permissible number of damaged fuel elements in case of DBA). The total probability of a severe accident (BDBA), for all initiating events, has to be lower than E-6/a.

In the reactor design, special attention was paid to IAEA recommendations, INSAG publications as well as European Utility Requirements (EUR), Revision C.

The following main safety characteristics are formulated:

- Prevention of abnormalities in operation, by reliable structures with high thermal inertia and high safety margins
- Minimization of general-cause failures and dependent failures
- Multifunctional emergency core cooling system, based on different principles and on combination of passive and active parts
- Localization system for radionuclides, in case of a severe accident, based on the containment
- Reduction of irradiation doses by appropriate design, materials and layout (this point refers to plant personnel)

The principle of protection in depth is implemented. There are multiple barriers to prevent emissions to the environment: Fuel matrix, fuel element cladding, primary circuit and containment.

## 4.2 Assessment

The quantitative probabilistic targets appear to be fulfilled by the NPP-2006.

However, it is not entirely clear from the EIA Report that the CDF and LRF values provided really include all plant states (full power, low power and shut-down) and all initiators (internal and external).

Even in case the PSA is “complete” in this sense it should be borne in mind that some factors cannot be taken into account, and some factors included are beset with large uncertainties. (This has already been pointed out in the section on selection of the NPP type.) Thus, quantitative probabilistic targets should not be ascribed too much significance.

Regarding general safety requirements, it is important to note that the NPP-2006 was developed from NPP-92, which is EUR certified. Thus, it is plausible that NPP-2006 also completely fulfills the EUR, as is explicitly stated in section 2.3 of the EIA Report.

The main safety characteristics and the concept of multiple barriers are described in a very general manner. In this form, they apply to many operating NPPs of Generation II.

## 4.3 Questions

11. Do the values for core damage frequency (CDF) and large release frequency (LRF) provided for the VVER-1200 in the EIA Report include all plant states (full power, low power and shut-down) and all initiators (internal and external)?
12. Which uncertainty is associated with the PSA results? In particular, can the 95% fractiles of CDF and LRF be provided?
13. It is stated in the EIA Report that the NPP-2006 fulfills the European Utility Requirements. Can more information be provided in this respect – in particular, regarding the source term which was assumed to check compliance with the “Criteria for Limited Impact” (CLI)?
14. Can the requirements the NPP has to fulfill (apart from the EUR) be specified in more detail?

# 5 ACCIDENT ANALYSIS

## 5.1 Treatment in the EIA Report

### 5.1.1 Normal operation

Radionuclide emissions from the ventilation stack during normal operation from different Russian reactors are given in the EIA report. (REPORT 2009, p. 67, table 23). Under the headline “Agriculture” the maximum resulting soil contamination of these emissions is presented. It is stated that the allowed limit for agricultural products (RAL-99) will not be exceeded, even under the assumption that the fallout of nuclides accumulates at grid-points of different distances on a fixed axis during the NPP's whole lifetime (REPORT 2009, p. 80f.):

### 5.1.2 Design basis accidents (DBA)

Acceptance criteria for design basis accidents (DBA) are presented as follows (REPORT 2009, p. 43).

In case of an accident with a probability:

- E-4 events/year the effective dose must be <1 mSv/event
- < E-4 events/year the effective dose must be <5 mSv/event

In section 4.3. Agriculture, the result of the assessment of emissions due to the “maximum projected damage” (= maximum DBA, according to the Russian edition) is presented, because “it is necessary to state sufficiently low densities of the soil contamination”:

A contamination of more than 0.37 kBq/m² of Cs-137 is predicted on an area of about 1,000 ha. For I-131, the area with the contamination density over 37 kBq/m² will make up about 700 ha. “Some limitations on the use of the agricultural products would be introduced near the axis of the trace at the distance of 500–7500m” from the NPP in the first vegetation period after the accident (REPORT 2009, p. 82):

In the EIA Report, the source term for the maximum design base accident (maximum DBA) is presented as:

- Iodine-131: 4,7 E11 Bq (0.47 TBq),
- Cesium-137: 2,7 E10 Bq. (0,027 TBq) (REPORT 2009, p.101)

For this maximum DBA emission the EIA report concludes that “there is no need in undertaking defensive measures, because the calculated forecasted doses of the radiation do not exceed the criterion for undertaking defensive measures (100 mSv over the whole body and/or 50 mGy on thyroid gland for the first 7 days following after the damage) – even in the worst meteorological conditions” (which are not specified in detail), (REPORT 2009, p. 101).

### 5.1.3 Beyond design basis accidents (BDBA)

The target characteristics for the project safety system configuration are specified as follows:

- calculated probability of severe core damage for all initiating events (CDF) ≤ E-5 per reactor-year calculated probability of limit(ed) radiation release in case of an BDBA ≤ E-7 per reactor-year (REPORT 2009, p. 39 , table 9).

In the EIA Report for the AES 2006 reactor, data for severe accident frequencies are given:

- CDF < 5.8 E-7 per reactor-year and the limit(ed) radiation release frequency (BDBA ) < E-8 per reactor-year (REPORT 2009, p. 31, table 6 )
- Probability of core damage (beyond the limits, specified for the accidents within the design basis) not more than E-6 for one reactor annually. (REPORT 2009, p. 46)

In case of an accident beyond the design basis, the equivalent radiation dose for the critical group on the border of the protective measures planning zone has to be < 5 mSv. (REPORT 2009, p. 43)

In chapter 5 under the headline "transboundary impact" severe accident source terms are presented:

- The BDBA source term for LNPP-2: (REPORT 2009, p. 107)
  - Iodine-131: E14 Bq (100 TBq)
  - Cesium-137: E13 Bq (10 TBq)
- The source term for the most severe BDBA (REPORT 2009, p. 116):
  - Iodine-131: 4.1 E14 Bq (410 TBq),
  - Cesium-137: 1.7 E13 Bq (17 TBq),
  - Strontium-90: 1.5 E12 Bq (1.5 TBq).

For Novoworonesh NPP-2 two source terms are given in the EIA Report: one is for the "maximum emergency emission through the passive ventilation system"; the other one is for a bypass scenario. For the bypass scenario Cesium and Iodine emissions are ten times the emissions through the ventilation stack.

The calculation of radioactive contamination spreading under BDBA and maximum DBA condition has been performed with the use of the two models:

- meso-scale model – up to100 km (was used for maximum DBA)
- trans-border model – ~ 1000 km (was used for BDBA).

These models calculate density fields for surface contamination as the result of dry/humid precipitation, resulting contamination is integrated over the time. The calculations are terminated, at maximum for the model distance or if the radionuclide concentration of the cloud has decreased to a certain limit.

## 5.2 Assessment

Information concerning accidents in the NPP is distributed over different parts of the EIA Report:

- Design limits for effective radiation dose in chapter 2.6 (table 6)
- Emission data for normal operation and DBA in chapter 4.3 Agriculture

- Chapter 4.9.3. Accident scenarios: 13 meteorological scenarios for the assessment of impacts from DBA and BDBA are considered.
- Chapter 5. transboundary emissions: short description of the dispersion calculation and BDBA source terms of different NPP (LNPP-2, NNPP-2) and a worst case.

There is no information about which DBA scenarios have been analyzed. Under the headline "Accident scenarios" only meteorological scenarios are presented (13 different weather scenarios have been analyzed to find out the worst case for the emergency planning zone). A source term is presented for the maximum DBA.

It is not well defined, which parameters are decisive for the design: the size of the contaminated area or the calculated effective dose.

Moreover there is some inconsistency concerning the dose limit for the DBA emissions:as it is cited above the maximum effective dose equivalent due to a DBA is 5mSv (REPORT 2009, p.43). In the chapter "Transboundary emissions" (REPORT 2009, p 101) the dose equivalent due to the maximum DBA emission is compared to a limit of 100 mSv.

From the description of the dispersion calculation can be concluded that meteorological worst case situations have been analyzed for the region near the NPP and for larger regions. The calculation uses a simplified long-range transport model.

Several BDBA source terms are presented, but without description of the initiating events and the progress of the emergency situation. Both Novoworonesh scenarios are described without details and without reference. The content of some tables is unclear "Предельный аварийный выброс" probably has to be translated as "limit for severe accident release"; the headline "maximum emergency emission" (table 29) is unclear (could mean this is the largest conceivable emission or this is the maximum emission permitted).

Further it is unclear whether the source terms are derived from deterministic or probabilistic assessments. That is also the case for the "worst case" BDBA emission scenario in Chapter 5 "Transboundary impact". For limited release scenarios the fulfillment of the EUR requirements is proved in the EIA Report. Severe accidents with a release exceeding this limitation are not considered in the EIA Report. No uncertainties of the presented results are given in the EIA Report.

The conclusion of the EIA Report, that no greater source terms than the presented limited releases could occur is not sufficiently substantiated. For all existing reactors and also for the new Generation III reactors now under construction, severe accidents with a release in the range of some percent of the Cs-137 inventory (2–20%) are not excluded. Even if the frequency of occurrence of accidents with a large release appears very small according to PSA, such severe accident source terms should be considered in the transboundary EIA (it should be taken into account that PSA results do not include all relevant factors, and some factors which are included are beset with large uncertainties – see chapter 6).

## 5.3 Questions

15. Which are the references for the source terms presented in the EIA Report ? Why are larger source terms not discussed?
16. Which source terms are worst case scenarios and which maximum permitted emissions?
17. Are results from preliminary safety reports of the NPPs Leningrad 2 and Novoworonesh 2 – NPP-2006 (VVER-1200/491) – under construction available to the authors of the EIA Report? Is there a level 2 PSA for these reactors?
18. Which DBA and BDBA scenarios have been analyzed by the designers of the NPP?
19. Is it possible to describe the accident management features and procedures which shall guarantee the limited emission in case of a BDBA?

# 6 RADIOACTIVE WASTE

## 6.1 Treatment in the EIA Report

### 6.1.1 Liquid and solid radioactive wastes

The handling of liquid and solid radioactive waste as well as methods for reducing the production of liquid radioactive waste and solidifying it are described very briefly (REPORT 2009, p. 50–52).

Solid radioactive waste amounts up to approximately 60 m³ per year per reactor unit after pressing and cutting, consisting of

- 76% low level active waste
- 23% medium level active waste
- 1% high level active waste (REPORT 2009, p. 51)

Solidified liquid radioactive waste makes up a volume of 30 m³ per unit and year, consisting of low and medium active waste (REPORT 2009, p. 51).

Spent fuel will be stored in the fuel pool in the reactor building. The capacity of the storage will be sufficient for keeping the spent fuel during ten years of the operation of the station. Besides, there will be provided place for unloading the core in case of an accident. After three years the spent fuel, may be removed from the pool in the reactor building. The spent nuclear fuel is to be removed to processing plants or to the supplier-country of the nuclear fuel in special shipping packaging sets (REPORT 2009, p. 50).

## 6.2 Assessment

There are no radioactivity levels indicated for the classification of liquid and solid radioactive wastes (high, medium, low level waste). The amounts of the solid and liquid radioactive waste are given only specified in m³.

The radioactive waste handling system is described without details but it is said that design and operation of the solid rad-waste handling system will be based on the EIA recommendations and the international practice (EUR version C).

No interim storage for the spent fuel and no plans for the radioactive waste disposal in Belarus are mentioned in the EIA Report.

## 6.3 Questions

20. What radioactivity levels do you use for the classification of radioactive wastes (high level, medium level, low level waste)?
21. Are there any plans for construction of an interim storage for spent fuel?
22. Are there plans for the construction of a disposal facility for operational nuclear waste in Belarus?

# 7 GLOSSARY

A ............................ year, Jahr
ABWR .................... Advanced Boiling Water Reactor
AES ....................... Atomnaja Electrostancija, translates into nuclear power plant (NPP)
AP-600, AP-1000 .... Reactor types by Westinghouse
APR. ...................... Advanced Power Reactor
APWR .................... Advanced Pressurized Water Reactor
BDBA ..................... Beyond Design Basis Accident
BNFI ...................... British Nuclear Fuels plc
CDF ....................... Core Damage Frequency
CLI ......................... Criteria for Limited Impact
Cs .......................... Caesium
DBA ....................... Design Basis Accident
DWR ...................... Druckwasserreaktor, English: PWR
ECCS .................... Emergency Core Cooling System
EIA ......................... Environmental Impact Assessment
EPR ....................... European Power Reactor
ESBWR ................. Economic Simplified Boiling Water Reactor
EUR ....................... European Utilities Requirements
g ............................ ground acceleration
I ............................. Iodine
IAEA ...................... International Atomic Energy Agency
ICRP ...................... International Commission on Radiation Protection
INSAG ................... International Nuclear Safety Group
kPa ........................ Kilo Pascal
LRF ........................ Large Release Frequency
mGy ....................... Milli Gray (energy dose)
mSv ....................... Milli Sievert (dose)
MW ........................ Megawatt
MWe ...................... Megawatt electric
NPP ....................... Nuclear Power Plant
NPP-92 .................. Russian type of PWR, predecessor model of NPP-2006 (=AES-92)
NPP-2006 ............... Russian type of PWR (= AES-2006)
NRC ....................... Nuclear Regulatory Commission (USA)
PSA ....................... Probabilistic Safety Assessment
PWR ...................... Pressurized Water Reactor
Sr ........................... Strontium
TBq ........................ Tera Becquerel
UVE ....................... Umweltverträglichkeitserklärung (English: EIA Report)
UVP ....................... Umweltverträglichkeitsprüfung (English: EIA)
VVER ..................... Vodo-Vodyanoy Energeticheskiy Reactor

# 8 REFERENCES

BMI (1976): Bundesministerium des Inneren: Richtlinie für den Schutz von Kernkraftwerken gegen Druckwellen aus chemischen Reaktionen durch Auslegung der Kernkraftwerke hinsichtlich ihrer Festigkeit und induzierter Schwingungen sowie durch Sicherheitsabstände; Bundesanzeiger 1976, Nr. 1979, 13. September 1976.

COSTAZ (1983): Costaz, J.L et al.: Evolution of the Concrete Containment in the French PWR Program; Transactions of SMiRT 7 – 7th International Conference on Structural Mechanics in Reactor Technology, Chicago, IL, August 22–26.

CSNI (2007a): OECD Nuclear Energy Agency: Nuclear Safety Research in OECD Countries – Support Facilities for Existing and Advanced Reactors (SFEAR); NEA/CSNI/R(2007)6.

CSNI (2007b): OECD Nuclear Energy Agency: OECD Research Programme on Fuel-Coolant Interaction – Steam Explosion Resolution for Nuclear Application – SERENA; NEA/CSNI/R(2007)11.

IAEA (2000): International Atomic Energy Agency: Safety of Nuclear Power Plants: Design; IAEA Safety Standards Series, No. NS-R-1, Vienna.

IAEA (2005): International Atomic Energy Agency: Design of the Reactor Core for Nuclear Power Plants; IAEA Safety Standards, No. NS-G-1.12, Vienna.

KRAEMER (2008): Krämer, J. & R. Richter: Die Entwicklung der Kernkraftwerkstechnik in Russland; atw 53. Jg. (2008) Heft 8/9, S. 520–520.

NEI (2009): Russia’s next VVER. Nuclear Engineering International, October 2009, p. 15-17.

NEK (2004): Bulgarian National Electric Company (NEK): Investment Proposal for Construction of Belene Nuclear Power Plant; Environmental Impact Assessment Report BNPP-EIA-PEC-NEK-0001-E3, March 2004.

REPORT (2009): Substantiation of investments in construction of the nuclear power plant in the republic of Belarus. Environmental impact assessment. Statement on possible environmental impact on Belarusian NPP (Preview Report on EIA on Belarusian NPP). Ministry of Power Engineering of the Republic of Belarus, Design Scientific-Research Republican Unitary Enterprise “Belnipienergoprom”.

RSK (2001): Reaktor-Sicherheitskommission des Bundesministeriums für Umwelt, Naturschutz und Reaktorsicherheit: 1. Stellungnahme zur Sicherheit deutscher Atomkraftwerke gegen gezielten Absturz von Großflugzeugen mit vollem Tankinhalt; 14.10.2001.

SEVON (2005): Sevón, T.: Molten Core-Concrete Interactions in Nuclear Accidents: Theory and Design of an Experimental Facility; Helsinki University of Technology, Department of Engineering, Physics and Mathematics, Espoo, 6 October 2005.

TVO (2008): Teollisuuden Voima Oyj: Nuclear Power Plant Olkiluoto 3; Eurajoki/Helsinki.

UMWELTBUNDESAMT (2008): Wenisch, A. et al.: NPP Loviisa-3 – Expert Statement to the EIA Report; ordered by the Federal Ministry for Agriculture, Forestry, Environment and Water Management, Report, REP-0167. Umweltbundesamt, Wien.

WNIH 2009: Nuclear Engineering International: World Nuclear Industry Handbook 2009; Progressive Media Markets Ltd, Sidcup (UK), 2009.

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