# SOVIET NUCLEAR POWER PLANT DESIGNS

## Key Facts

- Western-style plants employ the design principle of safety in depth, relying on a series of physical barriers—including a massive reinforced concrete structure called the containment—to prevent the release of radioactive material to the environment. With the exception of the VVER-1000 design, Soviet-designed reactors do not have such a containment structure.

- Soviet-designed reactors are essentially variations on two basic designs: the VVER—or pressurized light water—type, and the RBMK—the graphite moderated, channel reactor.

- Three generations of Soviet-designed VVER reactors—upgraded over time—are operating in Eastern Europe and the former Soviet Union. The first generation—the VVER-440 Model V230—operates at four plant sites in three countries: Russia, Bulgaria and the Slovak Republic. The second generation—the VVER-440 Model V213—operates at five plant sites in five countries: Russia, Ukraine, Hungary, the Czech Republic and the Slovak Republic. The third generation—the VVER-1000—operates at eight plant sites in three countries: Russia, Ukraine and Bulgaria.

- At the time of the collapse of the Soviet Union, two advanced versions of the VVER-1000 were under development. Russia has continued the development of an upgraded VVER-1000, and has developed a new design for a 640-megawatt reactor with enhanced safety features.

- Three generations of RBMK reactors are operating in the former Soviet Union: 11 units in Russia, one in Ukraine and two in Lithuania. Despite improvements to the RBMK design since the Chernobyl accident, concerns remain about these reactors, especially the first-generation ones.

## Major Difference Between Soviet-Designed and Other Nuclear Power Plants

Soviet-designed nuclear power plants differ from nuclear power plants of other countries in many respects, including plant instrumentation and controls, safety systems and fire protection systems.

While Soviet-designed plants—like other plants—employ the design principle known in the West as “defense in depth,” only one reactor design includes a containment structure as part of that principle.

In the unlikely event that safety systems fail, plants designed on the “defense in depth” principle rely on a series of physical barriers to prevent the release of radioactive material to the environment. At U.S. plants:

- The first barrier is the nuclear fuel itself, which is in the form of solid ceramic pellets. Most of the radioactive by-products of the fission process remain bound inside the fuel pellets.
- These pellets are then sealed in rods, made of special steel, about 12 feet long and half an inch in diameter.
- The fuel rods are inside a large steel pressure vessel, which has walls about eight inches thick.
- At most plants, this vessel is enclosed in a large, leak-tight shell of steel plate.
- All this is contained inside a massive steel and/or concrete structure—called the containment—with walls several feet thick.

Most Soviet-designed reactors employ similar features, but only the VVER-1000 design has a containment structure like that of most nuclear power plants elsewhere in the world. Without this protection, radioactive material could escape to the environment in the event of a serious accident.

## Plant Location and Design “Families”

At present, more than 70 commercial nuclear reactors of Soviet design are operating or under construction in Russia, Ukraine, Lithuania, Bulgaria, the Czech Republic, the Slovak Republic and Hungary.

A two-unit Soviet-designed nuclear plant in Finland was built using the VVER-440 Model 213 basic design, but was upgraded to include a Western instrumentation and control system and a containment structure.

With the exception of small nuclear units used for district steam heating and several small fast-breeder reactors—which produce fuel as they generate electricity—Soviet-designed commercial nuclear power plants are variations on two basic designs: the VVER—or pressurized light water—type, and the RBMK—the graphite moderated, channel reactor. There are no RBMK plants operating outside the former Soviet Union.

Like all nuclear units based on light water technology, the Soviet VVER design uses water to generate steam and to cool the reactor. Water also acts as a “moderator,” slowing down the atomic particles (neutrons) in the nuclear reaction to increase the chances of fissioning, or splitting. The “moderating” effect of the water adds safety, because a water loss will slow the nuclear chain reaction.

In the RBMK design, graphite is used in place of water as a moderator, surrounding vertical pressure tubes which hold the nuclear fuel and the water that will be boiled to steam. Unlike light water units, the RBMK’s nuclear chain reaction and power output increase when cooling water is lost. This design flaw—called a “positive void coefficient”—caused the uncontrollable power surge that led to the Chernobyl accident. The corrections and modifications made to all of the RBMKs since the Chernobyl accident are generally considered to be adequate to preclude the type of nuclear excursion—a sudden, rapid rise in power level—that occurred at Chernobyl Unit 4 in April 1986.

The Beloyarsk fast-breeder reactor in Russia is the second-largest such unit in the world, behind the French Super Phenix, and generates new fuel as it operates. The major components of the Beloyarsk unit are submerged in a large pool of liquid sodium, which acts as a moderator and transfers heat away from the reactor to boil water to make steam.

## The VVER: Three Generations of Light Water Reactors, Upgraded Over Time

Although it shares a basic engineering concept with its counterparts in the United States, France and Japan, the Soviet pressurized water—or VVER—design is very different and does not meet Western safety standards. However, second- and third-generation plants of this design—the VVER-440 Model V213 and VVER-1000—are widely viewed as having a design safety basis sufficiently comparable to that used in the West to justify short-term and long-term safety and performance upgrades on both safety and economic grounds. However, regulatory requirements and the extent of plant upgrading differ from country to country and plant to plant, resulting in varying levels of safety, even for plants of the same model.

### First-Generation VVERs

The earliest pressurized water nuclear plants were developed by the Soviets between 1956 and 1970. These plants include the following versions:

| | |
|---|---|
| VVER-210 (Prototype) | Novovoronezh 1<br>Russia (Shut down 1984) |
| VVER-365 (Prototype) | Novovoronezh 2<br>Russia (Shut down 1990) |

| | |
|---|---|
| VVER-440 | Novovoronezh 3 and 4 (First standardized model V230s) |
| VVER-440 Model V230 | 1) Kola 1 and 2, Russia<br>2) Medzamor 1 and 2 (Shut down 1989; Unit 2 restarted November 1995)<br>3) Kozloduy 1-4, Bulgaria<br>4) Bohunice 1 and 2, Slovak Republic<br>5) Greifswald 1-4 in the former East Germany (Shut down 1990) |

***Principal Strengths:***

- Six primary coolant loops (providing multiple paths for cooling the reactor), each with a horizontal steam generator (for better heat transfer), which together provide a large volume of coolant. In some respects this design is more forgiving than Western plant designs with two, three or four large vertical steam generators.

- Isolation valves that allow plant operators to take one or more of the six coolant loops out of service for repair while continuing to operate the plant. This feature is found in only a few Western plants.

- Ability to sustain a simultaneous loss of coolant and off-site power, due to coolant pumps and two internal power generators that "coast down" after a shutdown.

- Plant worker radiation levels reportedly lower than many Western plants, due to selection of materials, high-capacity primary coolant purification system, and water chemistry control.

- Ability to produce significant amounts of power despite design and instrumentation and control deficiencies.

***Principal Deficiencies:***

- Accident Localization System—which serves as a reactor confinement—designed to handle only one four-inch pipe rupture. If larger coolant pipes rupture, this system vents directly to the atmosphere through nine large vent valves. Western nuclear plants have containments designed for rupture of the largest pipes. In addition, the confinement has very small volume, very poor leak-tightness and poor hydrogen mitigation.

- No emergency core cooling systems or auxiliary feedwater systems similar to those required in Western nuclear plants.

- Major concern about embrittlement (gradual weakening) of the reactor pressure vessel surrounding nuclear fuel, due to lack of internal stainless-steel cladding and use of low-alloy steel with high levels of impurities.

- Plant instrumentation and controls, safety systems, fire protection systems, and protection for control room operators are below Western standards.

- Quality of materials, construction, operating procedures and personnel training are below Western standards.

## Second-Generation VVERs

The VVER-440 Model V213 was designed between 1970 and 1980. The development of this design coincided with the first uniform safety requirements drawn up by Soviet designers.

VVER-440 Model V213 units in the former Soviet Union include:

| | |
|---|---|
| Russia | Kola 3 and 4 |
| Ukraine | Rovno 1 and 2 |

VVER-440 Model V213 units in Central and Eastern Europe include:

| | |
|---|---|
| Hungary | Paks 1-4 |
| Czech Republic | Dukovany 1-4 |
| Slovak Republic | Bohunice 3 and 4 |
| Former East Germany | Greifswald 5 (Shut down) |
| Finland | Loviisa 1, 2 (includes Western I&C system and containment) |

Construction of a version of the Model V213 intended for export began in Cuba in 1983 but was suspended in 1992.

### *Principal Strengths:*

- Upgraded Accident Localization System vastly improved over the earlier VVER-440 Model V230 design, comparable to several Western plants, and using a vapor-suppression confinement structure called a “bubbler-condenser” tower.

- Addition of emergency core cooling and auxiliary feedwater systems.

- Reactor pressure vessel with stainless-steel internal lining to alleviate much concern about the vessel embrittlement associated with the earlier VVER-440 Model V230 design.

- Improved coolant pump, and continued use of six coolant loops (providing multiple paths for cooling the reactor) and horizontal steam generators (for better heat transfer) with large coolant volume.

- Standardization of plant components, providing extensive operating experience for many parts and making possible incremental improvements and backfits of components.

***Principal Deficiencies:***

- Plant instrumentation and controls—for example, reactor protection systems and diagnostics—behind Western standards. Significant variations exist among countries with VVER-440 Model V213 plants.

- Separation of plant safety systems (to help assure that an event in one system will not interfere with the operation of others), fire protection, and protection for control room operators improved over Model V230 plants, but generally below Western standards.

- Poor leak-tightness of confinement.

- Unknown quality of plant equipment and construction, due to lack of documentation on design, manufacturing and construction, and reported instances of poor-quality materials being re-worked at plant sites.

- Major variations in operating and emergency procedures, operator training, and operational safety (for example, use of control-room simulators) among plants. These aspects of plant operations depend primarily on the organization or country operating Model V213 plants rather than on the plant supplier. Some countries have added safety features to their Model V213 plants.

## Third-Generation VVERs

The VVER-1000 design was developed between 1975 and 1985 based on the requirements of a new Soviet nuclear standard that incorporated some international practices, particularly in the area of plant safety. The VVER-1000 design was intended to be used for many plants, and 18 units now operate in two former Soviet republics. Of these, two—Novovoronezh 5 and South Ukraine 1—are prototypes; three are Model V338s—Kalinin 1 and 2 and South Ukraine 2; and all the rest—Balakovo 1-4, Rovno 3, Khmelnitskiy 1, South Ukraine 3 and Zaporozhye 1-6—are Model V320s.

| | |
|---|---|
| Russia | Balakovo 1-4<br>Kalinin 1-2<br>Novovoronezh 5 |
| Ukraine | Rovno 3<br>Khmelnitskiy 1<br>South Ukraine 1-3<br>Zaporozhye 1-6 |

Two VVER-1000 units were built outside the former Soviet Union:

| | |
|---|---|
| Bulgaria | Kozloduy 5 and 6 |

Work was stopped on two other VVER-1000 units in Bulgaria (Belene 1 and 2) after public protests over claims of unsuitable soil and seismic conditions.

The Hungarian government canceled Paks 5 and 6 in 1989.

Construction of two VVER-1000 units at Stendal, in the former East Germany, was halted following reunification with West Germany.

Two VVER-1000 units under construction at Temelin in the Czech Republic are being upgraded with Western instrumentation and control equipment and fuel.

A total of 25 VVER-1000 units are at some stage of construction in the former Soviet Union—15 in Russia and 10 in Ukraine. But work on 12 of these units in Russia, and six in Ukraine, has reportedly been canceled or deferred indefinitely.

Of the VVER-1000 units earmarked for completion under the 1992 Russian plan, Kalinin 3—originally scheduled to come on line in 1995—is expected to be operational by 2000, according to a Ministry of Atomic Energy official. Other units expected to come on line by 2000 are Balakovo 5, a VVER-1000, and Rostov 1, a VVER-1000 that is reportedly 97 percent complete. A second unit at Rostov is said to be 95 percent complete, but there is local opposition to both projects. Russia's new energy law requires the approval of local authorities for plant construction.

Ukraine is seeking funding to complete the construction of two VVER-1000 units—Khmelnitskiy 2 and Rovno 4.

### *Principal Strengths:*

- Steel-lined, pre-stressed, large-volume concrete containment structure, similar in function to Western nuclear plants.

- "Evolutionary" design incorporating safety improvements over VVER-440 Model V213 plants. The Soviet approach to standardization was based on continued use of components that had performed well in earlier plants.

- Use of four coolant loops and horizontal steam generators—both considered improvements by Soviet designers.

- Redesigned fuel assemblies that allow better flow of coolant, and improved control rods.

- Plant worker radiation levels reportedly lower than in many Western plants, apparently due to selection of materials, high-capacity system for purifying primary coolant, and water chemistry control.

***Principal Deficiencies:***

- Substandard plant instrumentation and controls. Wiring of emergency electrical system and reactor protection system does not meet Western standards for separation—control and safety functions are interconnected in ways that may allow failure of a control system to prevent operation of a safety system.

- Fire protection systems that do not appear to differ substantially from earlier VVER models, which do not meet Western standards.

- Quality control, design and construction significantly deficient by U.S. standards.

- Protection measures for control-room operators essentially unchanged from earlier VVER-440 Model V213 design, which does not meet U.S. standards. Unlike all U.S. nuclear plants, and most in Western countries, VVER-1000s have no on-site “technical support center” to serve as a command post for stabilizing the plant in an emergency. Technical support centers were incorporated in U.S. and many Western nuclear plants following the accident at Three Mile Island Unit 2 in 1979.

- Operating and emergency procedures that fall far short of Western standards and vary greatly among operators of VVER-1000 plants.

- Higher power densities and the smaller volume of primary and secondary systems result in a somewhat less forgiving and stable reactor.

## VVER-1000 Derivatives

Even before the breakup of the Soviet Union, derivative versions of the VVER-1000 were under development.

In 1987, design work was begun on the VVER-1800, a VVER-1000 upgraded for greater safety and economy. The VVER-1800 design incorporated a lower-power reactor core, annual refueling, and more reliable control and protection systems.

In 1989, Finland and the Soviet Union jointly announced the start of development work on the VVER-91, a VVER-1000 version that would meet stringent Finnish nuclear plant design requirements. On paper, the Soviet

VVER-91 design is among the world's most advanced light water nuclear power plants.

Development of a new VVER-1000 design, the VVER-92, was expected to be carried out with Western assistance. The VVER-92 incorporated what one Finnish nuclear expert called "radically simplified" plant systems that included active safety systems, a reduced-power reactor core, and a double containment structure surrounding the nuclear reactor. However, the Ministry of Atomic Energy has reportedly diverted some funding for VVER-92 development to a pilot project for building a smaller advanced VVER, the VVER-640 or Model V407.

## The RBMK: The Chernobyl-Type Soviet Nuclear Power Plant

The former Soviet Union built 17 nuclear units based on the RBMK design used at the Chernobyl nuclear power plant, the site of the world's worst commercial nuclear accident. There are currently 14 RBMK reactors in operation: 11 units in Russia, one in Ukraine and two in Lithuania. These units were connected to the grid between 1973 (Leningrad 1) and 1990 (Smolensk 3). During these 17 years, the design evolved significantly. In addition, following the Chernobyl accident in 1986, some major safety upgrades were implemented. Today it is generally recognized that there are three generations of RBMK nuclear power plants, although even within a given generation the units can differ substantially.

RBMKs in the former Soviet republics include:

| | |
|---|---|
| Russia | Leningrad (Sosnovyy Bor) 1-4<br>Smolensk 1-3<br>Kursk 1-4 |
| Ukraine | Chernobyl 1-4<br>(Unit 4 was destroyed in the 1986 accident; Unit 2 was shut down in 1991; Unit 1 was shut down in 1996.) |

RBMKs of the 1,500-megawatt class include:

| | |
|---|---|
| Lithuania | Ignalina 1-2 |

Kursk 1 and 2 and Leningrad 1 and 2 are first-generation RBMKs. Kursk 3 and 4, Ignalina 1 and 2 and Leningrad 3 and 4 are second-generation RBMKs. Smolensk 3 is a third-generation RBMK.

At the time of the Chernobyl accident, six RBMK units were under construction in the U.S.S.R.: Kursk 5 and 6 and Smolensk 4 in Russia, Chernobyl 5 and 6 in Ukraine and Ignalina 3 in Lithuania. At the Kursk RBMK plant, Unit 5—originally scheduled to come on line in 1995—could be completed by 1998.

Since the Chernobyl accident, a considerable amount of work has been carried out—both by Russian institutions and by international groups—to improve RBMK reactor safety and to eliminate the root causes of the accident. Additional measures are planned or under way. But some concerns remain, particularly with respect to RBMKs of the first generation.

### *Principal Strengths:*

- The low core power density of RBMKs provides a unique ability to withstand station blackout and loss of power events of up to an hour with no expected core damage.
- The units can be refueled while operating, permitting a high level of availability.
- The graphite moderator design allows the use of fuel that is not generally suitable for use in conventional water-moderated reactors.

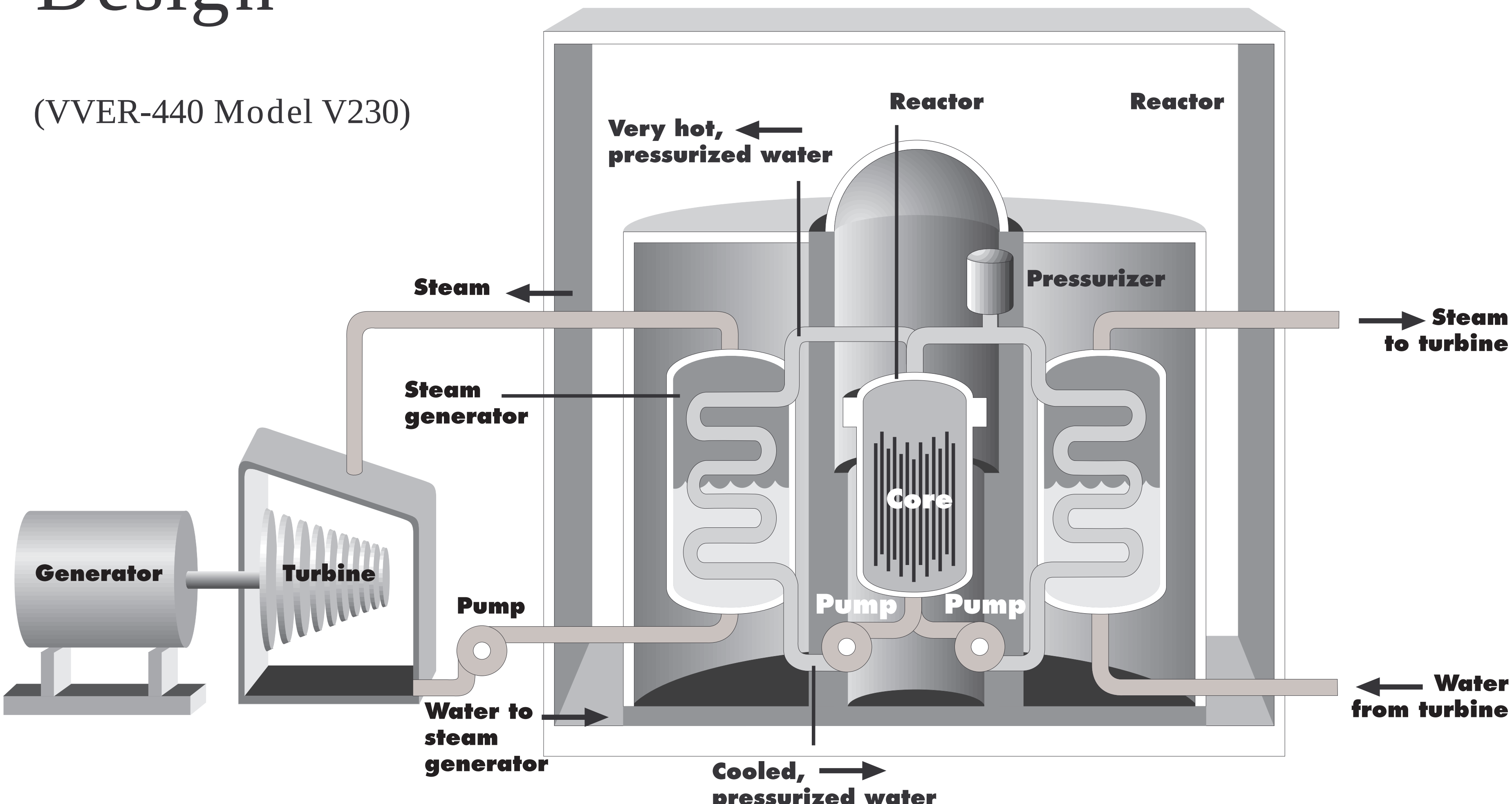
### *Principal Deficiencies:*

- The most significant difference between the RBMK design and most of the world's nuclear power plants is the RBMK's lack of a massive steel and/or concrete containment structure as the final barrier against large releases of radiation in an accident. The effectiveness of American-style reactor containments was shown in the 1979 Three Mile Island Unit 2 accident, when virtually all radiation was retained inside the containment building, despite considerable melting of the fuel. In the Chernobyl accident, the RBMK plant's accident localization system (the RBMK's version of containment) could not withstand the force of the accident. However, because the estimated energy released by the explosions was greater than most containment designs could withstand, it is highly unlikely that a containment structure could have prevented the release of radioactive material at Chernobyl.
- Accident mitigation systems are limited and ineffective.
- Reactor control systems are unforgiving to many potential system upsets, with a consequent potential difficulty of successful recovery.
- The reactor produces faster and less stable nuclear chain reactions—and power increases—when cooling water is lost. In technical terms, this characteristic is called a "positive void coefficient." Soviet engineers sought to mitigate this tendency by backfitting RBMKs with faster-acting control rods and other improvements. Modifications made to all RBMKs are generally considered to be adequate to maintain this positive void defect at a low enough level to preclude the type of nuclear excursion—a sudden, rapid rise in power level—that occurred at Chernobyl Unit 4. U.S.-style light water reactors are designed with just the opposite characteristic—a "negative void coefficient"—so that the nuclear chain reaction automatically stops when coolant is lost.

- Inadequate fire protection systems.
- Limited capability for steam suppression in the graphite stack.
- Flawed separation and redundancy of electrical and safety systems.
- Complicated piping arrangements.

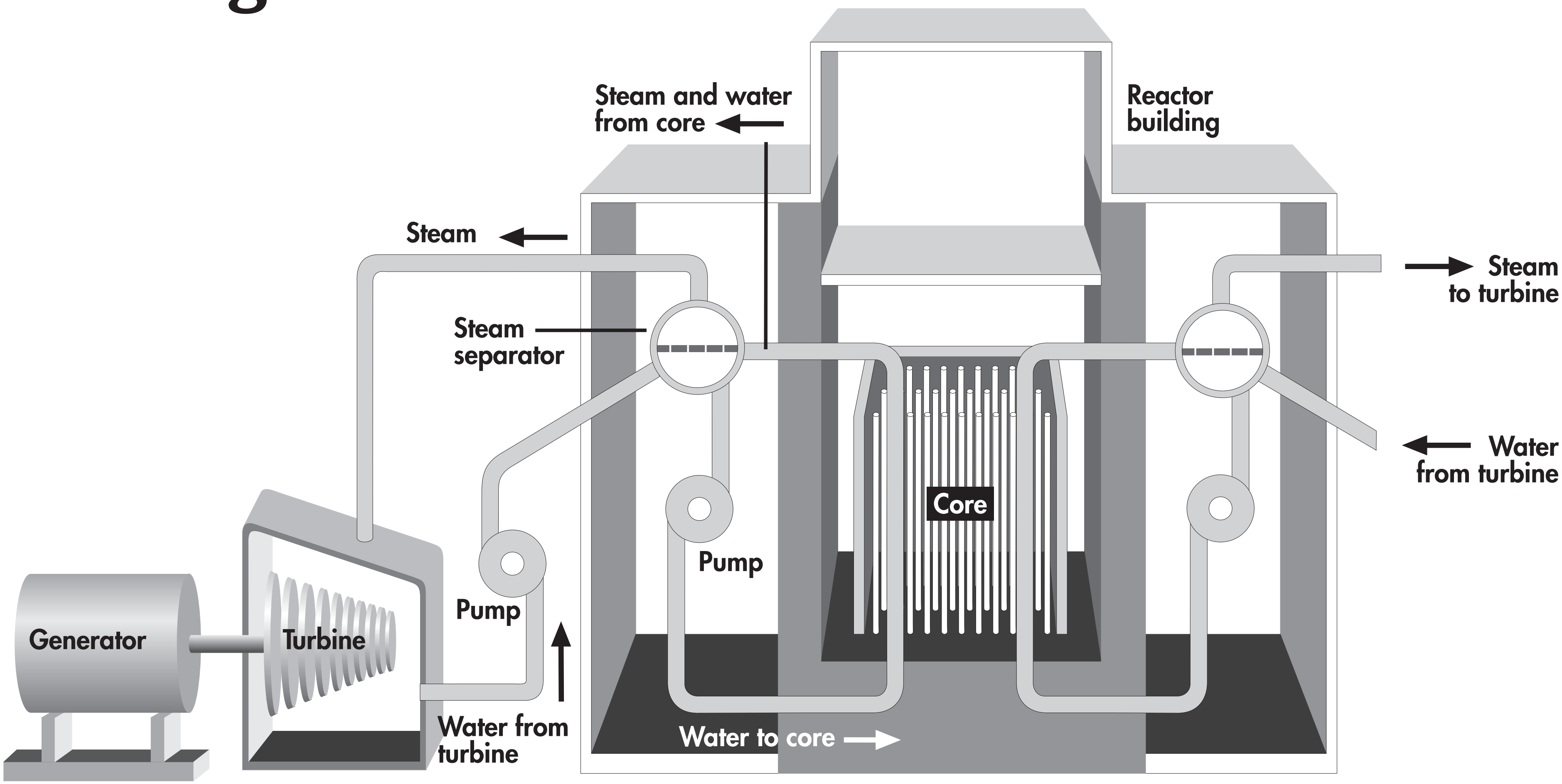
# VVER Reactor Design

(VVER-440 Model V230)




Source: Nuclear Energy Institute

# RBMK Reactor Design




Source: Nuclear Energy Institute

# THE INTERNATIONAL NUCLEAR EVENT SCALE

The objective of the scale is to promote self-assessment of the significance of an event at a nuclear facility, and to promote open communication between the nuclear community, the media and the public.

## Development of the Scale

Events at nuclear facilities are classified in most countries according to the categories and terminology of the International Nuclear Event Scale (INES).

The INES was developed under the auspices of the International Atomic Energy Agency (IAEA), an arm of the United Nations, and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD). The scale was field tested in 1991, formally adopted in 1992 and is applicable to all types of nuclear facilities.

INES was developed as a communication tool, not an emergency response tool. It is used to explain the safety significance of events at nuclear plants to the public and the news media.

The United States did not participate in the pilot stage of INES. However, in January 1993, the U.S. Nuclear Regulatory Commission (NRC) began a two-year limited participation in the use of INES. During the trial period, the NRC agreed to communicate, as part of a post-classification activity, the severity levels of U.S. reactor events to IAEA using the international scale. At the same time, the NRC continued to use the four-level response scale that categorizes the severity of an event for emergency planning purposes.

In May 1995, after the trial period ended, the NRC decided to continue the policy of limited participation for power reactors.

Fundamental differences in the U.S. and the INES systems for classifying events prevent the conversion of one classification system to the other.

- The INES is a tool to assess the severity of the safety consequences of events that occur at nuclear facilities. It relies on a sound technical basis—the probabilistic safety assessment—and employs user friendly procedures that are now computerized.

- The INES scale is primarily used to communicate information to the public. It has seven levels, ranging from Level 1, which is an anomaly with no off-site or on-site impact, to Level 7, which involves the release of large of amounts of radiation. Level 0 has no safety significance.

- The INES Information Service has been joined by 59 countries that are committed to the prompt communication of event consequences when they are significant for safety or for the public interest.

- The NRC scale categorizes the severity of an event to determine the appropriate emergency response, such as activation of response organizations and facilities. It has four levels: unusual event, alert, site area emergency and general emergency.

There is one other important difference. INES is a voluntary system of reporting information to IAEA and subsequently, to the 59 member states participating in the INES information service. The U.S. system is required by federal regulation.

## Classifying Nuclear Events with the INES

Had the INES existed at the time, these nuclear events would have been classified as follows:

***Chernobyl.*** The 1986 accident in Ukraine involved wide environmental and health effects and would have been classified as a Level 7 “Major Accident.”

***Three Mile Island.*** The 1979 accident that seriously damaged the core of Unit 2 at this nuclear power plant in Pennsylvania involved the release of very small amounts of radioactivity outside the plant and would have been classified a Level 5 “Accident With Off-Site Risks.”

**THE INTERNATIONAL NUCLEAR EVENT SCALE**
**for prompt communication of safety significance**

| LEVEL | DESCRIPTOR | CRITERIA | EXAMPLES |
|---|---|---|---|
| ACCIDENTS<br>7 | MAJOR ACCIDENT | • External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve a mixture of short- and long-lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands terabecquerels of iodine-131). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long-term environmental consequences. | Chernobyl NPP, USSR (now Ukraine), 1986 |
| 6 | SERIOUS ACCIDENT | • External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of iodine-131). Such a release would be likely to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects. | Kyshtym Reprocessing Plant, USSR (now in Russia), 1957 |
| 5 | ACCIDENT WITH OFF-SITE RISK | • External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of iodine-131). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects.<br><br>• Severe damage to the nuclear facility. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation. | Windscale Pile, UK, 1957<br><br>Three Mile Island, USA, 1979 |
| 4 | ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | • External release of radioactivity resulting in a dose to the most exposed individual off-site on the order of a few millisieverts. With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control.<br><br>• Significant damage to the nuclear facility. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations.<br><br>• Irradiation of one or more workers that results in an overexposure where a high probability of early death occurs. | Windscale Reprocessing Plant, UK, 1973<br>Saint-Laurent NPP, France, 1980<br><br>Buenos Aires Critical Assembly, Argentina, 1983 |
| INCIDENTS<br>3 | SERIOUS INCIDENT | • External release of radioactivity above authorized limits, resulting in a dose to the most exposed individual off site of the order of tenths of millisievert. With such a release, off-site protective measures may not be needed.<br><br>• On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination, for example, a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area.<br><br>• Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur. | Vandellos NPP, Spain, 1989 |
| 2 | INCIDENT | • Incidents with significant failure in safety provisions but with sufficient defense in depth remaining to cope with additional failures.<br><br>• An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event that leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and that requires corrective action. | |
| 1 | ANOMALY | • Anomaly beyond the authorized operating regime. This may be due to equipment failure, human error or procedural inadequacies. (Such anomalies should be distinguished from situations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. These are typically “below scale”). | |
| BELOW SCALE/ZERO | DEVIATION | NO SAFETY SIGNIFICANCE | |

## Basic structure of the scale

*(Criteria given in matrix are broad indicators only)*
*Detailed definitions are provided in the INES users' manual*

| CRITERIA OR SAFETY ATTRIBUTES | | | |
|---|---|---|---|
| | **Off-Site Impact** | **On-Site Impact** | **Defense In Depth Degradation** |
| 7<br>MAJOR ACCIDENT | MAJOR RELEASE:<br>WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6<br>SERIOUS ACCIDENT | SIGNIFICANT RELEASE:<br>LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5<br>ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE:<br>LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4<br>ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE:<br>PUBLIC EXPOSURE ON THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3<br>SERIOUS INCIDENT | VERY SMALL RELEASE:<br>PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT - NO SAFETY LAYERS REMAINING |
| 2<br>INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1<br>ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0<br>BELOW SCALE EVENT DEVIATION | NO SAFETY SIGNIFICANCE | | |
| OUT OF SCALE EVENT | NO SAFETY RELEVANCE | | |

# INTERNATIONAL ATOMIC ENERGY AGENCY

## Programs to Improve Nuclear Power Safety Around the World

The International Atomic Energy Agency (IAEA), a member of the United Nations' family of organizations, has played two important roles in Western efforts to improve the safety of Soviet-designed nuclear power plants. By serving as a forum where nuclear experts from the former Soviet Union and Eastern Europe can meet their Western counterparts and freely exchange views on plant design and operation, it has done much to develop and sustain an East-West dialogue. Through its program for reviewing nuclear plant design, operating practices and accident prevention programs, the IAEA has—with the full cooperation of the former Soviet government and Eastern European governments—provided an independent assessment of many of the nuclear plants in operation and under construction in these countries.

## International Meetings

In August 1986, the IAEA convened a special meeting on the Chernobyl international organizations. At that gathering, Valeriy Legasov, a member of the U.S.S.R. Academy of Sciences and a vice director of the Kurchatov Institute of Atomic Energy, gave a detailed description of the accident and what the Soviets had done to deal with its consequences and prevent a recurrence.

Legasov said the Soviet Union sought international cooperation aimed at improving the safety and operation of Soviet nuclear power plants.

Ten years after the accident, April 1-3, 1996, the IAEA and the United Nations' Department of Humanitarian Affairs sponsored an International Forum on Chernobyl's Nuclear Safety Aspects. Among the meeting's highlights:

- *Causes of the Accident.* Sufficient detailed information is available to identify the causes of the accident and take effective measures to prevent the repetition of such an event.

- *Safety of RBMKs.* Between 1987 and 1991, a first stage of safety upgrading was carried out at all RBMKs. The upgrades addressed the most serious problems identified: they reduced the void reactivity effect, increased the efficiency of the scram system and strengthened the operational organization.

- *Sarcophagus.* There is broad agreement on the risk of the sarcophagus' partial or total collapse during its design lifetime—about 30 years. The stabilization of the sarcophagus is thus a high-priority safety issue. Although the structure is currently safe from the point of view of criticality, configurations of fuel masses exist inside it that could reach a critical state when in contact with water. Thus, water entering the sarcophagus also is a significant safety issue.

The same month, the IAEA—together with the Commission of the European Union and the World Health Organization—sponsored an international conference on the consequences of the Chernobyl accident. More than 700 people attended the April 8-12, 1996 conference, which included updates on studies or projects undertaken by the three sponsoring organizations as well as the Organization for Economic Cooperation and Development's Nuclear Energy Agency and organizations in Germany, Japan and the United States.

Among the subjects addressed at the conference: initial response to the accident; releases and deposition of radioactive material; radiation doses; health effects, including the incidence of thyroid cancer, longer term effects and psychological consequences; environmental consequences; social, economic, institutional and political impact; nuclear safety and the sarcophagus (a summary of the results of the April 1-3 conference); and prognosis.

For more detail on the health effects, see **The Chernobyl Accident and Its Consequences** in the Ukraine section of this book.

## Evaluation of All Soviet-Designed Plants

In the late 1980s, the IAEA began receiving a growing number of requests for assistance in nuclear safety from countries operating Soviet-designed reactors. The IAEA responded in 1990 with an extrabudgetary program to evaluate the first generation of VVER-440 Model V230 reactors. The program's objective: to help countries operating Model 230s identify design and operational weaknesses, and to prioritize safety improvements. That program was expanded in 1992 to deal with VVER-440 Model V213, VVER-1000, and RBMK nuclear power plants in operation and under construction.

IAEA assistance focuses on three areas:

- Generic issues (for each reactor type)
  --Identification of safety weaknesses in design and operation based on current international safety standards and practices.
  --Categorization of safety issues according to their potential for degradation of the defense-in-depth safety concept.
  --Recommendation of the most effective safety improvements for reducing the overall risk of accidents.
  --Prioritization of recommended improvements for identified safety issues.

- Plant specific
  --Assistance to enable countries operating and/or building VVER and/or RBMK reactors to plan and evaluate safety modifications, and to verify that proposed modifications address the concerns identified through the generic activities.

- Training
  --Workshops on particular areas where safety deficiencies have been identified generically or on a plant-specific basis.

By late 1994, the program had identified design and operational shortcomings of VVER and RBMK nuclear power plants, and the related safety significance. The IAEA reached international consensus on the major safety issues for all Soviet reactor types, ranked according to urgency and significance with respect to the defense-in-depth concept.

For the VVER 440 Model V230, 97 safety issues related to plant design and operation were identified. For the VVER 440 Model V213, 87 issues were identified, for the VVER-1000, 84 issues, and for the third-generation RBMK, 58 issues.

The IAEA is now assisting countries to review safety improvements that have been proposed and/or implemented. IAEA member countries operating VVER 440 Model V213 and VVER-1000 plants, for instance, have asked the agency to review the completeness of proposed safety improvements with respect to the agreed list of safety issues.

## VVER-440 Model V230 Program

In September 1990, the IAEA launched the first phase of its VVER-440 Model V230 program, identifying site-specific and generic design and operational safety concerns. By the end of 1991, phase one was completed, with the publication of a document evaluating the significance of the identified safety issues. The document also provided a basis for the short- and long-term activities needed to improve plant safety.

The IAEA began phase two early in 1992. Phase two activities include:

- Evaluating potential modifications to plant hardware and operations,

- Reviewing matters of generic safety concern, and establishing a consensus on needed actions.

- Some of the major safety issues in the VVER-440 Model V230s are presented in the following sections. Fuller information is contained in the relevant IAEA reports.

### Reactor Vessel Integrity

A priority of the safety reviews and modifications has been the establishment of preventive measures that enhance “defense in depth.” An important part of preventing severe accidents is to ensure the integrity of the reactor pressure vessel. As such, all operating VVER-440 Model V230s—except Kozloduy 4 and Medzamor 2—have been annealed to restore the material properties of the reactor vessel degraded by radiation. Measures to further reduce damage from high neutron flux and to reduce loads from cooling transients also have been implemented in the plants.

The IAEA notes that further evaluation of the effects of annealing is needed. Similarly, regarding the structural response of the reactor pressure vessels, special attention has to be paid to the transients leading to pressurized thermal shock. Finally, concerning pressure vessel integrity, the agency says that completion of work to validate vessel assessment methods used to quantify safety margins and to demonstrate their conservatism is of utmost urgency.

The IAEA is preparing guidelines for use in evaluating pressurized thermal shock analysis.

### Primary Circuits

The integrity of the primary circuit of plants also has been scrutinized. In VVER-440 Model V230s, a limited functional capability of the emergency core cooling and containment systems means that they cannot cope with large primary circuit breaks. Therefore, the IAEA is providing assistance in applying the leak-before-break concept.

To date, leak-before-break application studies have been conducted on a plant-specific basis at Bohunice and are close to completion at Kozloduy; a generic study is under way for Kola and Novovoronezh. Monitoring systems have been installed or are planned to ensure prompt detection of leaks. Related integrity assessments —using the leak-before-break concept—are planned for steam and feedwater lines, to prevent damage to primary circuit and safety systems resulting from breaks in those lines.

Further work is needed to assure the integrity of secondary piping and to evaluate areas subject to restricted inspection. Also, measures to ensure integrity of pressurizer surge lines are required.

### Safety and Support Systems

Safety and support systems is another area of concern indicated by the IAEA. Improvements have been made or are being implemented in some plants to ensure sufficient steam generator water inventory during abnormal operation; to improve the capability and redundancy of the emergency feedwater system; and to improve redundancy and separation of the residual heat removal and related support systems. Bohunice has added an emergency feedwater system outside the turbine hall, and it has improved the pressurizer safety valves system. Novovoronezh, Kola and Kozloduy are planning to install additional redundant emergency feedwater systems protected against internal and external hazards.

Elsewhere, improvements have been completed or are planned to install remote shutdown panels, define the required post-accident monitoring instrumentation, assess the reliability of existing instrumentation and control equipment, improve physical separation, and review the control room layout. Bohunice, Kola and Novovoronezh plan to replace the reactor protection and engineered safety features actuation systems with new ones complying with current safety standards. They also plan to establish safety parameter display systems, and Bohunice intends to install a new redundant and seismically qualified water system able to provide sufficient heat sink to all safety systems.

Further work is needed, according to the IAEA, in the area of safety and support systems. The design basis of new emergency feedwater systems needs to be established. The design criteria to improve the emergency core cooling systems have to be detailed and clarified—and the plant response to all possible loss-of-coolant accidents further analyzed—to ensure short- and long-term cooling capability. Accident analysis covering all of the locations and break sizes is needed to establish the design bases for the emergency core cooling system improvement, and IAEA has developed guidelines for such analysis. The design criteria for the emergency remote shutdown panel have to be investigated further.

## Confinement

VVER-440 Model V230s suffer from very poor leak tightness due to their deficient containment capability. To address the problem, plants have launched a comprehensive program to detect and repair sources of confinement leaks. A first phase of improvements reduced the confinement leak rate by one order of magnitude. Improvements are planned to reduce the leak rate further. In addition, Bohunice is developing a design for confinement upgrading; Kola and Novovoronezh are considering similar modifications.

The IAEA has recommended further improvement of confinement leak tightness. The agency says that structural analyses are needed to determine the ultimate pressure capability of the confinement structure and to identify limiting points and expected failure modes. The spray systems should be modified to provide two redundant and separate trains to guarantee the performance of confinement. Additional studies on the prevention of hydrogen deflagration hazards are needed. And the adoption of a negative pressure approach should be considered to limit a post-accident release of high-level radiation.

## Conduct of Operation

As with most Soviet-designed plants, electricity production by the VVER-440 Model V230s came at the expense of safety. The lack of adequate operational and maintenance procedures and practices at the plants is being addressed through twinning programs and other technical exchange agreements with Western plants. Quality assurance programs are being implemented, radiation protection practices have been improved, and operating and administrative procedures have been or are being prepared and reviewed.

The housekeeping and material condition of the plants have improved. Most of the IAEA's recommendations on maintenance and surveillance testing practices have been addressed at Bohunice, Kola and Novovoronezh; a predictive maintenance program is planned for Kozloduy. A systematic approach for operator training has been adopted by Novovoronezh and computer-assisted training for Bohunice is under development. Kozloduy is about to begin operating its new training center. Emergency planning improvements have been carried out at Bohunice, Kola and Novovoronezh.

Although restoration of Kozloduy 1 and 2 has been carried out, improvements to the material condition of the other units are needed, and emergency planning at the plant needs to be reviewed. All of the VVER-440 Model V230s need control room design/human factors reviews. The lack of a full-scope site-specific simulator at Kola is impeding progress in the operator training program and the implementation of emergency operating procedures. Further improvements also are needed at all plants in the areas of maintenance procedures, maintenance personnel training, quality control procedures, and spare parts supply and control.

## Seismic Safety

In 1990, an IAEA mission to Kozloduy concluded that the plant lacked a sufficient safety margin for the estimated design basis earthquake. Some improvements have been made, but structural upgrading is still required for all four units.

At Bohunice, engineering and construction work has been ongoing since 1991, reducing the seismic risk of the units.

In 1993, at the request of the Armenian government, the IAEA established a technical cooperation program to assist in that country's efforts to restart the two units shut down shortly after the 1988 earthquake. Although the plant was undamaged, its location in an earthquake-prone area leaves three main problems to be solved:

1. Geological stability of the site should be demonstrated.

2. Seismic design basis for the site should be re-evaluated.

3. Seismic requalification of the plant's buildings and components to the new seismic design basis should be performed.

The first two problems have been resolved.

Through the IAEA technical cooperation program, Armenia is receiving assistance in strengthening its regulatory body, establishing a plan of action consistent with IAEA recommendations for VVER-440 Model V230s and addressing point three above—seismic requalification.

## VVER-440 Model V213 Program

The VVER-440 Model V213 reactor designs incorporate substantial safety improvements compared with their predecessor, the VVER-440 Model V230. Nonetheless, the 213s still lack many safety features found in Western nuclear power plants. So, in 1993, the IAEA launched a broad safety review of these plants. Much like the program to improve the 230s, this IAEA program looks at both generic and plant-specific issues. Some of the major safety issues in the VVER-440 Model V213s are presented in the following sections.

### Bubbler Condenser Containment Performance

VVER-440 Model V213 reactors are equipped with bubbler condenser-type containments, in which peak pressure after large-break loss-of-coolant accidents is reduced by a steam suppression system. While this approach to containment has several positive elements, it also has raised concerns. Specifically, experimental support for this type of construction is needed, since the containment design involved several original developments.

The IAEA has prepared guidelines for bubbler condenser structural evaluation. The application of these guidelines has confirmed that the structure needs to be reinforced to withstand the effects of an instantaneous guillotine break of the 500 millimeter reactor cooling pipes.

To address the safety concerns related to the bubbler condenser structure, large-scale experiments in natural geometry are also needed to investigate: maximum pressure difference on internal structures, uniformity of flows in the bubbler condenser structure, and pressure oscillations.

To date, experimental structures have been built—but are not yet operating—at Zugres in Ukraine and Bechovice in the Czech Republic.

IAEA also recommends that mechanical strength analyses now be performed on a plant-specific basis, and that regulatory authorities should determine the rules for bubbler condenser containment evaluation.

IAEA is preparing guidelines for containment evaluation.

### Protection of Emergency Feedwater Systems Against Common-Cause Failures

Despite being redundant and independent, the emergency feedwater systems in the 213s are wholly located within the turbine hall. As a result, the components of the systems are exposed to common-cause failures due to fire, flooding, steam line break, or a seismic event.

The IAEA has proposed that the emergency feedwater systems be located outside the turbine hall, with the routing of lines so that no common-cause hazards can damage more than one line.

### Handling of Large Primary to Secondary Leakage at the Steam Generators

VVER steam generators have, as a unique feature, a cylindrical primary header with a bolted flange. A rupture of this component would allow radioactive water steam to bypass the containment, because the primary water would go to the steam relief valves and to the environment—a scenario not considered in the initial design of the plant.

A number of corrective actions have been planned and partly implemented at the plants. They include: improving primary to secondary system leakage detection, increasing emergency core cooling system water reserves, and improving the pressurizer spray system.

### Protection of the Containment Sumps from Clogging

During a loss of coolant accident, the containment sumps should collect water escaping the reactor coolant system and make it possible for the emergency core cooling system to recirculate the water. However, strong jets of water or steam from broken pipes could tear thermal insulation from primary piping. This insulation could clog the containment sump filters, cutting off recirculation of water for core cooling.

One fix under consideration would be to replace the thermal insulation. However, that change would be costly, could introduce other problems, and would not guarantee an improvement of the situation. Other possible solutions are still being sought.

### Improvements in the Ventilation Systems of Control Rooms

Control rooms of VVER-440 Model V213s are not equipped with separate ventilation systems capable of filtering the intake air in case of radioactivity releases outside the containment. This is a major concern, since the safety of control room operators is required for proper management of an accident.

Improvements at all plants with 213-design reactors are planned. Redesigned ventilation systems should, according to the IAEA, be able to: supply the main and emergency control room with filtered air, free of radioactive material; and prevent contaminated air from entering the control room by maintaining overpressure in the rooms.

### Reconstruction of Instrumentation and Control

The instrumentation and control (I&C) systems of the 213s represent the technical level of the early 1970s. In addition to being outdated, the reliability of the systems is questionable, and they require an inordinate amount of effort to keep them in operation. Even then, the I&C systems do not always fulfill single-failure criteria, and the physical separation of redundant trains of the reactor protection system is inadequate.

Given the importance of I&C systems to the safety of nuclear power plants, reconstruction of the systems is planned at various VVER-440 Model V213s. The list of necessary work includes:

- Performing complex analyses of instrumentation and control performance in the plants.
- Replacing the most unreliable units.
- Analyzing physical separation between redundant I&C systems with respect to common-cause events.
- Checking full electrical isolation between control and protection functions.
- Exchanging the existing systems as needed, using qualified modern I&C apparatus with special attention to self-monitoring, testability and fault-tolerant design of the systems.
- Conducting necessary environmental studies and implementing needed backfitting measures.

### Fire Protection

Like most nuclear power plants developed during the early 1970s, the VVER-440 Model V213s lack sufficient attention to fire hazards. However, unlike most of those plants, the 213s have incorporated few of the improvements to fire protection, detection and suppression that swept through the rest of the world following the Browns Ferry fire in 1975.

The reduction of fire hazards is one of the most important tasks needed for improving the safety of these Soviet-designed plants. Systematic fire hazard analyses for each area of every 213 are needed. The analyses should identify the weak points of the fire barriers, show the need to separate redundant trains of safety important systems, and justify the acceptability of redundant train separation by distance. Additional analysis should be performed to identify the measures needed to improve fire prevention and fire suppression capability.

### Seismic Safety

At the request of its members, the IAEA initiated the Coordinated Research Program on the Benchmark Study for Seismic Testing of VVER Type Nuclear Power Plants in 1992. Two types of reactors—the VVER-440 Model V213 and the VVER-1000—were selected for a benchmarking study, which will be used to coordinate methods and criteria related to seismic safety. The Paks plant was selected as the study's 213 reference plant. The study includes a state-of-the-art seismic analysis and dynamic, full-scale testing, using explosions and/or vibration generators.

After an initial meeting at the Paks plant in 1993, on-site testing of the plant's equipment was performed; preparations for the full-scale dynamic testing are under way. Also in 1993, the IAEA reviewed seismic input at Paks and conducted two seismic safety missions to review the work already done on seismic input and seismic capacity.

The Mochovce nuclear plant, another VVER-440 Model V213, underwent a preliminary review of the re-evaluation of its seismic design basis in 1993.

## VVER-1000 Program

In February 1992, the IAEA was asked to expand its safety program on the VVER-440 Model 230 reactors to other Soviet designs. Bulgaria, Czechoslovakia and Ukraine separately requested that the agency initiate a more comprehensive safety evaluation of VVER-1000 nuclear power plants.

The VVER-1000 is a design that shares similarities with Western plants, in terms of design philosophy, design features and constructability. However, concerns remain about engineering design solutions, quality of manufacture, and reliability of equipment.

The strategy for improving the safety of VVER-1000s is similar to the IAEA's plan to upgrade the VVER-440 Model V213s. The main elements of the VVER-1000 program follow.

### Steam Generator Collector Integrity

Between 1986 and 1991, 24 VVER-1000 steam generators developed cracks in primary cold collectors; cracking occurred after 7,000-60,000 hours of operation, and was determined to be caused by environmentally assisted cracking at temperatures of about 280 degrees C. Although cracked collectors were generally replaced, and the cause identified, concern remains: As of November 1993, 19 operating VVER-1000s had been outfitted with 76 of the steam generators in question.

The rupture of steam generator collectors could initiate accidents of high safety significance in two ways: The radioactive primary coolant could be discharged to the environment through the main steam atmospheric dump; and the long-term cooling of the core cannot be assured in the event of loss of primary coolant water through the main steam atmospheric dump.

In addition to the existing corrective measures, the IAEA has suggested improvements related to detection, inspection, repair, material, manufacturing processes, stress relieving, accident mitigation, and operating conditions. A new, improved steam generator design is under consideration at Gidropress, a Russian nuclear components manufacturer. The following are other important future activities:

- All adopted measures should ensure a low probability of a catastrophic break of the collectors.
- The current estimates of the safety consequences of a steam generator rupture accident should be reviewed, with the aim of developing preventive and mitigative accident management procedures.
- In the short term, preference should be given to upgrading the main steam atmospheric dump valves for discharging of steam-water mixture and to developing procedures for better maintaining the water inventory.

### Fuel Assembly Structural Instability

Deformed fuel assemblies were discovered at Balakovo and Zaporozhye 1. The problem was observed after an irradiation of two years in the core. In addition, the distance between spacer grids was no longer uniform. Preliminary results of a post-irradiation examinations by Russia's Scientific Research Institute of Nuclear Plant Operations confirmed the deformation of whole fuel assemblies; the institute continued its study in 1994, and is looking into whether the cause is a design problem. The spacer grid movement may be the result of inadequate loading.

While a root cause analysis is under way, design modifications to make the fuel assembly structure more rigid and to provide dimensional stability are being considered by the Russian designer.

### Control Rod Insertion Reliability

During the refueling of Zaporozhye 1 in late 1992, it was discovered that eight control rod assemblies were not at the bottom position. Subsequently, the same problem was seen at Balakovo, Kalinin, Khmelnitskiy, Rovno and South Ukraine. In addition, an increased drop time exceeding the maximum design value was observed. Most of the problems have occurred during the third year of operating an assembly in the reactor.

Root cause investigations are being conducted. A preliminary conclusion links the problem to an increase in the friction between the control rods and their guide tubes in the fuel assemblies due to shape changes of the guide tubes or possible rubbed surface roughness. There appears to be a close correlation between the control rod insertion problem and the structural instability of fuel assemblies.

While the IAEA stresses the importance of determining the root cause and implementing measures to eliminate the problem, the agency notes that the final solution may rest on the new improved design of fuel and control assemblies.

### Seismic Safety

As part of the IAEA's benchmarking study on seismic safety, Kozloduy (units 5 and 6) was selected as the VVER-1000 reference plant. Extensive analysis and dynamic tests are planned. At Temelin, site safety has been documented, and progress review meetings were held on the topics of tectonics, microearthquake monitoring and hydrogeology.

## RBMK Program

Fourteen RBMK (Chernobyl-type) reactors are in operation in the former Soviet Union—11 units in Russia, one in Ukraine, and two in Lithuania. The design of the reactors evolved over the 17 years between construction of the first and last units, so it generally is recognized that there are three generations of RBMKs. For the generation of each operating reactor, see the RBMK section of **Soviet Nuclear Power Plant Designs**.

Major safety concerns exist with respect to the RBMK reactors—particularly the first-generation designs, which lack a dedicated emergency core cooling system and a pressure suppression system.

Since the 1986 Chernobyl accident, a number of safety-related improvements have been made to the RBMKs. Measures have been taken to reduce the void reactivity coefficient. The problem of reactor instability has been addressed with new operating procedures. The control rod design was improved. And some corrective measures have been implemented in the area of fire protection, detection and suppression.

Despite these safety enhancements, concerns about the RBMKs persist. The IAEA program has identified the design and operational shortcomings of third-generation RBMK nuclear power plants based on a review of Smolensk 3 and the Ignalina plant. The IAEA results include insights from other national, bilateral and multilateral projects. The following are the principal areas being addressed in the IAEA's RBMK program.

### Shutdown System

After the Chernobyl accident, the shutdown system of all RBMK reactors was improved by: modifying the control rod design to eliminate the positive scram effect; reducing the rods' insertion time; incorporating the short bottom rods to the shutdown system; and implementing 24 fast-acting scram rods.

The RBMK shutdown system consists of the fast-acting emergency protection system (which uses all fast-acting scram rods) and the emergency protection system (which uses all control rods). However, these subsystems do not fully meet the basic principles of shutdown system requirements as defined by the IAEA, nor do they comply with the licensing practices observed in Western countries.

To correct the problem, Russian designers intend to develop and modernize the RBMK control and protection system. The shutdown system would adhere to recommended IAEA safety standards.

### Multiple Pressure Tube Failure

The possibility of the rupture of multiple pressure tubes is one of the highest-priority safety issues related to channel-type reactors, including RBMKs. Scenarios have been developed that show the potential for such an accident.

Analysis of the outcome of such an accident is needed. In addition, further experimental information is needed to better understand the physical phenomena involved. The IAEA recommends collaboration among RBMK designers and experts in Western countries familiar with modern safety analysis techniques for channel reactors.

Issues to be addressed include the nature of low-flow transients in multiple parallel channels, thermal mechanical response of graphite to channel failures, and high-temperature pressure tube failure mechanisms.

In 1995, the IAEA issued a report on multiple pressure tube failure, and the agency has initiated an international exercise to validate computer codes used to analyze multiple pressure tube ruptures. The validation work is based on experimental results made available through the IAEA by the Japanese government.

## Planned IAEA Activities

The IAEA will provide assistance in 1997 and 1998 within the framework of three regional technical cooperation projects and its extrabudgetary program. The extrabudgetary program will be completed in 1998. After that, further IAEA assistance should be included in the agency's regular nuclear safety and technical cooperation programs.

***Technical Cooperation Projects.*** Under these projects, assistance will be provided to:

- enhance national capabilities for assessing the operational safety of nuclear power plants

- support the safety assessment of nuclear power plants, and

- strengthen nuclear safety legislative and regulatory infrastructures.

***Extrabudgetary Program.*** Two important areas of work are:

- keeping up to date the technical data base of IAEA findings and the plant-specific status of safety modifications, and helping the G-24 Nuclear Safety Assistance Coordinating Center—which relies on the data base—to identify gaps and overlaps in assistance projects.

- providing a forum for establishing international consensus and sharing information on technical measures required to resolve the highly significant safety issues identified in the nuclear plant evaluation program.

In addition, the IAEA has identified several design safety issues whose resolution requires consensus on the technical approaches to be used. These include:

- *Issues related to analysis tools and methods:* accident analysis, primary to secondary leaks, pressurized thermal shock analysis, quality and reliability of in-service inspection, integrated neutronic thermohydraulic 3D codes for RMBK core and system analysis, validation of thermal hydraulic best estimate codes for RBMK LOCA analysis

- *Issues related to containment/confinement integrity:* VVER 440 Model V213 containment strength, containment improvement options for Model 440 V230 nuclear power plants, RBMK confinement improvement

- *Issues related to reactor core:* control rod insertion reliability, RBMK fuel channel integrity, RBMK shutdown system modernization.

## Operational Safety Services

The IAEA carries out various types of nuclear plant services, including:

***Pre-OSART (Operational Safety Review Team)*** missions for plants under construction. The team examines construction quality, commissioning arrangements and preparations for operations that will have a bearing on eventual operational safety.

***OSART (Operational Safety Review Team***) missions, which focus on operational safety practices. Within 12-18 months of a Pre-OSART or OSART mission, IAEA conducts an on-site follow-up review to assess progress in implementing the initial proposal for improvement.

***ASSET (Assessment of Safety Significant Events Team)*** missions, which examine operating history and incident prevention programs. Under a new system introduced in 1996, much of the data on the direct cause and then the root cause of each safety-significant event—procedure, personnel or equipment—are gathered by the plant personnel, who then determine appropriate corrective actions. The ASSET mission then reviews the plant's self-assessment, draws conclusions and makes suggestions on enhancing operational safety and plant safety culture. The team may also suggest a follow-up ASSET mission.

Prior to the introduction of this peer review system, the IAEA team reviewed the data and recommended corrective actions itself. The agency sent an ASSET mission to help plant management implement the recommendations of the initial mission, and then sent a follow-up ASSET mission to determine the effect of the implementation of recommendations.

More than two dozen missions have been completed, including:

- One Pre-OSART (Belene), one OSART (Kozloduy) and one ASSET (Kozloduy) in Bulgaria
- One Pre-OSART (Temelin), one OSART (Dukovany) and two ASSETs (Dukovany) in the Czech Republic
- One Pre-OSART (Mochovce), one OSART (Bohunice) and one ASSET (Bohunice) in the Slovak Republic
- One ASSET (Greifswald) in former East Germany
- One OSART and one ASSET (Paks) in Hungary
- One Pre-OSART (Zarnowiec) in Poland
- One Independent Safety Review (including Pre-OSART) (Gorkiy), six OSARTs (Rovno, Kola, Novovoronezh, Ignalina, Zaporozhye and Bohunice), three Safety Review Missions (Temelin, Khmelnitskiy and South Ukraine), one Follow-Up Safety Review Mission (Bohunice) and 15 ASSETs (Kursk, Chernobyl, Ignalina, Khmelnitskiy, Balakovo, Leningrad

[Sosnovyy Bor], Smolensk, Novovoronezh, Kola, Rovno, Zaporozhye, South Ukraine and Kalinin) in the former U.S.S.R.

This list does not include missions conducted under the VVER-440 Model V230 project. Those missions, consisting of ASSET missions and Safety Review Missions (which covered both design and operational aspects), were conducted at Bohunice in the Slovak Republic, Kozloduy in Bulgaria, and Kola and Novovoronezh in the former U.S.S.R.

Several missions are scheduled for 1997 and 1998:

- One peer-review ASSET to Bulgaria: (Kozloduy, 1997)
- Three peer-review ASSETs to Russia: (Balakovo, 1997; Novovoronezh, 1998; and Kalinin, 1998)
- Four peer-review ASSETs to Ukraine: (Rovno, 1997; South Ukraine, 1998; Chernobyl, 1998; and Khmelnitskiy, 1998)
- One Follow-Up Safety Review Mission to Russia (Novovoronezh 5 [a VVER-1000], 1997)
- One Follow-Up OSART to Ukraine (Khmelnitskiy, 1997)
- One Follow-Up OSART to Lithuania (Ignalina, 1997)
- One Follow-Up OSART to the Slovak Republic (Bohunice).

# THE WORLD ASSOCIATION OF NUCLEAR OPERATORS

## Improving the Safety of Nuclear Plants Worldwide

### Background

On May 15, 1989, representatives of 144 electric utility organizations with operating nuclear power plants around the world gathered in Moscow to charter the World Association of Nuclear Operators (WANO), a new international organization intended to further enhance the safety and performance of nuclear power plants worldwide.

By taking this landmark step and committing their organizations to freely exchanging and using operating information, the founders built in part on the American model for achieving excellence in nuclear plant operations—the Institute of Nuclear Power Operations (INPO).

WANO was formed in response to the 1986 accident at the Chernobyl nuclear power plant in the Soviet Union. That event demonstrated the need for international cooperation and information exchange in nuclear plant operations. It also created a determination among nuclear utilities worldwide to work together for improved safety and reliability in plant operations.

WANO is based on the recognition that the ultimate responsibility for a nuclear plant's safety and reliability rests with the operator. Every organization in the world that operates a nuclear power plant is a member of WANO. The organization operates through four regional centers—in Atlanta, Moscow, Paris and Tokyo—and a small coordinating center in London.

## WANO Works to Enhance Nuclear Plant Safety and Performance Worldwide

WANO's mission is to maximize the safety and reliability of nuclear power plant operation by exchanging information and encouraging communication, comparison and emulation among its members. This mission is achieved through five main programs:

- ***The Peer Review Program***
  This program is designed to help WANO members compare their operational performance against the best international practices through an in-depth review of their operations by an independent team from outside their utility. The review, carried out at the request of the plant, is conducted by an international review team consisting of individuals from other WANO member nuclear power plants.

- ***The Operating Experience Information Exchange Program***
  This program enables members to learn from the operating experience of other plants. WANO screens and analyzes events that occur at nuclear power plants worldwide to identify possible precursors of more serious events, and disseminates the lessons learned to its members.

- ***The Performance Indicator Program***
  With each member providing data on its performance, WANO members can compare their performance with that of other plants around the world. Data is reported, trended and distributed for 10 performance indicators that relate to nuclear plant safety and reliability, plant efficiency and personnel safety.

- ***The Operator to Operator Exchange Program***
  This program enables members to directly share operating experience and ideas for improvement. This occurs through face-to-face communication—such as workshops and technical exchange visits—and through an electronic communications system used to transmit event reports and to exchange questions and answers on routine plant operations, maintenance or technical matters.

- ***The Good Practice Program***
  This program enables members to learn about a technique, program or process that has been proven particularly effective at one or more nuclear plants. Good practices are made available to members, who can implement them at their own plants if applicable.