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SLOVENSKÉ ELEKTRÁRNE, A.S.
NUCLEAR POWER PLANT MOCHOVCE
VVER 4×440 MW III CONSTRUCTION

Thematic Boxes

ANNEX 5.0

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5.1 CONSIDERATIONS ABOUT MO34 CLASSIFICATION

Mochovce 3-4 NPP is an “evolutionary design” (as defined by IAEA-TECDOC 936), like all the so-called Generation III reactors, since it is firmly based on proven and well-consolidated technology of currently-operating NPPs and, as appropriate, introduces significant safety and performance upgrades, implementing lessons learned from operating experience in order to ensure compliance with the latest international safety requirements and practices while putting a strong emphasis on maintaining proven design to minimize technological risks.

Generally Generation I plants are those developed at the dawn of the nuclear era, at the end of the ‘50s and at the beginning of the ‘60s. They had limited power output (100-300 MWe), simplified systems, no High Pressure Injection systems, no standardization, etc.

Generation II plants are those developed at the end of the ’60s through the end of the ‘70s. They featured a much higher power output (800-1100 MWe) and a degree of standardization. The US requirements for ECCS (10CFR50 App. K) were implemented. The safety was still totally based on deterministic analysis, PSA was not considered a design tool, no evaluation and no explicit consideration of severe accidents were involved. The large majority of the operating plants comply with this general picture.

After the Three Mile Island (TMI) accident in March 1979, the Safety Authorities required a number of backfitting to existing Generation II plants and an extensive revision of safety requirements were introduced for the plants to be built. The importance of the containment system to cope with even beyond design basis accident was underlined, as shown by the TMI scenario. The analysis of severe accident scenarios and the use of PSA for new plants became more and more normal practice. A PSA has been required to all plants as a verification of the safety margins to severe accidents. The Chernobyl accident in 1986, while not affecting directly the bases of LWR’s safety, further increased the attention to severe accidents, while new experiments and new computer codes allowed a more detailed and reliable assessment of severe accident prevention and mitigation scenarios.

All these considerations led at the end of the ‘80s and during the ‘90s to the development first of a set of utility requirements for a new generation of reactors (URD in the USA with participation of several other utilities and later EUR in Europe) and in parallel to the development of a set of new designs both of evolutionary and of passive natures (AP600, SBWR, EP1000, AP1000, VVER1000/92, EPR). This series of rectors are generally called Generation III plants and are currently available in the market for construction.

To complete the picture there are also the so-called Generation III+ plants or Near Term Deployment plants, which are in the pipeline of engineering and testing (no one is available yet) and will be an optimization of the existing ones and finally the Generation IV plants that will be available in the market from 2025 on.

VVER 440/213 is a reactor model that is in operation in a number of countries and in Slovakia too with very good performances and safety records. They have been backfitted according to the evolution of the technology, they have been assessed in depth by several groups of experts, including a massive work by

IAEA and OECD/NEA and a number of experiments and calculations have been carried out to prove the assumptions. All this work led to the conclusion that these plants are acceptable not only from the performance point of view, but also from the safety point of view. The European Union has not asked to any accessing country to shutdown this series of reactors.

In the case of MO34, while all the improvements implemented in plants of the same type will be obviously incorporated, the possibility exists of making further improvements during design completion and construction.

Therefore MO34 will become a kind of new generation of VVER 440/213, since it will feature for example:

- improved prevention of core melt accidents (e.g. dedicated full depressurization of the primary coolant system);
- capability to keep the corium, in case of core melt, inside the reactor vessel by external cooling, therefore greatly reducing the challenge to the containment;
- complete hydrogen management with autocatalytic recombiners and igniters, including consideration of hydrogen generated in a core melt accident;
- additional dedicated and totally independent containment spray system;
- improved redundancy and separation of safety system;
- use of PSA as a design tool;
- state of the art Instrumentation and Controls;
- improved containment leaktightness;
- compliance with IAEA requirements for new plants and WENRA;
- large compliance with EUR requirements;
- dose limits in case of accidents compatible with other new plants.

Therefore, MO34 will be finally not very different from many points of view from the modern Generation III plants, while it will capitalize the extensive operative experiences of the operation of many similar plants.

These considerations have been shared and agreed with an independent Safety Board setup by ENEL, whose six members are all well known and esteemed European experts in nuclear safety, who have played (and in some cases are playing) key roles of responsibility in Utilities, Safety Authorities, Academies of Science, International Organizations, Research Organizations and Universities.

After more than one year of activities, the Board has issued the positive statement that "*the Plant complies with the general principles of most recent international guidance, recommendations and requirements issued by international organizations. Moreover, the design has incorporated many of the principles and requirements included in the EUR's (European Utility Requirements), which are the requirements agreed by European Utilities for the advanced Nuclear Power Plants*".

5.2 OPERATION OF MULTIPLE UNITS

VVER-440 NPPs are designed usually as four unit plants. Each two units (1&2 and 3&4) are built as "twin units". In the case of Mochovce, all the four units are of the 213 type.

Units belonging to the "twin units" have certain common systems (fully or partially) and buildings, such as:

- demineralized water system;
- service water system;
- cooling water system;
- reserve 0.4 kV electrical bus (the units are reserve for each other);
- vent stack;
- diesel generator building;
- plant control room;
- reactor hall.

Systems common for the whole plant include:

- low pressure air supply system;
- make-up water preparation system;
- plant information centre;
- turbine hall (housing 8 turbines).

The above mentioned features determine the operational conditions of the plant. They influence also the organizational structure of the shift personnel. There are some organizational units that are common for two units or for the whole plant. Both advantages and disadvantages of this arrangement have been recognized.

Advantages

The advantages of multiple unit arrangement are both economical and safety related. The use of common systems, common organizational units, common maintenance and technical support are profitable from the economic point of view. From the operational safety point of view, the higher equipment/systems availability provided by the use of certain equipment of the other unit as a backup reserve seems to be an important advantage. For some accident management, multiple arrangement is very profitable. For instance, in the case of loss of off-site power, the probability of core damage in a multiple unit is much lower than for a one unit plant.

Likely disadvantages and their solution

Some disadvantages of multiple units arrangement may be safety related. Existing interconnections may permit an event that has occurred in one plant to be propagated to another unit (i.e. fire propagation).

Within the upgrading safety programme of EMO12 and MO34 some measures have been taken for eliminating or reducing these disadvantages. System reliability and safety studies have been undertaken to find appropriate solutions for all potential disadvantages of multiple unit arrangement. These activities concentrated on the following directions:

- Identification and elimination of the most hazardous interconnections, separation of the units;
- Reliability improvement in existing common systems;
- Installation of new common systems to increase the reliability;
- Overall plant safety improvement.

Examples of solutions that have been implemented are:

- Interconnection of the units at the 6 kV level;
- New technology for switch over from 400 kV line to 110 kV and installation of brand new electrical diagnostic system (DSE);
- For Units 3 and 4, installation of a common additional diesel generator unit (the standard configuration foresees 3 diesel generators for each Unit) that provides further electrical power supply for classified equipment required to mitigate consequences of severe accidents;
- Improvement of fire resistance in the area of the turbine hall roof support construction.

5.3 RADIATION PROTECTION AGAINST IONIZING RADIATION

Ionizing radiation

The term ***radiation*** is used to describe electromagnetic waves, such as light, radio waves and X-rays, and the particles emitted by radioactive materials as they disintegrate or decay to reach a non-radioactive state. These particles and the more energetic electromagnetic waves produce electrically charged particles, called ***ions***, in the materials they strike. This ***ionization*** can result in chemical changes; in living tissue, such changes can lead to injury to the organism.

Ionizing radiations are:

- ***α radiation*** (the nuclei of atoms of the element helium):

these particles are easily stopped and do not penetrate the skin; radioactive materials that emit alpha radiation can only be hazardous if swallowed or inhaled into the body, or if they enter the body through a break in the skin;

- ***β radiation*** (electrons):

these particles have greater penetrating powers than alpha particles but are stopped by relatively thin layers of water, glass or metal; radioactive materials that emit beta radiation can also be hazardous if taken into the body;

- ***γ radiation and X-rays*** (electromagnetic radiations):

these electromagnetic waves can penetrate relatively large thicknesses of matter before they are absorbed, but can be screened by a sufficient thickness of lead or concrete;

- ***neutrons*** (neutral particles present in all atomic nuclei except hydrogen):

these particles are also very penetrating but can be screened by thick layers of concrete or water.

Natural background radiation and human–made radiation sources

Natural background radiation comes from four primary sources: cosmic radiation, solar radiation, external terrestrial sources, and radon.

Human-made radiation sources are mainly represented by medical practice (diagnostic x –rays, use of radioisotopes, etc.); nuclear reactors for power generation; fallout from nuclear weapons testing; consumer products.

Radioactive decay

An important feature of all radioactive materials is that their activity decreases with time.

Radioactive decay is the process in which an unstable atomic nucleus loses energy by emitting radiation in the form of particles or electromagnetic waves. This decay, or loss of energy, results in an atom of one type, called the parent nuclide transforming to an atom of a different type, called the daughter nuclide.

Each material is characterised by a half-life, which is the time taken for half the radioactivity to decay. In two half-lives this is reduced to a quarter of its original level, and in ten half-lives to about one thousandth.

Half-lives of radioactive materials vary from fractions of a second to millions of years. In general, the most radioactive materials - those that emit intense penetrating radiation and require heavy shielding - decay to negligible levels relatively rapidly. Long-lived radioactive materials emit very little radiation, generally with low penetrating power; the hazard from such materials is principally associated with their being taken into the body.

Quantities of radioactivity are measured in ***becquerels*** (Bq). One becquerel of radioactivity corresponds to a rate (on average) 1 radioactive decay per second within the material of interest.

Dose quantities

The measure unit of **absorbed dose** is the ***gray*** (Gy), which corresponds to the deposition, in the matter, of 1 joule of energy per kilogram of material.

The units used to measure **equivalent dose** of radiation to individuals are the ***sievert*** (Sv), the ***millisievert*** (mSv) and the ***microsievert*** (µSv).

The sievert is a measure of the biological effect of radiation in humans exposed to ionizing radiation; it takes into account the way in which a particular type of radiation distributes energy in tissue so that we can allow for its relative effectiveness to cause biological harm. For gamma rays, X rays and beta particles, this radiation weighting factor is set to at 1, so the adsorbed dose and equivalent dose are numerically equal. For alpha particles, the factor is set at 20, so that the equivalent dose is deemed to be 20 time the adsorbed dose. Values of the radiation weighting factor for neutrons of various energies range from 5 to 20.

Instead, the **effective dose** is the **equivalent dose** weighted for the different harm to different tissue (by the tissue weighting factor). In fact, the risk of the various parts of human body varies from organ to organ. So, in case of partial irradiation of human body to different type of radiation the term **equivalent effective dose** (Sv) is used to quantify the overall equivalent impact on organs and body tissues.

One millisievert is one-thousandth of a sievert and one microsievert is one-millionth of a sievert. For example, doses of tens of sieverts to small regions of the body are used in radiotherapy to destroy cancerous growths while, in radioprotection, the international dose limit is fixed in 20 mSv/y for professionally exposed workers and 1 mSv/y for public.

It is sometimes useful to have a measure of the total radiation dose to groups of people or a whole population. The quantity used to express this total is the **collective effective dose**.

It is obtained by adding, for all exposed people, the effective dose that each person in that group or population has received from the radiation source of interest. For example, the effective dose from all sources of radiation is, on average, 2.4 mSv in a year. Since the world population is about 6,000 million, the annual collective effective dose to the whole population is the products of these two numbers, about 17,000,000 *man sievert,* symbol manSv.

It is common for effective dose to be abbreviated to *dose* and collective effective dose.

Biological effects

When ionization occurs in living tissue the resulting chemical changes can affect the behaviour of cells. The critical targets are the DNA molecules. These structures, present in every cell of the body, carry the information required for the development and division of cells and for the growth, proper function and reproduction of the organism. The damage to the DNA is often reparable, but in some cases can result in cell death or transformation.

Dead cells are normally absorbed or rejected by the organism. However, if a sufficient number of cells are killed, the functioning of the organism will be affected and it may die. Cell transformations (or mutations) do not necessarily lead to any deleterious effects. Indeed, many of such cellular changes occur normally during the lifetime of any organism. Very rarely, they result in a cancer or, in the case of the reproductive cells, in hereditary damage in later generations. Thus radiation can affect both the individual receiving the dose (somatic effects) and subsequent generations (hereditary effects).

Radiation Safety and ALARA

For all human actions that add to radiation exposure, or practices, ICRP recommends a system of radiological protection based on three central requirements. Each of these involves social considerations - explicitly in the first two and implicitly in the third - so there is considerable need for the use of judgement. They are the **Justification** of a practice, the **Optimization** of protection and the application of individual **Dose limit**.

ALARA is an acronym for “As Low As Reasonably Achievable”. This is a radiation safety principle for minimizing radiation doses and releases of radioactive materials by employing all reasonable methods. This policy is based on the principle that any amount of radiation exposure, no matter how small, can increase the chance of negative biological effects such as cancer, though perhaps by a negligible amount. It is also based on the principle that the probability of the occurrence of negative effects of radiation exposure increases with cumulative lifetime dose. At the same time, radiology and other practices that involve use of radiations bring benefits to population, so reducing radiation exposure can reduce the efficacy of a medical practice.

ALARA is not only a sound safety principle, but is a regulatory requirement for all radiation safety programs.

5.4 PRINCIPAL RADIOACTIVE DISCHARGE SOURCES

Operation of the NPP is typically cyclical. The reactor is designed to be run continuously for a certain period and then shut down annually, for one or two months, for routine maintenance, shuffling of fuel and partial refuelling.

Reactor Operation

Under normal operation, any leakage from, or partial failure of, the fuel cladding will lead to small amounts of fission products being released into the primary circuit. Tritium, produced in the fuel by fission, can be released through the cladding by diffusion and through any pin holes or defects. The amounts released depend on the design and quality of the fuel pins.

Small amounts of radioactive material may also be formed within the primary coolant as a result of neutron activation of fuel tubes, primary circuit and structural material surfaces.

Corrosion and erosion processes tend to release activation products from such materials into the primary coolant circuit. Tritium, generated from activation of boric acid in the primary coolant, is a particularly significant activation product. In addition, activation processes in the air surrounding the reactor pressure vessel produce small quantities of gaseous radioactive species including tritiated water vapour and noble gases.

A number of separate radioactive discharges from the reactor can be identified, concerned principally with chemical and volume control of the primary circuit coolant. Dissolved fission and activation products are removed from the coolant by an ion exchange process, which produces contaminated resins. The periodic removal and replacement of such resins generates both solid and liquid wastes. Periodically, some coolant is also discharged from the primary circuit in order to remove tritium, so that the activity concentration is maintained below a defined maximum operating limit. This discharge from the primary circuit also gives rise to a liquid waste stream.

Gases that grow up in the primary circuit during operation must be removed. This results in a gaseous waste stream. Atmospheric releases may also derive from the ventilation of fugitive emissions of primary circuit coolant through minor leakage. Such releases typically comprise tritiated water vapour, noble gases, aerosols and other vapours.

Estimates of the quantities of radioactivity present in the primary circuit coolant and the various waste streams have been made as part of the design basis of the reactor, using conservative assumptions. These estimates, together with consideration of the potential health impacts of any radioactive release, form a general basis for establishing operational limits in respect of emissions and waste management requirements. Information on discharges arising from normal operation, based on operating experience of other VVER-440 type reactors, illustrates that reactor operation can readily meet such discharge limits.

Indeed, in practice, plant performance against operational limits is routinely monitored by the regulatory authorities.

Refuelling and maintenance

At annual shut down, the cooling systems are depressurised, the primary circuit pressure vessel head removed, and one third of the fuel assemblies removed and transferred to a storage pond adjacent to the pressure vessel. The remaining two thirds are then rearranged to maintain optimum power densities and new fuel is inserted in the core. Typically, therefore, after the initial start-up period, each fuel assembly will remain in the reactor for three years.

In addition to the spent fuel, refuelling operations may give rise to active liquid effluents and atmospheric discharges that are of a similar nature to those derived from the primary circuit coolant during normal operation.

Repair and maintenance activities undertaken during shut-down also give rise to various contaminated solid wastes, caused by contact with activation products or by contact with contamination from the reactor primary circuit. Certain components, activated by neutron irradiation, may also be replaced, giving rise to solid wastes.

Classification of the sources of emissions

Activity sources within a nuclear reactor are generally classified as one of the following:

- fission products;
- corrosion products;
- activation products and actinides.

Fission products

Fission products are formed from nuclear fissions of atoms within the Uranium dioxide fuel. They comprise nearly 200 radioisotopes of some 40 different chemical elements (atomic numbers 30÷66) with diverse chemical and physical properties. Some are gases (e.g. the noble gases Krypton and Xenon), others are quite volatile at reactor temperature (such as Caesium and Iodine), and some are refractory metals (such as the Lanthanides). Cesium-137 is one of the most well-known fission products.

A considerable proportion of the fission product inventory is too short-lived to be of any environmental significance; these radionuclides decay rapidly before they are able to reach the environment in any significant quantity.

A series of barriers prevent release of fission products into the primary coolant. These include the fuel matrix itself, which serves to contain the majority of non-volatile fission products under normal operation. The volatile fission products tend to migrate through the fuel matrix and accumulate at the grain boundaries and its gaps within the fuel pin. The fuel cladding is designed to contain the volatile fission products within the fuel pin and to prevent contact between fuel and primary coolant.

Only gaseous fission product and the most volatile elements escape from the fuel matrix in any significant quantities and accumulate in the fuel pin gaps. The fuel cladding normally contains these radionuclides; however, some pins may develop small cladding defects as a result of mechanical or thermal stresses, corrosion or other causes. This can result in escape of some of the more volatile fission products into the primary coolant. Gross failure of cladding is also possible, but activity monitoring of the primary coolant ensures that these failed pins are detected and can be removed from the reactor.

Activation and corrosion products

The neutron flux in the reactor core results in activation of stable isotopes in various materials, including those found in the fuel cladding, structural components, coolant water and dissolved ions, dissolved air, and air in gaps in the reactor shaft. The active isotopes may be created by simple neutron capture or by secondary processes, such as neutron capture followed by α-decay. Some activation products are very short-lived and are not significant for environmental impact assessment or waste management; however, they may have implications for shielding in the reactor hall as a result of their penetrating γ rays.

Activation of materials used for structural components and fuel cladding, and the subsequent corrosion or erosion of these materials, can lead to the presence of radionuclides in the coolant in either soluble or particulate form. However, the chemical properties of the primary coolant are controlled to minimise corrosion. This entails routine monitoring of a wide range of parameters, including pH, conductivity, transparency, boric acid concentration, Potassium and Sodium concentrations, dissolved gases, Fluoride and Chloride concentrations and oil content.

The activation products in the coolant can be soluble or insoluble, and they are transported by water to all parts of the primary system. This presents problems with regard to accessibility and safe maintenance of various components because of radiation fields. Among those activated corrosion products, the γ-emitting activities (Co-60, Co-58, Zn-65, Mn-65 and Fe-59) are more important in creating the radiation field problems. The longer-lived species (Fe-55, Ni-63 and Co-60) are of more concern with the problems in the radioactive waste handling and disposal.

Activation of Boron dissolved in the primary coolant leads to the formation of Tritium. Although with a low radiotoxicity, Tritium is potentially significant from a radiological point of view because of its chemical behaviour as Hydrogen, which means that it readily forms water molecules where it is chemically indistinguishable and therefore extremely difficult to separate from 'normal water'. Tritium, in the form of tritiated water, is highly mobile in the environment and in living tissue. Its half-life is approximately 12 years; it will therefore always be present in the coolant in quantities determined by the power history of the reactor and the coolant replenishment cycle. This is the most significant source of Tritium; other sources, such as its formation as a ternary fission product or by activation of Boron carbide used in the control rods, require failure in the cladding of the fuel rod or control rod respectively in order to let Tritium reaching the primary circuit.

Instead of undergoing nuclear fission, Uranium in the reactor fuel may absorb neutrons to form actinides such as Plutonium; these can reach the coolant by slowly leaching from exposed fuel if a breach in the cladding occurs.

The most radiologically important radionuclide arising from neutron activation of air in the pressure vessel shaft is Ar-41, produced by activation of Ar-40. This is discharged via the hermetic zone ventilation system through Iodine and aerosol filters to the stack. Decay tanks at the gas cleaning station are used to allow the Ar-41 (half-life 1.8 hours) to decay before final discharge. N-16 is produced by nuclear reactions of the Oxygen in the primary coolant water. Although it has a short half-life (seven seconds) it is transported around the reactor coolant circuit to the heat exchangers. It is unlikely to represent a health hazard unless released under accident conditions. However it does produce very high energy gamma rays which are very penetrating and which require adequate shielding.

5.5 PRINCIPLES OF MICROMETEOROLOGY AND DISPERSION MODEL

Micrometerology

Meteorology is fundamental for the dispersion of pollutants as it is the primary factor determining the diluting effect of the atmosphere. Contaminants discharged into the air are transported over long distances by large-scale air-flows and dispersed by small-scale air-flows or turbulence, which mix contaminants with clean air.

The main parameters that characterizes local climate (micrometeorology) and affects air quality are represented by temperature, relative humidity, wind speed and direction, pressure, precipitation and solar radiation. The meteorological data set is derived from observation of the main parameters at the next surface station and from data that has to be inferred from other measurements as the atmospheric stability and the mixing height.

Atmospheric stability is a measure of the propensity for vertical motion and hence is an important indicator of the likely magnitude of pollutant dispersion.

Mixing Height or Mixing Depth is used to quantify the vertical height of mixing in the atmosphere. It is the height at which vertical mixing takes place. Forecasting of mixing height is done with the aid of the vertical temperature profile. A radiosonde is sent aloft and temperatures at various altitudes are radioed back. The altitude at which the dry adiabatic line intersects the radiosonde measurements is taken as the maximum mixing depth (MMD). The dry adiabatic line is defined as a decrease of about 1 centigrade over height of 100 m. The MMD is a function of Stability. In Unstable air the MMD is higher and in Stable air the MMD is lower.

Atmospheric stability and mixing height of the boundary layer are also required as extremely important issue in reference to air quality. We can tell how pollutant emissions are likely to disperse and what the likely ground level concentration patterns will be if we know how stable (or unstable) the atmosphere is at a given time. The stability of the atmosphere often dictates the behaviour of a plume in terms of the height it will rise and to what degree it will mix into the environment.

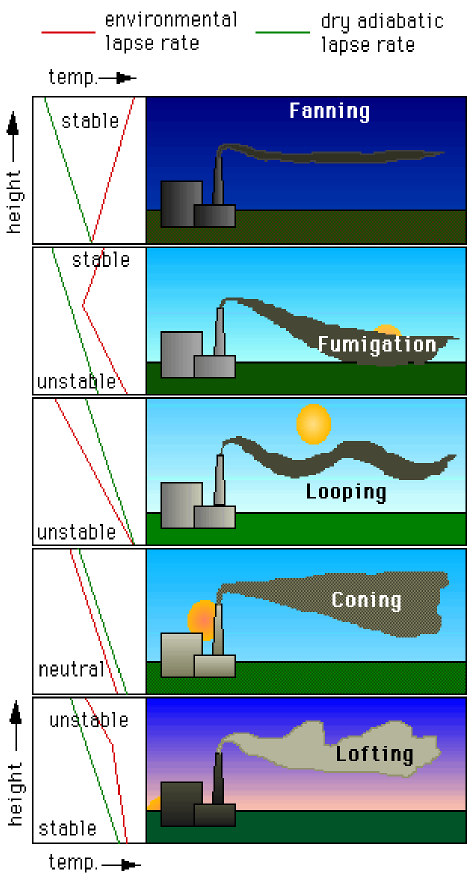
The oldest and, for a great many years, the most commonly used method of categorizing the amount of atmospheric turbulence present was the method developed by Pasquill in 1961, and later modified by Gifford.

Pasquill categorized the atmospheric turbulence into six stability classes named A, B, C, D, E and F with class A being the most unstable or most turbulent class, and class F the most stable or least turbulent class:

Unstable conditions promote the rapid dispersion of atmospheric contaminants and result in lower air concentrations compared with stable conditions.

P-G STABILITY CLASS	CONDITION	TIME of DAY
A	extremely unstable	day
B	moderately unstable	day
C	slightly unstable	day
D	neutral	day or night
E	stable	night
F	very stable	night

A plume released into an unstable atmosphere will display a looping pattern. Looping occurs when updrafts from warming air at the surface carry a segment of the plume upward while compensating downdrafts force the adjacent section downward. Coning occurs when a plume is released in the middle of a neutral layer. In addition to early evening, as the graphic illustrates, coning is common on over cast days and at night with the presence of strong winds. A stable atmosphere, commonly marked by an inversion on clear nights, yields a fanning pattern. A plume released into a stable atmosphere will not rise or mix unless it encounters turbulence. A fanning plume will often extend long distances downwind from the source.



Looping, coning, and fanning are characteristic of the more persistent conditions of stability, and for this reason, are observed for longer durations over a 24 - hour day. Fumigation and lofting, however, frequently characterize the transition periods between day and night and seldom last for more than a couple of hours. Fumigation occurs in the morning hours as the night-time inversion gradually disappears due to surface heating. As the surface heats, the air just above it warms. An unstable layer builds from the surface upward but remains capped by the inversion above. A plume released beneath the inversion is trapped near the surface until the inversion eventually disappears and is replaced by an unstable layer. Conversely, lofting occurs in the evening as soon as surface heating ceases and radiational cooling begins.

The behaviour of pollutants in the air is not only affected by the stability of the atmosphere. They are also affected by the direction the wind is coming from and the intensity at which it blows. Drafts caused by thermal and mechanical effects will blow the polluted air in that direction. All of these factors work together, and it is this motion that can be either stifled or accentuated by the stability of the air.

Atmospheric dispersion model

There is no complete theory that describes the relationship between ambient concentrations of air pollutants and the causative meteorological factors and processes. The dispersion by the wind is a very complex process due to the presence of different sized eddies in atmospheric flow.

Atmospheric dispersion modelling is the mathematical simulation of how air pollutants disperse in the ambient atmosphere. It is performed with computer programs that solve the mathematical equations and algorithms which simulate the pollutant dispersion. The dispersion models are used to estimate or to predict the downwind concentration of air pollutants emitted from sources such as industrial plants.

The dispersion models require the input of data which includes:

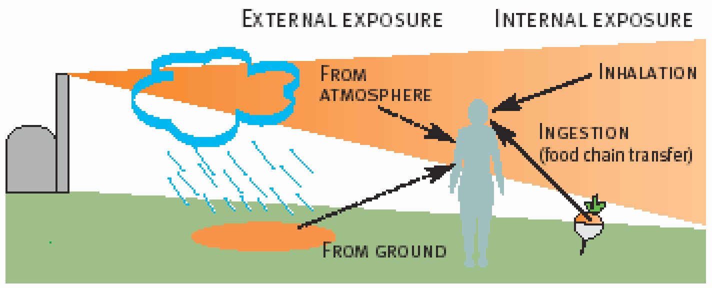
- Meteorological conditions such as wind speed and direction, the amount of atmospheric turbulence (as characterised by what is called the "stability class"), the ambient air temperature and the height to the bottom of any inversion aloft that may be present.
- Emissions parameters such as source location and height, source vent stack diameter and exit velocity, exit temperature and mass flow rate.
- Terrain elevations at the source location and at the receptor location.
- The location, height and width of any obstructions (such as buildings or other structures) in the path of the emitted gaseous plume.

The process of air pollution modelling contains 4 stages (data input, dispersion calculations, deriving concentrations and analysis). The accuracy and uncertainty of each stage must be known and evaluated to ensure a reliable assessment of significance of any potential adverse effects.

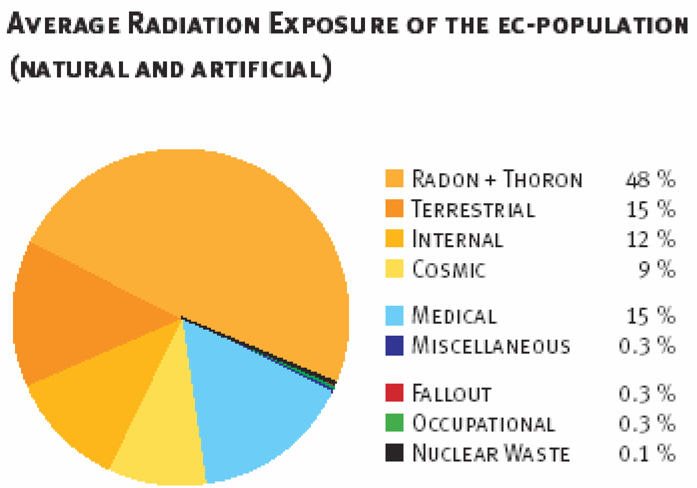
Currently, the most commonly used dispersion models are steady-state Gaussian plume models. These are based on mathematical approximation of plume behaviour and are the easiest models to use. They incorporate a simplistic description of the dispersion process and some fundamental assumptions are made that may not accurately reflect reality. However, even with these limitations, this type of model can provide reasonable results when used appropriately.

5.6 RADIOECOLOGY

Radioecology is a scientific discipline which studies how radioactive substances interact with nature; how different mechanisms affect the substances' migration and uptake in food chain and ecosystems. Investigations in radioecology might include aspects of field sampling, designed field and laboratory experiments and the development of predictive simulation models. This multi- and interdisciplinary science combines techniques from some of the more basic, traditional fields, such as physics, chemistry, mathematics, biology, and ecology, with applied concepts in radiation protection. Radioecological studies form the basis for estimating doses and assessing the consequences of radioactive pollution for human health and the environment.



The interest in radioecology increased after the Chernobyl accident in 1986 when large parts of Europe were contaminated with radioactive fallout. Areas, which received most deposition, were areas with heavy rainfall the period when the radioactive plume passed by. The measurement techniques (link) used to detect the radioactive substances are highly sensitive and today, almost 20 years later, it is still possible to detect the fallout. Other sources to artificial radioactivity in the environment are global fallout from nuclear weapons testing in the atmosphere in the 1950's and 60's, routine discharges from nuclear installations like the reprocessing facilities and possible leakage of radioactivity from sunken nuclear-powered submarines. It is necessary to underline that the average exposure to artificial radiation (excluding medical practice) is less than one percent of the total amount.



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Radionuclide tracers enabled a whole new field of research on the critical pathways of movement of pollutants in the environment and their potential for food chain discrimination or bioaccumulation in successively higher trophic levels. Sophisticated mathematical equations were developed which permitted calculation of the time dynamic (transient behaviour) of wholebody concentrations and equilibrium whole-body burdens from both acute and chronic ingestion.

Food chains inherently neither concentrate nor dilute pollutants, but this phenomenon continues to be misunderstood in the public's perception of the behaviour of hazardous materials in the environment. Food chain models have had important application in developing regulatory standards for environmental exposures (ingestion) and in developing risk analysis for hazardous releases.

5.7 CODE RDEMO©

For the estimation of the radiological consequences from discharged radioactive substances (to the atmosphere by ventilation stacks and to the hydrosphere – surface water, i.e. to river Hron) during normal and abnormal operation, the computing programme system RDEMO© was used.

The RDEMO© is one of group of four codes. RDETE, RDEDU, RDEBO, RDEMO. The first two are in use in Czech Republic (Temelin and Dukovany respectively).

The validation was performed on the basis of comparative analyses for reference tasks developed by Expert Commission No. 6 of SÚJB ČR in Prague for computing dispersion of radioactive materials. The comparative analyses are obligatory for programmes used in the ČR for this area. Conclusions are valid for all computing programmes RDxxx, because all the systems (RDEBO, RDEMO, RDEDU and EDETE) come out from uniform methodology and computing modules use the same algorithms and programme tools.

Moreover, the code RDEMO© has been validated also by comparison with the code NRCDose on November 2007. Both codes were used with the same input data for Mochovce NPP and surrounding area. Results of the comparison were in close agreement. Validation was performed by an independent organization and the validation protocol is given in Annex V.

Health Safety Slovak Authority gave permission to SE a.s. for using the code RDEMO© in its permission No. OOZPŽ/6274/2006 from 2nd November 2006.

Program set RDEMO© includes programs for preparation of input data files, calculation programs and programs for graphic and printed outputs with individual programs following from each other (outputs from one program form inputs for the following program).

Program enables calculation of annual individual effective and equivalent doses or 50(70)-year doses of collective effective and equivalent doses for six age categories (0 – 1, 1 – 2, 2 – 7, 7 – 12, 12 – 17, more than 17 years) for six body organs (gonads, bone marrow, lungs, thyroid gland, alimentary tract and skin) and for the whole body, for ten radiation routes:

- external exposure from the atmosphere from the plume and deposit,
- external exposure from the hydrosphere from bathing, sailing and from staying on sediments and on irrigated soils,
- internal exposure from inhalation,
- internal exposure from ingestion of food contaminated by atmospheric fallout (food chains: meat (beef, pork and poultry), milk, cereals, vegetables (green-stuff, root crop and potato), fruit and other crops (eggs, sugar, beer, oil crops),
- internal exposure from the hydrosphere - ingestion of drinking water, fish and food contaminated by irrigation.

Program also counts 50(70)-year bonds of collective effective doses for all zones – regional doses.

Program determines the critical population group (critical zone), critical radiation route and critical radio nuclides for individual radiation routes and total for atmosphere and hydrosphere including contributions by individual radio nuclides.

The area with 60km radius from Mochovce NPP is divided into 192 zones (0 – 1, 1 – 2, 2 – 3, 3 – 5, 5 – 7, 7 – 10, 10 – 15, 15 – 20, 20 – 30, 30 – 40, 40 – 50, 50 – 60km; direction N, NNE, NE, ENE, E, ESE, SE, SSE, S, SSW, SW, WSW, W, WNW, NW, NNW).

Programme RDEMO© is notably designed for evaluation of normal operation of NPP impact on the environment, but its use is also suitable for accident assessment of releases to the hydrosphere and assessment of radiological consequences in the intermediate and late phase of the accident.

NRCDose is a nuclear industry standard for calculation of inhabitant doses from routine radioactive releases from NPP operation.

NRCDose is a Microsoft Windows™ PC-based software which provides an interface for the industry standard LADTAP II, GASPAR II, and XOQDOQ programs. It is essentially a Windows™ version of NRC's 10CFR50, Appendix I Implementation codes. These codes implement NRC's current requirements for ALARA for radioactive effluents from nuclear power plants.

LADTAP II, GASPAR II, and XOQDOQ were originally written for mainframe computers, using the FORTRAN programming language. While still utilizing the proven FORTRAN computational modules, NRCDose allows the user to enter and retrieve data through a series of windows dialogs, making the use of the program much more user-friendly and efficient than its original design. This graphical interface also allows the user to create sets of data that can be named and retrieved at a later time for review or modification.

- LADTAP - Liquid Pathway Dose Modeling:
  - Regulatory Guide 1.109;
  - Fish and Invertebrate Ingestion;
  - Drinking Water;
  - Irrigated Crops;
  - Shoreline and Boating;
  - Recreational and Population Doses;
  - ALARA Cost-Benefit Evaluation;
- GASPAR - Gaseous Pathway Dose Modeling:Regulatory Guide 1.109;
  - Noble Gas Direct Exposure;
  - Inhalation Pathway;
  - Infant Milk Ingestion;
  - Deposition and Food Ingestion;
  - Individual and Population Doses;

- ALARA Cost-Benefit Evaluation;

- XOQDOQ - Atmospheric Dispersion Modeling:
  - Regulatory Guide 1.111;
  - *X/Q*, Annual Average Dispersion;
  - *D/Q*, Particulate and Radioiodine Deposition;
  - Intermittant releases - containment purge, decay tank releases;
  - Output formatted for direct input to GASPAR;

5.8 SAFETY ASSESSMENT RELATED TO ACCIDENTAL CONDITIONS

The safety evaluation of Mochovce NPP was performed on the basis of a structured approach, which is fully in line with both the IAEA fundamental principles and Western requirements and practices.

IAEA (Safety of Nuclear Power Plants: Design, Safety Standards Series No. NS-R-1, Vienna, 2000) states that "*A safety analysis of the plant design shall be conducted in which methods of both deterministic and probabilistic analysis shall be applied. On the basis of this analysis, the design basis for items important to safety shall be established and confirmed. It shall also be demonstrated that the plant as designed is capable of meeting any prescribed limits for radioactive releases and acceptable limits for potential radiation doses for each category of plant states, and that defence in depth has been effected.*"

The safety philosophy which is considered in the design and in the safety evaluation of the plant is aimed primarily at the prevention of accidents but also gives attention to the mitigation of the consequences of accidents that could give rise to major releases. The aim is to reduce both the probability of the events and their associated off-site consequences in order to avoid the need of extensive countermeasures and to offer the authorities the possibility of simplifying the offsite emergency planning.

For this purpose, the "defence-in-depth" concept is generally referred to, thus representing the basic framework for most of nuclear installation safety. To compensate for potential human and mechanical failures, the defence-in-depth concept is centred on several levels of protection, foreseeing successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the installation and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective. Defence in depth helps establish that the three basic safety functions (controlling the power, cooling the fuel and confining the radioactive material) are preserved, and that radioactive materials do not reach people or the environment.

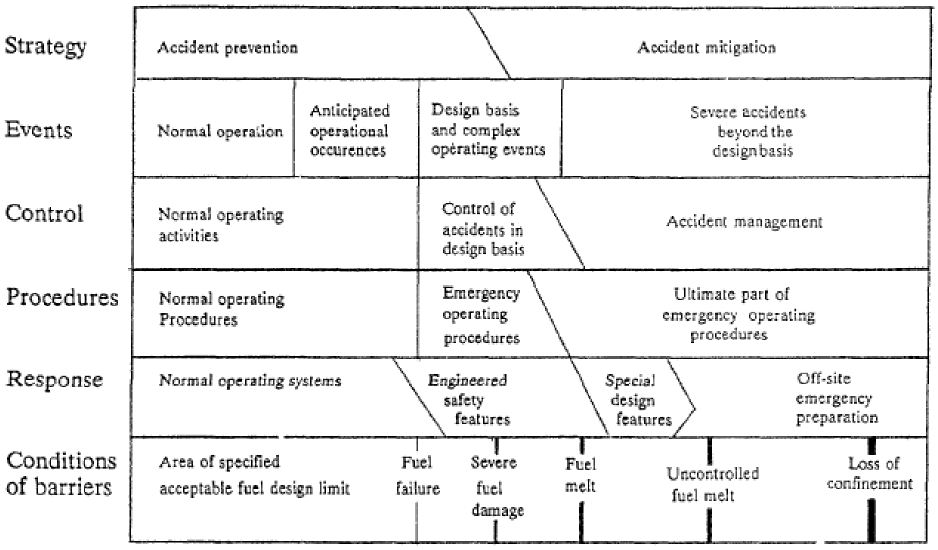

FIG. 1. Overwiew of defence in depth.

The assessment of the level of nuclear safety reached in a nuclear plant, and of the extent to which the defence-in-depth concept is implemented in the plant design, can be carried out both by deterministic and probabilistic analyses. The two approaches, nowadays generally combined, are briefly illustrated in the following.

Deterministic approach

In order to provide a robust demonstration of the fault tolerance of the plant and the effectiveness of its safety systems, in line with international practice, a deterministic analysis of the capabilities of the plant to cope with a representative, predetermined set of fault conditions is required. Suitable assessment tools are required (e.g., tests, calculations by validated computer codes or engineering analysis), and the overall approach is conservative, i.e., foresees the inclusion of suitable safety margins to take into account possible unfavourable combinations of failures which worsen the scenarios initially considered. The abnormal/accidental scenarios considered are classified in terms of their estimated frequency of occurrence, and acceptance criteria are ultimately expressed in terms of corresponding fission product release limits. Clearly, the design of the plant has to be such that the higher the frequency associated with an accident, the lower its radiological consequences.

However, the safety analysis generally incorporates both deterministic and probabilistic approaches. These approaches have been shown to complement each other and both are currently used in the decision making process on the safety and ability of the plant to be licensed.

Probabilistic approach

The probabilistic approach (typically referred to as Probabilistic Safety Assessment, or PSA) differs from deterministic safety analysis in that it provides a methodological approach to identifying accident sequences that can follow from a broad range of initiating events, and it includes the systematic and realistic determination of accident frequencies and consequences. A major advantage of PSA is that it allows for the quantification of uncertainties in safety assessments together with the quantification of expert opinion and judgement. Finally, PSA has been shown to provide important safety insights in addition to those provided by deterministic analysis. It is generally recognised that the primary objective of PSA studies is to evaluate the robustness of the actual safety status of a plant, and to identify priorities in safety upgrading measures.

In international practice, three levels of PSA have evolved:

- **Level 1:** The assessment of plant failures leading to the determination of core damage frequency.
- **Level 2:** The assessment of containment response leading, together with Level 1 results, to the determination of containment large early release frequencies.
- **Level 3:** The assessment of off-site consequences leading, together with the results of Level 2 analysis, to estimates of public risks.

The PSA determines all significant contributors to risk from the plant and evaluates the extent to which the design of the overall system configuration is well balanced, there are no risk outliers and the design meets basic probabilistic targets. For instance, considering the Level 1 PSA, the probabilistic target refers to the frequency of occurrence of core damage, as indicator of the effectiveness of the safety measures defined in the plant design for core damage prevention. In general, for a plant it is requested that all the combinations of equipment failures, human errors, human-induced events and natural events which, according to deterministic analyses, lead to core damage, have a sufficiently-low frequency of occurrence. It is recommended by IAEA that the sum of the frequencies of occurrence of all the sequences leading to core damage (in all plant states) be less than once in 10^4 years for operating plants (like EMO12) and once in 10^5 years for new plants (as MO34).

It is important to remark that the deterministic and the probabilistic approaches have both been considered in the design of EMO12: on the basis of a large amount of deterministic analyses, a Level 1 PSA has been prepared, with results in full compliance with the IAEA recommendations mentioned above. The same approach has been followed for MO34, for which a preliminary version of the Safety Analysis Report (PRESAR) and of the Level 1 PSA have been developed.

5.9 CODE RTARC©

The computing programme RTARC© (Real Time Accident Release Consequence) is used for analyses of radiological consequences of accident releases to the atmosphere. The code is designed notably for the estimation of the radiological situation during the early phase of an accident, i.e. for the period from the time when the potential for off-site exposure of the public is recognized to the time when significant amounts of radioactive material are released, but for later times too.

The code allows for inclusion in calculations on-line measurements from teledosimetric system (TDS) and measurements of mobile groups on source term reconstruction, but it can also be used in the event of a non-functional TDS. The data from TDS is then replaced with manually entered data on the meteorological situation. It comprises a package of programs and input databases.

RTARC© principal tasks include:

- forecast of concentrations, dose rates, and effective and equivalent doses;
- update and representation of the radiation situation course in graphic or tabular form;
- identification and representation or hazard zones calling for taking actions;
- identification and representation of hazard zones on taking protective actions;
- calculation and representation of trajectories or radioactive plume trace in changing meteorological conditions.

The programme is a standard component of tools for management and assessment of radiological accidents in the operational NPPs in SR and Czech Republic. The code calculations include atmospheric transport and diffusion, dose assessment, evaluation and display of the affected zones, evaluation of specific activity and deposition as well as dose rate in the air in selected area. The code RTARC© include prognosis of concentrations, dose rate, effective doses and equivalent doses for thyroid and/or for bone marrow for two age groups: adults and children to one year.

The databases and input data needed for calculations are: data charactering radionuclides (dose factors, half-time constants...), source terms characterizing release of radioactive materials (RM) to the atmosphere for selected accident, meteorological data and data about the countermeasures. The independent simulation of the urgent protective measures is involved – sheltering and iodine prophylaxis.

For internal exposure committed dose conversion factors for adults and infants are considered. Calculations of dispersion of RM in the atmosphere in the RTARC© programme are based to the Gaussian plume diffusion model:

$$X(x, y, z) = Q_R \, (2\pi \, \sigma_y \, \sigma_z \, u_i)^{-1} \exp(-y^2 / 2\, \sigma_y^2) \, S(h_i, x, z)$$

$$\text{kde} \quad S(h_i, x, z) = \exp[-(z + h_i)^2 / 2\, \sigma_z^2] + \exp[-(z - h_i)^2 / 2\, \sigma_z^2]$$

The physical processes which are to be considered in the prediction of doses to local groups from effluents released to the atmosphere are:

- dispersion by turbulent diffusion and the mean wind speed; vertical and horizontal standard deviations σy(x) a σz(x) correspond with the parameterization of Hosker;
- dry deposition onto the ground due to effects at the air ground interface;
- wet deposition due to washout and rainout as rain interacts with the plume;
- radioactive decay as the effluent decays;
- building wake effects due to the flow in the lee of large structures;
- for assessment of trajectory the ascending cloud is used model of Briggs;
- only heat elevation is assumed.

Dose calculation is solving by discrete of time variables q(t) by means of normalized function $f_Q(\Delta t)$, what meet the condition:

$$\sum_{l=1}^{n} f_Q(\Delta t_i) = 1$$

where n is number of time intervals. Subsequently source characterized by function q(t) is possible to considered that a source is composed from set of successive followings releases with constant rate Q_{oi} and time of duration Δt_i, where

$$Q_{0i} = \frac{f_Q(\Delta t_i)\, Q}{\Delta t_i}$$

and dose in the point (x,y) at the time t, calculated from the start of release is a sum

$$D(x,y,t) = \sum_{i=1}^{n} D(x,y,t_i)$$

where

$$t_i = t - \sum_{j=1}^{i-1} \Delta t_j$$

The computational analyses by the RTARC© programme are performed for 6 stability categories (categorization by Pasquill-Uhlig method to 6 categories A-F) of the atmosphere with typical wind speeds:

Atmosphere stability category	A	B	C	D	E	F
Typical wind speed [m s^{-1}]	1	2	5	5	3	2
Occurrence probability [%] (in 1997-2004)	2.5	12.9	26.2	40.1	8.4	9.9

The calculations of individual doses (effective and equivalent for thyroid – criterion parameters) by RTARC© programme are performed up to the distance of 40 km, exposure times: 2 hours, 1 day, 2 days, 7 and 15 days and 1 year for adults (the most numerous age group).

In the programme RTARC© those ways of exposure are considered which are the most important in the early phase of the accident, namely:

- external exposure by the passing radioactivity plume and by radioactivity deposited on the ground;
- internal exposure by inhalation which includes inhalation of radionuclides from the passing cloud and inhalation of radionuclides re-suspended from the ground.

The conservative approach at the dose calculations is done by assumption, that:

- height of release is 10, 25, respectively 43 m (not by stack, i.e. in height 125 m);
- sensible heat to 1E+7 cal/s;
- for design basis accidents the man is staying or moving 24 hours of the day in the axis of passing radioactive cloud (the sheltering is not assumed);
- for severe accidents normal living is assumed (by means of shielding factors: for cloud 0.14, for deposit 0.16 and reduction for inhalation 0.5);
- the stability class (the worst dispersion conditions) is not changed and the weather is stable in the whole year;
- comparison of the most greatest calculated doses (for the worst stability class of the atmosphere) with the criterion values;
- detail results are shown for the distance 2 km, this is the smallest distance of exclusion area border where is not permanent resettlement.

5.10 MACROSEISMIC SCALE AND MAGNITUDE SCALE

The Medvedev-Sponheuer-Karnik scale, also known as the MSK or MSK-64, is a macroseismic intensity scale used to evaluate the severity of ground shaking on the basis of observed effects in an area of the earthquake occurrence.

The scale was first proposed by Sergei Medvedev (USSR), Wilhelm Sponheuer (East Germany), and Vit Karnik (Czechoslovakia) in 1964. It was based on the experiences being available in the early 1960s from the application of the Modified Mercalli scale and the 1953 version of the Medvedev scale, known also as the GEOFIAN scale.

With minor modifications in the mid-1970s and early 1980s, the MSK scale became widely used in Europe and the USSR. In early 1990s, the European Seismological Commission (ESC) used many of the principles formulated in the MSK in the development of the European Macroseismic Scale, which is now a de-facto standard for evaluation of seismic intensity in European countries. In 1996 the XXV General Assembly of the ESC in Reykjavik passed a resolution recommending the adoption of the new scale by the member countries of the European Seismological Commission.

Unlike the earthquake magnitude scales, which express the seismic energy released by an earthquake, EMS 98 (*European Macroseismic Scale 1998*, European Seismological Commission, Luxembourg 1998) intensity denotes how strongly an earthquake affects a specific place.

MSK-64 is still being used in India, Israel, Russia, and throughout the Commonwealth of Independent States.

"*Modern macroseismic intensity scales have been developed and formally defined at the end of the XIX century as an empirical tool for measuring the strength of an earthquake, and deriving information on several physical characteristic of a seismic events, such as source parameters, attenuation, and site effects. Most important intensity scales used worldwide, such as the MCS, MM and MSK scales, are 12 degrees scales. Intensity scales are based on the effects of the earthquake. The effects on humans are the most important indicators of intensity up to the V degree. The assessment of intensity in the range between the VI and XII degree is based mostly on effects on man-made structures (damage) and on the environment (ground effects or environmental earthquake effects, EEE). This is true for all the early intensity scales "[21].*

The local magnitude ML scale (Richter magnitude scale) assigns a single number to quantify the amount of seismic energy released by an earthquake. It is a base-10 logarithmic scale obtained by calculating the logarithm of the combined horizontal amplitude of the largest displacement from zero on a seismometer output. Measurements have no limits and can be either positive or negative.

Earthquakes with magnitude of about 2.0 or less are usually called micro-earthquakes; they are not commonly felt by people and are generally recorded only on local seismographs. Events with magnitudes of about 4.5 or greater - there are several thousand such shocks annually - are strong enough to be recorded by sensitive seismographs all over the world. Great earthquakes, such as the 1964 Good Friday earthquake in Alaska, have magnitudes of 8.0 or higher. On the average, one earthquake of such size occurs somewhere in the world each year. The Richter Scale has no upper limit. Recently, another scale called the moment magnitude scale has been devised for more precise study of great earthquakes.

The Richter Scale is not used to express damage. An earthquake in a densely populated area which results in many deaths and considerable damage may have the same magnitude as a shock in a remote area that does nothing more than frighten wildlife. Large-magnitude earthquakes that occur beneath the oceans may not even be felt by humans.

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