RADIATION MONITORING

With the nuclear age came a need to monitor radiation levels for the protection of workers and the population, and monitoring instruments to fill this need were first developed for projects in government and university laboratories.

Early radiation monitors built in the 1950s were based on vacuum-tube electronics, which were then modified to use industrialized components that would withstand the environment of nuclear power plants. The cumbersome vacuum-tube instruments generated a lot of heat and required large, forced-air-cooled cabinets for housing readouts in the reactor control room.

During the late 1950s and early 1960s, vacuum tube–based electronics were replaced with transistor-based, solid-state electronics that reduced the required space for the control-room instruments. Each radiation detector had its own channel of electronics, including signal-conditioning circuitry, control-room readout, and associated power supplies. For a typical channel, the detector with a small amount of signal-conditioning electronics was located at the point of detection and the balance of the channel was mounted in or near the reactor control room. These analog-based, radiation-monitoring systems required a separate indicating system for each detection channel in the control room, and they had separate signal, high-voltage, low-voltage, and control-logic cables for each channel between the control room and the detector. This resulted in miles of multiconductor and shielded cabling in a typical plant to provide the required radiation indication at the control room.

In the late 1970s and early 1980s distributed microprocessors were developed, replacing the early analog systems with digital logic. These new microprocessor-based monitoring systems replaced the miles of multiconductor cables with single twisted-pair cables in a loop or star interconnecting configuration and replaced analog indicating instruments with a computer display.

Radiation is always around us and we are constantly being bombarded by radiation of subatomic particles and electromagnetic rays. Sources of the radiation from above us include our sun and solar system as well as the rest of the vast universe of space; and sources of radiation surrounding us include the soil and rocks in the earth and plants, animals, and people that are near us, as well as materials derived from those. Radiation includes both *ionizing* radiation in the form of X rays, gamma rays, alpha particles, beta particles, neutrons, protons, cosmic rays, etc., and *nonionizing* radiation in the form of the lower-energy portion of the electromagnetic spectrum, including electrical power in our homes, radio and video waves broadcast via cable or transmitted by air, visible light, and infrared energy emitted by bodies according to their temperature.

Radiation monitoring is concerned with measuring and monitoring ionizing radiation. Little is known about the harmful effects of ionizing radiation at levels that are typically encountered in our environment (natural background levels). Effects of high doses of ionizing radiation have been documented as a result of such incidents as the atomic bombs dropped on Japan, the atomic accident at Chernobyl, and observation of the effects of therapeutic uses of radiation. It is generally assumed that all radiation is harmful and that people should recieve the minimum radiation exposure for what needs to be accomplished. Naturally occurring radiation at historically typical background levels serves as the reference from which allowable additional radiation-exposure limits are set. Medical diagnostic radiation is the primary source of increased radiation over background for most people.

Radiation dose levels that cause death immediately or within a few weeks are well established. Radiation induced malignant tumors have been noted since the earliest use of X rays and other forms of ionizing radiation. Hypothetical increased risks forecasts for low-level radiation doses are based on linear or quadratic extrapolation of limited data where relatively large populations received very high doses. However, studies have not yet validated those models, and risk from low-level radiation may be lower than is generally postulated.

Background radiation levels are different at different places on the earth due to elevation, the makeup of the soil, and related factors. For example, background radiation in Denver, Colorado is about twice that of San Diego, primarily due to the higher elevation with less air mass to absorb radiation from space, such as cosmic rays. People are typically exposed to higher background radiation in masonry buildings than in wood structures because of naturally occurring radioactivity in the materials in cement, brick, or rock. The human body contains radioactivity in the form of naturally occurring potassium-40, carbon-14, and other radioactive isotopes. According to current government regulations, if a laboratory animal were injected with the amount of radioactivity in an average person's body, it would be considered radioactive (see *Reading List*).

Commonly encountered radioactive objects include: glow-in-the-dark radium watch dials made during the first half of the 20th century, thorium oxide coated gas-lamp mantles, smoke detectors containing about 1 microcurie of americium-241, fluorescent lamp starters containing a minute amount of krypton-85, porcelain tooth caps colored with metal oxides that contain uranium to give an improved reflective appearance, radon gas in tightly built homes over radium-containing soil or rock and in water supplies, and potassium-containing fertilizers. Though these may give off ionizing radiation at levels that would require careful accountability in a laboratory or industrial environment under state or federal regulations, they are not usually considered to be hazardous and are not required to be monitored.

In today's society there are areas where the use of ionizing radiation that must be monitored according to state or federal government regulations to ensure that radiation does not pose a hazard to personnel or to the environment: areas include radiation for instructional use in schools and universities, medicine, industrial gauging, sterilization of medical supplies or food, mining and milling of radioactive ores, steel mills, space flights, fusion facilities, and nuclear reactors.

The use of radiation sources at schools and universities may range from simple isotopic sources for demonstration purposes that are small enough to be exempt from regulations to fully operational nuclear test reactors where reactor physics is taught and isotopic sources are produced.

The use of ionizing radiation in medicine includes electron beams, X-ray sources, and gamma-ray sources for transmission imaging or for therapy and gamma-ray sources tagged to pharmaceuticals and injected into the patient for determining organ function in nuclear medicine. The types of sources include isotopic sources, X-ray tubes, and linear accelerators. X-ray tubes and linear accelerators generate ionizing radiation only when power is properly applied; however, isotopic sources always emit radiation as a result of the natural decay of the radioactive isotope. The physicians or medical technologists are responsible for properly administering radiation to patients; they use radiation monitoring devices to ensure that source strengths and photo energies are proper. Those who handle the isotopic sources should not be exposed inadvertently and receive doses above allowable limits.

Many industries use radiation sources for making routine measurements, such as material density, fill height of beverage containers, or material thickness. Radiation is also used to sterilize some medical supplies after the packaging is sealed. A typical radiation source is cobalt-60 with a half life of just over 5.3 years and a photon energy of just over a million electron volts; a typical sterilization dose is a million rads. Some foods are also exposed to gamma radiation to kill bacteria and prevent spoilage. Some seeds or bulbs are irradiated with lower doses to enhance growth and increase production. Some gemstones are irradiated to enhance color and brilliance. These industries and radiation facilities use radiation monitoring to calibrate and control exposure doses and to monitor personnel exposure.

Increased incidence of lung cancer among underground miners exposed to radon and radon daughters in their occupations has been demonstrated in epidemiological studies of the inhalation of radon gas and its effects on the lung epithelium. Efficient control of radon and radon daughters in underground mines has been difficult. It is the role of radiation monitoring to identify radiation exposure and assist in its control in underground mines (both uranium mines and non-uranium mines) where radon gases may be present.

The need for radiation monitoring in steel mills is fairly recent and is the result of 49 known incidents since 1973 where companies have inadvertently melted shielded radioactive sources, typically cobalt-60 or cesium-137. These incidents have not caused worker injuries but have resulted in economic harm to the companies with costs typically ranging from \$5 million to \$25 million per accidental melt of a radioactive source. These costs include loss of the melt, facility decontamination, and shut-down of steel production. The sources, generally, had been lost from licensees that had used them for industrial applications. It is also important to identify radioactively contaminated scrap, such as metal activated in a nuclear facility or contaminated in a melt from a lost source. To find a shielded source in a car load or truck load of scrap poses is very difficult.

Radiation monitors and detectors are placed on spacecraft for several purposes. Measuring secondary emissions, induced by the absorption of protons or neutrons, from the surface of planets or moons can identify the elements on the surface of the planets or moons. Monitoring the levels of radiation impinging on the spacecraft can provide information for predicting effects on the materials in the spacecraft and can be used to turn off sensitive electronic components during times of unexpectedly high-radiation exposure where radiation damage occurs when the components are powered.

Atomic fusion is another source of radiation; fusion combines light elements to create elements of greater atomic weight, neutrons, and excess energy, neutrons are eventually absorbed by the surrounding materials, typically resulting in radioactive isotopes. Radiation monitoring is required to measure levels of radiation during facility operation as well as radiation levels from neutron activation.

In the fission process in nuclear test reactors and nuclear power reactors, heavy atoms, such as uranium-235, absorb a neutron and split into fission products that include gamma rays, neutrons, alpha particles, beta particles, and lighter elements that are typically radioactive. Radioactive materials result from the fission process, as well as from activation from absorption of neutrons or other atomic particles. Radiation monitoring in nuclear power plants is discussed in detail later in this article.

Any of the facilities discussed prior that has one or more licensable radiation sources must have instruments that can measure the radiation from the source(s) and must provide personnel monitoring devices, such as film badges, ring badges, or thermoluminescent dosimeters (TLDs), to monitor the dose that personnel receive who work where they may be exposed to radiation from the source(s) that exceeds 10% of federally established limits. A personnel dosimeter badge is worn only by the individual to whom it is issued and, when not being worn, it is stored where it will not be exposed to radiation. This badge provides a record of radiation exposure for evaluating potential adverse effects and for ensuring that no worker exceeds established limits, such as the annual total body effective dose equivalent limit of 5 rem. Federally established limits in the United States of America can be found in the US Code of Federal Regulations, 10 CFR 20.

RADIATION MONITORING IN NUCLEAR POWER PLANTS

Radiation-monitoring (RM) systems are installed in nuclear power plants to satisfy U.S. Nuclear Regulatory Commission (USNRC) regulations and plant operating license requirements. The objective of those requirements is to protect both personnel and the environment from the effects of ionizing radiation. An installed system measures, displays, and records the presence and level of radiation and alerts plant personnel to excessive levels of radioactivity, and control actions are initiated automatically for required functions when levels exceed their limits.

Monitoring radiation in nuclear power plants is often divided into categories according to application. Typical categories include area monitoring (1) for determining radiation levels in areas where personnel may be working or may have a need to enter; process monitoring (2,3) for determining radiation levels in processes in the plant; effluent monitoring (4) for determining amounts of radiation leaving the plant through any pathway; and perimeter monitoring for identifying any increase in radiation level at the perimeter of a plant. Both process monitors and effluent monitors can be further separated in two categories according to whether they monitor a gaseous stream or a liquid stream. The instruments can be grouped according to the design as area γ-ray monitors, liquid monitors, and atmospheric gaseous, particulate, or iodine monitors.

A typical radiation-monitoring instrument has many functions to perform: it may detect, display, and record levels of radiation in the plant and provide alarms when selected radiation levels are exceeded; it may monitor process flow lines for detecting radioactive leakage; it may monitor effluent for recording radioactivity levels and inhibiting excessive releases from the plant; it may provide signals for control functions in other systems; it may provide samples for analysis for complying with USNRC Regulatory Guide 1.21; and it may provide postaccident monitoring in accordance with the requirements of NUREG 0737.

Electrical equipment in nuclear power plants is separated into safety categories according to the functions performed in order to establish quality requirements for procurement, installation, operation, and maintenance. The typical catgegories are *safety related* and *nonsafety related*. The safety-related category implies that the equipment is essential to ensure the integrity of the reactor-coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potentially major offsite exposures to the public. Safety-related equipment has the highest quality requirements and must not cease to perform its functions when any single credible failure occurs with the equipment. Other categories are introduced for postaccident monitoring equipment that must operate following a design basis accident. (The *design basis accident* for a nuclear power plant is the worst credible accident postulated for the plant for the purpose of evaluating risks and potential hazards associated with siting the plant.)

Most radiation-monitoring instrumentation is typically classed as nonsafety related, with some specific instruments identified either as safety-related or as postaccident monitors (5) that may have quality requirements similar to safety-related equipment. Requirements for safety-related equipment typically include demonstrated performance under normal and extreme service conditions and installation of redundant channels that are powered from redundant, safety-related power sources. Each channel may also be required to have demonstrated capability of performing its function under design basis service conditions following a design basis earthquake.

The USNRC has established the Standard Review Plan (SRP) (6) as a guide for reviewing designs of nuclear power plants against requirements, including a review of radiation-monitoring systems. An owner of a nuclear power plant provides a safety analysis review to respond to all points of the SRP. The installed RM system must meet the commitments made in the final safety analysis review and the requirements of documents referenced therein.

SYSTEM OVERVIEW

A nuclear power plant must have radiation monitors installed at strategic locations throughout the plant for monitoring radiation levels in order to meet regulatory requirements. In addition, laboratory instruments are used for analyzing collected samples of liquids or gases, and portable or handheld instruments are used for making surveys. Chemistry or health physics group members typically observe and record continuous radiation-monitor channels, but they typically rely on collecting and analyzing samples of effluents for final determination of the offsite dose, and likewise they rely on portable instruments for confirming radiation levels where people are working. Effluent samples are analyzed in laboratory instruments that measure ionizing radiation with excellent energy resolution for identifying the radioactive isotopes contained in the samples. These analyzers typically employ cryogenically cooled germanium detectors.

The descriptions of radiation-monitoring instruments contained herein relate to instruments installed in nuclear power plants for continuous monitoring and do not include laboratory or hand-held instruments. The installed systems typically monitor magnitudes of radiation and initiate alarms when radiation levels approach established set points; however, because of the need to operate continuously in the plant environment, the energy resolution of the detectors is much poorer than what is available with laboratory instruments.

Analog RM systems usually have a detector installed at each location where radiation levels are to be monitored with a small amount of signal-conditioning electronics, and the detectors are connected by long instrumentation cables to a cabinet at the reactor control room in which are installed signal-conditioning electronics, alarms, readout devices, and power supplies associated with each detector.

Digital RM systems are typically distributed computer systems that include a microcomputer and required power supplies located at or near each detector location, and a communication cable connects that location to a central computer at the control room and at a health physics office or other location where the information is to be used. The communication cable is usually a simple twisted-shielded pair cable.

Typical locations in a pressurized water reactor (PWR) nuclear power plant where area monitors may be installed include the control room, radiochemical laboratory, hot machine shop, sampling room, reactor-building personnel access, refueling bridge, in-core instrumentation area, fuel storage area, auxiliary-building-demineralizer area, waste-gas-decay-tank area, drumming area, waste-holdup-tank area, charging-pump area, turbine-building area, and main stream lines.

Typical monitoring points for gaseous process monitors include reactor-building-containment area for airborne gaseous and particulate monitoring, radioactive-waste-disposal-area vent, waste-gas header, fuel-handling-area vent, and control room for airborne-gaseous and -particulate monitoring. Monitoring points for liquid process monitors may include chemical- and volume-control-system-letdown line, radioactive-waste-condensate return, component cooling water, normal-sample-laboratory isolation, and main steam lines.

Gaseous effluent monitoring may be performed in the plant vent stack, in the condenser air ejector, and in the containment purge stack. Liquid-effluent monitors are typically installed in the radioactive-waste discharge line, in the neutralization sump discharge line, in the turbine plant area sump, and in the steam-generator blowdown. The locations listed above are typical for a PWR plant but may be different in each plant.

SOURCES OF RADIOACTIVE MATERIAL

The primary sources of radioactive material in light-water-cooled nuclear power plants are the fission process in the reactor core and neutron activation. The fission process, the splitting of uranium atoms, emits neutrons, γ rays, and β particles directly and creates radioactive elements in the fuel-pellet regions. Those radioactive materials then decay primarily by emission of γ and β radiation.

Neutron activation occurs whenever any atom absorbs one of the neutrons that is emitted from a splitting uranium atom. The absorbing atom gains atomic weight, thus becoming a new isotope of the same element, which may be unstable or radioactive and decay to a more stable state by emitting radiation. Neutron activation creates radioactive material in the fuel-pellet region; in the reactor-coolant region; in support structures, reactor vessel, and piping; and in the regions surrounding the reactor vessel including the air in that vicinity. In addition to neutron activation, some atoms may be activated by particle radiation emitted from fission products or from neutron-activated elements.

The plant is designed to keep radioactive material contained; however, if systems become unsealed, some radioactive material may leak into the coolant through the fuel cladding, then from the reactor coolant through the pressure boundary or from coolant purification and radioactive-waste processing systems into secondary systems. It is this leakage that radiation-monitoring systems are expected to detect during normal reactor operation. γ rays and β particles are the forms of radiation that are most readily detected.

The concentration and quantities of radioactive material in various regions of the plant depend on the balance among production, leakage, and removal of individual isotopes. In the fuel-pellet region, production processes include fission-product production directly from fissioning uranium atoms, parent-fission-product decay, and neutron activation. Removal processes include decay, neutron activation, and leakage through cladding defects into the coolant. In the coolant region, production processes include (1) leakage of fission and activation products from the fuel-pellet region and from fuel cladding and core structures, (2) parent decay in the coolant, and (3) neutron activation in the coolant materials. Removal is by decay, by coolant purification, by feed and bleed operations, and by leakage. The most abundant isotopes in the coolant are radioactive noble gases ^{85}Kr, ^{133}Xe, and ^{135}Xe during normal operation) and radioactive halogens (in particular ^{131}I).

Neutron activation leads to two isotopes of particular interest, ^{16}N and ^{14}C ^{16}N is produced by a neutron-proton reaction with ^{16}O and decays by higher-energy γ-ray decay with decay energies of 6.1 and 7.1 MeV. The half-life of ^{16}N is 7.3 s. While there is substantial decay of ^{16}N as it exits from the core and passes through the turbine, it must still be considered in the design of the turbine shielding for boiling water reactors (BWRs). The detection of ^{16}N in the secondary side of PWRs may be used to monitor changes in steam-generator leakage. ^{14}C is produced by neutron activation of ^{17}O and ^{14}N.

The principal generation of tritium (^{3}H) is from fission and neutron interaction with boron, lithium, and deuterium. The main leakage source is fission tritium released through fuel-cladding defects. Tritium produced in the coolant contributes directly to the tritium inventory, while tritium produced in control-element assemblies contributes only by leakage and corrosion.

Activation and corrosion of reactor core support structures produce corrosion products, forming a radioactive material commonly referred to as *radioactive crud*. Corrosion-product constituents are typically ^{60}Co, ^{58}Co, ^{54}Mn, ^{51}Cr, ^{59}Fe, and ^{95}Zr.

Leakage Sources

Any system containing radioactive materials in liquid form is a potential source of radioactive leakage, and radioactive leakage into the reactor-containment area comes from the reactor-coolant system and coolant-purification systems. Leakage from systems containing potentially radioactive liquids is collected and processed by liquid radioactive-waste systems. Noble gases that are dissolved in liquid leakage may go out of solution and into the local atmosphere.

Radioactive material can be released into effluents from secondary systems due to leakage. For PWRs, the amount of release depends on reactor-coolant radioactive material concentrations, reactor-coolant leakage rate, primary to secondary leakage rate, steam-generator blowdown rate, and secondary-system leakage rates. Abnormal leakage from the fuel region to the reactor coolant is commonly detected by monitoring the reactor-coolant-system (RCS) letdown stream, either continuously or on a sampling basis.

In PWRs, reactor coolant remains liquid under pressurization in a primary coolant loop and transfers its heat through a heat exchanger to a secondary coolant loop that is converted to steam in the steam generator. The secondary system is typically monitored on steam-generator-blowdown, component-cooling-water, and liquid-radioactive-waste-processing systems to check for leakage from primary to secondary coolant loops.

In BWRs, reactor coolant is converted directly to steam for use in the steam tubine. γ radiation levels external to the main steam lines are monitored to detect increased levels of radiation in the reactor coolant that may indicate problems such as significant fuel-cladding failure. A fuel-cladding failure would allow fission products, particularly noble gases, to be transported to the steam lines, which could cause the radiation level external to the steam lines to be well above normal background levels.

In both PWRs and BWRs, condenser exhaust is monitored.

Reactor-Coolant-System Leakage Detection

An increase in reactor-coolant-system leakage rate in a nuclear power plant of 1 gal/min must be identifiable within 1 h. USNRC Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Detection System*, outlines the means required for monitoring RCS leakage and indicates that this function must be provided also following a design basis earthquake. Three required means of monitoring leakage rate are (1) sump level and flow monitoring, (2) airborne-particulate radioactivity monitoring, and (3) either monitoring condensate flow rate from air coolers or monitoring airborne-gaseous radioactivity.

The sump flow rate and airborne-particulate channels are found to be capable of indicating an increase in RCS leakage of 1 gal/min within 1 h under most operating conditions. However, airborne-gaseous monitoring was found to have a much longer response time. This shortcoming was identified generically in a USNRC staff memorandum and is mainly due to the long half-life of ^{133}Xe, the major noble gas in the RCS, and the background radiation level from ^{41}Ar, which is created by neutron activation of air around the reactor vessel. Most RM systems use gross energy measurement methods for RCS leakage detection. Potential leakage-detection improvements using spectral capabilities of new γ-ray sensitive detectors for particulate and gaseous monitoring have been predicted, which might allow specific isotopes to be measured and separated from background radiation.

Liquid Effluent

Liquid-waste systems in a nuclear power plant collect and process radioactive liquid wastes generated during plant operation and reduce their radioactivity and chemical concentrations to levels of clean water acceptable for being discharged to the environment or recycled in the plant. Radioactivity removed from the liquids is concentrated in filters, ion-exchange resins, and evaporator bottoms, and these concentrated wastes are sent to a radioactive-waste solidification system for packaging and eventual shipment to an approved offsite disposal location. If the water is to be recycled to the reactor-coolant system, it must meet the water-purity requirements for reactor coolant. If the liquid is to be discharged, the activity level must be consistent with the discharge criteria of the U.S. Code of Federal Regulations, 10CFR20. These liquids normally pass through liquid radiation monitors prior to being recycled or discharged.

UNITS OF MEASURE IMPORTANT TO RADIATION MONITORING

Becquerel	The becquerel (Bq) was adopted in 1975 as unit of measure of activity, which is the measure of the rate of decay of a radioisotopic source. 1 Bq is one disintegration per second.
Curie	The curie (Ci) is a measure of the activity or number of disintegrations per second of a radioactive source. It was originally an estimate of the activity of 1 g of pure radium-226. 1 Ci is 3.7×10^{10} disintegrations per second (Bq).
Gray	The gray (Gy) is a measure of absorbed dose. 1 Gy is 1 joule per kilogram, or 100 rad.
Rad	The rad is a measure of absorbed dose in units of energy per unit mass of the absorbing material. 1 rad is 100 ergs per gram. The magnitude of dose will depend on material properties as well as on the radiation source. Air is typically used as the basis of measurement. When water is substituted for air, the absorbed dose is nearly the same because the atomic number of water is nearly the same as for air.
Rem	The rem is used to measure the effect of radiation on living organisms and was derived from the words *radiation equivalent in man* and is equal to the absorbed dose times a quality factor, Q. For γ rays and β particles, $Q = 1$ and 1 rem is equal to 1 rad. For charged particles, Q is much larger than 1.
Roentgen	The roentgen (R) is a measure of γ-ray exposure in terms of the charge due to ionization by the exposing radiation in a unit of mass of the material. 1 roentgen is 1 electrostatic unit of charge in 1 cubic centimeter of air at standard temperature and pressure.
Sievert	The sievert (Sv) is a measure of the effect of radiation on living organisms and is equal to 100 rem.

TYPICAL DETECTORS AND MONITOR TYPES

Area Radiation Monitors

Area radiation monitors continuously measure radiation levels at various locations within nuclear power plants including reactor-containment-building work areas and fuel-storage facilities for ensuring personnel safety. Area monitors have historically used Geiger-Muller (GM) tubes, ionization chambers, or scintillation crystals coupled to photomultiplier (PM) tubes, depending on the manufacturer and the sensitivity or range requirements. A block diagram of a typical GM-tube-based area monitor is shown in Fig. 1.

In a GM tube (7) an avalanche breakdown occurs in the gas in the tube each time ionizing radiation is detected and then self-extinguishes. The magnitude of the resulting signal (an electronic pulse) is independent of the number of original ion pairs that initiated the process and therefore independent of the energy of the detected ionizing radiation. The electronics in a GM-tube-based area monitor senses these pulses and converts them to a signal that is proportional to their rate of occurrence. However, present-day GM-tube monitoring systems use energy-compensated GM tubes for which the number of counts detected is nearly proportional to the total energy absorbed.

Thin-walled GM tubes used for area monitors are normally energy compensated for a linear response or $\pm 20\%$ or better for γ-ray energies of 60 keV to 1.25 MeV. The filter is designed to attenuate the lower-energy γ rays below approximately 100 keV and to increase the responses of higher-energy γ rays by the effect of the high-Z material used for the filter. This energy-compensation effect is due to the complex contribution of primary photon transmission or attenuation and secondary-particle production or attenuation at various depths in the GM-tube wall or outer energy filter (8).

In an ionization chamber (9) ionizing radiation is absorbed in the gas in the chamber, and the number of electron-ion pairs thus created is proportional to the energy of the absorbed radiation. The bias voltage on the ionization chamber sweeps the charge carriers to the electrodes, causing an electrical current to flow, and external circuitry typically measures the magnitude of the current from this ionization process. The current output signal from an ionization-chamber-based area monitor is proportional to the energy of the radiation absorbed in the gas in the chamber in the γ-ray flux field. These area radiation monitors are often calibrated in units of R/h.

When ionizing radiation is absorbed in a scintillator, a light pulse is produced. In turn the light pulse is converted into an electrical pulse in a PM tube that is optically coupled to the scintillator. Over a broad range of energies, the amplitude of the electrical pulse is proportional to the energy of the absorbed radiation. In a scintillator-based area monitor the output signal is proportional to the number of absorbed-radiation events in the detector and independent of energy if the electronics just counts events and provides a count-rate output signal. If the current from the PM tube is measured, the signal will be proportional to the energy of the absorbed radiation.

A typical pulse-counting area monitoring may have a range from 1 to 100,000 counts per minute. A typical ionization-chamber-based area monitor may have a range from 10^{-10} to 10^{-3} A, corresponding to a range from 1 to 10,000,000 rad/h. This high range may be used for applications such as postaccident monitoring inside containment.

Process Monitors

Process monitors provide information about radiation levels within the nuclear power plant's liquid, steam, and gaseous processing and storage systems. Liquid monitors may have the detectors mounted in the liquid or steam line or may have them mounted off-line with a sample stream taken from the main line and flowing through the detector sample chamber. Gaseous monitors too can have the detectors mounted in the stream or in a sample chamber. If the monitor is in a sample chamber, the gas to be monitored, which could be from a duct or from an open area such as the control room, passes through that sample chamber.

Process monitors typically measure radiation levels that are far below normal background levels, and, as a result, the detectors must be shielded from all external sources of radiation.

Effluent Monitors

Effluent monitors are similar to process monitors except that they monitor liquid or gaseous streams that leave the nuclear power plant boundary and may transport radioactivity.

The monitors described below are applicable to both process and effluent monitors according to the application to which they are dedicated.

Airborne Monitors

Off-Line Particulate and Noble-Gas β-Particle Scintillation Detectors. The β-particle sensitive particulate and noble-gas monitors incorporate plastic scintillators coupled to PM tubes. The gas-detector assemblies are of similar construction. These detectors use similar entrance windows, plastic scintillators, light pipes, and PM tubes. A typical particulate detector is built up from a plastic scintillator coupled to a PM tube and a 0.001-inch-thick aluminum entrance window used for β-particle detectors.

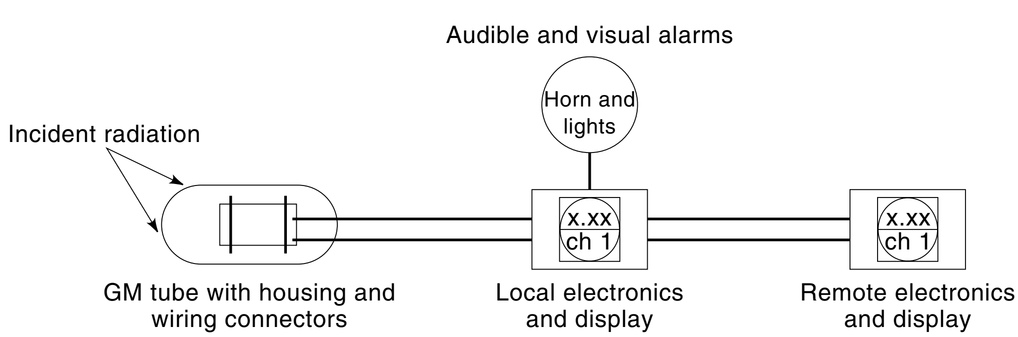

Figure 1. Typical block diagram of an area radiation monitor.

During original prototype evaluation, detectors are tested with solid β-particle sources to obtain a high signal-to-noise ratio when setting the discrimination level. Without changing the detector's alignment, responses are then obtained for calibrated solid or gaseous β-particle sources. The solid alignment sources are serialized and kept for future use. These sources are used for aligning production detectors and prototype detectors prior to isotopic calibration. After alignment of a production monitor to the same counting efficiency as the prototype detector that uses the same solid source, the production and prototype monitors have nearly identical responses to radioactive gases in the sample chamber or to activity on the filter.

After a monitor has been calibrated at a factory and shipped to a customer, corrections to the calibration may be required to account for atmospheric pressure affects because the response of the detector to β-particle radiation in the sample chamber is determined, not only by the total activity in the sample chamber, but also by self-absorption in the sample gas, which changes with gas pressure.

High-Temperature, In-Line, Noble-Gas, β-Particle Scintillation Detectors. These detectors are designed to operate at high temperatures and to be installed directly into an air duct. The detector assemblies are similar to the off-line β-particle scintillation detectors with the following exceptions:

A thin (0.007-inch-thick) CaF_2 (Eu) crystal or a high-temperature plastic scintillator (0.010 inch thick) and a quartz light pipe are used in place of the plastic scintillator with a Lucite light pipe, and a high-temperature PM tube is used. These detectors use the same 0.001-inch-thick aluminum entrance window as do the other β-particle detectors. Prototype detectors are aligned using the same methods described for the off-line β-particle scintillator. After alignment, a prototype detector is installed into a test fixture to simulate the geometry of the intended installation.

Iodine Detectors. Detectors used for the iodine channels normally consist of a 2-inch-diameter by 2-inch-long sodium iodide crystal with a 2-inch-diameter photomultiplier tube. The detectors are typically specified to have a maximum resolution of 8% for a 662 keV cesium-137 photopeak and may be supplied with an americium-241 pulser for pulse-height stabilization.

The electronics associated with a typical iodine channel has an energy-window discriminator with adjustable lower and upper thresholds. When the amplitude of a pulse signal from the detector lies between the lower and upper thresholds, the signal is counted as a valid event. The typical output is the number of events per unit time that fall within the energy window.

Since each detector has its own resolution and the system is operating as a single-channel analyzer, each detector's response will be unique. The response from a calibrated simulated iodine-131 source (barium-133) for each detector may be used when calculating the individual detector's expected responses for iodine-131. Upon completing the alignment of the iodine channel as a single-channel analyzer on the 356 keV photons of barium-133, the window will need to be readjusted to be centered on the 364 keV photons of iodine-131.

A typical pulse-height stabilizer consists of a small sodium iodide crystal. The doped crystal provides a constant source of γ-ray equivalent energy (GEE) in the form of light pulses. The light pulses are detected by the PM tube and converted to an electrical pulse by the PM tube and the preamplifier. The GEE is produced from an americium-241 5 MEV α-particle decay in the pulser crystal. These high-energy light pulsers are attenuated to an equivalent light energy of a 3 MEV γ-ray decay at the time the pulser crystal is imbedded into the mother crystal.

A typical preamplifier may contain three window circuits as shown in Fig. 1. One window is used to monitor the americium-241 stabilization signal from the detector, one is used to monitor the iodine peak, and one is used to monitor the background level at energies just above the iodine window. By monitoring the known stabilization source, compensation can be made for instabilities, such as from temperature variations in the gain of the PM tube and preamplifier. The background window can be used for active background subtraction.

Particulate, Iodine, and Noble-Gas Monitor. Often the measurement of airborne-particulate radiation, radioactive iodine, and radioactive noble gas is combined into a single instrument for such applications as airborne containment-building or vent-stack monitoring. A typical block diagram for such an instrument is shown in Fig. 2.

Typical Postaccident-Effluent, Wide-Range, Gas Monitor. A stack gas monitor typically is mounted near the nuclear power plant vent stack and receives a sample of the gas in the stack, which has been collected with isokinetic nozzles mounted in the stack. A normal-range monitor would be constructed as described previously and may cover a range of about five decades. A wide-range gas monitor may cover a range of about 12 decades through the use of multiple detectors.

An isokinetic nozzle is used to sample the air in the stack because, by keeping the velocity of the air entering the nozzle the same as the velocity of the gas bypassing the nozzle, a more representative sample of gas should be obtained for analysis in the monitor.

The low-range detector may be a β-particle scintillator and the preceding description applies. The mid- and high-range detectors may be solid-state detectors of a material such as cadmium telluride.

Liquid Monitors

Liquid monitors typically employ a sodium iodide scintillator coupled to a PM tube to measure γ radiation in the liquid stream. The liquid monitor may be mounted in-line with the stream to be monitored, or a sample stream may be extracted from the main stream and routed through a sample chamber into which the detector of the off-line liquid monitor is mounted.

An in-line liquid monitor typically is bolted directly into the liquid line with flanges on each end of the section of pipe that passes though the monitor. The monitor consists of a section of in-line pipe, a detector mounted adjacent to the pipe, and lead shielding surrounding the pipe and detector to prevent radiation from the surrounding area from entering the detector.

An off-line liquid monitor typically has a sample chamber into which the detector is inserted, and liquid enters and leaves the sample chamber through small-diameter pipes. Lead shielding surrounds the sample chamber and detector

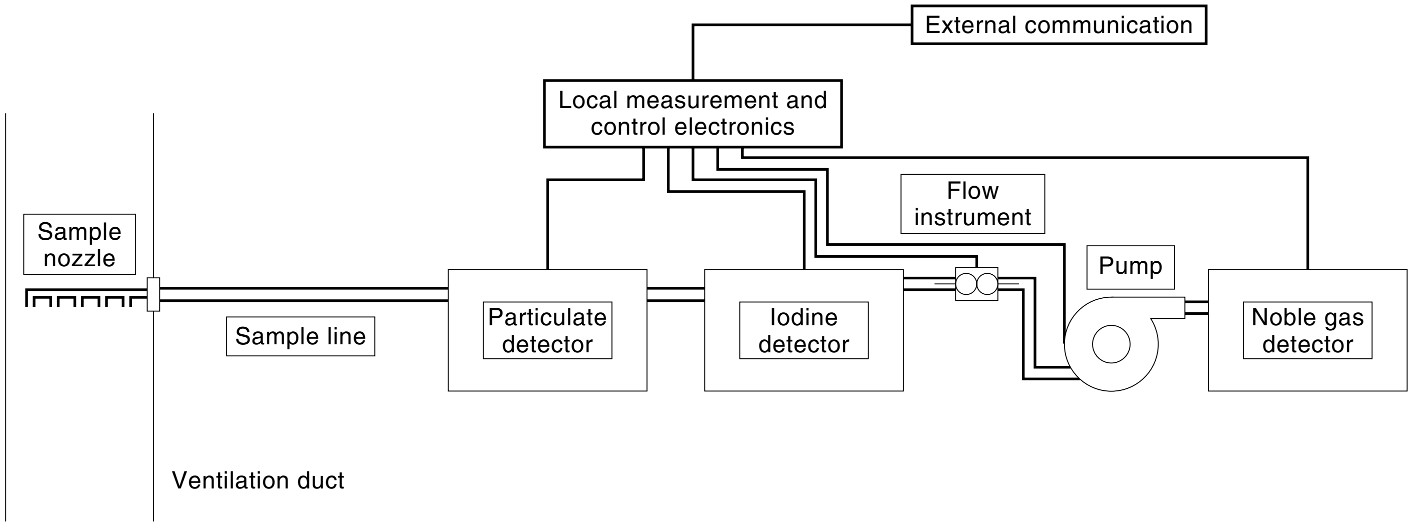

Figure 2. Block diagram of an airborne-particulate, iodine, and noble-gas monitor.

to prevent radiation from the area around the monitor from entering the detector.

The volume of water near the detector in a liquid monitor causes Compton scattering of the radioactivity in the liquid so that the signal seen by the detector includes not only the primary γ-ray energies but even more lower-energy signals from scattering of the primary photons. Therefore, there is typically no effort to distinguish specific energies of radiation in a liquid monitor.

Perimeter Monitors

Many nuclear power plants put radiation monitors around the perimeter of the plant site to measure dose levels at the site boundary. Communication with these monitors is often achieved by telephone lines or radio transmission. Perimeter monitors are typically high-sensitivity area monitors. At least one vendor has provided large-diameter, high-pressure ionization chambers and another vendor offers energy-compensated GM tubes. Typically requirements include an on-scale reading at normal background levels, a wide range for detecting significant radiation releases, and battery backup to avoid loss of data during a power outage.

MONITOR-PERFORMANCE PARAMETERS

In order for the monitors in the RM system to perform their required functions, they must operate within specific bounds of range, sensitivity, accuracy, and response time. The range is usually described in terms of the smallest and greatest magnitudes of activity or concentration for which the output signal is a valid representation and the radiation energies that may be included in the measurement. Sensitivity is a statement of how the output signal responds to a change in the measured variable. Accuracy is a measure of the uncertainty in the output signal. And response time is a measure of the length of time required for the output signal to change as a result of a change in the measured variable.

Range and Sensitivity

The range of an instrument may be limited on the low end by noise or by instrument precision and accuracy or it may be established by the scale and levels set for the output signal. The upper end of the range may be limited by the linearity or saturation characteristics of the detector or instrument.

The scales of most radiation monitors are logarithmic, and the span of an individual radiation channel is typically five decades. A decade is a factor of 10, so that a scale from 1 to 100,000 is five decades. The American National Standards Institute (ANSI) Standard N42.18 Section 5.4.2 recommends that the span be at least four decades above the minimum detectable level (MDL). Spans of more than six decades often require the use of multiple channels with overlapping ranges. Guidance for selecting the ranges for specific monitors is provided in ANSI N42.18 and ANSI/ANS-6.8.2 for effluent monitors, in ANSI/ANS-6.8.1 for area monitors, and in Regulatory Guide 1.97 for postaccident monitors.

For monitors that measure the concentration of radioactivity in gases or liquids, range and sensitivity are normally specified for a certain isotope or a distribution of isotopes as opposed to being specified over an interval of radiation energies. For monitors that measure dose rate, range and sensitivity are normally specified over an interval of radiation energies.

Sensitivity is normally determined by the characteristics of the detector. For monitors that measure dose rate, the sensitivity is often given as the ratio of the change in output signal to the change in radiation level that caused the signal to change, for example, for an ionization chamber, sensitivity is normally given in units of (A)/(R/h) or for a GM-tube-based area monitor, the sensitivity is usually given in units of (cpm)/(R/h), where cpm is counts per minute. For monitors that measure concentrations of radioactivity, the sensitivity is normally stated as the minimum detectable signal, which is a function of the detector characteristics, the effectiveness of radiation shielding, and the magnitude of background radiation.

Direct Measuring Instruments. A radioactive material concentration estimate A for a direct measurement is given by

$$A = \frac{y - \mathrm{Bkg}}{R_\mathrm{d}}$$

where y is the detector response to the concentration A plus the background, Bkg is the detector response to background, and R_d is the detector response per unit of concentration. A direct measurement is, for example, measurement of a gas or liquid volume in a fixed geometry. Typical units for y and Bkg are counts per minute, and typical units for R_d are counts per minute per μCi/cm^3.

The uncertainty in the concentration estimate due to counting statistics alone depends on the magnitudes of the total count and of the background count during some fixed time. The uncertainty due to counting statistics is then translated into an uncertainty in the concentration estimate by dividing by detector response.

The maximum sensitivity represents the lowest concentration of a specific radionuclide that can be measured at a given confidence level in a stated time (at a given flow rate, where applicable) under specific background radiation conditions (see ANSI 42.18, Section 5.3.1.4). The maximum sensitivity is commonly termed *minimum detectable level* (MDL) and is defined in terms of the uncertainty in interferences (termed *background* in radiation detection) and the response of the radiation detector:

$$\mathrm{MDL} = \frac{C_L s_b}{R_d}$$

where C_L is the confidence level desired in the measurement (unitless) and s_b is the background uncertainty in units of the detector output (e.g., counts per minute).

A MDL that is termed *minimum detectable concentration* (MDC) is based on ANSI N42.18:

$$\mathrm{MDC} = \frac{2 s_b}{R_d}$$

Another MDL is termed *lower limit of detection* (LLD) (10).

$$\mathrm{LLD} = \frac{4.66 s_b}{R_d}$$

Indirect Measuring Instruments. Radiation monitors that view a medium through which a sample has been drawn (e.g., particulate channels that monitor the radiation buildup on a filter) have additional characteristics for the establishment of range.

Detector response is stated in terms of output per unit of activity deposited on the filter medium. The quantity of activity on the filter for isotopes with half-lives much longer than the sample collection time is the product of concentration A times sample flow rate f times the sample collection time T. Then the concentration estimate becomes

$$A = \frac{y - \mathrm{Bkg}}{R_i f T}$$

where R_i is the detector output per unit activity on the filter [e.g., (counts per minute)/μCi]. Then

$$\mathrm{MDC} = \frac{2 s_b}{R_i f T}$$

and

$$\mathrm{LLD} = \frac{4.66 s_b}{R_i f T}$$

A specification of sensitivity also establishes the level above which the set-point value should be established. Set points should be well above MDLs in order to avoid spurious alarm/trip outputs due to statistical fluctuations in the measurement. See ANSI/ANS-HPSSC-6.8.2, Section 4.4.8.

Ambient background radiation is specified in order to determine the quantity of shielding required around the radiation detector, and this specified ambient level should be greater than the levels expected during plant operation. Detector-assembly performance can be stated in terms of detector response per unit background radiation for a specified isotope.

β-Particle Detectors for Airborne-Radiation Monitoring. Airborne effluent is frequently monitored by drawing a sample from the effluent stream into a lead-shielded sample chamber. The sample chamber is viewed by a detector with a thin (around 0.010 inch), predominantly β-particle-sensitive detector, since nearly all noble gases emit 1 β particle per disintegration.

β particles that are in excess of several hundred keV will lose about 100 keV for entrance normal to the detector face. The lower discriminator must be set above noise in the detection system. The upper discriminator needs to be open ended or set above 500 keV due to the high-energy tail of the energy distribution deposited by the β particles. This energy straggling for β particles results from infrequent, large-angle scattering by which the β particle can lose up to one half of its initial energy (11).

The decays per disintegration are the first factor in determining the response of a detector. The isotopic distribution of radioactive elements during normal reactor operation is significantly different from that postulated for accident releases. The use of a β-particle-sensitive detector has the advantage that the number of β particles per disintegration changes little from normal operation releases to the release postulated under accident conditions. γ-ray-sensitive detectors are at a disadvantage because the number of γ rays released per disintegration rises dramatically from normal operation to postulated initial-accident conditions and drops as noble gases decay. The element seen in normal operation is predominately ^{133}Xe, which undergoes γ-ray decay about 35% of the time. Initial-accident noble gas yields around 2 γ rays per disintegration.

Over Range Condition

An RM instrument should operate over its range within the required accuracy, and when radiation levels are significantly above the range (above full scale), the instrument must survive and continue to present an appropriate readout. When an instrument is in an over-range condition, it is important that the instrument output not fall below full scale with input levels up to 100 times greater than full scale.

Count-rate circuits are typically limited in their maximum counting rates by the resolving time required to distinguish two consecutive input pulses. That resolving time is sometimes referred to as *dead time*, and if a second pulse occurs during the dead time, it is missed and not counted. This is sometimes referred to as count-rate loss due to pulse pileup. If count-rate loss gets so severe that the output actually decreases while the input is still increasing, the condition is called *foldover.* Radiation emission from radioactive atoms is a random process and has a Poisson statistical distribution.

The resolving time of most count-rise instruments is equal to or greater than the width of the incoming pulse. As the input rate increases, the count-rate loss will increase until the output goes into saturation or even foldover. In some instruments, very high input rates have been able to freeze the circuit and cause the output rate to go toward zero. Such instruments are sometimes referred to as “paralyzable” and those that overcome this failing as “nonparalyzable.”

The instrument output count rate n can be calculated, as shown by Evans (12), from the input rate N and the instrument dead time p, during which the circuitry cannot respond to a second input pulse, by the equation

$$n = Ne^{-Np}$$

A useful method for estimating count-rate loss for a system that obeys Poisson statistics is to use the following approximation: when the output count rate is $A\%$ (any value below 10%) of the frequency represented by the inverse of the dead time, $1/p$, the instrument has a count-rate loss of approximately $A\%$.

Many design methods have been used to eliminate or reduce the effects of foldover and prevent the output of an instrument from going below full scale when the input levels are above full scale.

Accuracy

The definition of accuracy from ANSI Standard N42.18 is, “The degree of agreement [of the observed value] with the true [or correct] value of the quantity being measured.” Accuracy cannot be adjusted nor otherwise affected by calibration. It is a performance specification against which a channel is tested. For channels with multiple components, the individual accuracies are combined as part of the overall accuracy.

Accuracies, for monitors measuring concentration-related quantities, are normally specified for a certain isotope or a distribution of isotopes as opposed to radiation energies. Accuracies for monitors measuring dose rate are normally specified over an interval of radiation energies.

ANSI Standard N42.18, Section 5.4.4, provides a guideline that the instrument error for effluent monitors should not exceed ±20% of reading over the upper 80% of its dynamic range.

Response Time

For safety-related equipment, system response times used in safety analyses include the response times of the individual subsystems performing the protective function. This typically consists of instrument response time and mechanical system response time. The system response time must be allocated among the subsystems.

For a radiation-monitoring channel, the response time depends on the initial radiation levels, the increase in radiation level as a function of time, and the channel set point.

ANSI Standard N42.18 recommends that radiation-monitoring-channel response time be inversely proportional to the final count or exposure rate. This characteristic comes naturally with analog count-rate circuits. The time constant is typically established by a resistance and a capacitance in the feedback path of an operational amplifier, and for logarithmic circuits the resistance is typically the forward resistance of a diode at its current operating point. A factor of 10 increase in count rate will typically make the time response a factor of 10 faster. And, in these type circuits, if the change in input activity is a step function, the time response will be strictly a function of the end point and will not be affected by the starting point. Digital circuits, including software algorithms, are typically designed to emulate the time response of their analog counterparts. Thus fast response times can be provided at high radiation levels and longer response times at lower levels. Both types of circuits usually offer the ability to adjust the time constant to match requirements of the application.

Slow response times are needed to provide stable, smoothed outputs at the low end of the range. ANSI Standard N18.42 recommends that response times at low radiation levels should be long enough to maintain background readings within the required accuracy.

BIBLIOGRAPHY

1. ANSI/ANS Publication HPSSC-6.8.1-1981, *Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.*
2. ANSI/ISA S65.03-1962, ANSI Standard *Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection.*
3. USNRC Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Station Systems.*
4. American National Standards Institute Standard ANSI N42-18, *Specification and Performance at On-Site Instrumentation for Continuously Monitoring Radiation in Effluents.*
5. The requirements for postaccident monitoring equipment are outlined in USNRC Regulatory Guide 1.97, Revision 3, and NUREG 0737.
6. USNRC NUREG 0800, *Standard Review Plan,* Secs. 11.5, 12.3, and 12.4.
7. G. F. Knoll, *Radiation Detection and Measurement.* New York: Wiley, 1979, chap. 7.
8. D. J. Allard and A. M. Chabot, The N-16 gamma radiation response of Geiger-Mueller tubes, *Health Phys. Soc. Annu. Meet.,* Washington, D.C., 1991.
9. G. F. Knoll, *Radiation Detection and Measurement.* New York: Wiley, chap. 5.
10. USRCC NUREG/CR-4007, *Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurement,* 1984.
11. N. Tsoulfanidis, *Measurement and Detection of Radiation.* New York: Hemisphere Publish., 1983, chap. 13.
12. R. D. Evans, *The Atomic Nucleus.* New York: McGraw-Hill, 1955, p. 785.

Reading List

Fundamentals of Nuclear Medicine, N. P. Alazraki and F. S. Mishkin (eds.), New York: Soc. Nuclear Medicine.

CLINTON L. LINGREN
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RADIATION PATTERNS. See ANTENNA THEORY.
RADIATION PROTECTION. See DOSIMETRY.
RADIATION RESISTANCE OF ANTENNAS. See LINEAR ANTENNAS.
RADIATION, SPONTANEOUS EMISSION. See SPONTANEOUS EMISSION.
RADIO AND TV TRANSMISSION. See LOW-POWER BROADCASTING.
RADIO ANTENNAS. See ANTENNAS.
RADIO BROADCASTING. See ANTENNAS FOR HIGH-FREQUENCY BROADCASTING.