NUCLEAR POWER STATION DESIGN

Nuclear power plants are similar to other steam cycle power plants in that heat is utilized to boil water and then the energy in the steam is converted to mechanical work in a turbine. The rotational energy in the turbine powers a generator, resulting in electricity. The essential difference in nuclear plants is the utilization of nuclear fission as a source of heat. In nuclear power plants, the quantity of fuel to be handled is much less as compared to conventional plants, fuel cost fluctuations have minimal impact, the waste stream is relatively small, and certain emissions, such as carbon dioxide, are eliminated. On the other hand, a completely unique set of regulations and accident analyses apply, and issues like radiation must be addressed.

No new nuclear unit has been authorized in the United States since the mid 1970s. However, nuclear plants generate over 20% of the electrical power in the United States and almost 20% of the electrical power in the world. New plants are being built in Japan, South Korea, China, and other countries.

NUCLEAR LICENSING PROCESS

Commercial nuclear power plants are governed by a complex regulatory system. The process of addressing the regulatory requirements and documenting the conformance of the plant to the regulations is called *licensing*. This process is unique to nuclear plants in that it defines a large portion of the design criteria for the safety portion of the plants.

The licensing process is the collection of activities necessary to acquire an approval from regulatory agencies to build and operate a nuclear power plant(s) in accordance with the governing law. Regulation and licensing are required to assure and demonstrate adequacy of nuclear safety systems design to prevent accidents and to mitigate the consequences of accidents for public safety and health.

Most other countries utilize a process modeled on the US regulations. The US licensing process, codified into law in 10 CFR (Code of Federal Regulations) Part 50 is a two-step process: a step to acquire a construction permit (CP) and the other step to get an operating license (OL).

A Safety Analysis Report (SAR) and an Environmental Report (ER) are essential regulatory licensing documents which document the conformance of the plant design to the requirements. The USNRC has format and guidelines which specify what technical information shall be in each section and subsection of an SAR or ER. The requirements in the CFRs are law and are required. Additional documents like Regulatory Guides are guidelines to the acceptable implementation of the CFRs and subject to negotiation or interpretation. However, in practice, conformance to the Regulatory Guides is preferred because the high cost and risk of negotiating exceptions to the guidelines.

Title 10 of the Code of Federal Regulation (10 CFR) Section 50.34 defines the required contents of an application for a CP. The Nuclear Regulatory Commission (NRC) has issued two Regulatory Guides to assist a utility in preparing acceptable application, Regulatory Guide 1.70 (SAR) and Regulatory Guide 4.2 (ER). A typical Preliminary Safety Analysis Report (PSAR) may be 7000 or more pages long and may require about 1 year's preparation time. A typical ER often exceeds 1000 pages. Before the operating license is issued, the Final Safety Analysis Report (FSAR) is submitted and reviewed.

The difficulties experienced in licensing nuclear plants in the United States in the 1980s led to reevaluation of the licensing process. At the same time, the nuclear industry was working on standardized advanced nuclear plant designs. 10 CFR 52 was established as a more efficient licensing process. "Part 52" licensing is called one-step licensing because a specific plant is evaluated by the NRC only once, and public hearings are held only once. 10 CFR 52 has not been used for an actual plant application.

The Part 52 process is based on the assumption that nuclear plants will be standardized to the point that only minor changes are required to adapt them to candidate sites. When a standard design is certified by the NRC it can be placed on any site that meets certain preestablished criteria, such as seismic basis. When a plant owner proposes to build the unit, a Safety Analysis Report is submitted which is reviewed, and public hearings are held. If the application is approved, a combined construction and operating license is issued.

The license is based on constructing the plant as described in the application. This is established by Inspections, Tests, and Acceptance Criteria (ITAAC) which are part of the license. If the plant is completed as licensed, it may start up without additional review, public hearings, or other delays.

Operating nuclear plants are legally required to conform to the licensing requirements as committed to in the Updated Final Safety Analysis Report (UFSAR) and the Safety Evaluation Report (SER) issued by the NRC. Modifications to the plant or the operating procedures can be made only as allowed under 10 CFR 50.59 unless a license amendment is approved. Continued operation after the licensed life (generally about 40 years) is allowable only if the license is renewed. The renewal process has been established but no license extensions have yet been granted.

REACTOR TYPES

There are four major types of nuclear power plant reactors presently manufactured and operated in the world today: (1) the pressurized water reactor (PWR) manufactured by Westinghouse, Combustion Engineering, Babcock & Wilcox, Framatome, etc. (The Russian VVER and the internally developed Chinese plants are PWR designs.); (2) the boiling water reactor (BWR) manufactured by General Electric and licensees in countries, such as Japan, Germany, and Sweden; (3) the CANDU heavy water reactor manufactured by AECL; and

(4) the high-temperature gas-cooled reactor (HTGR) manufactured by General Atomics and similar older European gas reactor designs. Other designs, such as the Chernobyl type (RBMK) graphite-moderated, pressure-tube design have no potential future applications.

Advanced reactor designs are being developed in the United States, France, Germany, Japan, and other countries. These are variants of the existing designs with reduced requirements for active safety system functions and changes to reduce construction and operating costs. New concepts, such as breeder reactors (which convert ^{238}U to reactor fuel) have only limited support now. Work on fusion (using light elements, such as hydrogen, as fuel) continues, but commercialization is not expected before midway through the twenty-first century.

The PWR, BWR, and CANDU are water cooled, whereas the HTGR is cooled with helium. Because PWRs and BWRs are cooled and moderated with "light water" (which is ordinary water), they are called "light water reactors" (LWR). Heavy water, deuterium oxide (D_2O), has been used in research reactors in the United States and in the CANDU reactors. The primary advantage of the heavy water reactor is that it is more "economical" from the standpoint of utilizing neutrons and can use relatively low enrichment fuels. However, the cost of heavy water is high and negates much of this advantage now. Other than the use of heavy water, the CANDU units function much like the PWR plants.

THE FISSION PROCESS

The heat in a nuclear plant comes from a process known as fission, which takes place entirely within the nuclear fuel. Nuclear fission is an event where certain atomic nuclei "split" into fragments which are themselves nuclei of lower mass. When fission occurs, considerable energy is liberated primarily in the form of kinetic energy of the fission fragments. These fragments in turn impart their kinetic energy to the surrounding medium through collisions with other atoms, thus resulting in thermal energy or heat. The energy released via fission is obtained from the mass loss or "mass defect" between the original (parent) nucleus and the sum of the masses of the product nuclei (fragments or daughters). (Refer to nuclear engineering for a more complete description of the fission process.)

RADIATION AND SHIELDING DESIGN

One of the inevitable side effects of nuclear processes is nuclear radiation. Radiation is energy emitted as particles or rays. Radiation interacts with matter and transfers energy. The rays of the sun are common examples of this phenomenon. A substance that emits radiation is said to be radioactive. A material that contains unwanted radioactive material is said to be contaminated. Radiation control in nuclear power plants has two main goals: to protect personnel and equipment from direct exposure to radiation from the fission process and contamination and to minimize the spread of radioactive contamination.

Radioactivity

Nuclides that are unstable because they have been excited return to a ground state by emitting a particle or a quantity of electromagnetic energy. Such nuclides are said to be radioactive. In general, isotopes of most chemical elements that occur naturally are stable. However, a few are unstable, especially those of atomic number 84 and above.

The unit used to describe the amount of radioactivity, the becquerel (Bq), is defined as the emission from one disintegration per second. The type of radiation emitted in a disintegration is very important because some types of radiation are more damaging than others.

Types of Radiation

In addition to the disintegration rate (Bq), the type of radiation emitted in the decay is important. Unstable nuclides undergo spontaneous change at very definite rates by radioactive disintegration or radioactive decay. Decay results from emission of a charged alpha or beta particle, a gamma ray or a neutron, from the nucleus. Many nuclides decay into other unstable nuclides, resulting in a decay chain that continues until a stable isotope is formed.

The rate at which radioactive nuclides emit radiations is unaffected by temperature, pressure, or the presence of other elements that may dilute the radioactive substance. Each nucleus of a specific radioactive nuclide has the probability of decaying in a definite period of time. This is the "half-life" or the period of time over which the decay rate is reduced to one-half the original. Alpha and beta particles have very little penetrating power and are of little concern unless the source is inhaled or ingested. Gamma rays and neutrons have great penetrating power and require shielding or other protective features in the plant design.

Radiation Dose

The amount of radiation absorbed by a material is measured in units of grays. A gray is defined as 10,000 ergs per gram of material. Historically, the rad (0.01 gray) has been used for dose rate. Regulations in the United States are still based on rads. Because different types of radiation result in different biological effects, a unit called the rem has been introduced. A biological dose in rem is the absorbed dose in rads multiplied by a quality factor (QF). The QF is a function of the radiation type (alpha, beta, gamma ray, X ray, neutron) and at times the energy of the particle. For a constant energy level, the following quality factors may be used:

Gamma rays, X rays	1
Beta rays >0.03 MeV	1
Beta rays <0.03 MeV	1.7
Thermal neutrons	3
Fast neutrons	10

Radiation Standards. Radiation protection standards have been evolving since 1915. World wide [International Commission on Radiological Protection (ICRP)] and national bodies [National Council on Radiation Protection and Measurements (NCRP)] have developed and continue to review their recommended standards. These, in turn, have been adopted into the governing radiation protection regulations. These regulations govern the design and operation of nuclear power stations.

In the United States, the actions of nuclear power plant licensees are governed by Title 10 of the Code of Federal Regulations (10 CFR). In particular 10 CFR 20, Standards for

Protection Against Radiation, provides specific numerical guidelines for occupational exposure. From 10 CFR 20.101, the allowable exposures in rems per calendar quarter are as follows:

Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads	1-1/4
Hands and forearms; feet and ankles	18-3/4
Skin of whole body	7-1/2

10CFR20.106 defines the limits of radioactivity in effluents in the unrestricted area. Rule-making activities in the early 1970s addressed this and quantified its "as low as reasonably achievable" (ALARA) clause as Appendix I to 10CFR50 (1). Now the design objectives are to limit radioactivity in effluents from each reactor so that the annual dose to any individual in the unrestricted area (public exposure) is less than 5 mrem to the whole body and 15 mrem to any organ. (This may be compared with an expected natural background level of about 100 mrem per year.)

Shielding Design

The fission of a ^{235}U atom results in two new nuclides, generally neutron-rich and in an excited state. These nuclides subsequently decay by emitting (generally) beta particles (electrons) and gamma rays (electromagnetic radiation). Most of this radioactivity remains within the uranium oxide matrix of the individual fuel pellets. Thus the reactor itself requires the most shielding. However, some radioactivity does migrate from the fuel pellets through minute perforations in the fuel element cladding and finally gets into the primary coolant stream. From here, the radioactivity moves into all fluid streams serving the primary system and then into the remainder of the plant.

When certain nuclides are exposed to neutrons emanating from the reactor core, they undergo a nuclear reaction and result in a new nuclide which may require radiation protection. One such reaction occurs when ^{16}O in the reactor cooling water captures a neutron resulting in ^{16}N. ^{16}N then decays with a 7.13 s half-life releasing, among other things, a 6 MeV gamma ray. Although this half-life is short, it is sufficient to permit a large fraction of ^{16}N produced to be carried from a BWR core to the turbine and present itself as a severe turbine building and site shielding problem. ^{16}N is similarly produced in a PWR core, but it is contained within the primary coolant loop that in turn is located within primary containment. Thus its turbine does not see ^{16}N.

Certain radioactivity in the plant results from corrosion products in the primary coolant, particularly cobalt, which are then activated by neutrons from the reactor core. Two such nuclides are ^{58}Co and ^{60}Co whose half-lives are 71 days and 5.27 years, respectively. Both are gamma ray emitters. These ubiquitous nuclides plate out on piping and equipment throughout the plant. The buildup with time is evident in operating plants and manifests itself in increasing radiation doses in maintenance procedures. This source (and its concomitant radiation dose) can be reduced by chemically decontaminating the piping and equipment.

Having been given the mechanisms for determining sources, the next step is to quantify the sources in all piping and equipment that handles radioactive fluids. This includes consideration of equipment, such as filters and demineralizers, that accumulate radioactivity. This effort requires knowledge of all mechanical systems handling radioactive fluids.

Now given the radioactive sources, the geometry of source, and a design-basis dose rate, one can determine a bulk shield wall thickness. The distance from a point source reduces the dose rate by an inverse square relationship. Shielding effectiveness is additionally calculated using the exponential attenuation equation of the form

$$\text{Dose} = (\text{Unshielded dose}) \times e^{(-t/\lambda)}$$

where t is the thickness of shield and λ is the relaxation length. The relaxation length must include an adjustment for buildup and scattering of the radiation within the shield.

Unfortunately, the "ideal" shield wall must be penetrated by piping, HVAC ducts, doors, and conduits. If local dose rates are too high, compensatory shielding in some form may have to be added. Typical additions are labyrinths around doorways and formed-in-place penetration filler.

ALARA

Although it is not part of the shielding calculation, the equipment layout is also reviewed to assure that operational radiation exposures (for maintenance, inservice inspection, calibration, etc.) are as low as reasonably achievable (ALARA). Points to consider include assuring adequate space, permanent galleries, remote handling equipment, and separating nonradioactive from radioactive equipment. The basic techniques to reduce radiation exposure are Time (reducing time spent in high radiation areas), Distance (maintaining separation from radiation sources) and Shielding (adding absorbing material between the source and personnel. Additional guidance is included in USNRC Regulatory Guide 8.8.

SITE LAYOUT AND PLANT ARRANGEMENT

The site location considerations can be divided into two parts. The first is the location of the site itself and the second is the orientation of the plant within the selected site. Ideally, the location of the site should be at the center of the area where the power is to be consumed and at a source of adequate clean water for cooling. Additionally, the site should be located in an area where the length of required transmission lines can be kept to a minimum and the right-of-way for these transmission lines is readily available. These considerations, important as they are, may become secondary to the ecological and environment aspects of the site. In fact, a most important aspect in determining of the site is whether the site is ecologically acceptable, and in the case of a nuclear site, whether the plant can be licensed. The site cannot be located on a seismic fault or in an area susceptible to major seismic disturbances. It is also not desirable to locate a nuclear plant in a highly populated area. Regulations require that a large population exclusion area be provided around the plant.

The arrangement of buildings on the site and the arrangement of equipment within the buildings is affected by the number of units on the site, existing facilities and land use, and the safety issues to be addressed in the design of the plants. For example, concern about the potential damage from turbine fragments after a failure leads to a preference for a "peninsular" design where the turbine centerline is oriented so that these fragments are not ejected toward the safety-related equipment. Some key factors are discussed below.

Separation and Protection of Safety Systems

Nuclear plant safety systems are generally provided with two redundant, independent, and often identical trains (or divisions). To reach the desired level of safety, no single event should disable both trains. Therefore, the two trains need to be physically protected from internal and external hazards which could affect their operability and must be physically separated from each other to minimize damage to both from a single event.

This is achieved by placing each system within a designated area of a safety-related structure and dividing those structures into divisionally separated areas. The most important structures for safety-related equipment are the containment and the reactor or auxiliary building (which contains supporting safety-related equipment).

The electrical systems are divisionally separated except for one important special case. The control room and remote shutdown areas require connections to both divisions. Because of this, the control room is generally centrally located between the two divisions. Some plants have been built with a single control room to service multiple units. This complicates the separation and routing of electrical and control systems but allows some economy of design and convenience of operation.

ALARA and Maintainability Considerations

Nuclear plants must be constructed and operated with a policy of maintaining radiation exposure to personnel as low as reasonably achievable (ALARA). In addition to low radiation exposure, personnel safety and efficiency are important factors in plant operations and maintenance. Experience has shown that plants which are designed well for maintenance also have advantages in constructability. Although ALARA and maintainability are important considerations in the detailed design, the general arrangement has important influences on these issues.

The safety-related equipment outside containment is generally not highly radioactive in normal operation. The high radiation areas can be isolated to relatively small parts of each building. If these areas are located away from areas where personnnel normally are working, exposure and shielding requirements are reduced. In the general development process for the arrangement, personnel circulation plans are developed which define normal paths used by plant workers. Radioactive systems are located in areas removed from these paths.

Well-planned access and personnel movement patterns are important to minimize exposure. Separate paths are defined for entry into safety and nonsafety areas and into potentially high radiation or contaminated areas. This provides a method for implementing security and radiation control programs.

Constructability

A number of criteria are established for constructibility of the plant. The arrangement must allow equipment to approach the structure and cranes to be located within a reasonable distance of every point of the plant. As crane reach is extended, load capability is often reduced. A simple crane placement and equipment staging plan is usually developed for each proposed arrangement to evaluate the construction feasibility.

Aisles and passages are required in the design to allow equipment to be moved into place. It is common to leave openings in the walls and floors during construction to allow installing equipment easily. However, this involves a compromise because all openings must be closed at a later stage in construction, and this is an added cost over completing construction in one step. The routing of piping and cable trays is a major construction cost. An arrangement which allows routing to be done in designated pipe chases and tray paths is more economical to construct.

STRUCTURAL DESIGN

The design of safety-related structures for nuclear power plants differs from commercial or industrial design in several ways. The nonsafety structures are of a conventional design, but the safety-related structures are required to include loading combinations and events which are unique to nuclear units. Because of shielding and load-carrying requirements, nuclear plant structures are often massive. Conventional design methods may not be applicable to the increased thickness and sections of the concrete beams and slabs.

The complex seismic loading and size of the structures generally requires complex finite-element models of the buildings. In addition to the large seismic load and other external loads, such as wind, nuclear plant structures are also required to accommodate significant local loading. Pipe break, in addition to pressurization of local areas, may cause missiles and whipping pipe impact. External events, such as typhoons, also result in missiles.

The imposition of very large but very low probability loading on the structure leads to an economic need to minimize the safety factor (for the low probability loads) below that which might be used on a conventional building. To justify a reduced safety factor, the analysis must be more refined and well documented. Steel structures in nuclear plants are often part of a composite structure, requiring a more complex evaluation. Because of the high loads, steel design is carefully evaluated for torsional loading which may be difficult to withstand.

Containment Building

A containment building is a structure used to enclose the entire reactor vessel, the reactor coolant system, and other safety/auxiliary systems or portions thereof. The structure is designed to withstand all credible load conditions (normal, accident, and test) and those resulting from adverse environmental conditions (earthquakes, wind forces, tornados, aircraft impact, etc.). The basic safety design objective of the containment is to limit the release of radioactive fission products to the environment in the event of a postulated design basis accident (DBA) so that offsite doses comply with the Code of Federal Regulations (10CFR100) guidelines. In addition to limiting releases, the containment structure also provides radiation shielding to limit exposure in the areas outside containment. The containment is a pressure vessel which may be required to withstand pressurization as high as 500 kPa.

There are two general containment concepts or categories: dry containment (sometimes referred to as volumetric containment) and vapor or pressure suppression (wet) containment. Dry containments rely on a large volume to contain the energy introduced by a DBA after which the resultant peak pressure and temperature are reduced by an engineered safety feature (e.g., the containment spray system and containment fan coolers). Vapor or pressure suppression containments associated with BWRs use large heat sinks (water) to promptly absorb a portion of the energy released from a DBA by condensing steam, thus limiting the resultant pressure and temperature. Containment in BWRs (as it has been described) is called "primary containment" and is enclosed by another structure called the "secondary containment." There are a limited number of PWRs that use ice to suppress pressure. These containments, called ice condenser containments, have a section that is kept filled with ice in large baskets. In the event of a pipe break, the steam flows through the area of the baskets and is condensed in the ice baskets.

The most common type of concrete containment is the prestressed concrete design. The pre-stressing is implemented by posttensioning the containment with steel cable tendons. The containment walls and dome, which may be up to four feet thick, include tendon tunnels routed around the containment and over the dome. After construction, tendons are placed in the tunnels and tightened to preload the containment. Normally, the exterior of the structure has three or four tendon enclosures of siding which extend from the grade level to the Containment Building dome. Steel containments still require a concrete shielding structure outside the shell.

An airlock-type entrance is furnished for entering the containment building from the auxiliary building. Additional hatches are provided for removing large equipment, and an emergency personnel hatch is located in adidtion to the normal personnel hatch. Inside the containment, a shield wall called the primary shield or biological shield surrounds the reactor vessel. In a PWR, the steam generators and pressurizer are also inside shield walls. This shielding allows personnel entry for limited periods of time during operation. Because of the high level of radiation inside the biological shield during operation, personnel are not allowed in these areas when the unit is operating.

Auxiliary Building or Reactor Building

Normal facilities in the auxiliary or reactor building include the control room, cable routing and electrical equipment spaces, computer room, heating, ventilating and air-conditioning rooms, mechanical equipment spaces and other areas required to support the functioning of the safety-related systems. The structure may be steel or concrete. The structure is safety-related, and all exterior walls are designed to withstand tornado-generated missiles and other design basis events. Walls in the building may have specific requirements for fire rating, bullet resistance, or pressure resistance depending on the design requirements for the various areas in the building.

Turbine Building

The turbine building houses the turbine-generator (T/G) and associated operating equipment, such as the heaters, condensate pumps, bus-duct cooling unit. PWR turbine buildings are generally constructed with a substructure (below ground level) of reinforced concrete and a superstructure which is steel framing with metal siding. BWR turbine buildings generally require a more substantial superstructure to control releases. It is possible to construct a PWR with an outdoor turbine building. In this design, there is essentially no superstructure over the turbine operating floor. The turbine and other equipment are exposed to the weather. This design is feasible only in mild climates where freezing weather is not expected and is used on only a few units in the United States.

The turbine building at a nuclear plant may extend two or three levels (10 to 15 m) below the ground level. The height of the turbine building is set by the distance required to lift components with the turbine building cranes or the required elevation of the deaerator to provide feedwater pump suction pressure. A typical roof elevation is 30 m above grade. Because the turbine building does not have a safety-related function, it need not be designed for the design basis seismic event. However, in most plants the turbine building conforms to the Uniform Building Code (UBC) and is designed to withstand a certain seismic load.

Nuclear Fuel Handling Facilities

The function of this facility is to provide a place to store new fuel rods until needed for refueling and to store spent fuel that has been removed from the reactor core. The fuel handling building must be designed for the seismic design basis load. The function of retaining the water in the spent fuel pool is a safety-related function.

For a PWR nuclear facility a fuel handling building is normally provided which contains a spent fuel pool, new fuel storage vault, and decontamination facilities. The exterior walls and roof are designed to withstand tornado-generated missiles and are generally reinforced concrete. The building must be arranged so that the spent fuel from the containment can be transported through the fuel transfer canal and into the spent fuel while under water to maintain shielding. Because the refueling canal is generally a distance above grade, the fuel handling building must be tall. These buildings are generally constructed on a grade level concrete basemat.

POWER CYCLE

The purpose of any power plant is to convert heat energy from fossil fuels (coal, natural gas, oil), nuclear fission, or other sources into electrical energy. This is accomplished by setting up a closed thermodynamic cycle of working fluid, which is always water-steam in the power plants of interest. A cycle is a series of events in which the working fluid of a system experiences a number of processes that finally return the fluid to its initial state.

The cycle used today in steam turbine energy systems is based on the Rankine cycle which can be considered a practical modification of the Carnot cycle. The steam source generates heat from the fuel and converts high-pressure water into high-pressure, high-temperature steam. The steam enters the turbine where it expands to a low-pressure, low-temperature steam, and in doing so, works against the turbine blades, causing a rotation of the turbine shaft. Then the steam is condensed into water as it passes through the condenser, giving up its heat to cooling water supplied by a lake, ocean, or cooling tower. The condensed water is then pumped to a high pressure, after which it enters the boiler and starts the cycle again.

Typical large fossil-fired power plants produce superheated steam at 16 MPa and 550°C. A comparable nuclear steam supply system produces saturated steam at 7 MPa psig and 280°C. The lower pressure and temperature of steam produced in a nuclear power plant mean less available energy in each pound of steam. Hence, more steam is required for an equivalent power rating. This greater volume of steam requires larger passages throughout the turbine and longer blades. As steam expands through the turbine, the greater volume requires larger and longer blades. Most nuclear units in the United States currently use blades in the range of 1.0 m to 1.17 m long. Turbines with blades over 1.25 m have been designed for 50 Hz application outside the United States.

These requirements for large flow areas and longer blades make the 1800 rpm (1500 rpm for 50 cycle service) turbine design more applicable for nuclear turbines (3600/3000 rpm is typical for fossil turbines). To provide a flow path that passes the required volume of steam with the blade lengths available, the turbine designer uses the double-flow turbine, which passes half of the total flow to each of two sets of exhaust blades.

NUCLEAR POWER PLANT EQUIPMENT

Equipment in a nuclear power plant is often similar or identical to the equipment in fossil fueled plants. However, certain equipment is unique to the nuclear industry, either by design or by quality control and code requirements. Equipment at nuclear units is large compared to other power plants because of the economics of scale which have pushed the optimum nuclear plant size to 1000 MWe and beyond, whereas large fossil plants usually have multiple units with an individual size of about 600 MWe or less.

Equipment Classification

Equipment is classified as safety-related or nonsafety related. Safety-related equipment must be manufactured, procured, and installed under a quality assurance (QA) program meeting the requirements of 10CFR50 Appendix B. In addition there may be specific design requirements enforced by classification, such as seismic, environmental qualification, electrical Class 1E, or ASME Section III. The plant owner may introduce additional requirements by identifying equipment as critical for reliability or some other designation.

These requirements must be placed in the procurement specification for the equipment, the procedures for storage and handling at the site, and the construction/erection specifications. Safety-related nuclear plant equipment must have continuous documentation of the equipment to verify that damage or degradation has not occurred.

NUCLEAR PLANT SYSTEMS

Nuclear plants, like most major industrial facilities, are composed of systems that have defined functions and are linked at interfacial points. Systems are concepts that allow designing, operating, and evaluating the plant in manageable segments. Systems may be mechanical, electrical, or even structural, as well as a combination of all three. The boundary of a system is somewhat arbitrary and can differ at two otherwise identical plants. System divisions exist for convenience in identifying and classifying the plant components. A system is composed of a number of components which may be mechanical or electrical equipment or construction materials like piping.

System interfaces are one of the most important issues in nuclear plant design and operation. Although it is sometimes necessary to work with an isolated system, it is critical to remember that the only purpose of the systems is to support the overall operation of the plant. The most important element in system design is to correctly identify the system physical and functional requirements with respect to other systems.

System Classification

The design, construction, and operation requirements of a nuclear power plant is based heavily on the classification of the systems, structures, and components. There are several separate systems of classification which are combined to provide the requirements: safety classification, ASME Code classification, electrical classification, and seismic classification. These classifications are often related but are not consistently linked.

The safety classification is either safety-related (SR) or nonsafety related (NSR). In addition, some plants utilize classifications of reliability-related, regulatorily-related, or augmented quality to designate systems, structures, and components that are not required to be safety related but have important functions which warrant additional design, construction, or operational controls.

ASME Boiler and Pressure Vessel Code Section III governs the design of nuclear safety related fluid systems components. There are three categories of safety-related components in Section III:

Class I (or A) applies to components which are part of the primary system (reactor coolant) boundary and are required to prevent loss of inventory from the reactor coolant system.

Class II (or B) applies to components which are required to maintain the integrity of the secondary barrier (containment).

Class III (or C) applies to components which are safety related but do not fall under Class I or II.

In addition, ASME Boiler and Pressure Vessel Code Section B31.1 applies to nonsafety related power plant components. In many plants these components are referred to as Class D.

Electrical components and systems which must be safety-related are termed Class 1E components and those which are not safety-related are termed non-1E.

For convenience, the systems in a nuclear plant are organized into groups of systems that have related characteristics. Again, the grouping, like the system boundary definition, is somewhat arbitrary. However, it helps in understanding the functions of the systems.

Reactor Coolant and Fuel Systems

The reactor coolant system includes the components that contain the primary coolant which transfers heat directly from the core to the steam generators (in a PWR) or to the turbine (in a BWR). Reactor coolant system components are primarily made of stainless steel or an Inconel alloy. In some cases components may be made of carbon steel clad with stainless steel or Inconel. The PWR primary system operates at a pressure of about 16 MPa. Coolant leaving the core is slightly over 330°C. BWR operating conditions are about 7 MPa and 300°C.

The reactor is a large pressure vessel which contains the fuel. The reactor vessel is located in the approximate center of the containment structure and relatively low in the plant. This location minimizes the potential for a reactor coolant system boundary leak to drain the reactor vessel and leave the fuel uncovered.

Reactor coolant pumps circulate the coolant through the core and U-tube section of the steam generators in the PWR. The reactor feedwater pumps in a BWR have an analogous function to supply water to the core which boils and flows to the turbine. The PWR pumps are located close to the reactor on the "cold leg" or inlet piping. This location insures that the pumps handle the minimum temperature water in the loop and minimizes the potential for pump cavitation. The steam generators are large vertical heat exchangers. Westinghouse, Framatome and Combustion Engineering (as well as CANDU reactors) utilize U-Tube steam generators where primary coolant enters on side of the lower head and enters U-tubes, exiting on the other side of the lower head. Feedwater is introduced to the shell side of the steam generator where it flows over the tubes and boils.Westinghouse and Framatome PWR plants have two, three, or four steam generators and coolant loops and one pump per loop. Combustion Engineering units have only two loops and often have two reactor coolant pumps in parallel on each loop. Babcock & Wilcox units have two steam generators and are unique in that the steam generators are once-through designs instead of U-tube designs. The reactor coolant pumps typically circulate about 375,000 L/min and require up to 7.5 MW to drive them at full power.

PWR units have a pressurizer in the primary coolant system that keeps the system at constant pressure and prevents local boiling in the core. The pressurizer is a pressure vessel located relatively high in the containment and is connected to the rest of the system by a large diameter "surge line." In addition to the high location which adds a static head to the system, the pressurizer contains heaters which maintain a steam bubble in the top of the pressurizer. The pressurizer operates at about 25°C higher than the rest of the system. The high compressibility of the steam bubble dampens any pressure changes in the reactor coolant system.

The chemical and volume control system (CVCS) provides a means of adding fluid to the reactor coolant (RC) system during operation. At least part of the CVCS is safety-related in all plants. In some designs, it is part of the emergency core cooling system (ECCS) and provides a high pressure safety injection (SI) function. The CVCS also plays an important part in reactor power control by adjusting the chemical composition and mass of water in the reactor. As discussed previously, reactor power can be adjusted by varying the amount of boric acid in the coolant. The CVCS can add or remove boric acid. As a reactor heats up, the water expands and must be drained off to prevent overpressure. As the reactor cools off, the water contracts and must be supplemented to maintain water level. The CVCS also includes a water cleanup and treatment capability.

The letdown flow is additionally cooled after leaving the letdown regenerative heat exchanger. Then the coolant is filtered and demineralized in ion exchangers. If boron is to be removed from the coolant, it can be done in concentrators which boil off water to concentrate the boric acid or in a deborating ion exchanger.

Fuel handling during refueling and storage of spent fuel is a major factor in the economic operation of the plant. Refueling outages are the major portion of the time the plant is off-line. The limiting factor in outage time for a well-run plant is often the time required to move fuel. After shutdown, the reactor coolant system is cooled and depressurized and the bolts and seal welds holding the reactor head in place are removed. The upper portion of the containment (refueling cavity) is sealed and flooded to provide cooling and radiation shielding. When the plant is shut down, the fuel continues to generate decay heat for a long time. The spent fuel (about one-third of the core) and, sometimes, the entire core, is moved to the spent fuel pool, and new fuel is brought in to replace the spent fuel. Refueling outages are typically about 40 to 50 days but some plants have performed a full refueling in about 20 days. The CANDU units remain unique in their ability to refuel on line.

It may seem that the optimum approach would be to move only the fuel being replaced to the spent fuel pool. However, almost all fuel elements are moved, or "shuffled," as part of the core fuel management program. Some plants routinely inspect some or all of the used fuel to minimize the potential for cladding failure and major contamination of the primary system. If the entire core is off-loaded, radiation levels in the containment are significantly reduced and inspection or work on the reactor vessel can be done more efficiently. A system of fuel lifting and transport devices moves the fuel from the reactor vessel through the refueling canal and a containment penetration to the spent fuel pool. The storage pool contains all of the fuel off-loaded from the reactor until transfer is made to a disposal facility. Because of the failure in the United States to establish long-term, spent-fuel storage capability, many plants have redesigned the pools to accept more fuel and have developed dry storage technology outside the plant for the oldest fuel.

The fuel pool must maintain integrity and prevent leakage of water. About 30 feet of water cover over the fuel is required to provide radiation shielding and allow normal work in the area of the pool. Seismic design of the pool and leak detection systems control the loss of water from the pool. Safety-related systems are available to add water to the pool. The pool cooling system is required to provide cooling in a limiting case (such as a full core off-load) to keep the pool well below boiling. The storage configuration of the pool must prevent the spent fuel from approaching or reaching criticality under all pool conditions of water temperature, level, etc. A typical requirement is that for normal conditions $K_{\mathrm{eff}} < 0.95$ and for the limiting accident case, $K_{\mathrm{eff}} < 0.98$.

Engineered Safety Features

Engineered safety features (ESF) are provided to prevent or mitigate the consequences of postulated accidents. ESF systems have similar functions regardless of reactor type but the names and configuration of the system vary slightly with the different reactor manufacturers. ESF systems prevent the core from overheating, maintain the integrity of the reactor containment, and prevent release of radioactve material to the environment.

The term emergency core cooling system (ECCS) refers to a collection of systems which provide cooling water to the core after an accident or event. BWR and PWR systems use slightly different approaches because of the difference in fluid conditions in the reactor vessel.

Safety Injection System. PWR plants may utilize the safety injection and shutdown cooling systems or residual heat removal. The Safety Injection system (SIS) is the primary emergency core-cooling system (ECCS) in modern PWRs. The function of the safety injection system is to supply borated water to the reactor coolant system to absorb neutrons (provide additional shutdown margin) and to limit fuel cladding temperature in the unlikely event of a loss-of-coolant accident (LOCA). The SIS includes, as a minimum, three phases of system operation: passive accumulator injection; active pumped safety injection; and shutdown cooling (residual heat removal) pumped recirculation.

During the passive accumulator injection phase, accumulator tanks rapidly inject borated water, stored at approximately 650 psig, into the cold legs of the reactor coolant system (RCS). In normal operation, the accumulators are isolated from the primary system by check valves. Before the electrically driven intermediate (safety injection) and low head (shutdown cooling/residual heat removal) pumps are energized and deliver coolant, the accumulators provide rapid cooling of the reactor core for large breaks that would otherwise result in core uncovering and overheating. Because the intermediate-head safety injection pumps (and high-head safety injection pumps, if used) can maintain the pressure of the primary system above the accumulator pressure, accumulators would not function in small break accidents.

Shut-down Cooling/Residual Heat Removal. The shut-down cooling (SC) or residual heat removal system (RHR) transfers heat energy from the core and reactor coolant system during plant shutdown and refueling operations. As mentioned, the system is also employed in conjunction with the safety injection system for emergency core cooling when pipe ruptures occur. This system is a low pressure, high-flow system which operates at pressures up to about 400 psig. In some units, the shutdown cooling pumps are interchangeable with the containment spray pumps.

BWR Emergency Core Cooling Systems. The BWR emergency core cooling system (ECCS) generally consists of (1) a high-pressure core spray system; (2) a low-pressure core spray system; (3) an automatic depressurization system; and (4) the low-pressure coolant injection system, which is a function of the residual heat removal. The purpose of the high-pressure core spray system (HPCS) is to depressurize the vessel and primary coolant system, to provide makeup water in the event of a loss of reactor coolant, and to prevent fuel cladding damage in the event the core becomes uncovered due to loss of coolant. The low-pressure core spray (LPCS) system also prevents fuel cladding damage in the event the core is uncovered by a loss of coolant. The cooling effect is accomplished similarly to the high-pressure system by directing jets of water down into the fuel assemblies from spray nozzles mounted in a ring located above the reactor core. Automatic depressurization, or blowdown, through selected safety/relief valves, functions as a redundant system to the operation of the high-pressure core spray system. The blowdown depressurizes the reactor pressure vessel, permitting the operation of the low-pressure coolant injection system and/or the low-pressure core spray system. The low-pressure coolant injection system (LPCI) is a system redundant to the low-pressure core spray system. The low-pressure coolant injection system is one operating mode of the residual heat removal (RHR) system.

Containment Spray System. The containment spray system is an active ESF with several functions. It cools and depressurizes the containment after a high-energy line break by condensing steam out of the atmosphere. It also washes contaminants out of the containment atmosphere. Finally, it is designed to deliver enough sodium hydroxide (NaOH) to the containment to form an 8.3 pH solution when combined with the water collected at the bottom of the containment after a LOCA. This pH level prevents radioactive iodine from leaving the water solution and minimizes the potential for high off-site releases from containment leakage. The system consists of two entirely independent subsystems such that the required functions can be accomplished in the event of a single active failure in either of the subsystems. The containment spray system in PWRs reduces the temperature and pressure in the containment and provides the necessary iodine dose reduction factor following a LOCA. Generally, it is not used in BWRs for several reasons. One reason is that metals in the plant would react with the caustic spray. Therefore BWRs require a secondary containment and an standby gas treatment system (SGTS). In some BWR Mark III containments, a containment spray system utilizing only water is provided to aid in the reducing of post-LOCA pressure and temperature but not in the removal of radioactive iodine. The effects of chemical reactions on materials in the PWR is also a concern. The spray reacts with aluminum or galvanized material to release hydrogen. This is a major concern because of the fire or explosion potential. Therefore, use of these materials is restricted.

The containment spray system contains pumps that initially take supply from the refueling/safety injection water source and can be switched to take suction from the containment sump for long-term operation. If necessary, the spray water can be routed through heat exchangers (sometimes the shutdown cooling heat exchangers) to reduce the temperature before spraying. The spray is discharged through multiple spray nozzles mounted on ring headers in the top of the containment.

Standby Gas Treatment System. The standby gas treatment system for the BWR Mark I, II, and III systems (an active ESF) consists of two parallel full-sized systems designed to operate automatically, immediately following a LOCA. Both systems are located in the reactor building and process the reactor building atmosphere before discharge to the atmosphere via the elevated release point (ERP) stack. Each SGTS has a minimum capacity of 100% based on the total reactor building volume per day plus an amount to account for volumetric expansion due to a rise in reactor building temperature following isolation. Either system may be considered an installed spare. Each process system has sections (in order of air treatment) as follows: an air blower, a demister, a prefilter, an electric heating coil, a high-efficiency particulate air (HEPA) filter, an iodine filter, and another HEPA filter. The two SGTS filter trains are completely separate and are located at opposite sides of the reactor building.

Combustible Gas Control. Now, all containments are required to use a hydrogen removal system. This active ESF, called the combustible gas control system, recombines the hydrogen and oxygen produced during and after a LOCA. Hydrogen recombiner systems are often used for the BWRs and PWRs. Recombiners are devices, installed outside the containment, which take flow from the containment, heat it to very high temperatures (up to 1100°C), cool it, and return it to the containment. This process combines the free hydrogen with oxygen to eliminate the fire or explosion potential. Initially the recombiner uses electric heaters to begin the reaction and then uses a regenerative cooler to cool the exhaust and heat the intake air. The reaction becomes self-sustaining in the presence of significant amounts of hydrogen. The system is designed to keep hydrogen levels below 4%.

Another option is hydrogen ignitors which are essentially spark plugs or glow plugs mounted throughout the containment dome. These provide a controlled burn of hydrogen. A new option being suggested for advanced reactor designs is a passive catalytic hydrogen recombiner that consists of catalytic plates mounted in the containment which will cause hydrogen and oxygen to recombine when they come in contact with the surfaces. They have been demonstrated effectively in tests in simulated containment environments but have not been used in actual plants.

Auxiliary Systems

A number of systems support operation of a nuclear steam supply system (NSSS) but are not directly involved with responding to accidents or cooling the core. One of these, the auxiliary feedwater (or emergency feedwater) (AF) system, is sometimes considered an engineeered safety feature system because it is a safety-related system that responds to plant accidents or abnormal events. However, it cools the core indirectly through the steam generators. The steam generator blowdown (SD) system provides a means to clean the steam generators. The primary sampling (PS) system provides chemical and thermodynamic information about the reactor coolant system.

Auxiliary Feedwater System. The AF system (PWR plants only) is a safety-related system that provides water to the steam generators in when the main feedwater system is not operable. This allows removing the decay heat from the reactor by boiling the water in the steam generator and transferring the nonradioactive steam to the condenser or the atmosphere. The reactor coolant system is designed to provide decay heat removal capacity via natural (gravity-driven) circulation to the steam generators if the AF system provides cool water to the steam generators. This avoids the necessity of initiating safety injection, which imposes a thermal transient on the reactor vessel.

The AF system contains two to four pumps. Three pump systems are most common on older PWR units, and four pumps is the accepted current design. Each pump may have from 50% to 100% of the capacity needed for safe shutdown. In addition to safety division separation, diversity of pump drivers is required to minimize the potential for common mode failure. The most common pump drivers are electric motors and steam turbines, but some units have diesel engine drivers. The water is supplied from safety-related portions of the condensate storage tank in older plants and from dedicated AF storage tanks in the newest designs. The AF system piping distributes the flow to all steam generators via a connection to the feedwater piping or a dedicated line to a special nozzle on the steam generator. Depending on the system design, a pump may be capable of feeding all steam generators or some of the steam generators.

Cooling Water Systems

Important parts of any power plant include the cooling systems. Cooling is required for fluid systems and for equipment. Nuclear plants have unique requirements for those cooling systems that cool systems or equipment which are radioactive or potentially contaminated. Because cooling systems must ultimately transfer heat to the environment, either to the air or a body of water, the design must preclude release of radioactive materials. Cooling systems are also a required to provide certain amounts of cooling specifically to support the safety-related operations of the plant. Therefore, the cooling systems are required to be safety-related.

These requirements are generally met by using two systems, a closed cooling water system which directly cools the nuclear systems and equipment, and an open cooling system which transfers heat from the closed system to the environment. Under normal operations, the closed system will remain clean (uncontaminated). In the event of leakage from a radioactive system, radiation detectors are included in the closed system design, and the boundary between the open and closed system prevents release to the environment. The closed cooling water system is usually called the component cooling (CC) system and the open cooling water system is usually called the essential service water or nuclear service water (SX) system.

RADIOACTIVE WASTE PROCESSING

The operation of nuclear power plants produces moderate quantities of waste solids, liquids, and gases that contain, or are contaminated with, small quantities of radioactive material. Such wastes are called radwaste. Spent reactor fuel elements are not considered part of this category of radwaste. Processing of these wastes is required to prevent their uncontrolled release to the environment. The radioactive waste and corresponding treatment systems are normally subdivided into three major categories: (1) gaseous radwaste, (2) liquid radwaste, and (3) solid radwaste.

The gaseous radwaste system collects and processes radioactive (or potentially radioactive) waste gas. The system is designed to limit the release of gaseous radioactivity so that personnel exposure and activity releases in restricted and unrestricted areas are as low as is reasonably achievable

(ALARA). The design objectives are to limit the activity released in accordance with 10CFR50, Appendix I.

There are significant differences between the BWR and PWR gaseous source, and the methods for treating each are generally different. For a BWR, the dominant release point is the main condenser steam jet air ejector (SJAE) exhaust. In the early BWRs, the SJAE was designed to condense the water vapor, and the ejected mixture was discharged to a 30 minute delay pipe where most of the radioactive gases would decay. The delay was provided by a long, large pipe usually buried in the ground adjacent to the plant and release stack. A HEPA filter was located near the discharge point to remove any particulates in the system that were formed from the radioactive decay of xenon and krypton. The off-gas system currently favored for BWR units is a charcoal adsorption system.

For the PWR, the gaseous waste system is used to reduce the concentration of fission product gases in the primary coolant during operation and to remove these gases from the coolant when the reactor is shut down. Operating flow rates from this source are typically less than 10 scfm and may contain some hydrogen but no oxygen. Although PWRs today may have charcoal adsorption systems, most PWRs are equipped with pressurized gas decay systems. The number and size of the holdup tanks depends on the size of the unit and the holdup time required, usually 30 to 60 days.

The primary sources of liquid radwaste are leaks from the reactor coolant system and, to a lesser extent, from the fuel pool and the processes used to clean these plant process water systems. Both PWRs and BWRs have a reactor coolant cleanup system that continually removes a portion of the water from the rector, removes impurities, and returns it to the reactor. The processing of liquid radwaste depends on both its physical and chemical properties. Both PWR and BWR plants have facilities for collecting, sampling, removing suspended and dissolved solids, and analyzing effluent.

The basic methods, or unit operations, used in liquid waste processing are filtration, evaporation, and ion exchange. Filters remove solids suspended in the water, and ion exchange demineralizers or evaporators remove dissolved solids. Some systems may use a powdered resin precoat filter that filters and demineralizes in the same vessel.

One source of solid radwaste is the liquid radwaste system. Solid radwaste is the term applied to solid materials that are, or have become, radioactively contaminated through operation of the plant systems. These materials include contaminated filter sludge and media, spent demineralizer resins, and concentrates from evaporators. They are generally known as the wet solid wastes. Other solid wastes generated at the nuclear plant include such items as contaminated clothes, paper, rags, tools, and equipment, these are called dry active wastes (DAW).

Treating the wet solid radwaste at the station generally involves the tasks of remotely receiving, processing, packaging, storing, and loading the waste by a waste solidification system. This system receives the wet solid wastes on demand via interconnecting piping from the respective holding tanks. Some of these wastes are further concentrated by filtration to remove most of the liquid or are mixed with other wet solid waste to achieve the desired moisture content. Then these wastes are mixed with a solidification agent and delivered to a container to solidify. One of the most widely used solidification agents is Portland cement or one of its many variations involving various additives.

Trash or DAW is normally collected throughout the plant in plastic bags and is periodically brought to a central location for processing. At this location, the bags are manually fed into a dry waste compactor. These compactors compress as much as 225 kg of trash into each 200 L drum for off-site shipment.

Incineration is another technique for processing DAW, but it is not currently being practiced in any operating US plants. New plants are equipped with incinerators for DAW. The off-gas from this process is treated and released by conventional methods, and the resulting particulates and ash are collected and solidified.

Disposal of waste has become one of the most controversial issues of nuclear power. In the United States only a few sites accept low-level waste for disposal, and there are no operating sites for high-level waste (spent fuel). Although legislation has been passed to establish regional disposal sites for low-level waste, none of these sites has yet been developed. The cost of disposal has increased significantly adding economic pressure to the political pressure to minimize waste generation.

An attempt was made several years ago to add a third classification of waste, named Below Regulatory Concern (BRC). This would have applied to waste that has an activity level only slightly above normal background levels. Because this material would not significantly add to the exposure of the public, it was suggested that this material could be treated as nonradioactive waste. This proposal was politically unacceptable and was not incorporated into the disposal regulations but would have significantly reduced the dry waste volume.

Power Conversion Systems

Nuclear power plants utilize a relatively conventional steam power cycle to operate the turbine and generator. The utilzation of steam generators in the PWR to isolate the power conversion systems from the coolant that passes through the reactor core makes the PWR systems look much like a basic fossil fueled plant. The BWR design, of course, requires some shielding of the steam and feedwater systems and additional systems to handle the discharge of fluid from those systems.

In some ways nuclear systems are simpler than other power plants because nuclear units do not include superheaters or combined cycle designs. The basic nuclear plant power conversion systems consist of a condensate (CD) system, a feedwater (FW) system, a main steam (MS) system and, of course, the turbine generator. The circulating water system cools the condenser.

Condensate System

The condensate system takes suction from the hotwell of the condenser, where the condensed steam exhausted from the turbine is collected. Depending on the design of the main cycle and the characteristics of available pumps, the condensate system may consist of two pumps in series, a condensate booster and a condensate pump. These pumps supply water to the low pressure heaters and then to the suction of the feedwater pumps. The feedwater pumps are usually configured with three 50% capacity pumps. A combination of feedwater booster and feedwater pumps can also be used.

Condensate Polishing System

The condensate polishing (CP) system is a series of filtering and demineralizing beds which condition the feedwater and remove potentially harmful contaminants. Because of the nature of the steam generation cycle, most contaminants not removed in the CP system are eventually deposited in the PWR steam generators. The steam generator blowdown system removes some of the solid deposits from the steam generators. One of the major problems with operating PWRs is degradation of the steam generators, especially tube cracking. A significant number of units have required replacement or are expecting to require replacement. The total cost of replacing steam generators may be as high as $200 million. The predominant causes of degradation are material defects and poor chemical control in the feedwater system. Therefore, condensate polishing has become a very important system.

Circulating Water System

The circulating water (CW) system supplies water to the turbine condenser to remove heat rejected from the steam cycle. This rejected heat is rejected in turn to the environment by some type of final heat exchanger. The circulating water system can supply lesser amounts of auxiliary cooling water to other equipment and can act as a water reservoir for the fire protection system. The amount of heat rejected by the steam cycles is much greater than the amount of heat converted to electrical energy by the turbine-generator. The ratio of heat rejected to heat converted to work is about 2 : 1 for nuclear plants with a cycle efficiency of about 33%. The ratio of heat rejected to heat converted to work is about 1.5 : 1 for fossil-fuel plants with a cycle efficiency of about 40%.

Heat rejection by a circulating water system is called indirect cooling. Heat in the steam is exchanged with the environment indirectly via the circulating water. The environment may be a lake, river, ocean, or cooling tower. The circulating water system consists of the condenser, the circulating water pumps, the system piping, and the heat sink.

Condenser

The condenser receives the exhaust steam from the turbine and condenses it to water. The steam exhausted by the turbine is "wet" or saturated, but it still contains latent heat of vaporization. This heat is transferred to the circulating water. Power plant condensers are almost exclusively surface condensers, in which the steam and the circulating water are separated by a wall. The shell-and-tube condenser is the only type suited for large power plant applications. Steam is condensed to water on the outside surfaces of tubes through which the circulating water passes.

The design parameters for a condenser include steam flow or heat to be rejected, cooling water temperature, and tube cleanliness factors. The steam flow or heat load is determined by the turbine heat balances. The cooling water temperature is established by the site conditions, such as the mean lake or river temperature, or by the cooling tower performance specifications. The tube cleanliness factor is an assumption based on the type and quality of the circulating water.

Tube materials are admiralty brass, copper nickel, stainless steel, titanium, or other exotic alloys. Condenser tubes for freshwater applications are generally admiralty brass, 90-10 copper nickel, aluminum brass, or stainless steel. Condenser tubes for brackish water and seawater applications are generally admiralty brass or titanium.

There are always some noncondensable gases in the exhaust steam that collect in the condenser shell during normal operation. Because the condenser operates below atmospheric pressure, air leaks into the condenser shell through joints. These gases tend to destroy the vacuum effect, because they do not condense into a liquid. The noncondensable gases are removed from the condenser shell by the vacuum equipment. The vacuum equipment is usually one of two types: a steam jet air ejector (SJAE) or a vacuum pump. All condensers are equipped with vacuum breakers, which allows the condenser shell to be vented to the atmosphere in an emergency. Breaking the vacuum in the shell decreases the steam flow through the turbine, which helps prevent overspeeding the turbine.

NUCLEAR POWER PLANT ELECTRICAL SYSTEMS

A nuclear power plant has many electrical power systems that are required in the generation of electrical power. This section briefly discusses the major components within each of the following electrical power systems: main power (MP), auxiliary power (AP), diesel generator (DG), dc distribution system (DC), instrument and control power (IP), and low-voltage (LV) auxiliary power.

Main Power System

The MP system provides the power plant output to the utility network and electrical station auxiliaries.

The MP system receives electric power from the turbine generator (TG) system which converts mechanical energy into electrical energy. The MP system transforms or steps up the generator output voltage (typically 22,000 V) to a higher voltage (typically 345,000 V) to supply electrical power to the switchyard (SY) system and provides power to the AP system for distribution to station electrical loads.

The MP system consists of the main generator, the isolated phase bus including potential transformers, the main power transformer (MPT), the generator excitation and voltage regulation system, and the generator grounding transformer. Each subsystem includes accessories for control and protection.

The generator is driven by the steam turbine and produces electrical energy. The rotor carries the field winding, and the stator carries the three-phase armature winding which is typically wye-connected. The neutral of the generator is usually high-resistance grounded through a grounding transformer which is loaded by the resistor and includes ground fault protection relaying. Multiple sets of current transformers are installed on each phase bushing and on each neutral bushing of the generator to metering, voltage regulation, control, and protection equipment.

The generator terminals are connected to an isolated phase bus subsystem, which extends between the generator and the MPT. Taps permit connecting the unit auxiliary transformers, the potential transformer, and surge protection equipment to the isolated phase bus. Secondary potential transformers are provided for connecting metering, voltage regulator, control, and protection equipment. Surge protection equipment is provided to limit surges.

The MPT consists of three single-phase transformers or one large three-phase transformer, which are typically connected delta on the low-voltage side and wye on the high-voltage side. The MPT is used to step-up the voltage between the generator and the switchyard. Direct connected gas-insulated buses are used for the switchyard connection where gas-insulated switchgear (GIS) is used, or overhead conductors are used where there is an outdoor switchyard. Each high-voltage and low-voltage phase bushing is equipped with current transformers to connect metering and protection equipment. The high-side neutral is usually grounded through a disconnect switch to provide the capability of opening the grounded neutral for an ungrounded wye configuration. The high-side neutral is protected with a surge arrester when the neutral is isolated from ground and the disconnect switch is in the open position.

Some main power system designs use a generator circuit breaker (GCB) between the terminals of the generator and the MPT. This allows supplying power from the switchyard through the MPT and from the unit auxiliary transformers (UATs) to the loads when the generator is off-line.

Auxiliary Power System

The AP system provides a reliable source of power to all electrical loads in the station. The AP system receives power from the SY system and the generator terminals (MP system), transforms this power to appropriate voltage levels, and distributes it to loads located throughout the station.

Several voltage levels are used to match the requirements of various station loads. Large motors (2500 hp and larger) are typically supplied at a nominal voltage of 6,900 to 13,800 V. The remaining large motors and load centers are normally supplied at a nominal voltage of 4,160 V. Motors rated from 0.5 to 225 hp are typically supplied at a nominal voltage of 480 V. Motors rated less than 0.5 hp (with the exception of valve motor operators) are typically supplied from 120 V single phase distribution panels. These panels receive power from the 480 V system.

Power from the switchyard (typically 345 kV) is transformed by the start-up transformers (SUTs) to a lower voltage, typically 4,160 and either 6,900 V or 13,800. The SUT is provided with multiple secondary windings to power the 6,900 or 13,800 V and 4,160 V buses. To accommodate voltage variations in the 345 kV system voltage, the SUT maybe equipped with a load tap changer.

Power from the generator terminals is transformed by the unit auxiliary transformers (UATs), which also have multiple secondary windings, to provide 13,800 and 4,160 V supply to the non-Class 1E ac auxiliary power subsystem. The UATs do not supply the Class 1E ac AP subsystems unless a GCB is part of the main power system design.

Redundant Class 1E on-site emergency diesel generators (one per division) are provided to supply power to the Class 1E ac AP subsystem should the supply from the switchyard fail.

A non-Class 1E diesel generator may be provided to supply critical non-Class 1E loads separately from the Class 1E diesel generators if the normal sources of power fail.

The non-Class 1E ac auxiliary power subsystem supplies the non-Class 1E electrical loads, except for a limited number of critical loads requiring diesel generator backup which are fed from the Class 1E buses with appropriate isolation and separation. Large loads are typically supplied at 4,160 to 13,800 V by medium-voltage switchgear.

The non-Class 1E ac AP subsystem is typically divided into two independent divisions. Generally, parallel trains of equipment are fed from different divisions for increased reliability and operational flexibility. Where feasible, the equipment and cables of the two divisions are separated inside the power block to avoid damage to equipment in both divisions by a single event.

By selecting which redundant loads are operated, the operators can balance the load on the power sources to the auxiliary power system to optimize AP system operation.

The Class 1E AP subsystem supplies Class 1E electrical loads and a limited number of non-Class 1E electrical loads that require diesel generator backup. Large loads are fed at 4,160 V by medium-voltage switchgear. Smaller loads are fed at 480 V by load center transformers, load center switchgear, and MCCs. The medium-voltage switchgear and the load centers are normally located in the primary auxiliary building near the majority of the Class 1E loads. MCCs are located in the vicinity of their loads.

The Class 1E AP subsystem is typically divided into two redundant divisions. Redundant trains of mechanical equipment are fed from different divisions as dictated by the plant design and as stipulated by the NSSS supplier. The two divisions are physically and electrically separated to prevent damage to both divisions from a single event.

Selected non-Class 1E loads are typically supplied from extensions of the 4.16 kV bus of each division through a tie breaker. The tie breaker serves as an isolation device between the Class 1E system and the non-Class 1E loads. The non-Class 1E loads which are fed from these extensions are limited to loads that may require access to diesel generator power after an accident or loss of off-site power.

In the event of a complete loss of off-site power, the emergency DG system, in conjunction with the Class 1E seismic Category I AP system, provides an on-site standby source of ac electric power to the Class 1E loads, including the engineered safety feature (ESF) system and equipment required to (1) safely shut down the reactor, (2) maintain the reactor in a safe shutdown condition, and (3) ensure, in the event of a LOCA that off-site radiation doses are held below the requirement of 10 CFR 100 (5). In addition to these loads, the DG also supplies power to selected, critical non-Class 1E loads, per NUREG 0737, only after the operator has determined that generator has enough spare capacity to feed these loads.

In the event of a loss of off-site power (LOOP), the Class 1E AP system undervoltage relay trips the start-up transformer's main breaker and starts the DG. The DG attains rated voltage and frequency within 10 s. If a safety injection actuation signal (SIAS), auxiliary feedwater actuation signal (AFAS), or containment spray actuation signal (CSAS) occurs concurrently with a LOOP, the electrical output of the generator is fed to the 4.16 kV bus in timed, sequenced, loading steps to prevent stalling the generator and/or extreme voltage and