Table of Contents

12.0 Radiation Protection

12.1 Shielding
12.1.1 Design Objectives
12.1.2 Design Description
12.1.2.1 Station Site Area
12.1.2.2 Reactor Building
12.1.2.3 Auxiliary Building
12.1.2.4 Fuel Handling System
12.1.2.5 Design Basis Accident
12.1.2.6 Detailed Design Consideration
12.1.3 Source Terms
12.1.4 Area Monitoring
12.1.5 Operating Procedures
12.1.6 Estimates Of Exposure
12.1.6.1 Normal Operation, Exposures
12.1.6.2 Shielding Analysis
12.1.6.3 Plant Shielding Review Using NUREG-0578 Sources
12.1.7 References

12.2 Ventilation
12.2.1 Design Objectives
12.2.2 Design Description
12.2.3 Source Terms
12.2.4 Airborne Radioactivity Monitoring
12.2.5 Operating Procedures
12.2.6 Estimates of Inhalation Dose
12.2.6.1 Inhalation Doses in the Containment
12.2.6.2 Inhalation Doses in the Auxiliary Building
12.2.6.3 Inhalation Doses in the Turbine Building
12.2.7 References

12.3 Radiation Protection Program
12.3.1 Program Objectives
12.3.2 Facilities and Equipment
12.3.2.1 Radiation Protection and Chemistry Facilities
12.3.2.2 Additional Facilities and Access Provisions
12.3.2.3 Protective Clothing and Respiratory Protective Equipment
12.3.2.4 Portable and Laboratory Equipment
12.3.3 Personnel Dosimetry
12.3.4 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)
12.3.4.1 Policy Considerations
12.3.4.2 Design Considerations
12.3.4.3 ALARA Operational Considerations
12.3.5 References

List of Tables

Table 12-1. Design Radiation Zones

Table 12-2. Minimum Shield Thicknesses

Table 12-3. Primary Shield Description

Table 12-4. Parameters Used for Design Basis Accident Analysis of Control Room Direct Dose

Table 12-5. Design Basis Accident Containment Source Strength (gammas/cc-sec) vs (hours after release)

Table 12-6. Design Basis Source Strengths for Fluids.

Table 12-7. Design Basis Source Strengths for Demineralizers

Table 12-8. Design Basis Source Strengths For Filters

Table 12-9. Reactor Coolant System Nitrogen-16 Activity

Table 12-10. Design Source Strengths for Outside Storage Tanks.

Table 12-11. Area Radiation Monitors

Table 12-12. Annual Station Dose Estimates (person-rem/yr)

Table 12-13. Estimated Occupancy Times During Normal Operations and Anticipated Occurrences (man-hrs/wk)

Table 12-14. Estimated Occupancy Times During A Refueling Outage (man-hrs)

Table 12-15. Estimated Occupational Dose by Job Function

Table 12-16. Estimates of Airborne Radioactive Contamination in the Containment

Table 12-17. Estimates of Airborne Radioactive Contamination in the Auxiliary Building Room

Table 12-18. Estimates of Airborne Radioactive Contamination in the Turbine Building

Table 12-19. RCA and RCA Control Points

Table 12-20. Deleted Per 2011 Update

List of Figures

Figure 12-1. Radiation Zones at El. 695 + 0

Figure 12-2. Radiation Zones at El. 716 + 0

Figure 12-3. Radiation Zones at El. 733 + 0 and 739 + 0

Figure 12-4. Radiation Zones at El. 750 + 0

Figure 12-5. Radiation Zones at El. 760 + 6 and 767 + 0

Figure 12-6. Radiation Zones at El. 786 + 0

Figure 12-7. Radiation Zones Longitudinal Section

Figure 12-8. Control Room Isometric

12.0 Radiation Protection

The purpose of the information provided in this chapter is to demonstrate that external and internal radiation exposure to persons at the site boundary and to station personnel from station operation, including anticipated operational occurrences, and from postulated accidents, is kept “as low as reasonably achievable” (ALARA) within applicable limits.

Duke Energy management is firmly committed to the ALARA philosophy for all nuclear operations. This commitment is stated in the ALARA manual. A formal ALARA program has been established in order to convey and enforce Duke management's commitment. The program was established in conformance with the requirements of Regulatory Guide 8.8 to ensure that occupational exposures are kept ALARA within 10CFR 20 limits. ALARA is a major design consideration in accordance with Section C.1 of Regulatory Guide 8.8.

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12.1 Shielding

12.1.1 Design Objectives

The principal design objectives of shielding for the McGuire Nuclear Station are

1. To limit radiation exposure to the public and to station personnel which may result from normal operation including anticipated operational occurrences, and from maintenance during operation or shutdown, thereby meeting the applicable portion of NRC General Design Criterion 61 (10CFR 50, App. A).
2. To limit radiation exposure to the public and to plant personnel which may result from conditions following a postulated Design Basis Accident. Shielding is designed to limit the integrated whole body dose to personnel in the Control Room to less than 5 rem for the duration of the accident, as per Design Criterion 19. The direct external accident dose at the Exclusion Area boundary, combined with doses due to postulated radioactive releases, does not exceed the limits expressed in 10CFR 100.
3. To limit radiation exposure of vital structures, components and systems (Design Criterion 4) so that critical functions are not impaired and maintenance difficulties are reduced.

For the purpose of shielding design, the station is subdivided into radiation zones, depending on the anticipated frequency and duration of occupancy. Each of the zones, which are described below and summarized in Table 12-1, is assigned a limiting dose rate for anticipated operational occurrences. These maximum allowable dose rates are within the limits of 10CFR 20 (where applicable). Section 12.1.6.1 provides that estimates for external exposure are expected to be a factor of 10 to 100 below the design condition values.

Zone I is the designation for areas adjacent to the station site where Duke Energy does not normally exercise authority to control access. 10CFR20.1301 limits dose rates in these areas to 0.1 rem/yr. This is within the limits of 2 mrem in any one hour imposed by 20.1301(b).

Zone II includes areas within the station site where the station staff is expected to work continuously. For conservatism the limiting dose rate is selected as 0.5 mrem/hr. This is also within the limits imposed by 10CFR 20.1502 and 20.1902.

Zone III consists of areas within the station where the staff is expected to work for extended periods. In order to provide margin for possible non-routine exposure, such as might occur during unscheduled maintenance, the dose rate limit for Zone III is selected as 40 percent of the 10CFR20.1201(a)(1); this yields a design basis of 1.0 mrem/hr.

Zone IV is comparable to Zone III, except that staff occupancy is expected to be periodic rather than continuous. 10CFR20.1201(a)(1) limits dose rates in these areas to 2.5 mrem/hr.

Zone V includes infrequently occupied work locations where the dose rate exceeds continuous occupational levels but access need not be physically restricted. The limiting dose rate for this zone is designated as 15 mrem/hr. The precautions given in 10CFR 20.1902 for Radiation Areas are employed where local dose rate levels in Zone V warrant.

Zone VI encompasses all areas of the station where the dose rate exceeds that of Zone V. The precautions given in 10CFR 20.1601 for High Radiation Areas are employed where local dose rate levels in Zone VI warrant.

For clarity, the location of the zones which occur inside the station (Zones II-VI) are illustrated by presentation on Figure 12-1 through Figure 12-7. Radiation level gradients within each zone are expected. Some “hot spots” may exist in the broad zone areas designated on the figures, where design dose rates exceed the zone limit; these can be labelled or protected as necessary. More likely to occur are "cold" areas where radiation levels remain significantly below the specified limit even at design conditions. Under conditions of shutdown or maintenance, local dose rate levels vary significantly depending on the nature of the work being performed.

As stated above, these zones are for layout and shielding design purposes only. Actual radiation zone markings and personnel access control in the station are carried out under the Radiation Protection Program (Section 12.3), taking into account applicable regulations and existing radiation levels.

12.1.2 Design Description

Shielding at McGuire is of the necessary material and geometry to attenuate the source strengths of Section 12.1.3 to below the appropriate dose rate levels specified in Section 12.1.1. Shield thicknesses are given in Table 12-2; also shown are the elevation and column location, source model dimensions, and source term reference for each component.

12.1.2.1 Station Site Area

A detailed site plan for McGuire appears as Figure 2-41. This figure shows the orientation of the major buildings with respect to the boundary of the station site. Also indicated are the access roads and railroad spurs which serve the station and constitute paths for transport of packaged radioactive material.

Residual amounts of radioactivity are expected in the refueling water storage tanks and the reactor makeup water storage tanks. The Radwaste Facility may contain significantly higher levels of radioactivity. Because of this possibility, the Radwaste Facility is shielded by two-foot-thick concrete walls. Source terms are discussed in Section 12.1.3 and dose consequences in Section 12.1.6. Low level radioactive material is also stored in Warehouse 7 which is located outside the protected area on the east side of the plant. The limits for storage of radioactive material in Warehouse 7 are administratively controlled. The exterior walls of Warehouse 7 are two and a half foot thick concrete to limit exposures on the outside surface of the facility to ≤100mrem in a year per 10CFR20.1301, Dose Limits for Individual Members of the Public.

12.1.2.2 Reactor Building

Scale layouts and cross-sections of the Reactor Building are shown in Figure 1-9 through Figure 1-16. Shielding in this structure includes the primary and secondary shields, and others as described below.

The primary shield, described in Table 12-3 is designed to:

1. Attenuate the neutron and gamma fluxes from the reactor core and secondary gamma fluxes from the shield, in order to limit dose rates in the building and protect components and structures from excessive damage or activation.
2. Limit shutdown radiation levels in the building to permit access to Reactor Coolant System components.

The crane wall and the operating deck inside the Containment function as a secondary shield to limit the radiation from the Reactor Coolant System components outside the primary shield. The second shield is a reinforced concrete structure which surrounds the Reactor Coolant System equipment.

Lateral shielding is provided for radioactive portions of the steam generators and pressurizer which extend above the operating deck. Shielding integrity of the building in the area of the main steam and feedwater line penetrations is provided by external concrete doghouses. The regenerative and excess letdown heat exchangers are outside the crane wall and are shielded individually.

The Reactor Building wall, in conjunction with the primary and secondary shields and Containment vessel, is designed to limit radiation at its outside surface (as stated in Section 12.1.1) during full power operation and to protect station personnel from radiation sources inside the Containment following the postulated Design Basis Accident (Section 12.1.2.5).

12.1.2.3 Auxiliary Building

The Auxiliary Building contains a number of systems which handle potentially radioactive materials. Building space is organized into three general categories:

1. Pipe shafts - areas where piping is run between cubicles or from one floor to the next; limited no personnel access is expected (Zone VI).
2. Cubicles - rooms where individual components are located; classification is usually Zone VI during operation and Zone IV during system shutdown and isolation.
3. Corridors - areas where nonradioactive components are located or which provide personnel access to cubicle entrances; classified Zone IV.

Scale layout drawings of the Auxiliary Building are included as Figure 1-2 through Figure 1-7. Principal shielding is given in Table 12-2.

12.1.2.4 Fuel Handling System

Hardware and operation of the Fuel Handling System are discussed in Section 9.1.4. The principal shield for fuel handling operations is water, which has the advantages of good visibility and heat removal capability. Additional shielding is provided by structures along the fuel transfer route, such as concrete sidewalls of the refueling cavity, spent fuel pool, and around the fuel transfer tube which spans the annulus. The refueling cavity also accommodates the activated reactor vessel internals when they are removed from the vessel and stored.

12.1.2.5 Design Basis Accident

An area of the station which requires special consideration for the Design Basis Accident case is the Control Room, where the integrated whole-body dose must not exceed 5 rem for the duration of the accident (Section 15.4). The location of the Control Room is indicated on Figure 1-6, the functional layout is shown on Figure 7-29; and an isometric view is shown on Figure 12-8.

The principal shield for the Control Room during a postulated Design Basis Accident is the Reactor Building wall (3.0 ft concrete). Design Basis Accident Containment source strength as a function of time after release is provided in Table 12-5. Additional shielding is provided by the Reactor Building internal shielding (see Section 12.1.2.2); Control Room floor (12 in. concrete), ceiling (28 in. concrete), and wall (12 in. concrete); and other concrete structures such as the main steam pipe shaft. The parameters used for design basis accident analysis of control room direct dose is provided in Table 12-4. The Control Room direct accident dose analysis is presented in Reference 15.

During the time after a postulated Design Basis Accident, certain equipment might require access for inspection or maintenance. For this reason, shielding is provided around such components as the safety injection pumps, Containment spray pumps, and Containment spray heat exchangers.

Following the TMI accident NRC required that a detailed shielding review be conducted using radioactive sources recommended in NUREG 0578. This shielding review is described in Section 12.1.6.3.

12.1.2.6 Detailed Design Consideration

The majority of shielding throughout the station is concrete, with some use of lead and, to a lesser extent, steel. Concrete shields are composed of poured slabs (2.35 g/cc) or blocks (2.24 g/cc) of ordinary concrete. Lead shields are stacked bricks, laminated slabs, or lead shot encased in steel. Where blocks of either material are used instead of unit wall, care is taken to avoid unshielded paths through joints. In some locations, removable shielding is utilized to facilitate access for maintenance.

Where possible, piping penetrations of shielded equipment cubicles do not face in the direction of corridors or work areas. If necessary, gaps at penetrations are prevented with collars or extra local shielding. For large entranceways to cubicles, labyrinths are arranged to retain shielding integrity. Valve stations are placed outside Zone VI areas, with arrangements provided for remote control (as with reach rods).

In some locations in the station, shield materials also serve structural or other functions. Where these requirements predominate, the shielding exceeds that which is necessary and thereby incorporates additional conservatism.

12.1.3 Source Terms

The radioactive station components and systems to be shielded are modeled as idealized radiation sources for the purposes of radiation shielding analysis. The radiation source strengths attributed to various systems and components in the shielding analysis are conservatively estimated. Most source strengths to be shielded are derived directly or indirectly from design basis reactor coolant activity. FSAR Section 11.1 describes fully the reactor coolant source term. The sources used in shielding design are maximum short-term ones which might result from anticipated operational occurrences. A discussion of source terms as a function of operating history is included in Reference 16. Therefore, shielding source terms for each potentially radioactive system correspond to conditions in that system when one percent of the rated core thermal power is being generated by fuel rods with clad defects. These sources are listed in Table 12-6, Table 12-7 and Table 12-8.

In addition to the corrosion product and design basis fission product terms for reactor coolant, shielding for the Reactor Coolant system takes into account the coolant activation product Nitrogen-16. The concentrations of this isotope in the primary system are given in Table 12-9 and its production is discussed in Section 11.1.1.4.

Radioactive materials are found outside the Reactor Buildings and the Auxiliary Building in the refueling water storage tanks, the reactor makeup water storage tanks, the Low Level Solid waste warehouse, the Low Level Waste Storage Facility, Warehouse 7, the Radwaste Facility, and the Radiography Facility. In addition, other areas may be established for temporary radioactive material control. These areas may consist of storage containers, trailers, or established Radiation Control Zones. Activity in the refueling water storage tanks is modeled as design basis reactor coolant after cold shutdown diluted with refueling cavity water, i.e., demineralized water. The reactor makeup water storage tanks are assumed to contain the activity of the recycle evaporator condensate. The Radwaste Facility contains two tanks. One tank is assumed to contain activity equal to the waste evaporator feed tank. The second tank is assumed to contain activity equal to the floor drain tank. Source strengths are provided in Table 12-10. Dose consequences are discussed in Section 12.1.6. Average expected conditions are a factor of 10 to 100 below these design conditions.

Radioactive wastes leaving the station enroute to offsite disposal facilities are packaged in shipping casks or containers which meet NRC and Department of Transportation regulations for such shipment and thus do not constitute an offsite exposure hazard.

Main run process piping which carries or may carry radioactive material is identified early in the design and clearly marked on system flow diagrams with an appropriate symbol. This symbol indicates that the pipe must be routed in areas designated Radiation Zone IV or higher. All principal piping is detailed on engineering drawings with proper regard for shielding requirements. The small lines such as vents and drains which may be field-routed under the criteria of Section 3.9.2 tend to be closely associated with the principal piping and primarily run in the same cubicles or pipe chases. The Radiation Protection Manager reviewed field routing before initial plant operation to assure all station areas, subject to contain field run piping which handles radioactive fluids, are within allowable limits.

12.1.4 Area Monitoring

The Area Radiation Monitoring System consists of gamma sensitive detectors, signal conditioning and readout instrumentation, radiation level alarm sensing logic, audible and visual alarm devices and recording equipment.

The Area Radiation Monitoring System indicates radiation levels that may exist in various locations throughout the station where personnel are most likely to be exposed. Audible alarms are sounded at detector locations when the radiation level exceeds the alarm setpoint, with exception of the steam line monitors. At the same time audible and visual annunciation are initiated in the Control Room, with exception of the Technical Support Center (TSC) monitor. Indications and alarms from the area radiation monitors in conjunction with information from process monitors and operating procedures are used to assure that radiation exposure of personnel within the station does not exceed 10CFR 20 limits. The dynamic range of the area radiation monitoring equipment is adequate to indicate when personnel access is not permitted to a given area during abnormal or accident conditions. No control functions are initiated from the Area Radiation Monitoring System.

The majority of the Area Radiation Monitoring System is powered from the 240/120 VAC Auxiliary Control Power System (Section 8.3.2). The technical support center monitor is powered from the 120 VAC Technical Support Center Power System.

Locations of area monitoring detectors are given in Table 12-11. These locations do not necessarily represent points where comprehensive calculations of expected dose have been made. The detectors are situated to meet the following specific monitoring requirements:

1. Auxiliary Building corridors are areas where periodic occupancy of personnel is intended; these corridors pass adjacent to, above or below potentially radioactive rooms. The corridor monitors assure that, in the event of faulted conditions which might cause radiation fields above design levels, persons are not inadvertently sent into such areas and persons already in such areas are warned to exit, until the condition is corrected.

2. The sample rooms are monitored because personnel performing sampling operations are exposed to a widely varying array of potentially radioactive fluids. This same reasoning

applies to the laboratory, waste drumming room and for solid wastes, to the waste shipping area.

3. The Control Room is monitored because it has the highest occupancy of any location in the plant.
4. From time to time, contaminated or activated equipment requires servicing in the hot machine shop. The monitor located there assures that hot components are properly handled and stored.
5. There is limited regular access in the Reactor Building incore instrumentation rooms. An area monitor is provided to indicate that direct radiation levels are within allowable limits for such access.
6. Monitors are situated on the refueling bridges in both Containments and Spent Fuel Buildings. In the unlikely event of excessive radiation levels due to abnormal contamination or insufficient water shielding (accidental) of a fuel assembly, the alarm serves to alert personnel.

Exact location of area radiation monitors around the plant is a matter of judgment, taking into account the potential for both significant radiation levels and significant occupancy. Areas with high occupancy and little or no radiation (such as the Turbine Building) and areas with high potential for radiation but low occupancy (such as pipe chases) do not meet these criteria and, therefore, do not justify monitoring.

Electrical signals proportional to the radiation level at the detector location are displayed in the Control Room, with exception of the TSC monitor. When this signal exceeds a setpoint value, an audible and visual alarm are actuated in the control room (except TSC monitor) and an audible alarm (except steam line monitor) is sounded at the detector location. The TSC minotor readout module is located in the TSC and provides indication of radiation levels and a visual alarm within the TSC. Due to the variations in radiation levels at detector locations, the alarm setpoint can be adjusted throughout the range of each instrument. However, for normal operating conditions an alarm setpoint one-half decade above background is adequate.

Most of the detectors used in the Area Radiation Monitoring system are of Geiger-Mueller design. The span of the monitors is 5 decades giving a dynamic range of 10^{-1} mR/hr to 10^{+4} mR/hr.

Each low-level area monitoring channel is calibrated by exposing its detector to a calibrated source and verifying proper meter response. Calibration is performed at two points in the range of the channel that are separated by more than one decade.

A channel response check of each Geiger-Mueller channel of the Area Radiation Monitoring System can be performed using the checksource internal to each detector. The checksource can be remotely actuated and an upscale reading on the channel observed.

12.1.5 Operating Procedures

Operating procedures and administrative controls are provided to assure that external exposures are maintained within the limits of 10CFR 20 and as low as reasonably achievable during station operation and maintenance. These procedures comply with the intent of Regulatory Guide 8.8. The Radiation Protection Policy Manual, which is based on applicable regulations and regulatory guides, describes the radiation protection program for the station and contains procedures for implementation of this program. The information and procedures in the Radiation Protection Policy Manual, from which all station operating procedures related to radiological safety are derived, assure that external exposures will be kept as low as reasonably achievable during station operation and maintenance. These station operating procedures, which are either developed or reviewed by Radiation Protection, utilize specific reduction techniques outlined in Regulatory Guide 8.8. Procedures relating to radiological safety are also described in Chapter 16. Access to, and duration of occupancy in radiation control areas of the station are carefully controlled and a radiation work permit system is utilized to limit external radiation exposure during operation and maintenance work. Implementation of the Radiation Protection program (Section 12.3) and effective use of the area radioactivity monitoring system assures proper surveillance and control of personnel external exposure. In addition, operation and maintenance work is planned, reviewed and scheduled to minimize exposures. Provision is made for special temporary shielding as needed. Preliminary and periodic re-views of facility design, procedures, and dose records are conducted by Radiation Protection supervision and management, to assure that exposures are as low reason-ably achievable - see Section 12.3.4.

Radiation Protection management personnel review total dose, and doses relating to specific work groups and job functions, to determine locations where most exposures are being received, what work groups are receiving the highest exposures, and how personnel exposures can be reduced. Formal audits on the radiation control program are conducted periodically, to assure that the ALARA policy is being implemented.

Procedures and practices for achieving as-low-as-reasonably achievable exposures are reviewed, and radiation workers' job-performance is also reviewed, to assure that the workers are responsible, conscientious, and qualified to perform their work efficiently and safely. Postoperational debriefings are conducted to improve job performance. Permanent and/or removable shielding is designed to minimize radiation exposure. Mock-up training is conducted for high-exposure jobs. The development of special tools and remote-handling equipment is encouraged, to reduce external exposures (new and better ways to perform all radiological work with less exposure are diligently pursued).

The basic external radiation exposure control program described in the Radiation Protection Policy Manual has been used quite successfully at the Carolinas Virginia Tube Reactor (CVTR), and at the Oconee Nuclear Station. The program also utilizes procedures developed and used quite successfully by the nuclear Navy.

12.1.6 Estimates Of Exposure

12.1.6.1 Normal Operation, Exposures

The dose rate limits given in Section 12.1.1 represent realistic peak values for dose rates at corresponding locations inside and adjacent to the plant. The actual average dose rates in each radiation zone during normal operation depend on many parameters but are expected to be a factor of 10 to 100 below the respective design condition values. This is because the design values are based on conservative source strengths and pessimistic assumptions of operating modes.

An estimate of the annual in-station direct dose for design operation of both units is given in Table 12-12. Expected maximum dose rates have been assumed for operations in the separate station radiation zones so the estimated exposures are higher than those expected to be experienced. The exposure to outside contractors is estimated from industry experience.

Station personnel classifications as given in the station organizational chart are used to define the man-hours in the separate functional areas. Estimates of reasonable stay times in each zone were made based on experience gained at the Oconee Nuclear Station taking into account the design features and shielding of the McGuire Station which along with the station operating procedures provide assurance that external exposures are kept as low as reasonably achievable within applicable limits. Table 12-13 and Table 12-14 provide occupancy levels throughout the plant during normal operation and refueling outages. Table 12-15 provides estimated occupational dose by job function.

Direct external radiation exposure due to the operation of the McGuire station is expected to be well within applicable regulations and as low as reasonably achievable for the operating staff and maintenance personnel, and negligible for the population living in the vicinity of the station. The design condition direct dose rate at the Exclusion Area Boundary is conservatively estimated from stored activity outside the station buildings. An evaluation of the sources inside the station buildings according to the methods outlined in Section 12.1.6.2 has shown that their contribution to the Exclusion Area Boundary direct dose rate is far below that from the outside storage tanks. Levels of radioactivity stored outside of the Reactor Building and Auxiliary Building are discussed in Section 12.1.3. Using the computer code KAP VI and methods described in Section 12.1.6.2 direct dose rates at the Exclusion Area Boundary were calculated. Assuming all tanks contain 100% capacity of their design basis fluids, the Exclusion Area Boundary is conservatively estimated to be less than 10^{-5} mrem/hr.

12.1.6.2 Shielding Analysis

a. Initial Licensing and Analysis

Calculations to assess the adequacy of shielding were based on the source strengths of Section 12.1.3, and performed according to the methods described in this subdivision. The only major neutron source in the plant is the reactor core at power (the low neutron fluxes from irradiated fuel elements, due to alpha-neutron reactions and spontaneous fissions, are easily attenuated by the surrounding cooling water). Those portions of the primary shield out to and including the reactor vessel wall are designed and supplied by the NSSS vendor. The bulk arrangement of the reactor cavity concrete is similar to that used in other Westinghouse four-loop plants. As a check on the entire primary shield, the ANISN and SABINE codes have been utilized (see descriptions below).

Sources or potential sources of gamma radiation, as indicated in Section 12.1.3, are widely distributed throughout the Auxiliary and Reactor Buildings. These sources were analyzed with the code SHIELD or, where necessary, KAP-VI (See descriptions below). Analysis proceeds in the following typical sequence:

1. Determine the concentration of each principal nuclide in the source medium.
2. Adjust the concentration by accounting for accumulation, dilution, decay removal, etc.
3. Convert the resulting concentrations to a gamma source strength.
4. Select an idealized model or combination of models to represent the physical shape of the source container and all shields present.
5. Assemble the necessary data on the attenuation properties of the source and shield materials.
6. Perform the calculation for the desired dose point and tabulate the results for comparison with design objective dose rates.

Steps 1-3 were handled with the code N237BURP using the data of Section 11.1 (see code description below). Step 4 straightforward; tanks, demineralizers, filters, and pipes are represented by right circular cylinders; centrifugal pumps are spheres; point sources are assumed if source-receiver distance warrants. Step 5 is internal to the codes, except material densities which are input. Step 6 determines the adequacy of the shield, taking into account other sources which may contribute to dose at the same receiver point. Note that the codes compute dose rates and not shield thicknesses. This makes the design an iterative process, but is not a handicap because of the coarse, conservative shield sizes usually specified.

Care is taken to select representative receiver points and worst case source-to receiver paths. Cubicle ceilings and floors typically are as thick as their enclosing walls, so that scattered radiation arriving at points outside the cubicle by indirect paths is not a problem. Since component cubicles and the Containments are shielded overhead, air-scattered radiation (skyshine) from the plant is negligible.

Computer Code ANISN

ANISN performs shielding calculations by a discrete ordinates solution of the Boltzmann equation in one dimension. A CDC-6600 Fortran IV code was the version used. Through the use of transport theory with anisotropic scattering, ANISN is well suited to deep penetration problems. A 40-group coupled cross section set is utilized to account for both neutron attenuation and secondary gamma radiation. Calculations are made in cylindrical geometry. Detailed descriptions of the code are found in References 2 and 3.

Computer Code SABINE

SABINE, used on an IBM-370 solves neutron and gamma-ray shielding problems with removal-diffusion methods. The neutron/gamma production is a specified fission distribution in the source region. The code calculates neutron attenuation out through the shields using nineteen removal energy groups which in turn feed twenty-six groups for the diffusion calculation. Secondary gamma production in each shield region is output as a polynomial curve fit and gamma fluxes can be calculated. A description is given in Reference 4.

Computer Code KAP-VI

KAP-VI is a Fortran IV code used on the IBM-370 computer. It employs the point kernel technique to determine the dose rate from complex sources whose geometries can be described by second-order surface equations. An exponential attenuation function with build-up is employed for gammas; neutron attenuation functions are also available. Detailed descriptions of the code and its geometry routines are found in References 5 and 6.

Computer Code SHIELD

SHIELD is written in Fortran IV for the IBM-370 computer; because it is a Duke code it is described in detail here. Fluxes at receiver points are computed with integrals over simple source geometries (point, line, slab, sphere, cylinder) using the formulas of Reference 7. The gamma spectrum is subdivided into six energy groups. Input includes gamma source strength in each group, average energy in each group, source and shield geometry, and material densities. Whenever the spectrum of average energies changes, energy-dependent parameters are recalculated in the subroutine SHLDATA. It contains data as a function of energy on tissue flux-to-dose conversion factors (from Reference 8), mass attenuation coefficients (from Reference 9) for common source/shield materials (water, air, iron, lead, concrete, uranium dioxide), and Taylor-form dose buildup factor coefficients (from Reference 10, values below 0.5 MeV extrapolated). For combined shields the code automatically bases the build-up factor on the material exhibiting the greatest optical thickness in the lowest energy group. Certain calculations exceed the range of available data in SHIELD for cylindrical sources-in these instances warning messages are included with output. For this reason, dose rates at points near large tanks are calculated or checked with KAP-VI.

Computer Code N237BURP

Calculation of gamma radiation source strengths for given isotopic concentrations is handled by a Duke code called MIDIBURP. Written in Fortran IV for the IBM-370 computer, the code follows the processing of fluids containing fission and corrosion products through desired steps of dilution, purification, and decay. For nonzero input times N237BURP accounts for decay of each isotope including decay of one significant isotope to another; for the latter cast the Bateman equations, for two or three members chains with branching fractions, are solved. Individual removal fractions are input to account for passage across a demineralizer having isotopic microcuries per milliliter by considering particle yields and losses to conversion electrons. The code organizes the nearly 300 individual gamma emissions of these isotopes into six discrete energy groups whose boundaries remain fixed by whose average within-group energies are computed for each given spectrum of isotopes. This allows reasonably precise selection of energy-dependent shield material properties for attenuation calculations.

Computer Code G36ED

G36ED evaluates the uncollided flux at specified scatter points and multiplies it by the product of the differential cross-section for scattering toward the detector point and the number of electrons in the element volume associated with the scatter point. Attenuation through materials between the scatter point and the detector point is characteristic of the degraded energy as determined by the incident energy, the scattering angle, and the angle-energy relationship. Detector response is recorded as a function of incident and scattered energy both with and without application of buildup on the path between the scatter point and the detector point. A more detailed description is presented in Reference 13.

b. Current Shielding Analysis

MCNP, SCALE, and QAD-CGGP-A are the codes currently used for gamma and neutron shield analysis. They are all resident on desktop PC operating platforms.

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. MCNP for the PC operating platform was obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory.

The SCALE code package was developed for the Nuclear Regulatory Commission to satisfy a need flor a standardized method of analysis for the evaluation of nuclear fuel facility and package designs. In its present form, the code package has the capability to perform criticality, shielding, and heat transfer analyses using well-established functional modules tailored to the SCALE system. SCALE for the PC operating platform was obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory.

QAD-CGGP-A is a point-kernel code for calculating fast-neutron and gamma-ray penetration through various shield configurations defined by combinatorial geometry specifications. AECL developed this release by modifying the CC-493/QAD-CGGP package. The major improvements to the new version include the incorporation of a cubic spline interpolation scheme for the gamma attenuation coefficients, an added capability for source translation and rotation, correction of a potential error in evaluation of buildup factors at very deep penetration, and the adoption of complete free-format input reading routines from KENO-IV. QAD-CGGP-A for the PC operating platform was obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory.

12.1.6.3 Plant Shielding Review Using NUREG-0578 Sources

(Reference 13)

The design basis accident radioactivity sources are defined by TID-14844 and Regulatory Guide 1.4. The shielding analysis for McGuire Nuclear Station was performed under the assumption that in a LOCA 100% of the noble gases and 25% of the core iodine will become airborne and 50% of the core iodine and 1% of the remaining core fission products will be released in the liquid. These values are consistent with Regulatory Guide 1.7. Following the TMI accident, NRC staff issued a directive to conduct a shielding review using the following source terms:

Airborne:	100%	core inventory of noble gases
	25%	core inventory of Iodines
Liquid:	100%	core inventory of noble gases
	50%	core inventory of Iodines
	1%	core inventory of remaining fission products

The radioactivity released to the liquid is assumed to be homogenously distributed in the water volume consisting of:

1. Reactor Coolant System
2. Core flood tanks (Cold Leg Accumulators)
3. Water Injected by Safety Injection System, and
4. Water from the ice condenser melt.

These source terms defined by NUREG 0578 are extremely conservative as they include 100% of the noble gas inventory in the liquid.

Scope of Shielding Review

The purpose of this shielding review was to verify that radiation exposure to the personnel met the requirements of GDC 19 (Appendix A to 10CFR 50) and 10CFR 20 even with more conservative source terms recommended by NUREG 0578. In addition the safety equipment and systems are required to function in the environmental conditions to which they may be subjected to during the life of the plant (GDC4, Appendix A, 10CFR 50). The suitability of the plant equipment for their respective environment was determined by comparison of the calculated environmental conditions with the equipment design or qualification.

Summary of Description of Shielding Review

Plant systems or portions of systems which might contain significant levels of radioactivity as a result of a Design Basis Accident were selected for the station accessibility review. Included in the review were:

1. those portions of the Residual Heat Removal, Reactor Building Spray, Safety Injection, and Chemical and Volume Control Systems which could be aligned for recirculation of water from the containment sump to the Reactor Coolant System,
2. those portions of the Liquid Waste Recycle System which would collect and store leakage from the systems mentioned in item No. 1,
3. those portions of the Nuclear Sampling System which would be used in determining radiation levels inside containment or those systems mentioned above, and

4. those portions of the Chemical and Volume Control System which supply seal water to the Reactor Coolant Pump seals, and
5. the Waste Gas System and those portions of the Chemical and Volume Control System which could be used to degas the primary coolant.

To aid in identifying potential personnel access problems, the station was divided into post-LOCA radiation zones. Included in the radiation zones were all areas necessary for personnel access in controlling and mitigating a possible accident. Two types of area access have been designated: 1) continual occupancy and 2) infrequent occupancy.

Areas designated as requiring continual access are: 1) Control Room, 2) Onsite Technical Support Center (TSC), 3) Onsite Operational Support Center (OSC), 4) Operator Aided Computer (OAC), and 5) Personnel Access Portal (PAP). The Control Room, TSC, OSC, and OAC are located in the Control Complex. The PAP is located in the Administration Building and serves as the main personnel access point to the station proper. The Control Room serves as the initial onsite center of emergency control and is designed to evaluate, control, and respond to various accident conditions. A detailed description of Control Room design and functions is presented in Section 7.8 of the McGuire FSAR.

Areas requiring infrequent access are generally in Auxiliary Building corridors. Two exceptions to this are: 1) Containment Hydrogen Recombiner controls and 2) Emergency Diesel Generators. Redundant hydrogen recombiners are located in the Upper Containment. Power control panels for these recombiners are located in the Electrical Penetration Rooms. Redundant diesel generators are located in a section of the Auxiliary Building designated as the Diesel Generator Area. Various control functions associated with the diesel generators and supporting systems are located in the Diesel Generator Area. Typical functions centered in Auxiliary Building corridors are: 1) station radioactive waste control panels, 2) motor control centers, and 3) instrumentation panels for various station systems.

The major emphasis of the McGuire Nuclear Station plant shielding review was to assure that station personnel would be able to carry out their emergency procedures. The review featured the consistent use of the defined source terms in conjunction with the KAP-VI computer code.

Conclusions

The plant shielding review concluded that the plant is adequately shielded to permit personnel to carry out emergency procedures during a LOCA. The areas designed for continuous occupancy have radiation dose rate of less than 15 mrem/hr and the areas requiring infrequent access are designed to limit the radiation exposure to 5 Rem whole body or equivalent to any organ for the course of a LOCA.

Two areas were identified where the dose rates exceeded above mentioned limits. These areas are:

1. Unit 1 Sample Room
2. Floor Drain Tank Room

It is impractical to install additional shielding in these areas. An alternate method for collecting samples through tell-tale drain lines has been established to obviate the need to enter the Unit 1 sample room during a severe accident. The tell-tale valves are accessible during accident mitigation. The access to floor drain tank room is necessitated by the need to operate manual valves to isolate the floor drain tank or direct the contents of the tank elsewhere for storage. After an accident the contents of the floor drain tank can be highly radioactive. Reach rods have been installed to manipulate the manual valves without incurring high dose rates during a LOCA.

The environmental qualification review of Class 1E equipment is complete. This review includes the effects of NUREG-0588 radiation sources.

Environmental Qualification of Class 1E equipment is discussed in Chapter 3.

12.1.7 References

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12.2 Ventilation

The ventilation systems dealing with radiation protection are the Control Area Ventilation System, Auxiliary Building Ventilation System (including radwaste areas) and the Containment purge and Ventilation System. Design objectives and system descriptions are found in Section 9.4.

12.2.1 Design Objectives

In addition to the design objectives presented in Section 9.4, these ventilation systems assure that onsite inhalation exposures are kept as low as reasonably achievable and within the limits of 10CFR 20 at all times. The design philosophy for the Auxiliary Building Ventilation System best illustrates this. The system is designed to keep the Auxiliary Building at a negative pressure with respect to the outside, and also to keep the potentially contaminated areas at a negative by supplying ventilation air to the non-contaminated areas (corridors, etc.) and exhausting from the potentially contaminated areas (rooms, cubicles, etc.) through a filter train and out the unit vent. Slightly more air is exhausted than is supplied to give the negative relative pressure. This minimizes contamination in occupied areas.

Concentrations in areas of frequent personnel occupancy, such as the Control Room, are conservatively evaluated according to the limits of 10CFR 20.1201.

12.2.2 Design Description

These ventilation systems are fully described in Section 9.4 and in Reference 1. Additional information on building volumes used for estimating airborne exposure are found in tables 12-16, 12-17, and 15-33.

12.2.3 Source Terms

The sources of activity in the areas served by these ventilation systems are discussed in Sections 11.2.2 and 11.3.6.

12.2.4 Airborne Radioactivity Monitoring

Since the leakage source term is also the gaseous effluent source term, process monitoring, discussed in Section 11.4, indicates problem areas.

12.2.5 Operating Procedures

Operating procedures and administrative controls are provided to assure that internal exposures are maintained within the limits of 10CFR 20.1201. These procedures comply with the intent of Regulatory Guide 8.8. The Radiation Protection Policy Manual, which is based on applicable regulations and regulatory guides, describes the radiation protection program for the station and contains operating procedures that assure exposures (Total Effective Dose Equivalent, TEDE) will be kept as low as reasonably achievable during station operation and maintenance. Access to, and duration of occupancy in airborne radioactivity areas of the station are carefully controlled and a radiation work permit system is utilized to limit internal radiation exposures during operation and maintenance work. Implementation of the Radiation Protection program (Section 12.3) and effective airborne radioactivity sampling assures proper surveillance and control of personnel internal exposure. The basic internal radiation exposure control program described in the Radiation Protection Policy Manual has been successfully used to maintain inhalation doses as low as reasonably achievable at the Carolinas Virginia Tube Reactor (CVTR) and at the Oconee Nuclear Station. In addition operation and maintenance work are planned, reviewed and scheduled so as to minimize exposures. Provision is made for special temporary local exhaust ventilation as needed, and for the proper use of respiratory protective equipment.

In addition, it requires

1. Thoroughly-planned and carefully-performed work procedures, to minimize contamination;
2. The use of a Radiation Control Zone in the vicinity of each work area, and the use of containment within the work area, to prevent the spread of contamination;
3. Radiation Protection measurement and control of the airborne exposure of personnel; and
4. Whole-body and thyroid counting on-site, with frequency of measurement depending on amount of exposure and correlation of results with air-monitoring data to determine the effectiveness of, and the need for any additional control measures.

12.2.6 Estimates of Inhalation Dose

Inhalation doses are controlled to a great extent by operating procedures. The examples in the following paragraphs illustrate how concentration limits in 10CFR 20 are met by the ventilation systems in the presence of a continuous source. If significantly greater leakage or spills occur, the failed component is isolated and the air cleaned up prior to entry.

An evaluation of in-station population dose from airborne activity is made with the following assumptions:

1. The concentrations shown in Table 12-17 are typical of maintenance situations in component rooms in the Auxiliary Building.
2. The man-hours of exposure per week for component rooms in the Auxiliary Building are shown in Table 12-13.
3. The concentrations shown in Table 12-18 are typical of the Turbine Building.
4. The man-hours of exposure per week for the Turbine Building are shown in Table 12-14.
5. All other areas of the station have no airborne source term and all other man-hours are in these clean areas.

The following discussions give examples of situations which demonstrate the ability to control exposures. The annual in-station population dose resulting from the above assumptions is estimated to be no greater than 10 man-rem. Estimates of inhalation doses are based on experience gained at the Oconee Nuclear Station taking into account the design features and ventilation systems at McGuire Nuclear Station.

12.2.6.1 Inhalation Doses in the Containment

The Containment is a radiation area; therefore, the limits presented in 10CFR 20.1202 are applied. Since the Reactor Coolant System is located in the lower compartment of the Containment, the only source of activity in the upper compartment is from the transfer of air from the lower compartment. This transfer results in upper compartment concentrations much below those in the lower containment. In the event of a larger spill, activity can be eliminated by purge after recirculation through filters located in the lower compartment.

The incore instrumentation room has an independent purge and supply system and is not communicated with the rest of the lower compartment. Since there is rarely a source of activity in this area, concentrations are normally negligible in this room.

When it becomes necessary to enter the lower compartment during operation, the allowable period of occupancy is limited to much less than the forty hours upon which the limits in Appendix B, Table 1 of 10CFR 20 are based. If greater occupancy time is necessary, the reactor may be shut down. The concentrations and fractions of limit shown in Table 12-16 are calculated based on the assumptions at the bottom of the table and are provided as an example of continuous leak situation. The fractions of limit take into account occupancy time and are proportional to annual dose.

12.2.6.2 Inhalation Doses in the Auxiliary Building

Since they have no source of leakage and are maintained at a positive pressure relative to radiation areas to prevent in-leakage, the Control Room, the Equipment Room, and certain corridors are not considered as contributors to airborne exposures. All other spaces are identified as radiation areas and the limits of 10CFR 20.1202 are applied.

The radiation areas in the Auxiliary Building typically have possible sources of activity but less frequent occupancy. In the case of small continuous leaks, concentrations are low enough to permit entry for maintenance or inspection. The results of an analysis of a typical situation together with the assumptions used are presented in Table 12-17. The fractions of limit shown take into account occupancy time and are proportional to annual dose. In the case of much larger leaks and spills or the requirement for greater occupancy time, the faulty component can be isolated and the leak stopped. The ventilation system would then reduce the activity level to less than that shown in Table 12-17.

12.2.6.3 Inhalation Doses in the Turbine Building

The Turbine Building is conservatively evaluated according to the limits of the old 10CFR 20.106. These limits are consistent with the 10 times the EC limits specified in the new 10CFR 20, Appendix B, Table 2, Column 2 and the objectives of 10CFR 20.1302(b). Airborne contamination in the Turbine Building may result if there are fuel cladding defects, primary to secondary leakage and secondary system leakage to the Turbine Building atmosphere.

The results of an analysis of this situation are shown in Table 12-18 together with the assumptions used. The fractions of limit are proportional to continuous occupancy dose.

12.2.7 References

1. MCS-1577.VA-06-0001 Design Basis Specification for the VA System.

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12.3 Radiation Protection Program

12.3.1 Program Objectives

The three basic objectives of the Radiation Protection Program at McGuire are to:

1. Protect station personnel
2. Protect the public
3. Protect the station

Protection of personnel, means surveillance and control over the internal and external radiation exposure of personnel and maintaining the exposure of all personnel within permissible limits, and as low as reasonably achievable (ALARA) in compliance with applicable regulations and license conditions.

Protection of the public, means surveillance and control over all station conditions and operations that may affect the health and safety of the public. It includes such activities as radioactive gaseous, liquid and solid waste disposal and the shipment of radioactive materials. It also involves conducting an environmental radioactivity monitoring program and maintaining an effective emergency plan.

Protection of the station, means the continuous determination and evaluation of the radiological status of the station for operational safety and radiation exposure control purposes. This work is done in order to warn of possible detrimental changes and exposure hazards, to determine changes or improvement needed, and to note trends for planning future maintenance work.

The program organization is as follows.

The Station Manager is responsible for the protection of all persons against radiation and for compliance with NRC regulations and license conditions. This responsibility is in turn shared by all supervisors. Furthermore, all personnel are required to work safely and to follow the regulations, rules, and procedures that have been established for their protection.

The Duke Energy Technical System Manager - Radiation Protection establishes the Radiation Protection Program for McGuire that is designed to assure compliance with applicable regulations, licenses and regulatory guides, and provides technical guidance for conducting this program, audits the effectiveness and the results of the program and modifies it as required based on experience and regulatory changes. The Manager also provides technical assistance to the Senior Vice President, Nuclear Generation, who has management authority to implement the “as low as reasonably achievable” (ALARA) occupational exposure policy, to which Duke Energy is committed.

The Station Radiation Protection Manager is responsible for conducting the Radiation Protection Program that has been established for the station. The Station Radiation Protection Manager has the duty and the authority to measure and control the radiation exposure of personnel to a level that is as low as reasonably achievable within regulatory exposure limits; to continuously evaluate and review the radiological status of the station; to make recommendations for control or elimination of radiation hazards; to train personnel in radiation safety; to assist all personnel in carrying out their radiation safety responsibilities; and to protect the health and safety of the public both on-site and in the surrounding area.

In order to achieve the goals of the Radiation Protection Program and fulfill these responsibilities for radiation safety, radiation monitoring, survey and personnel exposure control work are performed on a continuous basis for all station operations and maintenance. This requires that there are several Radiation Protection Technicians on each operating shift. The extent of this surveillance is outlined below.

The Radiation Protection section performs the major portion of the Radiation Protection work for the station. Personnel in the Radiation Protection section provide radiological control coverage on all day shifts, four days a week, during periods of routine operation; and are supplemented on the back shifts for major maintenance, shutdown, and refueling work. The Radiation Protection Section is organized into two major units, headed by two Radiation Protection General Supervisors. These units are: (1) Shift, and (2) Field Operations.

For the purpose of defining and assigning work to be performed by the operating shifts and the Radiation Protection Section, the routine station radiation surveillance work can be described as consisting of radiation monitoring, radiation survey, radiation exposure control, and radioactive waste disposal activities.

The Radiation Protection Technicians on each shift perform radiation monitoring and exposure control work for the routine shift operations and maintenance, particularly on the back shifts (other than day shifts). An activities checklist designates other routine work to be performed.

The Radiation Protection Section also performs essentially all of the work necessary to calibrate and maintain the Counting Room instruments, the portable radiation monitoring instruments, and the fixed personnel contamination minotors.

Duties concerning radioactive liquid, gaseous and solid waste disposal are performed by the Radiation Protection and Chemistry sections and by the operating shifts under Radiation Protection direction. The detailed analyses and records required to characterize the nature of these releases, both qualitatively and quantitatively, are under the control of Radiation Protection. In addition, solid waste disposal and shipments of radioactive materials are under the control of Radiation Protection.

Training and qualification of personnel in Radiation Protection is the responsibility of the Station Radiation Protection Manager and is performed under his direction. All administrative aspects of training, such as scheduling and documentation are handled by McGuire Nuclear Training. McGuire Nuclear Training also administers the general employee standardized Radiation Protection Training.

The Radiation Protection Section is also responsible for the Offsite Radiological Monitoring Program for the station.

12.3.2 Facilities and Equipment

12.3.2.1 Radiation Protection and Chemistry Facilities

The Radiation Protection and Chemistry facilities are centrally located in the Auxiliary Building for efficiency of operation. Laboratory facilities consist of a conventional chemistry laboratory, a radio-chemistry laboratory, a shielded counting room, a radiological sample preparation laboratory, a radiation survey instrument calibration room, and a shielded radioactive source storage room. These facilities are equipped for conducting the radiation protection and chemistry programs for the station, for detecting, analyzing and measuring all types of radiation and for evaluating any radiological problem that may reasonably be expected. Equipment for performing measurements for internal personnel dosimetry purposes, and for radio-bioassay is also included and are located in the Administrative Building. In addition, a Radiation Protection operations office is provided in this location.

12.3.2.2 Additional Facilities and Access Provisions

Change room facilities are provided where personnel obtain clean protective clothing and other equipment required for station work. The change rooms service the reactor buildings, the Auxiliary Building, the spent fuel pools, and the Hot Machine Shop. These facilities are divided into clean and contaminated sections. The contaminated section of the change rooms is used for the removal and handling of contaminated protective clothing after use. Provisions for change and personnel decontamination are also available in Room 1004, 760' elevation in the Auxiliary Building and in the unit two Equipment Staging Building. Showers, sinks, and necessary radiation monitoring equipment are provided in all of the change rooms to aid in the decontamination of personnel.

Equipment decontamination facilities are also provided at the station for large and small items of station equipment, components and tools. In addition, a cask decontamination area is provided adjacent to each spent fuel pool which may also be used for the decon of plant equipment. A decontamination laundry and respiratory decon facility are also provided.

Decontamination of work areas throughout the station is facilitated by the provision of janitor's sinks on each floor level in the Auxiliary Building and in the reactor containments.

Drains from all these facilities go to appropriate radioactive liquid waste drain tanks. Written procedures govern the proper use of protective clothing, the change rooms, and the decontamination facilities.

In order to protect personnel from radiation and radioactive materials, the Restricted Area of the station is divided into areas of increasingly controlled access depending on radiation levels. Protection of personnel from access to radiation areas and high radiation areas that exist temporarily or permanently as a result of station operations and maintenance is by means of appropriate radiation warning signs, barricades, locked doors, audible and visual indicators and alarms, as required by 10CFR 20 and by the Technical Specifications. Administrative controls are also used in conjunction with the above and keys are issued to authorized station personnel for access to limited access areas within the Radiation Control Area under certain conditions.

Table 12-19 gives the boundaries of the Radiation Control Area (RCA) for each elevation in terms of the Radiation Zones given on Figure 12-1 through Figure 12-6. The primary personnel entrance/exit point to/from the RCA in the Auxiliary Building is located at QQ-56, elevation 774'. In addition, a secondary access point is located on the 784' elevation for some pre-determined routine entries (e.g., operator rounds). Contamination control checkpoints that are equipped with appropriate monitoring instrumentation are located at the primary access point. For Operator Time Critical Actions, entrances from the rear of the control room through the Unit 1 and Unit 2 Electrical Penetration rooms into the stairwells on the 767' elevation at CC-53 (Unit 1) and CC-59 (Unit 2) can be used during an emergency. Emergency electronic dosimeters with predetermined dose and dose rate setpoints are staged in the Control Room to use for Operator Time Critical Action situations. Incidental RCA access points are located at various points on the 733', 750', 760', and 774' elevations of the Auxiliary Building that can be used for equipment delivery and removal. Access through incidental entry points is continuously controlled by Radiation Protection personnel. All other personnel-access points into the RCA in the Auxiliary Building are protected by restricted-in/free-out doors, and are for emergency exits only. Stairs located on the north, south, east, and west sides of the Auxiliary Building are provided for personnel access from one elevation to another. Contamination control checkpoints are appropriately located at the necessary locations. Additional checkpoints are strategically placed throughout the RCA, to prevent the spread of high levels of contamination within this area.

The Radiation Work Permit system is also utilized to control access to high radiation areas.

Before leaving the Radiation Control Area, personnel are required to monitor themselves (with appropriate instruments, positioned near each exit door), to make sure that they are free of significant contamination.

Authorized personnel enter the Radiation Control Area through an RP access control point.

Personnel who are required to utilize protective clothing obtain these items in the Change Rooms. They first enter the Change Room on the “clean” side, don the required protective clothing, and then proceed to the job location. After completing work, they remove outer contaminated protective clothing, in the Radiation Control Zone set up at the work area. They then proceed to the “contaminated” side of the Change Room, where they monitor themselves, remove any remaining protective clothing items; and the proceed to the “clean” side, where they put on their personal clothing and leave.

All persons entering the Radiation Control Area of the station must wear the personnel monitoring equipment (TLD, electronic dosimeters, etc.) prescribed by the Station Radiation Protection Manager in accordance with NRC Regulations and must comply with applicable Radiation Work Permits.

All work on systems or in locations where radioactive contamination or external radiation is present requires a Radiation Work Permit prepared under the direction of the Station Radiation Protection Manager before work can begin. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit. The Radiation Work Permit lists the precautions to be taken during the performance of the work. The permit is issued for personnel use and a working copy is maintained by the Radiation Protection Section.

All persons working under a permit are required to read and understand the instructions on the permit. All entries into the Radiation Control Area/Radiation Control Zones require dose monitoring. This is normally done by entries into an electronic dose monitoring computer system. The data collected by the system serves, in part, as a personal monitoring record for the individuals involved.

The counting room is shielded on all sides to facilitate low level counting work. The radioactive source storage room also has shielded walls. In addition, extensive shielding of components has been utilized in the Auxiliary Building for the protection of personnel, both for routine operation and for maintenance.

12.3.2.3 Protective Clothing and Respiratory Protective Equipment

Special “protective” or “anti-contamination” clothing is furnished and worn as necessary to protect personnel against contact with radioactive contamination. This consists of coveralls, lab coats, hoods, gloves, and shoe covers. Change rooms are centrally located in the RCA for proper utilization of this protective clothing. Approved respiratory protective equipment is also available to supplement process containment and ventilation controls, for the protection of personnel against airborne radioactive contamination and the possibility of internal radiation exposure. This equipment consists of full face air purifying respirators and self-contained breathing apparatus. Also, a breathing air system has been installed in the station and respiratory protective equipment consisting of air-line full-face respirators, hoods, and plastic suits is provided, should its use become necessary or desirable.

Maintenance of the above equipment is in accordance with the manufacturer's recommendations, rules of good practice, and applicable regulations. The use and maintenance of protective clothing and respiratory protective equipment is under the direct control of the Radiation Protection Section and personnel are trained in the use of this equipment before using it in the performance of their work. The use of this equipment is in accordance with the Technical Specifications.

12.3.2.4 Portable and Laboratory Equipment

Different types of instruments are selected to cover the entire spectrum of radiation measurement requirements expected. This includes instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. These consist of counting room and portable radiation survey/monitoring instruments. These instruments are required to provide protection against radiation for station personnel (for surveys required by 10CFR 20.1501); to control the release of effluents for the protection of the health and safety of the public (compliance with NRC Regulatory Guide No. 1.21); and to provide for all other radiological measurements necessary for personnel and public safety and for the protection of property. Sufficient quantities are obtained to allow for use, calibration, maintenance, and repair.

Counting room instruments for radioactivity measurements include the following:

1. Computer-based multi-channel gamma analyzer with multiple Ge or solid state detectors, used for identification and measurement of gamma emitting radio nuclides in samples of reactor primary coolant, liquid and gaseous waste, airborne contaminants, and similar samples.
2. Automatic and manual beta-gamma counter-scalers used primarily for gross beta measurements of surface contamination on swipes.
3. Alpha counter-scaler used for gross alpha measurements such as uranium or plutonium in reactor primary coolant samples or alpha contamination from surface or air samples.
4. Liquid scintillation counters used for measurement of tritium in reactor primary coolant, liquid and gaseous wastes.
5. Shielded body-burden analyzers are used for measurement of possible internally deposited radio-isotopes for determination of internal dose of personnel.

Portable radiation and monitoring instruments for routine use are selected to cover the entire range from background to high levels for the radiation types of concern. These include (with nominal range characteristics as indicated):

1. Beta-gamma (GM) survey meters used for detection of radioactive contamination on surfaces and for low level dose rate measurements.
2. Beta-gamma ionization chamber survey meters used to cover the general range of dose rate measurements necessary for radiation protection purposes.
3. Neutron survey instruments used to measure neutron dose rates for radiation protection purposes.
4. Alpha scintillation counters used for measurement of alpha contamination on various surfaces that may result from any uranium or plutonium in the reactor primary coolant, for example.

The Process Monitoring System is relied upon for continuous monitoring of airborne radioactivity. This is supplemented by grab air samples collected and analyzed by Radiation Protection during maintenance and routine and abnormal operations where airborne radioactivity may be involved.

Airborne gaseous, particulate, and iodine samplers are also available for routine use as well as an assortment of special purpose and emergency type radiation survey instruments (including bubblers for tritium, gas sample containers, low volume air samplers, - 2-8 cfm, ~100 percent efficient air particulate filters, and activated charcoal and silver zeolite cartridges. All of this equipment is kept in the Radiation Protection Shift office or Instrument Issue Point. Necessary emergency instruments are also located in the Control Room and at remote assembly points.

In addition to the portable radiation monitoring instruments, appropriate fixed monitors are located at exits from the Radiation Control Area. These instruments are intended to prevent any contamination on personnel, materials, or equipment from being spread in the unrestricted secondary systems areas of the station. Appropriate monitoring instruments are also available at various locations within the Radiation Control Area for contamination control purposes. Portal monitors, are also used, as appropriate, to monitor personnel before leaving the station.

All of the above instruments are subjected to initial operational checks and calibration and to a continuing quality control program to assure the accuracy of all measurements of radioactivity and radiation levels. This program requires the routine use of primary and reference standards and instruments which are traceable to National Institute of Standards and Technology. Performance checks of Counting Room, portable, and fixed monitoring instrumentation are made on a scheduled frequency and these instruments are recalibrated with standards whenever their operation appears statistically to be out of the accepted limits. In addition, routine calibrations are performed periodically on all of this equipment and after all repairs. A shielded calibration range capable of exposure rates from essentially background to hundreds of R/Hr is used for calibration of radiation monitoring instruments. Also available is a small (mCi level) source for certain low level calibrations and a Pu-Be neutron source for neutron instrument calibration. The gamma sources are calibrated with an exposure rate meter traceable to the National Institute of Standards and Technology.

The body burden analyzers are calibrated using phantoms and radioactive standards to accurately quantify the radionuclides of concern (Co-60, Co-58, Cs-137, and I-131). These detectors are used in conjunction with a computer-based multi-channel gamma analyzer and associated readout to obtain a permanent record.

Records of all calibrations are kept and personnel dosimetry, survey and monitoring records, etc., are maintained as required by NRC regulations.

12.3.3 Personnel Dosimetry

Personnel monitoring equipment consisting of thermoluminescent dosimeters, (TLDs), and electronic dosimeters are worn by all personnel (employees and visitors) whose jobs involve radiation exposure as defined in 10CFR 20. Additional personnel dosimetry equipment such as extremity TLDs may be assigned as needed depending on the radiological conditions encountered.

Personnel whose jobs require them to frequently enter the Restricted Area of the station may ordinarily be assigned a permanent personnel monitoring badge (TLD). Personnel working under a specific Radiation Work Permit in a job situation where a sizeable fraction of the allowable dose may be received in a relatively short period of time may additionally be assigned extremity monitoring equipment, depending on job conditions. Extremity monitoring equipment is issued for jobs or situations where extremity dose is expected to be limiting or controlling or in excess of the whole body dose. The use of additional personnel monitoring equipment beyond that routinely used depends on the job and on existing radiological conditions as evaluated and determined by Station Radiation Protection.

Records of Radiation exposure history and current occupational exposure are maintained by the Radiation Protection Section for each individual for whom personnel monitoring is required. The external radiation dose to personnel is determined on a daily basis by means of electronic dosimeters. Personnel monitoring badges (TLDs) are normally processed semiannually but may be processed more frequently if necessary. The TLD processing lab is accredited under the National Voluntary Laboratory Accreditation Program.

A body burden analyzer system to determine internal exposure is available on-site. Outside services for radio-bioassay and whole body counting may be used as required for backup and support of the program. The station equipment is sufficiently sensitive to detect in the thyroid, lungs, or whole body the Derived Investigation Level for those gamma emitting radionuclides expected.

All individuals requiring initial entry or final exit from the protected area will receive a whole body count using either portal monitor or whole body analyzer.

Anyone onsite, whether badged or not, who was involved in a radiological accident where internal exposure was likely, would be given a body-burden analysis as soon as practicable. If radioactive material uptake had occurred, proper action would be taken as stated in the System Radiation Protection Procedures.

Personnel monitoring badges (TLD) are supplied by a central in-house service which is responsible for the calibration and maintenance of all TLD and TLD readout equipment. Electronic dosimeters are calibrated at the Central Calibration Facility and distributed to the site for use.

12.3.4 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

12.3.4.1 Policy Considerations

Duke Energy management is firmly committed to the “As Low As Reasonably Achievable” philosophy for all nuclear operations. This commitment is stated in the DPC ALARA Manual (Reference 1). A formal ALARA program has been established in order to convey and enforce Duke management's commitment to ALARA. It consists of:

1. a published ALARA Manual;
2. continued written appraisal of in-plant radiation and contamination conditions by the station and general office radiation protection staff;
3. an ALARA Committee at each station consisting of management and General Office radiation protection staff, whose purpose is to conduct and appraise the effectiveness of the ALARA program at the nuclear facility; and

The committee members have extensive background in nuclear plant radiation and exposure control, including such areas as layout, shielding, personnel access, ventilation, waste management, monitoring systems, operations, and maintenance.

Although upper level management is vested with the primary responsibility and authority for administering the Duke ALARA program, the responsibility for ALARA is extended through lower management to the individual employee. The specific responsibilities of the General Office and station radiation protection staffs are to ensure that:

1. An effective ALARA program is established at each Duke nuclear station that appropriately integrates Duke management philosophy and NRC regulatory requirements and guidance.

2. Plant design features, operating procedures and maintenance practices are in accordance with ALARA program guidelines; and that written reviews of the on-site radiation control program assure that objectives of the ALARA program are attained.
3. Pertinent information concerning radiation exposure of personnel from other operating LWR power stations within and outside of the Duke system, are reflected in the design and operation of Duke stations.
4. Appropriate experience gained during the operation of nuclear power stations relative to in-plant radiation control is factored into revisions of procedures to assure that the procedures indeed do meet the objectives of the ALARA program.
5. Necessary assistance is provided to insure that operations, maintenance, and decommissioning activities are planned and accomplished in accordance with ALARA objectives.
6. Trends in station personnel and job exposures are analyzed in order to permit corrective actions to be taken with respect to adverse trends.
7. Procedures and engineering controls will be used, to the extent practical, to ensure that occupational doses and doses to members of the public are ALARA.

Reports of the findings of the general office and station radiation protection staffs are also effectively conveyed to management.

Specific responsibilities of station personnel are to ensure that:

1. Activities are planned and accomplished in accordance with the objectives of the ALARA program.
2. Procedures and their revisions are implemented in accordance with the objectives of the ALARA program.
3. The general office radiation protection staff is consulted as necessary for assistance in meeting ALARA program objectives.

Other group and individual responsibilities to the ALARA program are outlined in Section II of the DPC ALARA Manual.

12.3.4.2 Design Considerations

Consideration was given to such factors as projected component dose rates, space, mobility, accessibility, etc., during the initial design and construction phases of McGuire Nuclear Station. There is a large degree of component separation between higher and lower radiation areas. Those components where the potential of exposure from CRUD exists are provided with flushing capability. In addition, the general office radiation protection staff reviews station and component design in light of state-of-the-art technology. Frequent station visits supplement a formal operational feedback program which is used to identify specific and/or generic problems and implement design improvements.

ALARA exposures receive further attention through the training of designers and in equipment selection. Piping designers attend training sessions where topics, such as methods of minimizing crud build-up in piping are covered.

These sessions provide designers with a working knowledge of radiation protection. Closely working with equipment vendors results in the purchase of low maintenance equipment with material properties suitable for minimizing corrosion.

12.3.4.3 ALARA Operational Considerations

Consistent with Duke Energy's overall commitment to keep occupational radiation exposures as low as reasonably achievable, (ALARA), specific plans and procedures are followed by station personnel to assure that ALARA goals are achieved. Operational ALARA policy statements are formulated at the corporate staff level in the Nuclear Production Department through the issuance of the Radiation Protection Policy Manual and are implemented at each nuclear plant by means of procedures. These statements and procedures are consistent with the intent of Regulatory Guides 8.8 and 8.10. Personnel and job exposure trends are reviewed by management at the station and in the general office, and appropriate action is taken. Summary reports of occupational exposure are provided that describe problem areas where high radiation doses are encountered and that identify which work group is accumulating the highest doses. Recommendations are then made for changes in operating, maintenance, and inspection procedures or for modifications to the station as appropriate to reduce doses.

Maintenance activities that could involve significant radiation exposure of personnel are carefully planned. They utilize any previous operating experience, and are carried out using well trained personnel and proper equipment. Radiation Work Permits (RWP's) for non-routine operations, or Standing Radiation Work Permits (SRWP's) for routine operations are issued for each job, listing Radiation Protection requirements that will be followed by all personnel working in the Radiation Control Area (RCA). Where applicable, specific radiation exposure reduction techniques, such as those set out in Regulatory Guide 8.8, are evaluated and used. Procedures for such radiation exposure related operations as maintenance, inservice inspection, radwaste handling, and refueling, are well planned and developed by cognizant groups, and are reviewed by the station radiation protection staff to ensure that exposures will be ALARA. Careful personnel radiation and contamination monitoring are integral parts of such maintenance activities. During and upon completion of major maintenance jobs, personnel radiation exposures are evaluated and assessed relative to estimated exposures so that appropriate changes can be made in techniques or procedures as soon as practicable for future jobs. The General Office radiation protection staff also conducts reviews of radiation exposure related activities to assure that procedures are adequate, that they are being followed properly, and that deficiencies are corrected as soon as practicable to ensure that exposures will be ALARA.

The station ALARA Committee carefully reviews operations and maintenance activities involving the major plant systems to further assure that occupational exposures are kept ALARA.

Information from these ongoing ALARA reviews is evaluated with regard to its relevance for the design of new plants, particularly those features and components dealing with radiation exposure and control. When problems are identified that are generic in nature, such as shielding of penetration leakage, requests are initiated by the Nuclear Production Department to the Site Engineering Department so that appropriate changes can be made in the design of this and all future stations.

12.3.5 References

1. Duke Energy Company, “ALARA Manual.”

2. Duke Energy Company, “Radiation Protection Policy Manual.”

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