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12.0 Radiation Protection

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

12.1.1 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 DE Fleet ALARA Manual. A formal ALARA program has been established in order to convey and enforce Duke management's commitment to ALARA. In accordance with the requirements of 10CFR20, procedures and engineering controls will be used, to the extent practicable, to ensure that occupational doses and doses to members of the public are ALARA. This program consists of the following:

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/or non-exempt representatives from the Operations, Maintenance, Engineering and Radiation Protection groups, including liaison from the 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. The ALARA program is implemented from initial planning through decommissioning of the station.

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 III of the DE Fleet ALARA Manual.

12.1.2 Design Considerations

In the design of Catawba Nuclear Station, maintaining occupational exposures as low as reasonably achievable (ALARA) is a major design consideration in accordance with Section C.1 of Regulatory Guide 8.8. Although the station's design is intended to result in ALARA exposures during operation, these design features will also facilitate decommissioning. One such example is the separation of radioactive components.

The station design is reviewed by the general office radiation protection staff. This assures the input of radiation protection professionals into the final station design. Design review not only entails examining layout and piping drawings, but also included inspection of a scale model of Catawba during various phases of initial station design and construction. Radiation Shielding personnel are kept aware of current or anticipated radiation protection problems by periodic visits to the Oconee and McGuire Nuclear Stations. These station visits provide valuable feedback for use in reviewing the Catawba design. A formal operational feedback program is used to identify 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 usage of radiation zones and methods of minimizing crud build-up in piping are covered. These sessions provide designers with a working knowledge of radiation protection. In addition, close work with equipment vendors results in the purchase of low maintenance equipment with material properties suitable for minimizing corrosion. Those components with the potential of exposure from CRUD are provided with flushing capability from either demineralized water or chemical decontamination. Also, equipment is being designed that separates highly radioactive portions from lower radiation level portions of a component. An example of this is the Westinghouse evaporator package. This design separates the evaporator into hot and cold skids. A two foot thick concrete wall is placed between the skids. With most valves on the cold side, the majority of evaporator maintenance will be done in a relatively low radiation field.

12.1.3 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 Generation Department through the issuance of the Radiation Protection Policy Manual and the Fleet ALARA manual and are implemented at each nuclear plant by means of procedures. These statements and procedures are consistent with the intent of Section C.1 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) are issued for routine operations and for specific jobs, 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 when appropriate, 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 determine if 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.

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12.2 Radiation Sources

12.2.1 Contained Sources

For analytical purposes, radioactive components and systems are represented as idealized, conservatively-based source terms. The source terms used in the Catawba shielding design are based on reactor coolant design basis activities. (See Section 11.1 for a description of the reactor coolant source term.) A listing of the shielding source terms is found in Table 12-1, Table 12-2, and Table 12-3. Locations and geometries of these source terms are found in Table 12-19.

In addition to design basis corrosion and fission product terms, shielding for the Reactor Coolant System also accounts for N16 production, and full power gamma and neutron fluxes. Concentrations of N16 throughout the primary system are given in Table 12-4. A discussion of N16 production is found in Section 11.1. Four group gamma and neutron fluxes at the inside face of the primary shield concrete are presented in Table 12-5. These fluxes are based upon an axial peaking factor of 1.3 and are average values on the core center plane.

The only radioactivity - other than radiography sealed sources - to be stored outside the Reactor Building, Waste Monitor Tank Building, Auxiliary Building or contaminated parts warehouse is in the refueling water storage tanks, the reactor makeup water storage tanks, the steam generator drain tanks, the Retired Steam Generator Storage Facility (RSGSF) and designated radiation control zones (radwaste and materials) Activity in the refueling water tank is modeled as design basis reactor coolant after cold shutdown diluted with refueling cavity water, i.e., demineralized water. The reactor makeup water tank is assumed to have the activity of the recycle evaporator condensate. The steam generator drain tanks will contain the initial drain and flush of the secondary side of a steam generator. The activity of the drain is assumed to be design basis degassed reactor coolant. The flush is assumed to contain ten percent of the drain activity. Source strengths are presented in Table 12-6. Average expected conditions are a factor of 10 to 100 below these design values. The retired Unit 1 steam generators, at the time of removal, contained the corrosion and fission products listed in Table 12-41. The listed radionuclides and quantities were used to determine the RSGSF shield requirements.

Radioactive wastes leaving the station enroute to offsite disposal facilities are packaged in shipping casks or containers that meet applicable NRC and DOT regulations.

Gamma sources for spent fuel are presented in Table 12-7. These sources are used to establish shielding requirements during refueling operations. Transfer shielding is provided by a 5 foot thick poured concrete shield in the annulus. An irradiation time of $10E^8$ seconds at 109 kw/liter is the assumed operating condition. The source is homogenized over the volume of a fuel assembly.

All process piping that carries or may carry radioactive material is identified early in the station design. These lines are appropriately marked on flow diagrams, indicating that routing is to be in restricted areas of the station. All pipe routing is done with consideration of shielding requirements. Field run piping receives the same consideration for shielding as principal process lines. (See Section 3.9.2 for field run piping criteria.) All field run piping is reviewed before station operation to assure all areas are within allowable limits. Station radiation protection personnel review all field run piping that will carry radioactive fluids for potential exposure problems to ensure that ALARA principles are applied where appropriate.

12.2.2 Airborne Radioactive Material Sources

Airborne radioactivity is introduced within the plant principally via the following mechanisms: (1) leakage of radioactive fluids through pump seals, valve stems, and flanges; (2) evaporation of tritiated water; and (3) recirculation of contaminated air discharged from the plant. Each of these mechanisms is discussed below.

12.2.2.1 Leakage

Leakage of radioactive fluids is minimized by the use of special design features as described in Section 11.2.1. Nonetheless, the leakage source term cannot be realistically reduced to zero. In Section 11.3, the effluent releases due to equipment leakage are based on the recommendations of NUREG-0017. These same releases are used as the source term in determining airborne concentrations within the plant. For the Auxiliary Building, the fraction of the total source term which is released into each separate cubicle or corridor is an estimate based on the type of equipment housed, the number and type of valves, the number of flanges, and the level of radioactivity in the fluid streams. The leakage source term for selected Auxiliary Building cubicles is listed in Table 12-8. Leakage from the RSGSF is not expected since the retired steam generators have all nozzles sealed and all surface contamination fixed. The RSGSF design includes a sump to collect water for sampling prior to release if required.

12.2.2.2 Evaporation

The spent fuel pool and, during refueling, the reactor transfer canal are the two significant sources of airborne tritium due to evaporation. The actual rate of evaporation at any time is dependent upon pool temperature, air velocity across the pool, and relative humidity. However, based on the values in Section 11.3, tritium can be calculated to appear at an average rate of 84,000 µCi/hr in the spent fuel building atmosphere and, during refueling, in the containment atmosphere.

12.2.2.3 External Airborne Radioactivity

The release of low levels of radioactivity from the unit vents is a continuous process. In general, the airborne material will rise above the plant due to the momentum and heat content of the exhaust and will be carried away by wind currents. However, during certain meteorological conditions, compounded by the wake effect of nearby structures, this material may linger close to the station where it can be drawn in by the various building ventilation systems. The method of calculating atmospheric dispersion and the dispersion coefficient (X/Q) for various ventilation intakes is discussed in Section 2.3. Airborne releases from the RSGSF are not expected since the retired steam generators have all nozzles sealed and all surface contamination fixed. The RSGSF design includes a passive HEPA system to prevent releases from the facility.

12.2.2.4 Inplant Concentrations

The levels of airborne radioactivity within the plant during normal operation are based on estimates of the above sources. It is assumed that in areas where there are no potential sources of radioactive leakage or evaporation the concentration of radioactivity is equal to the concentration in the air external to the ventilation intakes. This is reasonable since the design ventilation system is such that air flows from areas of low potential airborne radioactivity to areas of higher airborne radioactivity. For those areas with sources of leakage or evaporation, the concentration is calculated by

$$C = Co + \frac{Q}{1.7X10^{6} F}$$

where

C	=	room concentration (μCi/ml)
Co	=	outside air concentration for the appropriate ventilation system (μCi/ml)
Q	=	source term (μCi/hr)
F	=	room exhaust flow rate (ft^3/min)
$1.7 X 10^6$	=	conversion of (ft^3 - hr - min^{-1}) to (ml)

Decay has conservatively been neglected, with the exception of the containment activity during operation, to be consistent with the assumptions inherent in the effluent release calculations in Section 11.3. The containment concentrations during operation are based on a two-control volume model, assuming an upper volume of 670,000 ft^3, a lower volume of 380,000 ft^3, and an average exhaust flow from the upper volume of 50 scfm. The estimated inplant concentrations are presented in Table 12-9 through Table 12-14 based on the flowrates provided in Section 9.4.

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12.3 Radiation Protection and Design Features

12.3.1 Facility Design Features

Although the basic design of Catawba pre-dates Regulatory Guide 8.8, maintaining occupational exposures ALARA is a major design consideration. To aid in layout and shielding design, the station is divided into radiation zones. These zones indicate maximum dose rates based on design basis activites only. (See Section 12.2.1) The zone limits are summarized in Table 12-15, while their bases are listed below.

Zone I	-	areas adjacent to the station where Duke Power Company does not normally control access. 10CFR 20.1301 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	-	areas within the station where personnel are expected to be continuously present. Dose rates in these areas are limited to 100 mrem in 7 days. This is also within the corresponding limits imposed by 20.1502 and 20.1902.
Zone III	-	areas within the station where personnel are expected to be present for extended periods of time. The 1.0 mrem/hr limit is 40 percent of 10CFR 20.1201(a)(1)
Zone IV	-	analogous to Zone 3 except occupancy levels are lower. 10CFR 20.1201(a)(1) limits dose rate in these areas to 2.5 mrem/hr.
Zone V	-	infrequently occupied areas where dose rates exceed continuous occupational levels. 10CFR 20, sections 1601, 1602, 1901 and 1902 precautions are applied where warranted. Limit set at 15 mrem/hr.
Zone VI	-	access is limited. 10CFR 20, sections 1601, 1602, 1901 and 1902 precautions are applied where warranted. Limit set at 100 mrem/hr.
Zone VII	-	all areas of the station where dose rates exceed Zone VI. Access is physically restricted and 10CFR 20, sections 1601, 1602, 1901 and 1902 precautions are applied where warranted.

For clarity, the radiation zones are depicted in Figure 12-1 through Figure 12-28. The associated radiation zone designation for access during normal operation and shutdown conditions is shown in Table 12-17 and Table 12-29. Gradients are expected in radiation levels within each zone. Although "hot spots" that exceed zone limits may exist, most areas are expected to be significantly below these zone limits. Actual zone markings and access control will be carried out under the Radiation Protection Program (Section 12.5), taking into account applicable regulations and existing radiation levels.

Obtaining ALARA exposures as a result of design is basically a three step process. 1) Prevent the creation or buildup of a radioactive source, 2) if prevention is not successful, isolate the source from personnel, 3) if complete isolation is not possible, minimize the effect of the source on personnel.

Maintenance of equipment containing radio-cobalt results in a large percentage of a station's man-rem expenditure. Therefore, source prevention must start with the Reactor Coolant System. Co^{58} is produced by activation of nickel released by corrosion. The largest amount of nickel in the primary system is the Inconel 600 steam generator tubes. To reduce tube corrosion and to reduce crud retention proportions, Westinghouse steam generator tubes are bright annealed. Major sources of Co^{60} are activation of cobalt impurities in system materials and grindings from hard facing materials. To reduce these sources, the use of Stellite as a hard facing material is restricted to surfaces where excessive wear would otherwise occur. Strict cobalt impurity limits are placed on primary system materials as an additional reduction in cobalt sources. A list of cobalt content limits is in Table 12-16. However, these methods only reduce the level of activiated corrosion products. As a result, efforts must be made to prevent the buildup of activity in components and piping.

Many of the suggestions in Regulatory Guide 8.8 have been used in reducing the buildup of corrosion products. In routing pipe, care is taken to avoid dead legs. Valves and pipe connections are oriented so as to avoid the possibility of crud traps. Spent resin lines slope downward and flow by gravity from the demineralizers to the storage tanks. Evaporator concentrate and spent resin lines contain clean-out connections to handle plugging. Solid radwaste system lines use large radius bends and butt welds on valves. In addition, all tank bottoms slope towards the tank outlet. Also, most major components can be remotely flushed.

In some cases components and process piping become highly radioactive and must be isolated from station personnel. Filters, demineralizers, and the radwaste batching tank, for example, are accessible only by a hatch in their cubicle ceiling. Long runs of radioactive process piping are located in shielded pipe chases whenever possible. Access to areas containing highly radioactive components is restricted by locked doors. Valve galleries are provided to allow remote operation of valves via reach rods or air operators. For components that will continuously contain high levels of radioactivity, a valve room is generally provided to allow valve maintenance in a lower radiation field. These valve rooms usually have valve galleries adjacent to them.

Additionally, in instances where exposure could occur, steps are taken to minimize exposure. Radioactive process piping is lead shielded when it crosses areas designated Radiation Zone 4 or lower. Major components are placed in separate cubicles to reduce maintenance exposures from other equipment (see Section 12.3.2.2 and Table 12-19 for shield thickness). Canned pumps are used whenever possible to reduce the possibility of leakage. Floor drains are placed in most cubicles to prevent the spread of any contaminated leakage. Also, special floor coatings are used to aid in clean-up operations following any spills or leakage. Ventilation is designed so that air flows from noncontaminated areas into areas of potential airborne activity (see Table 12-8 and Table 12-9 for location and magnitude). Some cubicles that contain two radioactive components have a shield wall between the two, e.g., evaporator feed pumps. Some equipment is designed with shielding spaces that separate hot and cold portions of a component, e.g. evaporator package and waste gas hydrogen recombiners. Finally, a radiation monitoring system is provided to alert personnel to any sudden changes in area radiation levels. A description and the location of these monitoring systems are provided in Section 12.3.4 and Table 12-29, respectively.

The design basis radiation level in the counting room (elevation 594, column lines QQ-PP, 58-60) during normal operation and anticipated operational occurrences is 0.05 mR/Hr from outside sources of radioactivity. See Section 1.8 for additional sampling capability.

12.3.2 Shielding

12.3.2.1 Shielding Analysis

Calcuations to determine the adequacy of station shielding are based on Section 12.2.1 source strengths and the methods outlined below. Dose points are selected inside and outside cubicles containing radioactive equipment. Cubicle ceilings and floors are generally the same thickness as cubicle walls. Skyshine from the station is negligible because rooms or cubicles containing radioactive material are shielded overhead.

The only major neutron source in the station is the reactor core at full power. The section of the primary shield out to and including the reactor vessel wall is designed by Westinghouse. The reactor cavity concrete arrangement, designed by Duke, is similar to other Westinghouse four loop nuclear stations. The codes ANISN, DOT, MORSE, and SABINE were used to verify the effectiveness of the primary shield for initial licensing.

Sources of gamma radiation are distributed throughout the Auxiliary and Reactor Buildings. The codes SHIELD and KAP-VI were used to verify gamma source shielding for initial licensing. The following sequence typifies a gamma source shielding analysis:

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 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 through 3 were done with the code N237BURP and data from Chapter 11. Step 4 is self explanatory; tanks, demineralizers, filters, and pipes are modeled as a right circular cylinder, etc. Except for the input of material densities, Step 5 is code internal. Step 6 simply determines the adequacy of the shielding. Shield design is an iterative process. However, due to the coarse, conservative nature of shield size, this poses no problem.

All the computer codes described below were used on Duke's IBM 370 computer, with the exception of ANISN, for initial licensing of Catawba.

ANISN performs shielding calculations by a discrete ordinates solution of the Boltzman equation in one direction. Through 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 are found in References 1 and 2.

SABINE 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 through shields using nineteen removal energy groups that in turn feed twenty-six groups for the diffusion calculation. Secondary gamma production in each shield region is output as a polynomial curve fit. Gamma fluxes are also calculated. Further details are in Reference 3.

KAP VI employs the point kernel technique to determine dose rates from complex sources whose geometries can be described by second order surface equations. An exponential attenuation function with buildup is employed for gammas. Neutron attenuation functions are also available. Detailed descriptions of the code and its geometry routines are found in References 4 and 5.

SHIELD is an in-house code that calculates fluxes at receiver points with integrals over simple geometries, using the equations of Reference 6. The gamma spectrum is divided into six energy groups. Input includes group specific source strength and average energy, source and shield geometries, and material densities. Whenever the spectrum of average energies changes, energy dependent parameters are recalculated. The code contains energy dependent data on tissue flux-to-dose conversion factors (Reference 7), mass attenuation coefficients (Reference 8) for common source/shield materials and Taylor-form buildup factor coefficients (Reference 9). For combined shields the buildup factor is automatically based on the material with the greatest optical thickness in the lowest energy group. When calculations exceed code-internal data, appropriate warning statements are output.

MORSE is a multi-purpose neutron and gamma ray Monte Carlo transport code. Through the use of multigroup cross-sections, either forward or adjoint solutions of neutron, gamma ray, or coupled neutron-gamma ray problems may be obtained. Time dependence for both shielding and criticality problems is provided. Three dimensional as well as specialized one dimensional geometry descriptions may be used. An albedo option is available at each material surface. Also available is isotropic or anisotropic scattering up to a P^{16} expansion of the angular distribution. A complete description of the code is in Reference 10.

DOT solves the Boltzman transport equation in two dimensional geometries by use of the discrete ordinates method. Balance equations are solved for the density of particles moving along discrete directions in each cell of a two dimensional mesh. Anisotropic scattering is treated using a Legendre expansion of arbitrary order. Both homogeneous and external source problems can be solved. Albedo boundary conditions are available. A more detailed description is presented in Reference 11.

N237BURP is an in-house code that calculates the accumulated activity on demineralizer resins or filters and the resultant activity of the process stream. This is accomplished by solving a pair of coupled, first order differential equations. Required input is isotopic removal efficiencies and operation time. Gamma source strengths are obtained from the calculated specific activities by considering gamma yield and losses due to conversion electrons. The nearly 300 individual gamma emissions of these isotopes are divided into six discrete energy groups. Group boundaries remain fixed, but the average group energy is calculated for each spectrum of isotopes. This allows reasonably precise selections of energy dependent shield material properties for attenuation properties.

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 for 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 Computional 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 CCC-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.

12.3.2.2 Shielding Design

Most of the station shielding is poured ordinary concrete. Some lead, and to a lesser extent steel, is also used as shielding. For shield design, a concrete density of 2.35 gm/cc is used. Lead shields are either stacked bricks or laminated slabs. Whenever concrete or lead blocks are used as a shield, care is taken to avoid unshielded paths through joints. Section 3.8.1 addresses the use of Regulatory Guide 1.69 for concrete shields.

Design features, as discussed in Regulatory Guide 8.8, are utilized throughout the plant, specifically, shield walls separating Zone 4 areas from higher radiation areas are not penetrated. If such a wall must be penetrated, care is taken to reduce the consequences of the penetration by avoiding paths for radiation streaming. Cubicle entrances retain their shielding integrity with labyrinths. Valve galleries are provided to allow remote operation of valves from a shielded area. Minimum shield thicknesses for Catawba are given in Table 12-19. Descriptions of areas requiring shielding are presented below.

Station Site

A detailed plan of the Catawba site is shown in Figure 1-19. Residual amounts of radioactivity are expected in the refueling water storage tanks and the reactor makeup water storage tanks. (See Section 12.2.1) The steam generator drain tank may contain significantly higher levels of radioactivity and is shielded by two-foot thick, concrete walls. Dose consequences are discussed in Section 12.4. The retired unit one steam generators are stored in the Retired Steam Generator Storage Facility. Each steam generator contained approximately 70 curies at time of removal. The RSGSF design includes 2.5 foot thick concrete walls. The shield is designed to limit exposures on the outside surface of the facility to less than or equal to 100 mrem in a year per 10CFR20.1301, Dose Limits For Individual Members of the Public.

Reactor Building

Scale layouts and cross-sections of the Reactor Building are shown in Figure 1-10 through Figure 1-18 Shielding consists of the primary shield, crane wall, compartments and the Reactor Building.

The primary shield, described in Table 12-20 and Figure 12-29, is designed to:

1. Attenuate the neutron and gamma fluxes from the reactor core and secondary gammas from the shield to allow limited access to the containment during normal operation and to protect components and structures from excessive damage or activation.

2. Limit shutdown radiation levels in the building, permitting limited access to Reactor Coolant System components.

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

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

The Reactor Building wall, in conjunction with the primary and secondary shields and the Containment Building, is designed to limit radiation at its outside surfaces, as required by 10CFR 20 and 10CFR 100, during both full power operation and the postulated Design Basis Accident.

Auxiliary Building

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

1. Pipe Shafts - areas where piping is run between cubicles or from one floor to another; no personnel access is expected; Zone 6.
2. Cubicles - rooms where individual components are located; usually Zone 6 during operation and Zone 4 during system shutdown and isolation.
3. Corridors - areas where non-radioactive components are located or that provide personnel access to cubicle entrances; Zone 4.

Scale drawings of the Auxiliary Building are shown in Figure 1-2 through Figure 1-9. Minimum shield thicknesses of major components are given in Table 12-19.

Fuel Handling System

The Fuel Handling System is described in Section 9.1. Water, which has the advantages of good visibility and heat removal capability, is the principal shield for fuel handling operations. Additional shielding is provided by structures along the fuel transfer route, such as concrete sidewalls of the refueling cavity and the spent fuel pool. The annulus portion of the fuel transfer tube is permanently shielded with five feet of poured concrete to reduce the annulus dose rate during fuel transfer operations. Access to identified High Radiation Areas will be controlled per applicable procedures. Access to the fuel transfer tube bellows seal is controlled by a locked gate to insure coordination between Radiation Protection, inspection, and spent fuel transfer operations personnel. Details of the shielding can be found in Figure 12-2 through Figure 12-4. The activated reactor vessel internals are stored in the refueling cavity after removal from the vessel.

Design Basis Accident

An area of the station that requires special consideration for the Design Basis Accident analysis is the control room. The whole body dose must not exceed 5 rem for the duration of the accident.

Principal shielding for the control room is the Reactor Building wall (3 ft. concrete). Additional shielding is provided by the Reactor Building's internal shielding and the control room floor (1 ft. concrete), ceiling (2 ft. concrete), and wall (1 ft. concrete). Other concrete structures, such as the main steam pipe shaft, also provide shielding. The parameters and sources used in the control room direct dose analysis are presented in Table 12-21 and Table 12-22.

After a postulated design basis accident, certain equipment may require inspection or maintenance. For this reason, shielding is also provided for safety injection pumps, containment spray pumps, and containment spray heat exchangers.

12.3.3 Ventilation

The ventilation systems dealing with radiation protection are the Control Room Area Pressurizing Air Sub-System, the Auxiliary Building Ventilation System (including the Radwaste Area), the Lower Containment Clean-up Sub-System, the Containment Purge Exhaust Sub-System, the Fuel Handling Area Exhaust Sub-System and the Annulus Ventilation System.

12.3.3.1 Design Objectives

These ventilation systems are designed to assure that concentrations of airborne radioactivity are kept as low as reasonably achievable below the limits specified in 10CFR 20. In addition, the systems have adequate capacity to reduce concentrations of airborne radioactivity in areas not normally occupied. Where maintenance or in-service inspection is to be performed, levels are in compliance with the requirements of 10CFR 20. Filter systems capable of controlling airborne radioactivity are easily maintained and will not create a radiation hazard to personnel. A typical filter arrangement is shown in Figure 12-30.

12.3.3.2 Design Description

The ventilation systems indicated in Section 12.3.3 are described in detail in Sections 6.4 and 9.4.

12.3.3.3 Air Flow Control

The ventilation systems indicated in Section 12.3.3, with the exception of the Control Room Area Pressurizing Air Sub-System, are designed to supply air to the clean areas of the station and exhaust air from areas of potentially higher airborne radioactivity through filters. The Control Room Area Ventilation System is designed to provide sufficient outside air through an air clean-up filter system to pressurize and thus ensure that all control room area leakage is effluent. Air that is removed from potentially contaminated areas is passed through clean-up units and exhausted by way of the unit vent. The only filtered air that is recirculated is containment air, which is passed through the containment auxiliary carbon filter units, and the portion of the annulus air which is not exhausted.

12.3.3.4 Typical System

The Auxiliary Building Ventilation System is used as an illustrative example of a typical air clean-up system design.

The Auxiliary Building general ventilation supply subsystem has a major impact on the personnel protection features incorporated into the design of the ventilation system. To control airborne activity, the Auxiliary Building ventilation supply air is delivered to the "clean" areas and areas of general personnel occupancy. This air is then routed to areas of greater contamination potential by pressure gradients induced by the exhaust system. Air is supplied and exhausted from the various areas of the Auxiliary Building as shown in Figure 9-121, Figure 9-122, Figure 9-123, and Figure 9-124. As shown in Figure 9-121, Figure 9-122, Figure 9-123, and Figure 9-124. some of the potentially contaminated areas of the Auxiliary Building require a direct supply of clean air in order to maintain the desired environment. However, in all such cases the quantity of air exhausted exceeds the amount supplied directly, thus confining any airborne contaminants to the subject space(s).

Clean outside air is supplied to areas of the Auxiliary Building through a bank of 2-inch deep prefilters and a bank of cartridge filters with an average efficiency of not less than 45% by the NBS test method for atmospheric dust. Each filter bank is provided with static pressure taps and an indicating gauge to measure the pressure drop across the filters. Filter change out is conducted in accordance with the manufacturer's recommendations. All exhaust from contaminated areas may be exhausted through filter trains, all of which contain, as a minimum, prefilters, HEPA filters, and a carbon adsorber in that order. In addition, all filter trains which are classified as Nuclear Safety-Related have a moisture eliminator and preheater upstream of the prefilters and another bank of HEPA filters down-stream of the carbon adsorber. A layout of a typical safety-related filter train is illustrated in Figure 12-30. Prefilters, moisture eliminators, and HEPA filters are provided with static pressure taps and indicating gauges to measure the pressure drop across the filters. Filter change-out is conducted in accordance with the filter train manufacturer's recommendations. Testing and changing of carbon is conducted in accordance with ANSI N510-1980.

The design of the Nuclear Safety-Related filter systems has been compared with the 1978 edition of Regulatory Guide 1.52. This comparison can be found in Table 12-23, Table 12-24, Table 12-25 and Table 12-26.

A description of the ESF Filter System design parameters is given in Table 12-27.

The design of the Containment Purge Filter System has been compared with Regulatory Guide 1.52, Revision 2 in Table 12-28.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring System

The Area Radiation Monitoring System is provided to monitor radiation levels in various plant locations that are potential radiation exposure areas. Indications and alarms from this system are used as an aid in conjunction with information from process radiation monitors (Section 11.5), plant operating procedures, and administrative controls to assure that station personnel exposure remains as low as reasonably achievable within 10CFR 20 limits. Additionally, the Area Radiation Monitoring System assists in compliance with General Design Criteria 19, 63, and 64, and Regulatory Guides 8.2 and 8.8. A discussion of compliance with Regulatory Guide 1.97 is provided in Section 7.1.2. The only control function associated with the Area Radiation Monitoring System deals with the area monitors on the Refueling Bridge in the Spent Fuel Buildings (1EMF15, 2EMF4). If radiation exceeds a monitor's setpoint, the monitor will prevent the new fuel elevator from raising fuel assemblies.

12.3.4.1.1 Description

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 outputs available for recording. Geiger-Mueller detectors, each with a range of 10^{-1} to 10^{4} mR/hr, are used in 33 area radiation monitoring channels. Four ion chambers, each with a range of 10^{-1} to 10^{4} R/hr, are installed in high-level monitoring channels. The location, sensitivity, range, and accuracy for each detector are shown in Table 12-29.

All supporting equipment for the detectors (i.e., analyzer/rate meter modules, annunciators, power supplies, auxiliary relays, module test pulse generators, etc.) is located in the control room, except for the supporting equipment associated with the technical support center monitor which is located in the Technical Support Center.

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.

12.3.4.1.2 Locations and Criteria

The location of the low-level area radiation monitors is based on the potential for significant radiation levels in an area and the expected occupancy of that area. The locations of the monitors are provided in Table 12-29.

Areas of the plant with high occupancy but little or no radiation potential (e.g., Turbine Building) and areas with high radiation potential but no occupancy (e.g. pipe chases) do not meet the above criteria and are not monitored.

The four radiation monitors located in the new fuel storage area are described in Table 12-29. These monitors meet the requirements of 10CFR50.68(b)(6).

Catawba is exempt from the requirements of 10CFR70.24. For more information on 10CFR70.24, see the letter from the NRC to M.S. Tuckman dated July 29, 1997 (Reference 12).

The eight Geiger-Mueller radiation monitors are located adjacent to the main steam lines to detect secondary radioactivity due to a steam generator tube rupture.

The location of the four high-level ion chamber radiation monitors is based on the potential for radioactivity accumulation in the reactor coolant filters to exceed the shielding capacity of the spent filter transfer cask.

12.3.4.1.3 Alarms and Indicators

Each detector, except the technical support center detector, is provided with an audible and visual alarm in the control room. The technical support center monitor is provided with an audible and visual alarm locally in the Technical Support Center. Additionally, each low-level area monitor (with the exception of the control room monitor) provides a local audible and visual alarm. When any alarm condition is acknowledged, the audible alarm will cease; however, the visual alarm will remain until the alarm condition clears. The Area Radiation Monitoring System is also provided with a self-checking feature. This self-checking feature initiates the alarms associated with the particular monitoring channel upon loss of detector signal, loss of power, high voltage, or loss of circuit continuity between the detector and the analyzer/rate module.

Meter indication of radiation level is provided locally and in the control room for each low level area radiation monitoring channel, except the technical support center monitor. The technical support center area monitor provides local meter indication in the Technical Support Center. Meters are calibrated from 10^{-1} to 10^{4} mR/hr.

12.3.4.1.4 Testing and Calibration

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.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

The Airborne Radioactivity Monitoring Instrumentation is a part of the Process and Effluent Radiological Monitoring System. A complete description of the airborne radioactivity monitoring instrumentation is presented in Section 11.5. See Section 1.7 for discussion of Regulatory Guide 1.21 compliance.

12.3.5 References

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12.4 Dose Assessment

Individual exposure and the summation of occupational exposure varies with the size of the station staff, radiation levels, and time spent in radiation areas. The time factor is dependent upon the complexity of the job, the frequency of the job, and the efficiency of the personnel assigned to the job. In order to obtain ALARA exposures and realistic dose estimates, all of these factors must be considered in accordance with Regulatory Guide 8.8.

Table 12-30 provides the expected station organization and the occupancy of each individual group in major work locations (i.e., office, field, and control area) based on operating experience at Oconee Nuclear Station and proposed operation at Catawba. Occupancy in the field is divided into seven radiation work areas; the number of personnel and time spent in these areas are provided in Table 12-31. Routine operation dose assessment for each work group, work area, and the total station is a combination occupancy and work area radiation dose rate, and is provided in Table 12-32. Dose rates in this table are based on the design basis source terms of Section 12.2 and the radiation zoning diagrams of Section 12.3.

The annual radiation exposures that could be received by plant personnel during normal operation, inspection, and anticipated operational maintenance due to direct radiation have been estimated for major work functions. These estimates include the following:

1. Plant operation
2. Routine patrol
3. Periodic tests and inspections
4. Waste processing
5. Inservice inspection
6. Anticipated maintenance
7. Refueling

These radiation exposure estimates were developed from exposure models for each of the major jobs in the above operations and functions. Each exposure model has been developed by breaking the job into individual tasks and identifying expected radiation fields, time spent in each radiation field, and number of men required for each task. The estimates for each task are based primarily on feedback from operating plants. The feedback used to develop these estimates includes frequency of the operation, background exposure rate, contact exposure rate, time required for the operation, number of men required for the operation, and total exposure accumulated during the operation. The time required for a given task includes ingress and egress time from the specified radiation area as well as orientation time and setup time for the job. In many cases, the exposure models have been developed from a number of radiation/time studies from different plants. Engineering judgment has been used to define typical values for each parameter in the exposure model. As such, the resultant exposure estimates should be used as typical values keeping in mind the variability of the input data from which the estimates were calculated. In many cases, the Catawba design differs from that at operating plants from which the operating data has been accumulated. In these cases, engineering judgment has been applied to the field data to develop adjusted exposure estimates for the new system and component designs.

Several assumptions have been made in the development of the exposure models which reflect balance-of-plant design considerations and utility operational considerations. These assumptions include:

1. Radiation levels

   The radiation levels in the vicinity of systems and components are dependent on plant layout, permanent and temporary shielding, and operational considerations (e.g., general housekeeping, etc.).

   The assignment of a particular radiation level(s) to a particular task is based on field exposure, considering layout and shielding reflected in operating plants.

2. As low as is reasonably achievable (ALARA) techniques

   Tasks and operations performed in radiation areas assumed to be greater than 100 mrem/hr are assumed to take advantage of applicable ALARA techniques. Such techniques include flushing and draining of tanks, pumps, and fluid lines; installation of temporary shielding; and use of remote and semi-remote handling equipment.

In addition to the above assumptions, several other assumptions have been made in the development of the exposure models and subsequent exposure estimates. These assumptions include:

1. Exposure estimates are calculated assuming that the personnel perform all their required operations in areas of constant dose rate during each individual task, but may be in several areas of different dose level in performing the overall task.
2. Exposure times are based on estimates of the average time and average staff required to perform the designated task. These tasks include periodic tests and inspections, scheduled maintenance, refueling, routine plant operation, and anticipated operational maintenance.
3. Routine maintenance operations are divided between mechanical, instrumentation and control, and electrical personnel as follows: 40, 40, and 20 percent. Components are isolated by the auxiliary operator and this operation takes 1.0 man-hour per task.

The resultant estimates of expected annual radiation exposure by function to plant personnel from direct radiation developed from the exposure models are given in Table 12-33 through Table 12-39.

1. Table 12-33 summarizes the estimated annual occupational radiation exposures by activity, according to the format of Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates".
2. Table 12-34 summarizes the annual occupational radiation exposure for the reactor operations and survelliance category.
3. Table 12-35 summarizes the annual occupational radiation exposures for the routine maintenance category.
4. Table 12-36 summarizes the annual occupational radiation exposure for the waste processing category.
5. Table 12-37 summarizes the annual occupational radiation exposures for the refueling category.
6. Table 12-38 summarizes the annual occupational radiation exposures for the inservice inspection category.
7. Table 12-39 summarizes the annual occupational radiation exposures for the special maintenance category.

The occupational radiation exposure estimates for the inservice inspection category have been annualized over a 10-year period of operation. The in-service inspection requirements specified in the ASME Code, Section XI, and Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes", and in the technical specifications are generally amenable to averaging over a 10-year period.

The occupational radiation exposure estimates for the special maintenance category reflect dose estimates for projected special maintenance and repair tasks based on historical data. They do not include dose estimates for unique tasks that may be performed on a limited basis (i.e., a small number of the total plants in operation) such as unforseen major repair tasks or unusual inspection efforts.

The dose rate values given in Table 12-33 through Table 12-39 are averaged dose rates that will be seen by the worker performing the overall task. During the course of the task, the worker may experience dose rates above or below the average dose rate for various time increments.

A detailed summary of an exposure model is given for the refueling category in Table 12-37. This table provides a task-by-task description of the refueling operation along with expected dose rate and manpower requirements for each task. This model is typical of those used to evaluate the other major operations given in Table 12-33 through Table 12-39.

As a result of the exposure estimate models, several changes to the nuclear steam supply system equipment have been made. One of these design changes involves the installation of the conoseals following the reactor vessel head replacement during refueling. The previous conoseal installation required a person to perform several bolting operations in blind positions. The conoseal clamp has been redesigned to allow installation operations to be performed in full view of the person performing the operation. This has reduced projected installation time and minimizes the chances for erroneous installation, thus reducing the overall exposure associated with this operation.

The exposure estimates delineated above that are used in maintaining occupational radiation exposures at ALARA levels are in compliance with the guidance and considerations given in Regulatory Guide 8.19, Rev. 1.

In Section 12.2.2, concentrations of airborne radioactive material within the plant during normal operation were estimated. Using these values, the contribution of airborne radioactivity to the integrated personnel exposure is calculated to be 12 man-rem and 1.5 man-thyroid-rem. It has been assumed that the concentrations of noble gases and iodines in Column 3 of Table 1 in Appendix B to 10CFR 20 result in doses of 0.1 rem whole body and 0.6 rem thyroid, respectively, for a forty hour week.

The dose rate at the site boundary is conservatively estimated to be less than 10^{-5} mrem/hr. This assumes all outside tanks contain 100% capacity of their design basis fluids. This dose rate is based on source terms discussed in Section 12.2.1. A discussion of airborne contributions is provided in Section 11.3.

If McGuire or Oconee spent fuel assemblies are stored in the Catawba spent fuel pools as discussed in Section 9.1.2.4, additional occupational exposures of 0.029 man-rem per shipment could be expected. This would not contribute significantly to annual occupational exposures at Catawba.

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12.5 Radiation Protection Program

12.5.1 Organization

The administrative organization of the Radiation Protection program and the qualifications of the personnel responsible for the program and for handling and surveying radioactive material are in compliance with Regulatory Guide 1.8 and are discussed in FSAR Section 13.1, with exceptions as noted in Section 1.7.1.1.

This administrative organization is responsible for and has appropriate authority for assuring that the three basic objectives of the Radiation Protection Program at the Catawba Nuclear Station are achieved. These objectives are to:

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

Protection of personnel, means surveillance and control over internal and external radiation exposure 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 portions of the station 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.

This administrative organization is also responsible for and has appropriate authority for maintaining occupational exposures as far below the specified limits as is reasonably achievable by assuring that:

1. Station personnel are made aware of management's commitment to keep occupational exposures as low as is reasonably achievable,
2. Formal reviews are performed periodically to determine how exposures might be lowered,
3. There is a well-supervised radiation protection capability with well-defined responsibilities,
4. Station workers receive sufficient ALARA training,
5. Radiation Protection is provided with sufficient authority to enforce safe station operation.
6. Modifications to operating and maintenance procedures and to station equipment and facilities are made where they will substantially reduce exposures at a reasonable cost,
7. The radiation protection staff understands the origins of radiation exposures in the station and seeks ways to reduce exposures,
8. Adequate equipment and supplies for radiation protection work are provided.

The administrative 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 Power Company Technical System Manager, Radiation Protection, establishes the Radiation Protection Program including the program for handling and monitoring radioactive material for Catawba that is designed to assure compliance with applicable regulations, technical specifications, and regulatory guides. The Duke Energy Fleet Radiation Protection Manager also provides technical guidance and support for conducting this program, reviews the effectiveness and the results of the program and modifies it as required to assure that occupational radiation exposure and exposure to the general public are maintained as low as reasonably achievable. This position also provides Technical assistance to the Vice President, Nuclear Generation, who has management authority to implement the "as low as reasonably achievable" (ALARA) occupational exposure policy, to which Duke Power Company is committed.

The Station Radiation Protection Manager at Catawba 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 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 protection; to assist all personnel in carrying out their radiation protection 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 protection; radiological monitoring, survey and personnel exposure control work are performed on a continuing basis for station operations and maintenance.

The Radiation Protection Section performs the major portion of the radiation protection work for the station. The majority of Radiation Protection personnel work regular weekday schedules. In addition, Radiation Protection Technicians are supplied to each operating shift at all times. The Radiation Protection Section is typically organized into major areas, such as Surveillance and Control, Staff, Radioactive Materials Control and Shift.

The routine station radiation surveillance work consist of radiation monitoring and surveys, 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, particularly on the back shifts (other than day shifts). Several Radiation Protection procedures designate the 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. Either the Radiation Protection Section or the Central Calibration/Repair Facility calibrates and maintains the radiation monitoring instruments.

Duties concerning radioactive liquid, gaseous and solid waste disposal are performed by the Radiation Protection Section and by Chemistry and the operating shifts. 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 shipments of radioactive materials are under the control of Radiation Protection.

Training and qualification of personnel in Radiation Protection are the responsibilities of the Station Radiation Protection Manager and are performed by the Radiation Protection Section, or by the Training Group, under the Station Radiation Protection Manager's direction.

The Nuclear General Office Radiation Protection Laboratory also conducts the Offsite Radiological Monitoring Program for the Station.

The CNS radiation protection program organization of responsibilities is defined in compliance with Regulatory Guides 8.2, 8.8, and 8.10 (with same exceptions as previously noted for Regulatory Guide 1.8 in Section 1.7.1.1).

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 Portable and Laboratory Equipment and Instrumentation

The various types of laboratory equipment, portable radiation monitoring instruments and equipment and personnel monitoring equipment used in the radiation protection program were selected to provide the appropriate detection capabilities, ranges, sensitivities and accuracies required for the anticipated types and levels of radiation expected at Catawba Nuclear Station during normal operations and emergency conditions. The instruments are used to detect and measure alpha, beta, gamma and neutron radiation within the operational limits specified in Regulatory Guide 1.97 and to comply with Regulatory Guides 8.4, 8.8 and 8.15.

These instruments are required to perform radiation surveys (required by 10CFR 20.1501) for the protection of station personnel; to control the release of effluents for the protection of the health and safety of the public (for compliance with Regulatory Guide 1.21); and to provide all other radiological measurements necessary for personnel and public safety and for the protection of property. Sufficient amounts of equipment are obtained to allow for use, calibration, maintenance and repair.

12.5.2.1.1 Laboratory Equipment

Instruments used to measure radioactivity are located in the Radiation Protection Counting Room and other plant locations. The types of instruments and their operating characteristics are as follows:

A multi-channel gamma analysis system using germanium detectors having at least 12% efficiency is used to identify and measure gamma emitting radionuclides. Some of the sample types analyzed are primary reactor coolant, liquid and gaseous waste and airborne contaminants.

Automatic and manual GM counters are used to measure gross beta and gamma surface contamination.

Low background counters that can discriminate for alpha particles are used to measure gross alpha surface contamination and airborne contamination.

Liquid scintillation counters are used to measure tritium. Some of the sample types analyzed are liquid and gaseous waste and on-site environmental samples.

A shielded body-burden analyzer is used to identify and measure internally deposited radionuclides for determination of internal dose. It is sufficiently sensitive to detect a few percent of the Annual Limit on Intake (ALI) for most gamma emitting nuclides in the thyroid, lungs and whole body. It is located in the Administration Building. Outside services are used, as required, for backup and support of the bioassay program. This method is in accordance with Regulatory Guide 8.9.

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Laundry monitors may be used to measure gross radioactivity on laundered protective clothing prior to making it available for use.

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12.5.2.1.2 Portable Radiation Monitoring Instruments and Equipment

Routinely used portable radiation monitoring instruments are selected to detect background to high radiation levels. Most instruments are located in the Radiation Protection Respiratory/Instrument Control area.

Beta-gamma GM count rate survey meters are used to detect radioactive contamination on surfaces and for low level exposure rate measurements.

Low and high range beta-gamma GM and ion chamber survey meters are used to measure the range of dose rates necessary for radiation protection purposes.

Neutron survey instruments measure thermal, intermediate and fast neutron dose rates for radiation protection purposes.

Respiratory protective equipment such as breathing air, air-line, and self-contained breathing apparatus respirators are used per regulatory guidance provided in Regulatory Guide 8.15, 10CFR 20.1702 and 10CFR 20.1704.

Radiation survey instruments, recovery equipment and respiratory equipment for use in emergency situations are stored in emergency kits located in various areas such as the Operations Support Center, the Technical Support Center and Emergency Equipment Storage Room.

Various portable samplers for collecting airborne gaseous, particulate and iodine samples are available for routine use. Special purpose and emergency-type sampling devices and sample collection media such as bubblers for tritium, low volume air samplers, 100% efficiency particulate filters, silver zeolite cartridges and activated charcoal cartridges are also available. This equipment is controlled by Radiation Protection. Equipment necessary for emergency situations is located at various assembly points.

12.5.2.1.3 Personnel Monitoring Equipment

Fixed personnel monitoring equipment such as whole body monitors, hand and foot monitors, Small Article Monitors (SAMs) and portal monitors are located at selected exits from the Radiation Control Area, at various locations within the RCA and at the exit from the restricted area of the station. These instruments are used to prevent and control the spread of contamination from personnel, materials or equipment to areas outside the restricted area.

Small Article Monitors (SAMs) or vendor services are used to measure gross radioactivity on laundered protective clothing prior to making it available for use.

Thermoluminescent dosimeters (TLD's) and secondary dosimetry giving real time dose information are worn, as required by 10CFR 20, by personnel who work in the Radiation Control Area or whose job involves significant radiation exposure. TLD's measure dose attributable to beta, gamma and neutron radiation. Secondary dosimeters recording gamma radiation exposure and, when necessary, calculations of beta and neutron radiation exposures are used to track exposure until primary TLD's are processed. In addition, special dosimeters are available for measuring dose to extremities. TLD's and electronic dosimeters are issued from the Radiation Protection Dosimetry and Records Control area or temporary dosimetry issue areas. Electronic dosimeters, serving as secondary monitoring devices, are obtained as needed from self service storage areas or from Radiation Protection.

The TLD's have a sensitivity of ten millirem (gamma deep dose) and can measure doses up to one thousand rem. TLD's and the instrumentation used to process them are subjected to a continuing quality control program. The QC program includes the use of a computer program that compares TLD values with secondary dosimeter values recorded during the same monitoring period. Periodically, NIST traceable calibrations, instrument checks and evaluations and other manual checks are performed. Duke Energy also participates in NRC required performance testing programs and NVLAP accreditation. Secondary dosimeters and related instruments are also subject to annual calibrations and, if applicable, leak tests.

TLD's are supplied by a centralized in-house Dosimetry Laboratory which meets all applicable requirements for sensitivity, range and accuracy of measurement. Conformance with appropriate standards is also required for NVLAP accreditation. The Dosimetry Laboratory is capable of providing dosimetry for both routine and emergency conditions. Dosimetry processing is performed in accordance with applicable standards and regulations.

12.5.2.1.4 Instrument Calibration and Operational Checks

All of the aforementioned instruments are calibrated and subjected to operational checks prior to initial use and are included in a continuing quality program to assure accuracy of all levels of radioactivity and radiation measurements per Regulatory Guide 1.33.

Calibration verifications are performed on all laboratory equipment annually and after equipment repairs or other changes that could affect their operation. Equipment is calibrated when equipment performance or calibration verifications indicate it is necessary. Calibration solutions and standards traceable to the National Institute of Standards and Technology (NIST) are used to perform calibrations and calibration verifications. Resolution and energy linearity checks are performed on the multi-channel gamma analysis systems and the body-burden analyzer system daily or prior to use. Operational checks are performed on all laboratory equipment daily or prior to use. The quality control program ensures proper equipment performance.

Portable radiation monitoring instruments and equipment are calibrated periodically or whenever instrument operational verifications fall outside of statistically acceptable limits. Calibrations are performed at the Duke Energy Central Calibration/Repair Facility or on site using appropriate calibration equipment and sources. The calibration gamma sources are calibrated to ±5% with instruments (secondary standards) having calibrations traceable to the National Institute of Standards and Technology (NIST). Operational checks and source response checks, using appropriate sources, are performed daily or prior to use.

Calibration verifications are performed on personnel monitoring equipment annually and after equipment repairs or other changes that could affect their operation. Equipment is calibrated when equipment performance or calibration verifications indicate it is necessary. The calibrations are performed using standards traceable to the National Institute of Standards and Technology (NIST).

12.5.2.2 Inplant Radiation Monitoring

12.5.2.2.1 Sampling and Analysis System

Inplant radiation monitoring systems provide station personnel with the capability to assess the radiological situation in various areas of significance during normal operations as well as during an off-normal or an emergency situation. The monitoring systems include the area radiation monitoring system, airborne radioactivity monitoring system, and portable radiation monitoring equipment. Conformance to Regulatory Guide 1.97 is discussed in Section 1.7.

The Area Radiation Monitoring System is provided to monitor radiation levels in various plant locations that are potentially significant personnel exposure areas. This system consists of gamma-sensitive detectors, signal conditioning and readout instrumentation, radiation level alarm sensing logic, audible and visual alarm devices, and outputs available for recording. Geiger-Mueller detectors, each with a range of 10^{-1} to 10^{4} mR/hr, are used in 40 area radiation monitoring channels. Ion chambers, each with a range of 10^{-1} to 10^{4} R/hr, are installed in four high-level monitoring channels. A complete description of the location, sensitivity, range, and accuracy of this system is presented in Section 12.3.4. Indications and alarms from this system are used as an aid in conjunction with information from process radiation monitors, plant operating procedures, and administrative controls to assure that station personnel exposure remains as low as reasonably achievable within 10CFR 20 limits.

Compliance with Regulatory Guide 1.97 is discussed in Section 7.1.2.

The spent fuel pool and fuel transfer canal areas have area radiation monitors located on the fuel handling bridges.

The process airborne monitoring system is provided in part to monitor ventilated areas that are potential sources of airborne radioactivity. These monitors provide indication of the airborne radioactivity in the areas monitored and provide alarms in the control room. This system monitors primary and secondary systems during normal operation, including anticipated operational occurrences. Additionally, some of the monitors (e.g. unit vent airborne monitor, control room air intake) perform control functions during postulated accident conditions. A complete description of the process airborne monitoring system including its range, sensitivity, setpoint, and detector type is presented in Section 11.5.1.2.2.

In addition, portable or mobile continuous air monitors (CAM's) are available for use during operations that may require a long period of time for completion, and/or where there is a probability that airborne contamination may be a longer-term problem. These monitors are equipped with a particulate filter and a detector, which collect and measure gross activity concentrations of airborne particulates.

Radiation Protection personnel are knowledgeable in the appropriate station procedures and are trained in the use of equipment required to determine airborne iodine concentrations in the plant under all conditions. Counting equipment is available for performing detailed sample analysis. A procedure to determine airborne radioiodine concentrations is also established which does not rely on the availability of a counting room. This procedure utilizes portable "survey-type" instrumentation to determine the iodine concentration. This instrumentation in conjunction with silver zeolite cartridges is a fully adequate method to monitor iodine in-plant.

12.5.2.2.2 Portable Post Accident Sampling and Analysis System

During post accident conditions, the station plans to use portable air samplers with charcoal cartridges to determine iodine levels in vital areas. Samples will be counted in the counting room with germanium detectors if the equipment is not needed for other samples.

To reduce counting system saturation, sample sizes will be varied to minimize counting system problems. If very small samples are still too radioactive to count in "normal" geometries near the detectors, the samples will be moved away from the detector to reduce the counting efficiency.

The iodine cartridge sample will be purged with a sufficient volume of nitrogen or clean air equivalent to remove noble gas attached to cartridge. Nitrogen purging of germanium detector shields can be used in addition to the counting room ventilation filtration to reduce airborne activity interferences.

All detectors are shielded, samples are shielded or removed from counting room, and the counting room has been designed with shielding to reduce radiation levels to 0.02 mrem/hr from plant sources during normal operation.

No samples will be collected if personnel exposures could exceed legal limits. Personnel collecting portable samples will be provided self reading dosimeters to monitor exposure rates in areas where samples are being collected. Radiological survey and radiation protection monitoring will be conducted for personnel entering plant areas listed in Section 1.8.1.

12.5.2.3 Description and Location of Facilities

12.5.2.3.1 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 sample preparation laboratory, and a shielded radiation survey instrument calibration 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. A counting area in the administration building is provided for performing measurements for internal personnel dosimetry purposes.

12.5.2.3.2 Personnel Change/Decontamination Areas

Change room facilities are provided where personnel obtain clean protective clothing and other equipment required for station work. The change rooms serve the reactor buildings, the Auxiliary Building, the Spent Fuel Pools, and the Hot Machine Shop. In addition, change rooms are provided in the Single Point of Access (SPA) facility for personnel that are working in the Unit 2 UHI Building for the MOX Fuel Project. A change room is also provided for female employees. 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 the first aid room in the Radiation Control Area). The change rooms in the SPA for the Unit 2 UHI Building have separate male and female <u>clean</u> sections only. Provisions for removal and handling of contaminated protective clothing will be performed in an enclosed area between the UHI Building and the SPA. Showers, sinks, and radiation monitoring equipment are provided in all of the change rooms to aid in the decontamination of personnel. The change rooms in the SPA for the UHI Building do not contain any showers or sinks.

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, at the exit of the Radiation Control Zone set up about the work area. They then proceed to the "contaminated" side of the change room, where they remove inner protective clothing items, monitor themselves; and then proceed to the "clean" side, where they put on their personal clothing and leave.

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 conveniently located in the Radiation Control Area of the station 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. This equipment consists of a breathing air system for use with air-supplied full-facepiece respirators and one piece suit. In addition, full-facepiece air-purifying (filter) respirators and Self-Contained Breathing Apparatus (SCBA) are available for use in certain situations.

Maintenance of the respiratory protective equipment is in accordance with the manufacturer's recommendations and Regulatory Guide 8.15. 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 them in the performance of their work. The use of respiratory protective equipment is in accordance with appropriate NRC regulations (10CFR 20) and regulatory guides.

12.5.2.3.3 Equipment Decontamination Areas and Contaminated Laundry

An equipment decontamination facility is 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. A decontamination laundry and a respiratory protective equipment cleaning and repair 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 of 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.

12.5.2.3.4 Control Points for Entrance to and Exit from the Radiation Control Area (RCA).

Table 12-40 describes the boundaries of the RCA for each elevation and designates the entrance/exit points. The main personnel entrance/exit point and other points (designated in Table 12-40) to/from the RCA are provided with contamination control checkpoints that are equipped with appropriate monitoring instrumentation. All other personnel access points into the RCA in the Auxiliary Building will be minimized and are protected by restricted in/free-out doors. 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 strategically placed throughout the RCA to prevent the spread of contamination within this area.

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

Authorized personnel enter the Radiation Control Area through access doors, usually on the Turbine-Building Service-Building Control-Room side of the station, and leave through these doors after completing a whole body monitor check when exiting the Radiation Control Area.

12.5.3 Procedures

Routine radiological monitoring to detect radiation, radioactive contamination, and airborne radioactivity will be performed throughout the plant on periodic schedules. Monitoring frequencies will be determined by the Station Radiation Protection Manager based upon the actual or potential radiological conditions. Schedules of routine monitoring will be issued to the technicians who will initial or sign the schedule when the routine is completed. As plant conditions change, the schedule will be updated. Radiological surveys are performed before personnel enter potential or actual high radiation areas where RP personnel have sufficient doubt as to the existing conditions. Radiological surveys are also performed as a backup to routine monitoring when conditions change. All survey and routine monitoring data is recorded and filed in files or electronically per site administrative standards. Retention of survey and monitoring records follows the requirements of 10CFR 20, Regulatory Guide 1.88 and The Duke Quality Assurance Program.

All work on systems or in locations where radioactive contamination or external radiation is present requires a Radiation Work Permit (RWP) prepared and approved under the direction of Station Radiation Protection supervision before work can begin. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit.

Keeping exposures ALARA is a major consideration. The Radiation Work Permit lists the precautions to be taken including, as appropriate and protective clothing to be worn, respiratory requirements and special dosimetry. The permit is available for review by people who perform the work. The Radiation Protection Section maintains original permit information.

All persons working under a permit are required to read the instructions on the permit and to complete appropriate entry and exit transactions before entering and after leaving the Radiation Control Zone. The information from the transaction is entered into the RM&C computer programs and serves, in part, as a personnel monitoring record for the individuals involved.

In order to protect personnel from radiation and radioactive materials, the Radiation Control 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, audible and visual indicators and alarms, etc., as required by 10CFR 20. Administrative controls are also used in conjunction with the above and keys are issued to authorized station personnel for access areas within the Radiation Control Area, under certain conditions.

Section 12.1.1 defines the Duke Energy overall ALARA program. Inplant procedures involving radiological conditions are written such that keeping exposures ALARA is a major consideration. The guidance of Regulatory Guides 8.2, 8.8, and 8.10 are utilized in formulating the radiological protection program and are used in the preparation and review of operating procedures. The knowledge and experience gained from other Duke Energy operating nuclear stations, as well as other utilities, are factored into the program, also.

All persons entering the Radiation Control Area of the station must wear personnel monitoring equipment as prescribed by the Station Radiation Protection Manager in accordance with NRC Regulations and must comply with applicable Radiation Work Permits.

Personnel whose jobs require them to enter the Radiation Control Area of the station ordinarily are assigned a TLD personnel monitoring badge which normally provides the record value for radiation dose and a secondary dosimeter which provides real time radiation dose information. Extremity monitoring equipment is issued for jobs or situations where necessary. The additional required personnel monitoring equipment beyond that routinely used, is job coupled and depends on radiological conditions as evaluated and determined by Radiation Protection personnel for those persons working under a specific Radiation Work Permit.

Individual Occupational Radiation Exposure records are filed and retained for each individual in accordance with the recommendations of Regulatory Guide 8.7.

The Sentinel computer program provides useful information needed to efficiently and effectively maintain daily personnel dose records. The Sentinel computer program maintains personnel dose information equivalent to the information required on a NRC-5 form. The computer program categorizes dose according to work group and job function. The Sentinel program also provides a report listing those cases where poor correlation is encountered between TLD badge results and secondary dosimeter totals reported for the same time period. These computer programs are designed to facilitate conformance with Regulatory Guides 8.2, 8.8, 8.10, and the Duke ALARA program.

Duke employees and contract service employees issued a personnel monitoring badge are given a body-burden analysis or passively monitored when the badge is initially issued and when employment is terminated or alternatively, when the person is transferred to a non-radiological assignment. Visitors are generally given a body-burden analysis or passively monitored each time a monitoring badge is issued and at the termination of the station visit. In addition, badged station personnel and appropriate other Duke system personnel participate in a routine passive monitoring program upon each exit from the protected area. The Station Radiation Protection Manager may waive the requirement for any analysis or passive monitoring on a case by case basis if it is determined that the analysis is inappropriate or impracticable.

Anyone onsite, whether badged or not, who is involved in a radiological accident where internal exposure is likely, is given a body-burden analysis as soon as practicable thereafter.

Body-burdens exceeding a small fraction of an Annual Limit on Intake (ALI), identified by passive monitoring, can be referred to a more definitive measurement facility to quantify the activity. Quantified body-burden analysis results are used when determining and assessing the resulting dose more accurately. The bioassay program at CNS was developed following the guidelines of Regulatory Guide 8.9.

Potential airborne radioactivity concentrations are kept to a minimum by process and engineering controls, proper preventive measures and good housekeeping techniques in conformance with Regulatory Guide 1.39. The frequency of routine surveys at selected areas for the assessment of radiation-field, radioactive contamination, and airborne radioactivity levels will be determined by the Station Radiation Protection personnel and will be based upon the actual or potential hazard, station status, tasks to be performed, and occupancy factors to ensure ALARA exposures within 10CFR part 20 limits. The frequency of routine measurements for airborne radioactivity may be weekly, monthly, or continuously depending upon the location, operating conditions and actual or potential hazard. All survey results are recorded, filed and may be posted locally to assure adequate radiological controls. Caution placards will be posted locally to comply with 10CFR 20.1901 and 10CFR 20.1902 requirements.

Control of airborne radioactivity levels will be assured through the use of the station's heating, ventilation and air conditioning systems and portable air movers and filters. At times when airborne radioactivity problems exist, prompt assessment of the airborne activity levels is required. Air samples with high efficiency filter media are used to sample for particulate activity. The filter media are screened and may be submitted for gamma spectral analysis, to determine the exposure conditions. Gaseous air samples are obtained using gas chambers, charcoal, or other absorbent media (such as water for tritium) and analyzed by radiometric counting or gamma analysis. Radioiodine sampling cartridges are used for sampling air when noble gas interference is suspected. Radiation Protection personnel are knowledgeable in the appropriate station procedures and are trained in the equipment required to determine airborne iodine concentrations in the plant. To be used as a final alternative, respiratory protection equipment will be available for use in those situations where airborne radioactivity hazards exist and other control measures are inadequate at the location and time. Section 12.5.2.2 provides information on the in-plant radiation monitoring systems.

The primary objective of the Respiratory Protection Program is to limit the intake of airborne radioactive materials by personnel. The preferred method for achieving this objective is the application of engineering controls (process containment, ventilation systems, and local exhaust equipment). When additional engineering controls are impractical or cannot be applied, respiratory-protective equipment is evaluated to determine if its use will result in the least TEDE doses.

Written procedures for implementing the requirements of the respiratory program are utilized by the station in conformance with Regulatory Guide 1.33. These detailed procedures include: (1) selection and supervision of personnel who are to be qualified in the use of respiratory equipment, (2) training of personnel by qualified and knowledgeable instructors, (3) methods for ensuring an adequate fit, (4) a maintenance program that includes decontaminating, cleaning, disinfecting, inspection, repair, and storage, and (5) the administrative controls for the issuance, usage, and handling after usage.

Personnel are screened by a physician to ensure that they are medically able to use respiratory equipment. Personnel passing the screening test are then given a test fitting.

Qualified and knowledgeable personnel are used for training personnel and their supervisors in the proper use of respiratory protective equipment. Retraining is conducted annually to ensure a high proficiency in the use of respiratory protective equipment.

Respiratory-protective equipment is properly selected and used to ensure that peak concentrations of airborne radioactive materials inhaled by an individual wearing the equipment do not exceed the limits and requirements specified in 10CFR 20. The program is conducted in accordance with Regulatory Guide 8.15.

Reusable respiratory-protective equipment is monitored for contamination, decontaminated as necessary and otherwise cleaned and disinfected. Equipment inspections are made before and after each use to ensure that it is functioning properly. Inspections are performed only by qualified and knowledgeable personnel.

Respiratory-protective equipment is available in designated locations for use by qualified personnel. Normally, this equipment is issued by appropriate Radiation Protection personnel from a respiratory equipment facility.

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The following radiation-safety requirements are provided to ensure that adequate safeguards are used for handling and storing sealed and unsealed source, special nuclear and byproduct materials. The Station Radiation Protection Manager, or other appropriate Radiation Protection personnel, is notified prior to ordering radioactive sources and other such materials so that the necessary arrangements for adequate protective measures and ALARA considerations can be made. Upon receipt of radioactive material at the site, Radiation Protection is immediately notified. When required by regulations or proceudres, Radiation Protection then properly monitors, records, delivers, opens, labels or posts, and assigns a custodian, before the source is stored or used. The custodian is responsible for the safekeeping, proper use, storage, and handling of all radioactive material assigned to him. He also accounts for the material at regular intervals whenever an inventory check is made by Radiation Protection.

All radioactive material is stored in appropriate locations in the RCA or RCZ's outside the RCA and is posted or labeled in accordance with 10CFR 20 regulations. Sealed sources containing more than 100 microcuries of beta-gamma activity or more than 5 microcuries of alpha activity are leak-tested once every six months. This testing is performed by Radiation Protection. Radiation Protection is informed of any change in storage locations or change in custodian.

The Station Radiation Protection Manager approves the applicable radioactive shipping procedures authorizing appropriately trained Radiation Protection personnel to ship radioactive material and waste from the station. All shipments are monitored to ensure proper packaging and labeling, and to complete the shipment-record forms and logs, in accordance with Department of Transportation (DOT) and NRC regulations and other requirements.

Contaminated material and equipment to be removed from the RCA for storage, repair, or use, are first monitored and tagged by Radiation Protection. Contaminated material and equipment transferred to another location within the RCA for storage, repair, or use, are monitored and tagged by Radiation Protection if necessary to ensure protection of workers. Exceptions to this are tools designated for contaminated use only, protective clothing, scaffolding with associated parts designated for contaminated use only and chemistry samples being transported to a laboratory for analysis. In all cases, restrictions apply for radiation and contamination levels. Handling and control of contaminated material and equipment is affected by the use of tags and labels. Contaminated material and equipment are properly packaged to prevent the spread of contamination.

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