12.0 RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

12.1.1 Policy Considerations

Administrative programs and procedures, in conjunction with facility design, embrace the as low as is reasonably achievable (ALARA) philosophy. The applicable Regulatory Guides (RG) used to develop the administrative programs and procedures are referenced in Table 3.12-1.

12.1.1.1 Management Commitment. STPEGS is committed to the ALARA philosophy. In keeping with this commitment, an ALARA policy statement has been promulgated by STPEGS management and provides the basis for the ALARA program at STPEGS.

During the design phase, instructions to designers, constructors and vendors reflected the commitment to ALARA which were, in turn, reflected in station design features. Now, during operations, HL&P's ALARA commitment is reflected in written procedures, training, and instructions to contractors, vendors and station personnel.

As with industrial safety, ALARA is a shared responsibility of all management and employees. Consequently, the ALARA program includes an ALARA Review Committee (ARC) composed of department management. The responsibilities of this committee include:

1. Providing oversight for the ALARA program to ensure effective implementation consistent with plant operations.

2. Reviewing and approving site, departmental, and outage exposure goals.

The Health Physics Division under the direction of the Health Physics Manager ensures that an effective ALARA program is implemented. The Health Physics Manager is independent of other station divisions, such as operations, maintenance, or technical support and reports to the Plant General Manager. The Health Physics Manager delegates specific responsibilities, as appropriate, to various individuals and staffs. The responsibilities of the Health Physics Division include: CN-3023

1. Aiding in establishment of an integrated ALARA program that combines management philosophy, regulatory requirements, and lessons learned.

2. Assuring that an effective measurement program is established to evaluate the degree of success achieved by station operations with regard to ALARA goals and objectives.

3. Reviewing measurement results on a periodic basis, providing the measurement results to the ARC for consideration, and initiating corrective actions when necessary.

4. Identifying locations, operations, and conditions that have potential for exposure through a program of radiation surveys.

5. Administering a radiation exposure and access control program.

6. Approving training programs related to radiation protection.

7. Developing plans, procedures, and methods for keeping radiation exposure ALARA.

8. Reviewing and recommending changes to procedures and processes to ensure ALARA is considered.

Responsibilities of other managers, including the Plant General Manager with respect to ALARA include:

1. Ensuring support from their personnel

2. Participating in the selection of specific goals and objectives for the plant.

3. Supporting the Health Physics Manager in implementing the ALARA program

12.1.1.2 Design and Construction Policies. The ALARA philosophy was applied during the initial design of the plant and implemented through internal design reviews and documentation. In addition, the design has been reviewed for ALARA considerations by STPEGS personnel. These reviews are also documented consistent with the recommendations of RG 8.8.

The plant design was reviewed, updated, and modified as necessary during the design and construction phases. The plant design integrates the layout, shielding, ventilation, and monitoring designs with traffic control, security, access control, and health physics aspects to ensure that the overall design produces a facility conducive to maintaining low personnel exposures.

Inspection and testing of plant shielding was conducted during startup to verify that the shielding performs its function of reducing radiation to design levels. During initial power operations, radiation surveys were conducted to ensure that there were no defects in the shielding that might seriously affect personnel exposures during normal operation and maintenance of the plant.

12.1.1.3 Operation Policies. An ALARA policy, as indicated in 12.1.1.1, has been promulgated by the President & Chief Executive Officer. That policy is implemented primarily through plant procedures. These procedures incorporate the requirements of 10CFR19 and 10CFR20 and, as appropriate, the guidance of RGs 8.10 and 8.13. ALARA policy is further disseminated by training programs, maintenance, and operating procedures.

STPEGS supervision and management are responsible for assuring that the radiological practices of company employees and contractors comply with plant procedures and that exposures are maintained ALARA. Health Physics supervisors and technicians have been delegated the authority to halt any operation which, in their judgment, is not being performed in accordance with established radiological controls. They are also charged with promptly notifying the Health Physics Manager of any unsafe practices which exceed their authority to correct.

Training programs for plant personnel and contractors were initiated prior to fuel load. Employees and contractors who work in radiologically controlled areas receive instruction in and are tested on the principles of health physics commensurate with the safe performance of their jobs.

All written procedures are screened as part of their 10CFR50.59 review to determine whether a formal ALARA evaluation is required in accordance with station guidelines. Similar screening, with subsequent ALARA evaluation as necessary, is applied to system and plant modifications to ensure that the ALARA philosophy is translated into implementing actions.

12.1.2 Design Considerations

During the design phase prior to and during construction, design considerations to ensure that occupational radiation exposure is ALARA are based upon the guidance provided by RG 8.8, Rev. 1, and the collective experience of the Nuclear Steam Supply System (NSSS) vendor engineers and designers. This experience resulted from participation in the design and operation of several nuclear power plants. In addition to design requirements specified by the NSSS vendor, additional direction in the form of NRC regulations, RGs, and industry standards was distributed to each design discipline. Radiation protection specialists were available to the designers for discussions and guidance on specific questions associated with the reduction of radiation exposure and radioactive releases. The NSSS vendor reviewed published accounts of operating experience and other information to determine design features which could be applied to this facility to prevent problems identified elsewhere. Utility review of engineering documents, such as drawings, system design descriptions, and specifications defining components, layout, or operation of systems containing or controlling radioactivity, were performed to ensure that occupational radiation exposures are ALARA.

12.1.2.1 Facility. The Reactor Containment Building (RCB), Fuel Handling Building (FHB), and Mechanical Auxiliary Building (MAB) were designed to ensure that during routine operations, personnel are shielded from significant radiation sources. Every effort was made to route radioactive piping within radioactive pipe chases. Where possible, multiple radiation sources were located in individually shielded cubicles. Adequate space for ease of maintenance and other operations was provided which permits the tasks to be completed more quickly, thereby reducing the length of exposure. Platforms are provided, where practicable, to avoid the need for temporary scaffolding in radiation areas.

12.1.2.2 Equipment. Materials used to fabricate components of the Reactor Coolant System (RCS) and steam generator (SG) tubes were selected by the NSSS vendor. Specifications for these materials, and other reactor components, restrict the amount of cobalt contained in the base metal which reduces the amount of cobalt-60 formed as a corrosion product. In addition, the materials were selected to minimize corrosion and maintenance requirements.

The equipment associated with systems containing radiation sources was designed to reduce the need for maintenance and other operations in radiation fields. In accordance with the NSSS vendor's design recommendations, corrosion- and maintenance-free and long-wearing materials are specified for systems handling radioactive fluids.

Personnel radiation exposure is greatly reduced by the provision of remote operation capability and by reducing the time required for work in radiation fields. Remote valve operation was provided in radioactive systems, allowing the operator to remain in a safe area during valve manipulation. Similarly, remote inspection techniques are considered for components that contain radioactive crud or which have induced radioactivity. In either case, the operator performs the actual inspection while isolated from the high-radiation fields.

When maintenance or replacement of radioactive equipment is required, the systems were designed, when possible, so that they may be isolated, drained, and flushed to reduce the contained residual activity.

12.1.3 Operational Considerations

The radiation exposures of plant personnel are kept ALARA by means of the Health Physics program discussed in Section 12.5. The radiation protection principles and practices contained therein are promulgated through the training program discussed in Section 13.2, through the STPEGS Radiation Protection Procedures discussed in Section 12.5, and through operating and maintenance procedures.

Procedures for radiation exposure-related operations (such as maintenance, inservice inspections, radwaste handling, and refueling) that routinely occur at an operating pressurized water reactor (PWR) were written and approved for both units prior to their initial fuel loading. Radiation-related procedures were reviewed by Health Physics Division personnel to ascertain that ALARA concepts had been included.

For unusual or first-time operations that involve significant radiation exposure, procedures are normally prepared by or with the assistance of the group doing the work and then reviewed by the Health Physics Division. The Plant Manager has final responsibility at the plant to see that the ALARA concept has been taken into consideration.

Based upon the experience gained during operation, information learned from other utilities and periodic reviews, procedures are revised for improved ALARA performance.

Some of the common dose reduction techniques that are used when working with systems that contain, collect, store, or transport radioactive gases, liquids, or solids are discussed in Section 12.5.3. Approved procedures incorporate these techniques where applicable. Such techniques are normally followed unless inconsistent with ALARA.

12.1.3.1 Design Features for Dose Reduction. Several design features are incorporated to reduce radiation exposure associated with operation and maintenance of systems processing and conveying contaminated fluids as a result of corrosion product buildups. The following are examples of such features included in the STPEGS design.

1. To reduce the corrosion rate in the RCS, the dissolved oxygen concentration and pH are maintained within prescribed limits. Makeup water from the reactor makeup water storage tank (RMWST) to the volume control tank (VCT) is monitored for dissolved oxygen and pH.

2. The STPEGS design includes improved refueling features that reduce the total time required for refueling and result in a reduction in operational exposures during refueling. These features are described in Section 9.1.4.

3. Permanent shielding is provided, so that workers may stay behind walls or in areas of lower radiation level when not actively involved in work in radiation areas. For some jobs, temporary shielding, such as lead blankets is used. Temporary shielding will be used if the total exposure, which includes the exposure received during installation and removal of shielding, is consistent with ALARA.

4. To facilitate decontamination, the walls of rooms which have the potential to become radioactively contaminated are painted with suitable smooth surface coatings. These coatings are applied to the walls to a minimum of 40 inches above the floor surface which is also coated in these areas.

5. Lighting fixtures used in high radiation areas are provided with extended life incandescent lamps to reduce maintenance time required for relamping. Where possible, each room or area has at least two fixtures which are actuated from outside the room or area, or from a low radiation area to minimize radiation exposure.

12.1.3.2 ALARA Considerations for Steam Generator Repairs. Consistent with ALARA considerations, SG repairs are made using state of the art methods, thus reducing time spent by personnel inside the channel heads.

12.1.3.3 Specific ALARA Considerations for Reactor Head Removal and Installation. Applicable techniques in Section 12.5.3.2 normally are used when removing and installing the reactor head and performing the other associated activities necessary for refueling. The rapid refueling system significantly reduces personnel exposure. This system allows the rapid disengagement of the reactor studs, which in the past has caused considerable exposure to personnel, to be done on a remote basis. Also, with this system it is not necessary to disconnect the electrical cables to the reactor head, which saves time and exposure.

A permanent welded seal ring, rather than a bolted one, is used to seal the cavity prior to flooding, thus reducing both direct exposure and airborne activity during installation. Because of the above considerations, time spent in the refueling cavity has become minimal. When jobs of long duration are required, however, communications are provided up to the refueling floor, thus reducing lost time and exposure to personnel. When jobs of long duration are to be performed in areas with high radiation levels, temporary shielding is provided for personnel protection consistent with ALARA.

12.1.3.4 Specific ALARA Considerations for Inservice Inspections. Applicable techniques in Section 12.5.3.2 are normally used when performing inservice inspections (ISIs). Where feasible, remote testing devices are used. An example of this is the remote eddy current probe positioner and probe indexer used to examine the SG tubes. Other remotely operated devices are typically utilized to perform inspections consistent with ALARA.

12.1.3.5 Specific ALARA Considerations for Other Exposure-Related Jobs. Other operations such as refueling, radwaste handling, spent fuel handling, loading and shipping, routine maintenance, sampling, and calibration are discussed in Section 12.5.3. Various combinations of the techniques in Section 12.5.3.2, as applicable to the particular job, are used on these jobs.

12.1.3.6 Similar Operational Considerations at Other Facilities. Many of the policies and practices described in Sections 12.1.3.2 through 12.1.3.5 have been used with success at San Onofre Unit 1, Yankee Atomic, Nine Mile Point, Connecticut Yankee, and military nuclear power plants. The effectiveness of these and similar operating practices and procedures at other facilities has been demonstrated.

12.1.3.7 ALARA Considerations for Decommissioning Operations. The majority of the features included in the STPEGS design for maintenance of operational radiation exposures to ALARA also assist in maintaining radiation exposures to ALARA during decommissioning operations. Examples of these features are:

- Crud control
- Isolation and decontamination
- Material Selection
- Access and space requirements
- Equipment Selection
- Provisions for in-place decontamination

All of these features are closely interrelated and the inclusion of any one feature also provides for one or more of the other listed features.

Reduction of crud deposits during operation will reduce radiation exposures during decommissioning operations. In the STPEGS design, crud control is maintained by system arrangement, material selection, and equipment design.

Isolation and decontamination of system components is provided for in the STPEGS design. As shown in the facility arrangement drawings listed as Figures 1.2-12 through 1.2-46 in Table 1.2-1, maximum separation of radiation sources is provided in the RCB, MAB, and FHB. Separation of tanks, pumps, and valve rooms from each other will reduce radiation exposures during decommissioning operations by reducing radiation exposure from adjacent radiation sources. Provisions are included in the design of process systems to isolate equipment and to drain, flush, and decontaminate the system components in order to reduce radioactivity levels prior to decommissioning.

The installed decontamination facility in the MAB provides equipment necessary for the decontamination of plant components in a manner that allows control of access to the area, control of decontamination chemicals, and ease of decontamination. These features result in a reduction of radiation exposures during decommissioning.

Access and space requirements are considered in the STPEGS design, and adequate space is provided for access during operation and maintenance. These provisions, in addition to allowing ease for access during operation, also allow ease of access and adequate space for removal of system components. The use of monorails installed during construction to assist in removal and installation of equipment reduces radiation exposure during maintenance and decommissioning. Installed platforms in the RCB provided for SG and reactor coolant pump maintenance will also aid decommissioning operations and reduce radiation exposure at that time.

Equipment design features considered in the equipment selection process are those design features that provide for low corrosion rates, minimum crud-collection pockets, draining and flushing, and ease of removal for maintenance and replacement. These features, in addition to providing lower operational radiation exposure, also reduce radiation exposure during equipment removal at the time of decommissioning.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

The shielding design source terms are based upon the three general plant conditions of normal full-power operation, shutdown, and design bases events.

12.2.1.1 Sources for Normal Full Power Operation.

12.2.1.1.1 Reactor Core: For the normal full power condition, a fission rate sufficient to sustain a power production of 4,100 MWt was used for the 14-foot-high active core to determine the neutron source strength. The radial source strength variation was based upon information provided in Chapter 4. The prompt and short-lived fission gamma sources were also calculated for the 4,100 MWt core using data from Reference 12.2.1-1.

Irradiated incore detector and drive cable source strengths are used in determining shielding requirements and area radiation levels. These source strengths can be found in Table 12.2.1-3A.

12.2.1.1.2 General, Non-Core Sources: The main sources of activity outside the primary shield complex during normal full power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. All shielding includes, as a design basis, the maximum case of clad defects in fuel rods producing 1.0 percent of a core thermal power of 4,100 MWt (Section 12.3.2.2.2).

Each plant system is shielded according to the amount of activity present and the adjacent zoning and access criteria. These systems include the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Boron Recycle System (BRS), Spent Fuel Pool Cooling and Cleanup System (SFPCCS), and the Condensate Polishing System. Those sources that are contained in equipment of the Solid, Liquid, and Gaseous Waste Processing Systems and the RCS Vacuum Degassing System are described in Chapter 11. The following sections give the activities or radiation source terms in the reactor coolant, in the auxiliary system process streams, and in the various plant components.

12.2.1.1.3 Reactor Coolant: The N-16 activity of the coolant in the primary loop compartments is the controlling radiation source in the design of the RCS secondary shielding and is given in Table 12.2.1-1 as a function of transport time in a reactor coolant loop (RCL). The reactor coolant fission and corrosion product activities are given in Table 11.1-2.

12.2.1.1.4 Steam Generators and Pressurizer: The steam generators (SGs) are located within the secondary shield in the Reactor Containment Building (RCB). The principal activity in the SG during operation is the N-16 activity given in Table 12.2.1-1. The deposited fission and corrosion product activity is important during shutdown and is discussed in Section 12.2.1.2.

The pressurizer is located in a shield cubicle just outside the secondary shield in the RCB. In addition, the pressurizer relief tank (PRT) is located just below the pressurizer. The radiation sources in the pressurizer and the PRT are tabulated in Tables 12.2.1-2A and 12.2.1-2B.

12.2.1.1.5 Chemical and Volume Control System: One of the functions of the CVCS is to provide purification of the reactor coolant.

The major equipment items include regenerative and letdown heat exchangers (HXs), mixed-bed and cation-bed demineralizers, reactor coolant filter, volume control tank (VCT), and charging pumps. The Boron Thermal Regeneration (BTR) Subsystem includes three BTR HXs and the BTR demineralizers. The Seal Water Subsystem for the reactor coolant pumps (RCPs) includes the injection and return filters and the seal water HX.

Since the CVCS processes reactor coolant, the activity in the coolant going into the system (the letdown flow) is that given in Table 11.1-2. Table 12.2.1-4 gives a summation of this source by energy group. The contribution to this source from the N-16 isotope is not included because sufficient delay time is provided by the routing of the letdown line.

The radiation sources in the demineralizers, filters, VCT, and HXs of the CVCS are given in Table 12.2.1-5. The source volume in each piece of equipment used in the shielding analysis is given in parentheses. A brief discussion of the location, function, and source of activity in each component is given in the following paragraphs.

All of the CVCS demeralizers and filters are clustered in one general area in the Mechanical Auxiliary Building (MAB) at the 41-ft level.

The mixed-bed demineralizers retain the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation-bed demineralizers are used for pH control and supplement the mixed bed in removing yttrium, cesium, molybdenum, and the crud metals.

The specific source strengths of the reactor coolant, seal water return, seal water injection filters, and letdown filters are included in Table 12.2.1-5. The source for the reactor coolant filter corresponds to a deep dose equivalent rate of 500 rem/hr contact. The source for the remaining filters corresponds to a deep dose equivalent rate of 100 rem/hr contact. The filters were assumed to be drained of process fluid and were considered to be homogeneous sources.

The BTR beds' function is to regulate the boron concentration in the reactor coolant. They may be utilized during load follow operations and in removing boron from the coolant as the nuclear fuel is depleted. These demineralizers also collect radioactive anions, such as iodine, which may have passed through the mixed bed. The specific source strengths for these beds can be found in Table 12.2.1-5.

The VCT is located in the MAB on the 41-ft level. The sources (Table 12.2.1-5) in the VCT correspond to a nominal operating level in the tank of 300 ft^3 in the liquid phase and 300 ft^3 in the vapor phase.

The regenerative and excess letdown HXs are located in the RCB on El. 37 ft-3 in. and 52 ft, respectively. They provide the initial cooling for the reactor coolant letdown, and their sources (Table 12.2.1-5) include N-16 activity. The balance of the CVCS HXs are located in the MAB, where N-16 activity is not a significant factor. The letdown HX, located on the 10-ft level of the MAB, provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The activity at this point is that of reactor coolant letdown, but it does not include the contribution from the N-16 isotope. Sufficient delay time is provided in the letdown line to make the N-16 source insignificant at the letdown HX. The specific source strengths can be found in Table 12.2.1-5.

The BTR HXs include the moderating, letdown chiller, and letdown reheat units, which are all located on the 10-ft level of the MAB. The radiation sources (Table 12.2.1-5) in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

12.2.1.1.6 Boron Recycle System: The radiation sources in the BRS are listed in Table 12.2.1-6. The source volume in each piece of equipment used in the shielding analysis is given in parentheses. The major equipment items included in this system are the recycle holdup tanks (RHTs) and the recycle evaporator with its associated equipment, i.e., feed demineralizers and filter, condensate demineralizer and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

All of the BRS demineralizers and filters are clustered in one general area on the 41-ft level of the MAB.

The evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed-bed resins to remove nongaseous activity from the reactor coolant directed to the holdup tanks.

The evaporator condensate demineralizer function is to remove any boron and iodine activity which may be carried over with the evaporator condensate.

The RHTs, located on the 10-ft level of the MAB, are each equipped with a diaphragm to retain gases which flash from the reactor coolant letdown. Periodically, the gases are vented to the Gaseous Waste Processing System (GWPS). The radiation sources in the holdup tanks are based upon 50 percent of the gaseous activity flashing into the vapor phase.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers. Their design function is to retain particulates and any resin fines which may escape from the demineralizers.

The maximum radiation sources on these filters are given in Table 12.2.1-6. The sources for the feed filter correspond to a deep dose equivalent rate of 100 rem/hr contact. The condensate filter sources result in deep dose equivalent rates of less than 1 rem/hr contact. The resultant radiation sources on the concentrates filter correspond to a deep dose equivalent rate of approximately 10 rem/hr contact.

12.2.1.1.7 Spent Fuel Pool Cooling and Cleanup System: The radiation sources in the SFPCCS are given in Table 12.2.1-7. The SFPCCS demineralizers and filters are located at El. 51 ft-6 in. of the MAB. The demineralizers and filters design function is to maintain water clarity and remove activity released during refueling operations and the subsequent fuel cooling period. The filter sources correspond to a deep dose equivalent rate of 100 rem/hr at contact. The SFPCCS pumps and HXs are located at El. 21 ft-11 in., and 42 ft-6 in., respectively, of the Fuel Handling Building.

12.2.1.1.8 Condensate Polishing System: The design bases radiation sources for the Condensate Polishing System are given in Table 12.2.1-8. The sources are based on the removal of radioactive contaminants by the demineralizer beds assuming 1 gal/min primary-to-secondary SG leakage, 1 percent fuel clad defects, and a demineralizer run time of 42 days for the mixed beds and three days for the cation beds.

The demineralizers and cation and anion regeneration tanks are located at El. 29-ft of the Turbine Generator Building (TGB). There are three high total dissolved solids (TDS) tanks, one for the cation-bed demineralizers and two for the mixed bed demineralizers. Each tank is assumed to contain one batch of regeneration solution. Ninety percent of the activity in the regeneration tanks is assumed to pass to the high TDS tanks with the rest going to the low TDS tank. The high TDS tanks are in the yard near the TGB.

12.2.1.2 Sources for Shutdown Conditions. The core gamma sources (after shutdown) are used to establish radiation shielding requirements during refueling operations and during shipment of spent fuel. The bases for the core average source strengths is a three-region equilibrium cycle core at end-of-life (Section 4.3.2). These source strengths, per unit volume of homogenized core, are tabulated in Table 12.2.1-3B for various times after shutdown.

The deposited corrosion product activities as a function of time for the RCS are given in Table 12.2.1-9.

An additional system requiring shielding is the Residual Heat Removal (RHR) System. The specific source strengths in the RHR loop are given in Table 12.2.1-10. The sources are maximum values with credit taken for 8 hours of activity decay and purification.

The Old Steam Generator Storage Facility (OSGSF) is a reinforced concrete building which provides long-term, interim storage of and shielding for the old steam generators removed from the Reactor Containment Buildings. The facility is located outside the protected area, but is within the exclusion area and site boundary, as determined from Figure 1.2-3. The shielding analysis for the OSGSF used the maximum measured dose rate of 80 mR/hr, at contact on the outside of the generators, and isotopic surveys of waste samples to conservatively characterize the source inside the generators. The isotopic surveys identified Co-60 as the dominant gamma-emitting isotope for shielding calculations.

The total activity initially stored in the steam generators when they were discharged is estimated at 170 curies per Unit 1 steam generator and 205 curies per Unit 2 steam generator. Using this activity estimate, a typical assay of STP solid active waste, and the decay time for the Unit 1 steam generators, an approximate breakdown of the activity stored in the OSGSF as of October 2002 is presented in Table 12.2.1-11. Table 12.2.1-12 describes the inputs used to estimate the OSGSF curie content, including the isotopic characterization.

Because of decay time between the replacement of the Unit 1 steam generators and the Unit 2 steam generators, the curie content of the OSGSF was approximately 940 curies at the time the Unit 2 steam generators were placed in storage.

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12.2.1.3 Sources for Design Bases Events. The sources for the Design Basis Accident (DBA) used to determine the 30-day exposures to control room personnel, the 2-hour exclusion zone boundary dose, and low population zone doses are presented in Sections 6.4 and 15.6.5, respectively.

12.2.1.4 Field Run Process Piping. All piping containing radioactive fluids is shown on piping diagrams and piping composites. Routing of all piping shown on the composite drawings was performed and checked by engineering prior to installation.

12.2.2 Airborne Radioactive Material Sources

This section deals with the models, parameters, and sources required to evaluate airborne concentrations of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected.

Leakage sources are dependent upon the concentrations of radionuclides in the primary system, secondary system, spent fuel pool, and the refueling pool. The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various applicable regions are listed in Table 12.2.2-1. The CVCS and the SFPCCS were designed to purify reactor coolant through ion exchangers after reactor shutdown and cooldown. This ensures that the effect of activity spikes will not significantly contribute to the Containment airborne activity during refueling operations. The contribution to airborne activity due to reactor vessel head removal is considered negligible as the RCS Vacuum Degassing System (Section 11.3) is designed to remove this activity prior to head removal. The detailed listing of the expected airborne isotopic concentrations in typical accessible regions is presented in Table 12.2.2-2. The final design of the plant ensures that the expected airborne isotopic concentrations in the typical accessible regions do not contribute significantly to the total effective dose equivalent of the work force.

12.2.2.1 Model for Calculating Airborne Concentrations.

Plant areas with airborne radioactivity are characterized by a constant leak rate of a radioactive source at a constant source strength with a constant exhaust rate of the contaminant. This leads to a peak or equilibrium airborne concentration of the radioisotope in the regions as calculated by the following equation:

$$C_i(t) = (LR)_i \, A_i \, (PF)_i \, \frac{\left(l - e^{-\lambda_{Ti} t}\right)}{\left(V\lambda_{Ti}\right)} \qquad \text{(Eq. 12.2.2-1)}$$

where:

$(LR)_i$ = leak or evaporation rate of the ith radioisotope in g/s, in the applicable region,

and

A_i = activity concentration of the ith leaking or evaporating radioisotope in μCi/g

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the ith radioisotope

λ_{Ti} = total removal rate constant for the ith radioisotope in sec^{-1} from the applicable region

= $(\lambda_{di} + \lambda_e)$

(λ_{di} and λ_e are the removal rate constants in sec^{-1} due to radioactive decay and the exhaust from the applicable region respectively for the i^{th} radioisotope)

t = time interval between the start of the leak and the time at which the concentration is evaluated in seconds

V = free volume of the region in which the leak occurs in cm^3

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in $\mu Ci/cm^3$ in the applicable region

From the above equation, it is evident that the peak or equilibrium concentration, $C_{Eq,i}$ of the i^{th} radioisotope in the applicable region will be given by the following expression:

$$C_{Eq,i} = (LR)_i\, A_i\, (PF)_i \,/\, (V\lambda_{Ti}) \qquad \text{(Eq. 12.2.2-2)}$$

With high exhaust rates, this peak concentration will be reached within a few hours.

12.2.3 The Impact of Extended Burnup and VANTAGE 5H Fuel on Source Terms

The source terms presented in Sections 12.2.1 and 12.2.2 are based on an equilibrium fuel cycle using discharge burnup of 33,000 MWD/MTU. The use of extended burnup fuel at STPEGS has been reviewed in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (References 11.1-4, 11.1-5) and has been determined to not significantly change the results previously presented in safety analysis reports based on operation to 33,000 MWD/MTU discharge burnup.

For VANTAGE 5H fuel, source terms based on an equilibrium fuel cycle using batch average burnups of 20,000 MWD/MTU, 40,000 MWD/MTU, and 60,000 MWD/MTU (each at 1/3 core size) with fuel enriched to a nominal 5.0 w/o U-235 have been evaluated. The results do not significantly change the results previously presented.

REFERENCES

Section 12.2:

12.2.1-1 DLC-75/Bugle-80 Coupled 47-neutron, 20-Gamma Ray-Group, P3, Cross Section Library for LWR Shielding Calculation

TABLE 12.2.1-1

REACTOR COOLANT N-16 ACTIVITY

Position in Loop	Loop Transit Time (sec)	N-16 Activity (μCi/g)
Leaving core	0	206
Leaving reactor vessel	1.1	185
Entering steam generator	1.5	179
Leaving steam generator	7.0	104
Entering reactor coolant pump	7.6	98
Entering reactor vessel	8.6	89
Entering core	10.8	80
Leaving core	11.7	206

TABLE 12.2.1-2A

PRESSURIZER SOURCES FOR SHIELD DESIGN

	Pressurizer Sources		
Photon Energy (Mev)	Liquid Phase* 3.6×10^7 cm^3 (Mev/g-sec)	Steam Phase* 2.4×10^7 cm^3 (Mev/cm^3-sec)	Liquid N-16 (Mev/g-sec)
0.4	4.4×10^5	7.4×10^5	---
0.9	6.3×10^5	3.9×10^4	---
1.35	2.8×10^5	1.1×10^4	---
1.8	1.7×10^5	3.2×10^4	9.5×10^2
2.2	1.7×10^5	5.4×10^4	---
2.6	1.8×10^5	1.1×10^5	---
3.0	3.1×10^4	6.9×10^2	8.7×10^3
4.0	9.1×10^3	4.6×10^2	---
7.0	1.8×10^6	2.4×10^2	1.8×10^6
7.5	1.5×10^5	2.0×10^1	1.5×10^5

* Includes N-16 sources

TABLE 12.2.1-2B

PRESSURIZER RELIEF TANK SOURCE

Photon Energy (Mev)	Liquid Phase 4.4×10^7 cm^3 (Mev/cm^3-sec)	Gas 1.5×10^7 cm^3 (Mev/cm^3-sec)
0.4	1.7×10^2	1.0×10^6
0.9	1.7×10^3	6.5×10^3
1.35	6.3×10^2	9.7×10^2
1.80	3.7×10^2	2.9×10^3
2.20	4.3×10^1	5.1×10^3
2.60	1.6×10^1	1.1×10^4
3.0	---	5.4×10^1
4.0	---	1.8×10^1

TABLE 12.2.1-3A

INCORE DETECTOR AND DRIVE CABLE
MAXIMUM SOURCE STRENGTHS

Energy Group (Mev)	Incore Detector (Mev/cm^3-sec)	Drive Cable (Mev/cm^3-sec)
0.4	3.8×10^{10}	6.0×10^{8}
0.9	1.6×10^{11}	5.1×10^{10}
1.35	1.1×10^{11}	1.6×10^{10}
1.8	1.1×10^{11}	3.1×10^{8}
2.2	2.9×10^{10}	3.8×10^{10}
2.6	3.1×10^{10}	1.3×10^{9}
3.0	1.6×10^{10}	1.3×10^{9}
4.0	2.1×10^{10}	3.1×10^{8}
5.0	1.5×10^{10}	---
6.0	1.4×10^{9}	---

TABLE 12.2.1-3B

CORE SHUTDOWN SOURCES

Photon Energy (Mev)	Time after Shutdown*				
	12 Hours	1 Day	1 Week	1 Month	3 Months
0.4	1.9×10^{11}	1.6×10^{11}	6.2×10^{10}	1.5×10^{10}	5.0×10^{9}
0.9	1.2×10^{12}	1.0×10^{12}	5.7×10^{11}	3.6×10^{11}	1.9×10^{11}
1.35	2.0×10^{11}	1.3×10^{11}	4.5×10^{10}	1.3×10^{10}	3.2×10^{9}
1.8	4.1×10^{11}	3.6×10^{11}	2.5×10^{11}	7.1×10^{10}	4.1×10^{9}
2.2	2.8×10^{10}	1.9×10^{10}	9.9×10^{9}	4.0×10^{9}	1.7×10^{9}
2.6	2.7×10^{10}	2.0×10^{10}	1.5×10^{10}	4.3×10^{9}	1.6×10^{8}
3.0	7.0×10^{8}	3.6×10^{8}	2.6×10^{8}	7.4×10^{7}	2.9×10^{7}
4.0	4.8×10^{8}	1.4×10^{8}	1.0×10^{8}	2.9×10^{7}	1.1×10^{6}

* All sources in units of Mev/cm^3-sec of homogenized core

TABLE 12.2.1-4

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Letdown Coolant Sources

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/g-sec)
0.4	4.4×10^5
0.9	6.3×10^5
1.35	2.8×10^5
1.8	1.7×10^5
2.2	1.7×10^5
2.6	1.8×10^5
3.0	2.2×10^4
4.0	9.1×10^3

TABLE 12.2.1-5

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Mixed-Bed Demineralizer
(2.1×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	8.1×10^7
0.9	3.7×10^8
1.35	3.7×10^7
1.8	1.3×10^7
2.2	7.3×10^5
2.6	4.2×10^5
3.0	1.0×10^5
4.0	3.1×10^4

Cation-Bed Demineralizer
(2.1×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	3.5×10^6
0.9	3.1×10^8
1.35	2.1×10^7
1.8	7.5×10^6
2.2	3.0×10^5
2.6	1.0×10^5
3.0	1.0×10^5
4.0	2.4×10^4

TABLE 12.2.1-5 (Continued)

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Boron Thermal Regeneration Demineralizers
(2.5×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	2.6×10^6
0.9	2.0×10^6
1.35	6.8×10^5
1.8	3.9×10^5
2.2	3.5×10^4
2.6	2.3×10^4

Reactor Coolant Filter
(1.1×10^4 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.9	5.7×10^7
1.35	1.5×10^7

Seal Water Injection Filter
(2.0×10^3 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.9	1.4×10^8
1.35	2.8×10^7

TABLE 12.2.1-5 (Continued)

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Letdown Filter and Seal Water Return Filter
(1.1×10^4 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.9	1.1×10^7
1.35	3.0×10^6

Volume Control Tank Vapor Phase
(8.5×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	5.1×10^6
0.9	4.3×10^5
1.35	1.0×10^5
1.8	3.0×10^5
2.2	5.3×10^5
2.6	1.1×10^6
3.0	6.1×10^3
4.0	3.0×10^3

TABLE 12.2.1-5 (Continued)

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Liquid Phase
(8.5×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/q-sec)
0.4	3.7×10^5
0.9	3.1×10^5
1.35	1.6×10^5
1.8	6.8×10^4
2.2	1.1×10^5
2.6	6.6×10^4
3.0	2.1×10^4
4.0	6.3×10^3

Regenerative HX (9.4×10^7 cm^3), Tubes
Regenerative HX (3.5×10^5 cm^3), Shell
Excess Letdown HX (2.4×10^4 cm^3), Tubes
Specific Source Strength Mev/q-sec

Gamma Energy (Mev/γ)	Regenerative – Shell Excess Letdown – Tubes	Regenerative -Tubes
0.4	4.4×10^5	3.7×10^5
0.9	6.3×10^5	3.1×10^5
1.35	2.8×10^5	1.6×10^5
1.8	1.7×10^5	6.8×10^4
2.2	1.7×10^5	1.1×10^5
2.6	1.8×10^5	6.6×10^4
3.0	1.0×10^5	2.1×10^4
4.0	9.1×10^3	6.3×10^3
7.0	1.6×10^7	----
7.5	1.3×10^6	----

TABLE 12.2.1-5 (Continued)

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Letdown HX (5.7×10^5 cm^3), Tubes
Seal Water HX (1.6×10^5 cm^3), Tubes
Specific Sources Strength, Mev/g-sec

Gamma Energy (Mev/γ)	Letdown HX – Tube Side Seal Water HX – Tube Side
0.4	4.4×10^5
0.9	6.3×10^5
1.35	2.8×10^5
1.8	1.7×10^5
2.2	1.7×10^5
2.6	1.8×10^5
3.0	2.2×10^4
4.0	9.1×10^3

Boron Thermal Regeneration System
Moderating HX (1.8×10^6 cm^3)
Letdown Chiller HX (1.8×10^6 cm^3)
Letdown Reheat HX (1.2×10^5 cm^3)

Gamma Energy (Mev/γ)	Specific Source* Strength (Mev/g-sec)	Specific Source** Strength (Mev/g-sec)
0.4	4.1×10^5	4.4×10^5
0.9	3.5×10^5	6.3×10^5
1.35	1.7×10^5	2.8×10^5
1.8	1.0×10^5	1.7×10^5
2.2	1.6×10^5	1.7×10^5

* Moderating HX, Tube and Shell Side
Letdown Chiller HX, Tube Side
Letdown Reheat HX, Shell Side

** Letdown Reheat HX, Tube Side

TABLE 12.2.1-5 (Continued)

RADIATION SOURCES
CHEMICAL AND VOLUME CONTROL SYSTEM

Boron Thermal Regeneration System
Moderating HX (1.8×10^6 cm^3)
Letdown Chiller HX (1.8×10^6 cm^3)
Letdown Reheat HX (1.2×10^5 cm^3)

(Continued)

Gamma Energy (Mev/γ)	Specific Source* Strength (Mev/g-sec)	Specific Source** Strength (Mev/g-sec)
2.6	1.7×10^5	1.8×10^5
3.0	2.2×10^4	2.2×10^4
4.0	8.3×10^3	9.1×10^3

* Moderating HX, Tube and Shell Side
Letdown Chiller HX, Tube Side
Letdown Reheat HX, Shell Side

** Letdown Reheat HX, Tube Side

TABLE 12.2.1-6

RADIATION SOURCES
BORON RECYCLE SYSTEM

Evaporator Feed Demineralizer
(2.1×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	4.2×10^6
0.9	1.5×10^8
1.35	2.0×10^7
1.8	3.6×10^6
2.2	2.1×10^4
2.6	1.4×10^4

Recycle Evaporator Condensate Demineralizer
(8.5×10^5 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	3.0×10^4
0.9	2.0×10^4
1.35	4.0×10^3
1.8	1.9×10^3
2.2	1.7×10^2
2.6	1.1×10^2

TABLE 12.2.1-6 (Continued)

RADIATION SOURCES
BORON RECYCLE SYSTEM

Recycle Holdup Tanks
Vapor Phase
(7×10^6 cm^3, each tank)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.6×10^6
0.9	2.2×10^5
1.35	6.9×10^4
1.8	2.1×10^5
2.2	3.5×10^5
2.6	6.6×10^5
3.0	6.9×10^3
4.0	9.1×10^3

Liquid Phase
(3.0×10^8 cm^3, each tank)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	3.8×10^5
0.9	1.2×10^5
1.35	4.3×10^4
1.8	6.2×10^4
2.2	9.2×10^4
2.6	1.6×10^5
3.0	3.7×10^3
4.0	2.9×10^3

TABLE 12.2.1-6 (Continued)

RADIATION SOURCES
BORON RECYCLE SYSTEM

Recycle Evaporator Feed Filter
(1.1×10^4 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.9	1.1×10^7
1.35	3.0×10^6

Recycle Evaporator Condensate Filter
(1.5×10^3 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	3.0×10^4
0.9	2.0×10^4
1.35	4.0×10^3
1.8	1.9×10^3
2.2	1.7×10^2
2.6	1.1×10^2

Recycle Evaporator Concentrates Filter
(1.5×10^3 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.6×10^5
0.9	8.6×10^5
1.35	4.0×10^5
1.8	4.6×10^4
2.2	3.5×10^3
2.6	2.1×10^3

TABLE 12.2.1-6 (Continued)

RADIATION SOURCES
BORON RECYCLE SYSTEM

Recycle Evaporator Vent Condenser Vapor
(1.1×10^5 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.2×10^7
0.9	1.8×10^6
1.35	5.4×10^5
1.8	1.7×10^6
2.2	2.7×10^6
2.6	5.2×10^6
3.0	5.5×10^4
4.0	7.2×10^4

Evaporator Concentrates
(2.1×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.6×10^5
0.9	8.6×10^5
1.35	4.0×10^5
1.8	4.6×10^4
2.2	3.5×10^3
2.6	2.1×10^3

TABLE 12.2.1-7

RADIATION SOURCES
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Demineralizer
(2.1×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.0×10^4
0.9	6.5×10^5
1.35	3.0×10^5
1.8	5.9×10^3

Spent Fuel Pool and Spent Fuel Pool Skimmer Filter
(1.1×10^4 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.9	1.1×10^7
1.35	3.0×10^6

TABLE 12.2.1-8

RADIATION SOURCES
CONDENSATE POLISHING SYSTEM

Cation Demineralizer (6.0×10^6 cm^3)
Cation Regeneration Tank (6.0×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	4.02×10^2
0.8	2.85×10^3
1.3	6.51×10^2
1.7	9.83×10^1
2.2	1.75×10^1
2.5	2.01×10^0
3.5	4.53×10^{-1}

Mixed Bed Demineralizer (6.0×10^6 cm^3)
Anion Regeneration Tank (6.0×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.53×10^4
0.8	9.60×10^3
1.3	2.27×10^3
1.7	5.42×10^2
2.2	2.92×10^2
2.5	2.86×10^1
3.5	5.84×10^{-1}

TABLE 12.2.1-8 (Continued)

RADIATION SOURCES
CONDENSATE POLISHING SYSTEM

Cation High Total Dissolved Solids Tank (5.2×10^7 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	4.61×10^1
0.8	3.28×10^2
1.3	7.46×10^1
1.7	1.13×10^1
2.2	2.01×10^0
2.5	2.31×10^{-1}
3.5	5.21×10^{-2}

Mixed Bed High Total Dissolved Solids Tank (8.9×10^7 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	1.03×10^3
0.8	6.45×10^2
1.3	1.52×10^2
1.7	3.63×10^1
2.2	1.96×10^1
2.5	1.90×10^0
3.5	3.90×10^{-2}

TABLE 12.2.1-9

DEPOSITED CORROSION PRODUCT ACTIVITY
IN THE REACTOR COOLANT SYSTEM

Isotope	Operating Time, Years			
	1*	2*	5*	10*
Mn-54	1.0	1.1	1.3	1.4
Co-58	12.0	12.0	12.0	12.0
Fe-59	0.5	0.5	0.5	0.5
Co-60	1.5	2.3	4.0	6.0

* All units in μCi/mg

TABLE 12.2.1-10

RADIATION SOURCES
RESIDUAL HEAT REMOVAL SYSTEM

RHR Heat Exchanger
(2.6×10^6 cm^3)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cm^3-sec)
0.4	3.0×10^5
0.9	8.3×10^4
1.35	3.6×10^4
1.8	7.9×10^3
2.2	6.3×10^3
2.6	6.4×10^3
3.0	4.7×10^2
4.0	1.8×10^2

TABLE 12.2.1-11

ISOTOPE INVENTORY – OLD STEAM GENERATOR STORAGE FACILITY
(Historical Information as of October 2002

CN-3072

Isotope	Estimated Curies In OSGSF
Cr-51	1. 63E+02
Mn-54	9.30E+00
Fe-55	2.22E+02
Co-58	3.44E+02
Co-60	9.02E+01
Ni-63	4.00E+01
Nb-95	1.64E+01
Zr-95	1.32E+01
Sb-124	4.20E+01
Total	940 Ci

TABLE 12.2.1-12

PARAMETERS AND ASSUMPTIONS USED FOR CALCULATING OSGSF CURIE CONTENT

Steam Generator Activity
Unit 1 steam generator survey average: 2.85E-7 Ci/cm^2
Unit 2 steam generator survey average: 3.43E-7 Ci/cm^2
Assumed survey efficiency: 10%

Estimated Activity
Unit 1: 2.85E-6 Ci/cm^2
Unit 2: 3.43E-6 Ci/cm^2

Estimated steam generator primary side surface area:
5.93E+7 cm^2 (Unit 1)
5.91E+7 cm^2 (Unit 2)

Estimated total activity AT DISCHARGE:
Unit 1: 170 Ci
Unit 2: 205 Ci

Isotopic Data

From a May 1996 assay used for replacement steam generator dose and shielding calculations:

Isotope	μCi/sample	% of Total	Decay Constant (λ) day^{-1}
Cr-51	1.6E-1	19.9	2.5023E-02
Mn-54	7.85E-3	1.0	2.2167E-03
Fe-55	1.51E-1	18.8	7.0286E-04
Co-58	3.37E-1	42.0	9.7902E-03
Co-60	5.57E-2	6.9	3.6003E-04
Ni-63	2.13E-2	2.7	1.8958E-05
Nb-95	1.64E-2	2.0	8.0040E-03
Zr-95	1.27E-2	1.6	1.0827E-02
Sb-124	4.07E-2	5.1	1.1514E-02
Total	8.03E-1	100.0	----

Unit 1 steam generators were replaced in March 2000; Unit 2 steam generators were replaced in October 2002.

As of removal ofthe Unit 2 steam generators, the Unit 1 steam generators had approximately 2.5 years of decay.

To approximate the curie count for Table 12.2.1-11, apply the isotopic percentages to the 170 Ci/Unit 1 steam generator estimation and decay the Unit 1 values 2.5 years. Apply the isotopic percentages to the 205 Ci/Unit 2 steam generator estimation. Add 4 times the decayed Unit 1 values to 4 times the Unit 2 values. This represents a reasonable characterization of the Model E steam generator activity in the STP OSGSF as of October 2002.

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TABLE 12.2.2-1

PARAMETERS AND ASSUMPTIONS USED
FOR CALCULATING AIRBORNE RADIOACTIVITY

1. Reactor Containment Building (RCB)

   a. Reactor coolant activity concentrations are listed in Table 11.1-7

   b. Release rates from RCS:

      Noble Gas 1%/day
      Iodines .001%/day
      Particulates .0001%/day

   c. Containment volume is 3.58 x 10^6 ft^3

   d. Purge rates:

      Normal operation – continuous 4,500 scfm (**See Note)
      Prior and during refueling – 40,000 scfm

   e. During operation, 4 hours per week are assumed to be needed for Containment access.

   f. Refueling source term is due to evaporation of the fuel pool.
      (Area = 1827 ft^2, 120°F, Air flow = 75 ft/min

      Evaporation Rate Assumed – 1.25 gal/min

   g. Refueling source term:

      1 μCi/gm – Tritium, 5.9 x 10^{-3} μCi/gm – I-131
      Tritium PF=1, I-131 PF = .01

2. Mechanical Auxiliary Building (MAB)

   a. Reactor coolant activity concentrations are listed in Table 11.1-7

   b. MAB total leakage is assumed to be 160 lb/day. All is assumed to be primary coolant.

   c. Nuclide Partition Factors:

      Noble Gas 1.0
      Halogen .0075
      Particulate .001

** 5,000 scfm was used for assumption

TABLE 12.2.2-1 (Continued)

PARAMETERS AND ASSUMPTIONS USED
FOR CALCULATING AIRBORNE RADIOACTIVITY

d. Two cases are analyzed using the above common assumptions and the following case specific assumptions:

i) General Area: Volume = 2.45×10^6 ft^3
HVAC Flow = 196,300 scfm

ii) Worst Potential Room: Volume = 329 ft^3
HVAC Flow = 100 scfm
(Leakage for this case is taken to be 5% of the 160 lb/day)

3. Turbine Generator Building (TGB)

a. Secondary activity is taken from Table 11.1-7
b. Leakage is taken to be 1,700 lb/hr of secondary steam
c. A partition factor of 1 was assumed for all nuclides
d. TGB volume is 3.01×10^6 ft^3
e. TGB HVAC flow rate was assumed to be 855,000 scfm

4. Fuel Handling Building (FHB)

a. Source of activity is the fuel pool where the following sources are assumed:

1 μCi/gm – Tritium, 5.9×10^{-3} μCi/gm I-131

b. Partition factors are:

Tritium	1.0
I-131	0.01

c. Fuel Pool evaporation rate is assumed to be 1.02 gal/min
(Area = 1,482 ft^2, 120°F, Air flow = 75 ft/min)

d. FHB volume is 1.26×10^6 ft^3

e. FHB HVAC flow rate is 28,260 scfm

TABLE 12.2.2-2

AIRBORNE RADIOACTIVITY CONCENTRATIONS

Nuclide	DAC (µCi/ml)	RCB Normal Conc. (µCi/ml)	RCB Refueling Conc. (µCi/ml)	MAB General Area Concen. (µCi/ml)	Worst MAB Room Conc. (µCi/ml)	TCB 1 Conc. (µCi/ml)	FHB Conc. (µCi/ml)
H-3	$2x10^{-6}$	$5.62x10^{-6}$	$4.2x10^{-6}$	-	-	$5.31x10^{-10}$	$4.83x10^{-6}$
Kr-85M	$6x10^{-5}$	$3.85x10^{-7}$	--	$7.59x10^{-10}$	$7.89x10^{-8}$	$1.05x10^{-14}$	-
Kr-85	$1x10^{-4}$	$5.88x10^{-8}$	--	$4.14x10^{-11}$	$4.21x10^{-9}$	$5.84x10^{-16}$	-
Kr-87	$5x10^{-6}$	$9.69x10^{-8}$	--	$4.64x10^{-10}$	$5.05x10^{-8}$	$6.69x10^{-15}$	-
Kr-88	$2x10^{-6}$	$5.53x10^{-7}$	--	$1.47x10^{-9}$	$1.54x10^{-7}$	$2.04x10^{-14}$	-
Xe-131M	$4x10^{-4}$	$1.73x10^{-7}$	--	$1.26x10^{-10}$	$1.28x10^{-8}$	$1.75x10^{-15}$	-
Xe-133M	$1x10^{-4}$	$8.99x10^{-7}$	-	$7.27x10^{-10}$	$7.41x10^{-8}$	$1.00x10^{-14}$	-
Xe-135	$1x10^{-5}$	$1.68x10^{-6}$	-	$2.21x10^{-9}$	$2.27x10^{-7}$	$3.06x10^{-14}$	-
Xe-133	$1x10^{-4}$	$4.68x10^{-5}$	-	$3.50x10^{-8}$	$3.57x10^{-6}$	$4.83x10^{-13}$	-
Ar-41	$3x10^{-6}$	$2.16x10^{-7}$	-	-	-	-	-
Br-83	$3x10^{-5}$	$1.41x10^{-11}$	-	$3.13x10^{-13}$	$3.30x10^{-11}$	$7.31x10^{-16}$	-
I-131	$2x10^{-8}$	$2.69x10^{-9}$	$2.46x10^{-10}$	$1.48x10^{-11}$	$1.51x10^{-9}$	$1.64x10^{-13}$	$2.84x10^{-10}$
I-132	$3x10^{-6}$	$2.77x10^{-10}$	-	$6.37x10^{-12}$	$6.74x10^{-10}$	$5.22x10^{-14}$	-
I-133	$1x10^{-7}$	$3.11x10^{-9}$	-	$2.28x10^{-11}$	$2.33x10^{-9}$	$1.64x10^{-13}$	-
I-134	$2x10^{-5}$	$6.00x10^{-11}$	-	$2.87x10^{-12}$	$3.21x10^{-10}$	$2.94x10^{-15}$	-
I-135	$7x10^{-7}$	$1.02x10^{-9}$	-	$1.19x10^{-11}$	$1.22x10^{-9}$	$5.22x10^{-14}$	-
Rb-86	$3x10^{-7}$	$8.91x10^{-14}$	-	$6.38x10^{-16}$	$6.50x10^{-14}$	$5.84x10^{-18}$	-
Cs-134	$4x10^{-8}$	$2.68x10^{-11}$	-	$1.89x10^{-13}$	$1.92x10^{-11}$	$1.59x10^{-15}$	-
Cs-136	$3x10^{-7}$	$137x10^{-11}$	-	$9.88x10^{-14}$	$1.00x10^{-11}$	$7.44x10^{-16}$	-
Cs-137	$6x10^{-8}$	$1.91x10^{-11}$	-	$1.35x10^{-13}$	$1.37x10^{-11}$	$1.06x10^{-15}$	-
Cr-51	$8x10^{-6}$	$2.02x10^{-12}$	-	$1.44x10^{-14}$	$1.46x10^{-12}$	$1.06x10^{-16}$	-
Mn-54	$3x10^{-7}$	$3.19x10^{-13}$	-	$2.25x10^{-15}$	$2.28x10^{-13}$	$2.60x10^{-17}$	-
Fe-55	$2x10^{-6}$	$1.66x10^{-12}$	-	$1.17x10^{-14}$	$1.19x10^{-12}$	$9.03x10^{-17}$	-
Fe-59	$1x10^{-7}$	$1.02x10^{-12}$	-	$7.28x10^{-15}$	$7.41x10^{-13}$	$6.37x10^{-17}$	-
Co-58	$3x10^{-7}$	$1.65x10^{-11}$	-	$1.17x10^{-13}$	$1.19x10^{-11}$	$9.03x10^{-16}$	-
Co-60	$1x10^{-8}$	$2.04x10^{-12}$	-	$1.44x10^{-14}$	$1.46x10^{-12}$	$1.16x10^{-16}$	-
Sr-89	$6x10^{-8}$	$3.55x10^{-13}$	-	$2.52x10^{-15}$	$2.56x10^{-13}$	$2.65x10^{-17}$	-
Sr-90	$2x10^{-9}$	$1.03x10^{-14}$	-	$7.28x10^{-17}$	$7.41x10^{-15}$	$6.37x10^{-19}$	-
Sr-91	$1x10^{-6}$	$4.19x10^{-13}$	-	$5.41x10^{-15}$	$5.56x10^{-13}$	$2.01x10^{-17}$	-
Y-90	$3x10^{-7}$	$1.13x10^{-15}$	-	$8.97x10^{-18}$	$9.15x10^{-16}$	$3.55x10^{-19}$	-
Y-91M	$1x10^{-4}$	$4.47x10^{-14}$	-	$2.95x10^{-15}$	$3.31x10^{-13}$	$1.52x10^{-17}$	-
Y-91	$5x10^{-8}$	$7.50x10^{-14}$		$5.31x10^{-16}$	$5.40x10^{-14}$	$3.93x10^{-18}$	-
Y-93	$1x10^{-6}$	$2.25x10^{-14}$	-	$2.84x10^{-16}$	$2.91x10^{-14}$	$1.32x10^{-18}$	-
Zr-95	$1x10^{-7}$	$6.23x10^{-14}$	-	$4.41x10^{-16}$	$4.48x10^{-14}$	$3.93x10^{-18}$	-
Nb-95	$5x10^{-7}$	$5.19x10^{-14}$	-	$3.69x10^{-16}$	$3.75x10^{-14}$	$3.98x10^{-18}$	-
Mo-99	$6x10^{-7}$	$8.07x10^{-11}$	-	$6.37x10^{-13}$	$6.49x10^{-11}$	$5.31x10^{-15}$	-
Tc-99M	$6x10^{-5}$	$5.88x10^{-11}$	-	$4.14x10^{-13}$	$4.21x10^{-11}$	$1.16x10^{-14}$	-
Ru-103	$3x10^{-7}$	$4.69x10^{-14}$	-	$3.33x10^{-16}$	$3.38x10^{-14}$	$2.65x10^{-18}$	-
Ru-106	$5x10^{-9}$	$1.03x10^{-14}$	-	$7.28x10^{-17}$	$7.41x10^{-15}$	$6.37x10^{-19}$	-
DAC Fraction		1.48	0.22	0.002	0.25	$3.7x10^{-5}$	0.26

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The radiation shielding described in Section 12.3.2, and shown in the general arrangement drawings listed as Figures 1.2-12 through 1.2-46, 1.2-60 in Table 1.2-1 and on Figures 12.3.1-1 through 12.3.1-36, protects plant operating personnel and the general public against radiation exposure from in-plant sources of ionizing radiation. In addition to the radiation shielding, the South Texas Project Electric Generating Station (STPEGS) has many facility design features which ensure that occupational radiation exposures are as low as is reasonably achievable (ALARA), in accordance with the guidance given in Regulatory Guide (RG) 8.8. Some of these features are incorporated into the shielding design and are discussed in Section 12.3.2. Examples of other facility design features are given below:

12.3.1.1 Equipment Layout.

1. The routing of all piping containing radioactive fluids is controlled through the use of piping diagrams and piping composite and instrument piping drawings. The routing is established on these drawings, and the piping is installed accordingly. Changes in routing from that shown on the composite drawings are reviewed by engineering and radiation protection specialists prior to installation.

2. Local instrumentation readouts are brought to points outside shielding walls, where the resulting exposures are lowest within considerations dictated by the requirement for closeness of the particular instrument to the associated equipment, component, or process line and by sloping requirements.

3. Permanent platforms with ladders or stairs are used at various heights in numerous areas to facilitate maintenance and inservice inspection (ISI) and to keep occupational exposures ALARA. As an example, platforms are provided below each steam generator (SG) to facilitate access to the primary side manholes. A platform is also provided around each of the reactor coolant pumps (RCPs) to allow work on the pump seals. The pressurizer has a platform near the relief valves and one below the heaters, with other intermediate platforms as required. The equipment itself is oriented so that the expected servicing tasks can be performed efficiently.

4. Radiation damage to equipment is limited through proper materials selection as well as by the use of shielding. Special attention is given to the selection of organic and other radiosensitive materials such as electric cable insulation and connectors, solid-state electronics, gaskets and sealants, seals, packings and diaphragms for valves, lubricants, and tank linings. Where practicable, components containing these materials are located outside shielded cubicles in lower radiation areas.

5. The waste and recycle evaporator designs provide that the arrangement of components requiring maintenance are separated from components that are potentially high radioactive sources.

6. Whenever possible radioactive equipment is located in individually shielded cubicles so that access to a cubicle is not restricted by radiation from an adjacent radioactive piece of equipment.

12.3.1.2 Equipment Design.

1. Liquid filters are designed to be removable from the top using a minimum number of tools. Filter bolt lead-in for tool entry is provided to facilitate remote loosening. Lifting bails are located in the center of the heads to allow the removal of the cartridge assemblies without the requirement for visual observation.

2. The pumps associated with systems that contain radiation sources are equipped with very high quality shaft seals, or a canned pump is used. The canned pumps are designed to prevent any leakage of radioactive fluids.

3. The reactor pressure vessel (RPV) and the RPV thermal insulation are designed for ISI of the RPV welds with a minimum of occupational radiation exposure to personnel. The RPV thermal insulation is a reflective metallic type that is installed in easily handled sections equipped with quick-disconnect fasteners. An optimum design configuration of the RPV is provided to readily accommodate the ultrasonic testing sensor. For example, the RPV nozzle is tapered along the reinforced areas to assure a smooth transition. Other equipment design features that facilitate ISI are discussed in Section 5.2.4.

4. STPEGS is equipped to use the Westinghouse Rapid Refueling System. This system simplifies or eliminates many refueling operations. It features a new technique for rapidly closing the reactor vessel to the vessel head, utilizing closure studs modified with breech-block lugs (Roto-Lok) for attachment to the vessel. The system also includes a single-lift upper package which combines missile shield removal, control rod drive mechanism (CRDM) cooling duct removal, and upper core support structure removal. The Fuel Handling System is described more fully in Section 9.1.4.

5. The radiation protection design features of the STPEGS Gaseous Waste Management System (GWMS) are described in Section 11.3.

6. Process sampling lines leading to the primary and radioactive waste sampling panel are routed through radioactive pipe chases, and local shielding is provided to protect personnel at the sample panel. More detailed information on the Process Sampling System (PSS) is presented in Section 9.3.2.

7. The Mechanical Auxiliary Building (MAB) is designed with a centrally located radioactive pipe chase occupying a large part of the El. 29-ft level. The systems in the MAB are thus able to transfer radioactive fluids from one shielded cubicle to any other shielded cubicle through pipes that are shielded along the entire route.

12.3.1.3 Reduction of Radiation Sources (Where Operations Must Be Performed).

1. The production of activation products in the Reactor Coolant System (RCS) is minimized through the use of corrosion-resistant materials (stainless steel, zirconium, and inconel) and by controlling the oxygen and pH in the reactor coolant. Oxygen is controlled during startup and during power operation. Control of pH is accomplished with additions of lithium hydroxide to the reactor coolant. Compatibility of the reactor coolant with materials which are in contact with the coolant is discussed more fully in Section 5.2.3.

2. The distribution of activation products is reduced by many methods, a few of which are described below.

   a. During normal operation, the Chemical and Volume Control System (CVCS) (Section 9.3.4) continuously purifies the reactor coolant letdown stream by the filtering, demineralizing, and purging of radiogases. In this way, the activity of the reactor coolant is controlled, as is the activity of any leakage. Since essentially all leakage is collected and processed by the waste handling systems (Chapter 11), which are operated and maintained by plant personnel, reduction of reactor coolant activity contributes to minimizing occupational radiation exposures.

   b. The spread of activation products, and the resulting occupational radiation exposures during refueling operations, is reduced by decontamination of the reactor coolant.

   c. The spread of contamination is limited by providing shielded cubicles with a threshold coaming to contain small spills within the cubicle, by painting surfaces for ease of decontamination, and by routing of drains to sumps for disposal (Section 9.3.3). In addition, the tank cubicles are designed to contain radioactive material in the unlikely event of a component rupture. Each component also has been provided with an adequate drainage system.

   d. The Heating, Ventilating, and Air Conditioning (HVAC) Systems are designed to maintain negative pressure in shielded cubicles relative to the surrounding clean areas in order to provide air flow from the clean area to the potentially contaminated area. A more complete description of the Mechanical Auxiliary Building HVAC System may be found in Section 9.4.3.

3. The retention of activation products is reduced by one or more of the following methods:

   Prior to refueling, the reactor coolant is normally oxygenated under controlled conditions while CVCS is operating to remove activation products such as cobalt-58.

   Liquid Waste Processing System (LWPS) waste evaporator surfaces that contact the concentrates are made of highly polished Incoloy 825 so that they may be easily decontaminated prior to maintenance operations.

   Tanks containing potentially radioactive liquids have bottom surfaces that are sloped towards the drain, thus allowing ease in flushing of radioactive sediments.

4. Prior to removal of the reactor vessel head, steps may be taken, if necessary, to reduce the radioactivity in the reactor coolant, thereby reducing airborne releases and radioactive contamination of the refueling cavity water.

   Decontamination of the RCP rotating parts in other nuclear power plants has been successful in reducing the radiation field when maintenance has been required. As a result, radiation levels have been significantly reduced, resulting in a significant reduction in man-rem exposure to maintenance personnel. The plant decontamination facilities are described in Section 12.5 and are shown in the general arrangement drawings listed as Figures 1.2-28 and 1.2-29 in Table 1.2-1.

12.3.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation. Area radiation and airborne radioactivity monitoring is provided to alert personnel to the presence of abnormal radiation levels. This is discussed more fully in Section 12.3.4 and Table 12.3.4-1.

12.3.1.5 Process and Effluent Radiological Monitoring Instrumentation. This instrumentation is designed in part to give early warning of increasing radiation levels in process streams and to initiate corrective action either automatically or through operator response on high radioactivity levels. A more complete description is found in Section 11.5.

12.3.1.6 Control of Access, Posting of Radiation Sources and Areas. The guidelines contained in NUREG-0452 Section 6.12 were applied to STPEGS during the design phase with regard to control of access and posting of radiation sources and areas. For example, restrictive personnel barriers, such as lockable doors, were provided at entrances to projected high-radiation areas to ensure ALARA occupational radiation exposures. Table 12.3.2-1 illustrates the radiation zones and the posting requirements based on 10 CFR 20 and access control methods. Radiation sources and radiation zones are shown on Figures 12.3.1-1 through 12.3.1-16.

The location and layout of the Health Physics clean area in the Mechanical-Electrical Auxiliaries Building (MEAB) facilitate control and monitoring of personnel entering and leaving the main radiologically controlled areas, as shown on Figure 12.3.1-9. The Health Physics Program is discussed in Section 12.5.

12.3.2 Shielding

12.3.2.1 Design Objectives. The primary design objective of the plant radiation shielding is to protect plant operating personnel and members of the public against radiation exposure from the various sources of ionizing radiation in the plant during normal operating conditions, anticipated operational occurrences, postulated accident conditions, and maintenance.

This objective is accomplished by designing the shielding to perform the following functions:

1. Limit in-plant radiation exposures of plant personnel, contractors, and authorized site visitors to ALARA for normal operation or shutdown, including anticipated operational occurrences and maintenance consistent with the guidance given in RG 8.8.

2. Limit offsite exposures of members of the public in unrestricted areas from direct and air-scattered radiation due to contained sources generated during normal operation, shutdown, and anticipated operational occurrences to a small fraction of the limits specified in 10CFR20.

3. Limit radiation exposure in vital areas as defined by NUREG-0737, in the unlikely event of an accident, to allow access or habitability as specified in 10CFR50, Appendix A, General Design Criterion (GDC) 19, and NUREG-0737.

4. Limit offsite exposures of members of the public, during the unlikely event of a DBA, from all direct and air-scattered radiation from contained sources on the site to ensure that the total radiation exposure will be within the limits specified in 10CFR100

5. Ensure, in accordance with 10CFR50, Appendix A, GDC 4, that the structures, components, and systems important to safety are protected from excessive radiation damage, neutron activation, and heat generation. The protective shielding ensures that critical functions of the structures, components, and systems are not impaired and thus minimizes maintenance requirements.

To comply with the above functions and objectives, the plant shielding is designed to attenuate radiation throughout the plant from direct and scattered radiation to below the values specified in: (1) Table 12.3.2-1 during normal full-power operation and anticipated operational occurrences when operating with design basis fuel defects, and (2) Table 12.3.2-2 for postulated accident conditions. Anticipated access control requirements are also indicated in Table 12.3.2-1.

12.3.2.2 Design Description.

12.3.2.2.1 Design Criteria: Application of the following design criteria in the facility shield design ensures that occupational radiation exposure is ALARA:

1. The station radiation shield design is based upon either the operating or shutdown condition of a system, whichever is more restrictive, and takes into consideration residual radiation.

2. Major sources of radiation (Chapter 11 and Section 12.2.1) are located in individually shielded cubicles to ensure that personnel exposures are ALARA during inspection and maintenance functions. Labyrinths are used to reduce radiation streaming through access openings in shielded cubicles.

3. Where applicable, pumps and other support equipment for components which contain radioactive material are located outside the component cubicle. Cubicles are provided for the support equipment containing radioactive material to ensure that the radiation exposure of personnel performing inspection or maintenance functions is ALARA. To minimize service time, and hence exposure, shielding cubicles housing active equipment (e.g., pumps and valves) and equipment requiring service are designed so that work can be performed without being hampered by the lack of space.

4. To ensure that personnel exposures are ALARA, shielded valve cubicles are used where feasible to allow valve maintenance without drainage or decontamination of associated equipment. To further reduce personnel exposure, remotely operated valves are used where

practicable, and if manual valves are required, extended handwheels through a shield wall to a lower radiation area are provided.

5. The systems containing radioactive material are appropriately arranged relative to each other to optimize the radiation shielding required for personnel protection.

6. Penetrations in shield walls are located to reduce direct radiation streaming from major components containing radioactive material. Where this could not be accomplished, penetration shielding has been provided.

7. Piping containing radioactive material is appropriately routed to reduce exposure to plant personnel. This involves:

   a. Routing of piping containing radioactive material through radioactive pipe chases or behind shield walls (cubicles) suitably designed to ensure that the design dose rates in radiation shielding zones are not exceeded. These zones are described in Table 12.3.2-1.

   b. Wherever possible, separating piping which normally contains radioactive material from piping containing nonradioactive material. This ensures that exposures to personnel performing inspection or maintenance on nonradioactive systems are ALARA.

   c. Pipelines that are sloped, wherever possible, so that the slope will assist in removing crud deposits from the line prior to maintenance operations. Pipes that handle spent demineralizer resin have a minimum 5-diameter-bend radii.

   d. Thermal expansion loops that are horizontally or vertically upwards rather than vertically downwards and eccentric reducers to reduce crud traps in the piping lines.

   e. The letdown line routed within the Reactor Containment Building (RCB) secondary shield, providing sufficient delay time for decay of the reactor coolant N-16 activity so that N-16 is not a factor in shielding outside the RCB.

8. The principal shielding material is concrete with a dry density of 136 lb/ft^3. Where necessary to save space, steel is used to supplement the concrete shielding.

9. Procedures for concrete shield wall construction present implementation methods that correspond to the position of RG 1.69, which generally endorses the requirements of American National Standards Institute (ANSI) N101.6-1972.

10. Consideration has been given to provision for shielding of major sources of radiation during inservice inspection to allow access and minimize radiation exposure of personnel (Section 6.6).

11. Shield plugs, concrete hatch covers, and shield doors to potentially highly radioactive components are provided with offsets to reduce radiation streaming.

12. Temporary shielding may be used in order to reduce the doses during activities in radiation areas. Consideration has been given, as part of the ALARA program during design and construction, for allowing room where possible for the placement and use of temporary shielding.

In addition to the above criteria, plant radioactive systems and shielding designs were constantly reviewed, updated, and modified as necessary during plant design and construction.

12.3.2.2.2 Description of Plant Shielding: The general plant yard areas are within the restricted area and separate from areas accessible to members of the public. Layout arrangement drawings of all buildings containing systems handling radioactive material are shown on Figures 12.3.1-1 through 12.3.1-16. These figures indicate the design maximum and expected radiation levels for normal full power operation conditions (Table 12.3.2-1). The shielding design bases for normal operation are 4,100 MWt and 1 percent failed fuel. Exception may be taken to the 1 percent failed fuel fraction in those cases where, due to the advanced stage of construction, the modifications would require excessive shielding changes. In this instance the design basis may change from 1 percent failed fuel to 0.25 percent failed fuel on a case-by-case basis. Figures 12.3.1-1 through 12.3.1-6 identify the RCB radiation access zones for after-shutdown conditions. All other areas inside the restricted area boundary not described in the Figures are categorized as Zone 1 which are not expected to exceed 0.5 mrem/hr under normal operation conditions or operational occurrences.

12.3.2.2.2.1 Primary Shield – The concrete primary shield surrounds the RPV located inside the RCB and extends upward from the Containment floor to form a portion of the refueling cavity. The primary shield:

1. Adequately reduces, in conjunction with the secondary shield, the radiation level due to sources within the RPV complex to allow controlled, limited access into the RCB during full power operation and occupational access after reactor shutdown.

2. Limits activation of equipment and structural materials inside the RCB over the life of the facility.

The 7-foot-thick (minimum) primary shield, together with the 3.5-foot-thick secondary shield wall, ensures that the deep dose equivalent rate outside the secondary shield from core and RCS sources does not exceed 100 mrem/hr during full power operation and is less than 2.5 mrem/hr upon access to the RCB after shutdown. Figures 12.3.1-1 through 12.3.1-6 define the RCB design radiation zones for normal operation and shutdown.

12.3.2.2.2.2 Secondary Shield - The secondary shield is located inside the RCB and is a concrete wall-slab structure that surrounds the RCS equipment, including piping, pumps, SGs, and pressurizer. The secondary shield is designed to:

1. Protect personnel during full power operation and shutdown from direct gamma radiation emanating from the primary coolant.

2. Adequately attenuate N-16 activity in the primary coolant loops during reactor operation.

3. Supplement the primary shield in attenuating radiation emanating from the RPV complex to permit limited, controlled, periodic access to the RCB during full power operation and maintenance of equipment in certain areas during shutdown.

12.3.2.2.2.3 Reactor Containment Building Shield – The RCB shield is a prestressed, cylindrical concrete structure with a hemispherical dome, which is lined on the inside with steel. The structure completely surrounds the Nuclear Steam Supply System (NSSS) and also serves as a radiation shield for both normal operation and accident conditions. This shield:

1. Adequately protects personnel outside the RCB from all radiation sources in the RCB during full power operation and reactor shutdown.

2. Adequately attenuates radiation from all sources in the RCB following a DBA to ensure that site exclusion boundary and low population zone (LPZ) doses from these sources are sufficiently low to limit the total exposures to those specified in Table 12.3.2-2. Analyses of site exclusion boundary and LPZ doses are presented in Chapter 15.

3. Together with the control room shielding, adequately attenuates radiation from all sources in the RCB following a DBA to ensure that the dose rate inside the shielded control room from these sources is sufficiently low to limit the integrated deep dose equivalent (based upon 30-day residence) to that specified in Table 12.3.2-2. Analyses of control room doses are presented in Section 6.4.

The cylindrical portion of the RCB concrete shield is 4 ft thick, and the hemispherical dome is 3 ft thick. These concrete thicknesses adequately reduce the radiation from sources in the RCB to achieve the above objectives. The primary and secondary shields, together with the RCB shield, are designed to ensure that the deep dose equivalent rate outside the Containment from sources inside the Containment does not exceed 0.5 mrem/hr during full power operation.

12.3.2.2.2.4 Fuel Transfer Shield – The fuel transfer shielding comprises all structures that provide the required attenuation of radiation during fuel handling, storage, and transfer, and includes the water in the refueling cavity, the temporary fuel storage pool, the transfer canal, and the spent fuel storage pool. The shielding also includes the concrete walls of the refueling cavity, the in-Containment temporary fuel storage area, the transfer canal, and the spent fuel storage pool. In addition, the Fuel Transfer System (FTS) shielding is arranged to provide laydown space for the reactor internals.

The fuel transfer shielding also includes all structures in the Fuel Handling Building (FHB). These structures are designed to ensure that the deep dose equivalent rates on the exterior walls of the building do not exceed 0.5 mrem/hr. The building arrangements are given on Figures 12.3.1-13 through 12.3.1-16.

All spent fuel is remotely handled under water to ensure that the deep dose equivalent rate above the refueling pool and spent fuel pool (SFP) is less than 2.5 mrem/hr from the fuel assembly being transferred. Inside the RCB, a minimum of 5 ft of concrete around the fuel transfer canal and temporary spent fuel storage area ensures that the deep dose equivalent rates during refueling do not exceed 2.5 mrem/hr in normally accessible areas.

In the FHB, the SFP north, east, south, and west walls at the storage elevation are 6.0, 5.25, 7.25, and 6.5 ft thick, respectively, to ensure that the radiation levels in the FHB access areas and outside the buildings are within the values depicted on Figures 12.3.1-13 through 12.3.1-16.

There are two access paths to the unshielded area adjacent to the fuel transfer tube in the FHB:

1. Via a ladder up from Room FHB-011 at El. (-) 13 ft-0 in. to Room FHB-112 at El. 21 ft-2 ½ in.

2. Via a ladder down from Room FHB-304 at El. 68 ft-0 in. to Room FHB-112 at El. 21 ft-2 ½ in.

To restrict access to the area adjacent to the fuel transfer tube, STPEGS will:

1. Provide lockable barriers at the points of access to the ladders in Rooms FHB-011 and FHB-304

2. Ensure, through procedural requirements, that the appropriate barriers and doors to Room FHB-112 are locked closed prior to and during the transfer of irradiated fuel through the fuel transfer tube

3. Conspicuously post these barriers in accordance with 10CFR20 and other applicable project requirements

Administrative control of keys to these areas are prescribed by plant procedures.

The thicknesses of concrete shielding around the cask loading area, which range from 18 in. to 60 in., ensure that the expected access zone radiation levels around the cask loading area are not exceeded during cask loading operations.

12.3.2.2.2.5 Auxiliaries Shielding - The auxiliary shielding comprises wall-slab structures which provide the necessary attentuation for radioactive sources in all auxiliary system equipment. Certain equipment located in the Mechanical Auxiliary Building (MAB), RCB, Turbine Generator Building (TGB), yard next to the TGB, and FHB contains radioactive material either during normal operation or under special conditions. Those components were analyzed to establish their maximum radiation levels, and shield thicknesses subsequently were defined to ensure that the prescribed radiation levels in the access zones defined on Figures 12.3.1-1 through 12.3.1-16 are not exceeded.

Auxiliary shielding has been designed for the radioactive components in the CVCS, Boron Recycle System (BRS), Residual Heat Removal System (RHRS), Spent Fuel Pool Cooling and Cleanup System (SFPCCS), Waste Processing Systems, Condensate Polishing System, Emergency Core Cooling System (ECCS), and Containment Spray System (CSS).

Figures 12.3.1-1 through 12.3.1-16 illustrate that the facility shield design and arrangement has implemented the design criteria specified in Section 12.3.2.2.1. Radioactive components in the MAB, FHB, TGB, and RCB are placed inside individual cubicles where necessary, whereas associated support equipment (pumps, valves, etc.) is placed outside the cubicles whenever possible. Potentially radioactive support equipment is placed inside separate cubicles to ensure that personnel in corridors and valve operating areas, (i.e., areas containing extended valve handwheels) are adequately protected from radiation emanating from this equipment. Access opening to areas that were projected to have dose rates of 1 rem/hr or above are provided with lockable doors.

Summarized below are the placements and shielding thicknesses incorporated into the STPEGS design, as they relate to personnel radiation protection. The discussion considers the examples of the VCT, reactor coolant filter, mixed-bed demineralizer, boron recycle evaporator, high total dissolved solids (TDS) tank, charging pumps, and PSS sampling room. Since these components may contain varying quantities of radioactive material during the life of the facility, they have generally been located in the MAB, and precautions were instituted to ensure adequate personnel radiation protection in all activities associated with the components.

12.3.2.2.2.5.1 Placements – In general, components expected to contain significant amounts of radioactive material are placed in separately shielded cubicles, where necessary, to maintain access radiation levels defined on Figures 12.3.1-1 through 12.3.1-16, with sufficient space available for personnel access for inspection and maintenance. To the extent practicable, piping, valves, and other support equipment associated with each component are located outside the component cubicle and in turn are enclosed in separate shielded cubicles. Piping and duct penetrations through the component cubicle walls are located and shielded to reduce radiation streaming.

The VCT is located in the CVCS upstream of the charging pumps. Since a function of the CVCS is to purify letdown from the RCS by removal of corrosion and fission products, the VCT contains a varying quantity of radioactive material throughout the life of the plant. The placement of the VCT is shown on the general arrangement drawing listed as Figure 1.2-28 in Table 1.2-1. The cylindrical VCT (with elliptical heads) is positioned vertically with respect to the major axis, to minimize the surface area available for crud buildup. This also aids in decontamination of the component, if required. Access to the VCT cubicle is through a watertight door, which in turn is enclosed in a shield labyrinth to ensure minimal radiation streaming from the cubicle into access corridors.

The reactor coolant filters are located in the CVCS upstream of the VCT and are used to remove particulate radioactive material from the CVCS prior to entering the VCT. The placement of the filters is depicted on Figure 12.3.1-9 and on the general arrangement drawing listed as Figure 1.2-28 in Table 1.2-1. Concrete shield hatches with offsets were placed 9in the cubicle tops over the filters for personnel access and filter replacement.

The mixed-bed demineralizer is located in the CVCS upstream of the reactor coolant filters. Its function is to remove and retain the fission product activity, both cation and anion, and corrosion product metals from the reactor coolant letdown. The placement of the demineralizer is shown on Figures 12.3.1-9 and 1.2-28. Offset concrete hatches are placed in the top of the cubicle for personnel access and demineralizer maintenance, if required. To ensure ALARA personnel exposures, resin handling is accomplished remotely, i.e., behind shield walls. The new resin is loaded from the resin fill tank located on the floor above the demineralizer.

The boron recycle evaporator is located in the BRS downstream of the boron recycle evaporator feed pump. The projected radioactive sources associated with the evaporator are those of the recycle holdup tank (RHT) liquid. The fission gases are removed in the evaporator stripping column before the liquid enters the evaporator shell. See Figure 12.3.1-7 and the general arrangement drawing listed as Figure 1.2-26 in Table 1.2-1 for layout of the package. The personnel access doorway into the cubicle is provided with a labyrinth shield to reduce radiation streaming to normally accessible areas. A block wall is used to separate the more radioactive components of the recycle evaporator package.

The high TDS tanks are located in the yard near the TGB. Although these tanks contain mildly radioactive regenerate solution from the resin regeneration tank, shielding for normal operations is not required. However, space has been left to accommodate shielding if warranted by primary to secondary leakage and fuel failures. The charging pumps are located in the CVCS downstream of the VCT and are used to return the letdown primary coolant to the RCS. Figure 12.3.1-7 and the general arrangement drawing listed as Figure 1.2-26 in Table 1.2-1 depict the placement of the pumps. Access to each charging pump room is through a lockable door which is adequately shielded with a labyrinth to reduce radiation streaming to normally accessed areas. A section of one cubicle wall is constructed of concrete blocks for removal of each pump, if required.

Where practicable, process samples of radioactive material are obtained in the sample room located in the MAB (Figure 12.3.1-9 and the general arrangement drawing listed as Figure 1.2-28 in Table 1.2-1). The radioactive process sample lines in the room are located behind shielding, and the room layout was selected to ensure ALARA exposures to personnel obtaining the samples. The PSS is described in detail in Section 9.3.2.

12.3.2.2.2.5.2 Shielding - Sufficient shielding thickness has been provided for cubicles defined above to ensure that dose rates in adjacent areas do not exceed design zone values. The shielding thicknesses required have been calculated with industry-accepted and verified shield design codes using 136 lb/ft^3 dry density concrete.

The VCT and cation tank sources used are shown in Tables 12.2.1-5 and 12.2.1-8, respectively, and are consistent with purified letdown originally containing a source distribution dictated by 1 percent failed fuel.

The reactor coolant filter sources are shown in Table 12.2.1-5 and are consistent with a corrosion and fission product source distribution and strength required to obtain 500 R/hr contact, the maximum expected radiation level at time of filter removal.

The mixed-bed demineralizer sources are shown in Table 12.2.1-5 and are consistent with source buildup on the resin bed anticipated with 1 percent failed fuel.

The boron recycle evaporator package sources are shown in Table 12.2.1-6. The package components are positioned in the cubicle as shown on Figure 12.3.1-7.

The high TDS tank sources are shown in Table 12.2.1-8. The bases for these sources are discussed in Section 12.2.1.1.8.

The sampling room is shown on Figure 12.3.1-9. The room walls are designed so that the radiation levels in adjacent occupational areas will not exceed 2.5 mrem/hr.

12.3.2.2.2.6 Control Room Shielding – Control Room Shielding – Shielding of the control room has been designed in accordance with the requirements of 10CFR50, Appendix A, GDC 19, which requires access to, and full habitability of, the control room following a Loss-of-Coolant

Accident (LOCA). The accident analyses in Chapter 15 show the dose to personnel in the control room for each accident (except the sample line break and the small line break outside containment).

Shielding of the control room for direct radiation is provided by concrete walls, ceiling, and floor. The control room is surrounded with the following effective thicknesses of concrete shielding: north –2 ft, south – more than 8 ft, east – 4 ft, west – 7 ft, and ceiling – 4 ft. These structures are depicted in the layout of the control room shown on Figure 12.3.1-9.

The control room shielding has been analyzed for various accident sources. Calculations were made to confirm the shielding adequacy for a post-LOCA containment airborne source, for a post-accident airborne source outside the Electrical Auxiliary Building (EAB), and for post-accident doses due to the control room filters. For the sum of these doses the 30-day integrated dose was below GDC 19 limits. Further discussion of post-accident control room calculations can be found in Chapter 15 for each accident (except the sample line break and the small line break outside containment). Control room habitability is discussed in Section 6.4.

12.3.2.2.2.7 Penetration Shielding - As defined in the shield design criteria, the shielding is arranged to ensure that occupational radiation exposures are maintained ALARA. Precautions are taken to minimize the contribution of radiation streaming through penetrations. Wherever possible, penetrations are located so that they do not pass through shield walls in a direct line with a radiation source.

Penetrations in the RCB have been clustered in four areas. Each of these areas is shielded to isolate the penetrations from the rest of the plant.

12.3.2.2.2.8 Design Criteria for Routing Radioactive Process Piping - Item 7 of Section 12.3.2.2.1 discusses the philosophy used in the routing of radioactive process piping. All significantly radioactive piping is shielded to ensure that the design radiation level of the zone (Table 12.3.2-1) in which the piping is located is not exceeded. Some process piping associated with the Condensate Polishing Demineralizer System (CPDS) resin regeneration unit is routed inside the TGB and in the yard adjacent to the TGB. These areas are designed not to exceed a radiation level of 0.5 mrem/hr. The major portion of the radioactive piping is located in the radioactive pipe chase shown on Figure 12.3.1-8.

12.3.2.2.2.9 Old Steam Generator Storage Facility Features

The Old Steam Generator Storage Facility (OSGSF) is a reinforced concrete building which provides long-term, interim storage of and shielding for the old steam generators removed from the Reactor Containment Buildings. The OSGSF storage area has an 18 inch concrete roof and 30 inch concrete walls. A labyrinth type vestibule, access to which is through a lockable personnel access door, is designed to minimize radiation streaming beyond the outside surface of the facility. The design of the vestibule accommodates personnel access for surveillance of an electrical panel without entry into the steam generator storage area. The general arrangement of the OSGSF is listed as Figure 1.2-60 in Table 1.2-1.

The OSGSF has been designed such that the dose rates at the exterior of the facility (walls and roof) are within the dose limits of 10 CFR Part 20. Exterior areas of the facility and the area in the vestibule used for personnel access to the electrical panel is Zone 1. The storage area is assigned a Zone 5.

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12.3.2.2.2.10 Refueling Equipment Building - Deleted

12.3.2.2.2.11 Other Plant Areas - The general plant yard areas are not routinely accessible to members of the public. All areas within the protected area boundary are Zone 1 (design value less than 0.5 mrem/hr under normal operating conditions) unless otherwise indicated on Figures 12.3.1-1 through 12.3.1-16.

In the event the SGs develop tube leaks allowing N-16 and other activation, corrosion, and fission products to enter the Steam System, the steam line dose rates between the secondary shield and TGB will increase slightly. The maximum expected radiation level 1 ft (30 cm) from the steam line at the SG is less than 0.5 mrem/hr with 1 percent failed fuel, and a 1 gal/min primary-to-secondary tube leak rate. Thus, supplemental shields are not required for the steam lines.

Power operation with the unlikely occurrence of SG tube leakage coincident with failed fuel cladding will also result in above design value (0.5 mrem/hr) radiation levels near the condensate polisher demineralizers and the associated resin regeneration unit. Should this occur, access to these units will be temporarily restricted and controlled in accordance with plant procedures.

12.3.2.2.3 Methods of Shield Design - The methods of calculation used to verify the shield thicknesses depend upon the complexity of the source-shield-dose point configuration. The computer code QAD-CG was used in the secondary shield analysis to depict the SG sources and concrete shielding. ANISN (Ref. 12.3-6) was used to calculate primary gamma, neutron, and secondary gamma dose rates on the midplane of the core. In most of the auxiliary systems shield design, such as smaller components and piping, the methods used in shielding analysis are those of References 12.3-8 and 12.3-9. A general description of the codes is given in Table 12.3.2-4.

12.3.2.2.4 Shield Design Calculations: In the RCB the main sources of radiation during power operation are the reactor vessel and the primary loop components consisting of the SGs, the pressurizer, the RCPs, and associated piping. The reactor vessel is shielded by the concrete primary shield.

The 7 ft primary shield was analyzed using the computer code ANISN. The direct radiation level from the vessel outside the primary shield was calculated to be well below 100 mrem/hr. Streaming through penetrations in the primary shield and up the gap around the reactor vessel was calculated using the code MORSE-CG (Ref.12.3-5). Results from the MORSE-CG indicate that limited personnel access to the operating deck is permissible during power operation.

The 3.5 ft secondary shield was analyzed using the computer code QAD-CG (Ref. 12.3-4). The main sources are the SGs and the reactor coolant piping, with the primary radionuclide source being N-16. Results indicate that direct radiation levels outside the secondary shield will be below 100 mrem/hr.

All other areas containing radioactive sources were analyzed for compliance with zoning criteria. Most radioactive components are located in either the RCB or the MAB. For sources other than the reactor vessel, bulk shielding was analyzed via the methods in Reference 12.3-8 and by use of the computer code QAD-CG (Ref. 12.3-4) or other point kernel codes. Gamma ray scattering analysis was achieved using the methods in Reference 12.3-9 and by use of the computer code G-33 (Ref. 12.3-7). For a list of the primary codes used see Table 12.3.2-4.

12.3.3 Ventilation

The plant HVAC systems are designed to provide a suitable environment for personnel and equipment during normal operation and events of moderate frequency or certain infrequent events. Parts of the plant HVAC systems perform safety-related functions.

12.3.3.1 Design Objectives. The plant HVAC systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of 10CFR20 and 10CFR50.

12.3.3.2 Design Criteria. Design Criteria for the plant HVAC systems include:

1. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed are ALARA and within the concentrations of 10CFR20 Appendix B for normally occupied areas.

2. During normal operations and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary is ALARA and within the limits specified in 10CFR20 and 10CFR50, Appendix I.

3. The plant site dose limitations of 10CFR100 (for the sample line break and small line break outside containment) or 10CFR50.67 (for all other Chapter 15 accidents) will be satisfied following those hypothetical accidents described in Chapter 15.

4. The dose to control room personnel will not exceed the limits specified in GDC 19 of Appendix A to 10CFR50 following those hypothetical accidents described in Chapter 15.

12.3.3.3 Design Guidelines. To accomplish the design objectives, the following guidelines are followed wherever practical.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity:

1. Access control and traffic patterns were considered in the basic plant layout to minimize the spread of contamination.

2. Where leakage was anticipated, equipment vents and drains were piped directly to a collection device connected to the collection system instead of allowing any radioactive fluid to flow across the floor to the floor drain.

3. Suitable coatings were applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.

4. Design of potentially contaminated equipment incorporated features to reduce the potential for airborne radioactivity during maintenance operations. These features included flush connections on pump casings for draining and flushing the pump prior to maintenance, or flush connections on piping systems that could become highly radioactive.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity:

1. The airflow was designed to be from areas with lesser potential for contamination to areas with greater potential for contamination.

2. In building compartments with a potential for contamination, the exhaust was designed for greater volumetric flow than the supply to the area to reduce uncontrolled exfiltration from the area.

3. Air cleaning systems criteria for emergency systems are discussed under RG 1.52 in Section 6.5.1. System design for both normal and emergency systems is described in Section 9.4.

4. Means were provided to isolate the Containment and filter the fuel handling buildings upon indication of radioactive contamination to prevent the discharge of contaminants to the environment.

5. Means were provided to isolate and pressurize the control room envelope to minimize inleakage of contaminated air to the control room operators.

6. Suitable containment isolation valves were installed to ensure that the Containment integrity is maintained. See additional discussion in Section 6.2.4.

7. Redundancy and seismic Category I classification features were provided for safety-related components of HVAC systems.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment:

Access and service of ventilation systems in potentially radioactive areas were provided by component location to reduce personnel exposure during maintenance, inspection, and testing as follows:

1. Ventilation equipment rooms for outside air supply units and building exhaust system components were generally located in radiation Zone 2 and accessible to the operators. Work space was provided around each unit for anticipated maintenance, testing , and inspection. Safety-related filter-adsorber units generally are consistent with the recommendations of RG 1.52 for access and service requirements.

2. Local HVAC equipment that services the normal building requirements was located in areas projected to have low contamination.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The instrumentation provided for area and airborne radioactivity monitoring is an integral part of the Radiation Monitoring System (RMS) (see Section 11.5) and was designed considering the guidance of RGs 1.21, 1.97, and 8.8, and ANSI N13.1-1969. In conjunction with the Health Physics Program (Section 12.5) the instrumentation aids in preventing plant personnel from unknowingly entering into areas with excessive airborne radioactivity or background radiation.

12.3.4.1 Area Radiation Monitoring. The area radiation monitors provided with the RMS were designed to obtain accurate and reliable information concerning the gamma radiation levels in selected plant areas and to immediately inform plant personnel when predetermined radiation levels in these areas are exceeded.

12.3.4.1.1 Monitor Necessity and Location Criteria: In determining the need and actual location of area monitors within the plant, the following guidelines were used:

1. Detectors were located in areas where personnel may perform regular (once a day or more frequently) duties and where changes in plant operating conditions may cause variations in radiation levels above normal ambient levels.

2. Detectors were located in areas where personnel may perform infrequent duties and where significant increases in radiation levels may occur due to crud buildup or operational transients.

3. Detectors were located in areas where radiation levels may be affected by equipment or process system operation, as required to ensure proper operation of the equipment or system.

4. Detectors were located to best measure the representative radiation levels within a specific area and to avoid shielding of the detector by structural materials.

5. Detectors were located to provide easy access so that minimal maintenance equipment would be required and to provide an uncluttered area near the detector to permit field alignment and calibration.

12.3.4.1.2 Design Criteria: The basic design criteria of the area radiation monitors were identical to those of the RMS, as discussed in Section 11.5, and supplemented by the following:

1. Each monitor has two local audible and visible alarms initiated on a HIGH or ALERT alarm at the detection unit or at an area where the local visual alarm can be seen before entry into the affected area.

2. A monitor's detectors can be separated from the electronics.

3. There are two RCB monitors which have post-accident monitoring capability for a Condition IV accident, as discussed in Chapter 15. These (RCB High Range Monitors) are considered safety-related and meet the guidelines of RG 1.97 and NUREG-0737. They are physically located as shown on Figure 12.3.1-6. These monitors do not have the overrange provisions; the nonsaturating design described in Section 11.5.1.2.11 is not applicable to these monitors.

4. The range of the monitors was selected such that the maximum zone radiation level and any anticipated operational occurrences which would increase the radiation level could be detected.

5. Portable area radiation monitoring capability was provided which may be used to monitor the area radiation level in locations not normally monitored or in maintenance areas.

6. RG 1.97 radiation monitoring guidelines are addressed in Section 7.5 and Appendix 7B.

7. A minimum accumulative accuracy of ± 25 percent of the actual intensity value criterion was used for monitor selection.

8. The design of the fuel pool racks and administrative controls preclude criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10CFR70.24 and RG 8.12, are not needed.

12.3.4.1.3 Equipment Description: The area monitors are an integral part of the RMS, as discussed in Section 11.5. The area monitors have the identical alarm, display, and recording capabilities, locally and in the main control room, as described for the process and effluent monitors in Section 11.5.

Table 12.3.4-1 lists the location, range, and alarm setpoints of the area monitors. The location of the monitors is also shown on Figures 12.3.1-1 through 12.3.1-16.

12.3.4.1.4 Monitor, Calibration and Testing: The Area RMS is calibrated using one or more reference standards traceable to the National Institute for Standards and Technology (NIST). The calibration is verified for at least three points. A channel calibration that includes a channel functional test is performed periodically on the non safety-related area monitors. A channel calibration that includes a channel functional test is performed on the safety-related area monitors at least once every 18 months or during the refueling outage if the detector is not readily accessible. In the event a calibration is questionable, the channel can be isolated and a more thorough calibration performed.

12.3.4.2 Airborne Radioactivity Monitoring. The monitors provided in the RMS for measuring plant airborne radioactivity are identical in design and installation to the process and effluent airborne monitors described in Section 11.5.2.3. Each is an off-line type monitor which pulls an air sample from the HVAC System. Alarm, display, and recording capabilities, as described in Section 11.5, are provided for the airborne radioactivity monitors.

The airborne radioactivity monitors provide a continuous surveillance of the airborne radioactivity level within selected plant areas to warn if airborne activity exceeds selected values. Selection of areas to be monitored was based on the potential airborne radioactivity sources in the areas and the frequency with which the area is occupied.

Supplemental mobile airborne radioactivity monitors were provided in each unit. The mobile monitors each have particulate, iodine, and gaseous detector channels, as described in Section 11.5.2.3.1, and can be used for measuring the airborne radioactivity level in any specific area. Local alarm, display, and recording capabilities of a monitor, as described in Section 11.5, are provided.

The range of the airborne radioactivity monitors was chosen to achieve the low level radiation detection required for monitoring projected occupational concentrations at or below 10 CFR 20 values. Design parameters for the airborne radiation monitors are shown in Table 12.3.4-2.

The calibration and testing of the fixed and mobile airborne monitors are identical to those of the process and effluent monitors, as described in Section 11.5.2.1.5.

12.3.4.3 In-Plant Sampling.

1. Criteria

The criteria for the selection of air sampling and monitoring instrumentation used to determine the concentrations of airborne radioactivity in plant areas and effluents are given in Section 12.5.2.2, item 5.

2. Methods

Airborne radioactivity concentrations in areas accessible to personnel may be determined by both RMS monitors and air sampling as a part of the health physics program.

The fixed monitors sample the exhaust air in the HVAC ducts from selected cubicles. These monitors are described in Section 12.3.4.2 and the locations are shown on Figures 9.4.3-2 and 9.4.3-4.

The mobile monitors are designed to collect representative samples of airborne radioactivity. In addition, the mobile monitors may be used to determine the exact source of airborne radioactivity when the fixed monitors have reached an alarm condition.

Air sampling as part of the health physics program is described in Section 12.5.2.2 and is used as needed to determine airborne radioactivity concentrations in plant work areas prior to and during work activities.

Constant flow, low volume air samplers may also be used to monitor airborne radioactivity concentrations over a long period of time as part of the health physics program.

REFERENCES

Section 12.3:

12.3.1 "Byproduct Material"; Title 10, Code of Federal Regulations.

12.3-2 "Special Nuclear Material"; Title 10, Code of Federal Regulations.

12.3-3 "Source Material"; Title 10, Code of Federal Regulations.

12.3-4 R.E. Malenfant, "QAD, A Series of Point-Kernel General-Purpose Shielding Programs", LA3573, Los Alamos Scientific Laboratory, (October 1966).

12.3-5 E.A. Straker, P.N. Stevens, D.C. Irving, and V.R. Crain, "Morse – A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code", ORNL-4585, Oak Ridge National Laboratory (September 1970).

12.3-6 W.W. Engle, Jr., "A Users Manual for ANISN: A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering", Report No. K-1693, Union Carbide Corp. (1967).

12.3-7 R.E. Malenfant, "G-33: A General Purpose Gamma Ray Scattering Program", LA5176, Los Alamos Scientific Laboratory (June 1973).

12.3-8 T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., (New York, 1956).

12.3-9 N.M. Schaeffer, "Reactor Shielding for Nuclear Engineers", TID-25951 (1967).

TABLE 12.3.2-1

PLANT RADIATION SHIELDING ZONES FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL

Zone Number	Design Radiation Level (mrem/hr)	Posting Required[2]	Anticipated Access
1	0.5	No	Inside restricted area boundary and OSGSF Access without time limitation
2	2.5	No	Limited[3] Occupational access
3	15.0	Yes[4]	Limited Periodic access
4	100	Yes	Limited Occasional access
5	>100	Yes	Limited Infrequent[5]

* 1. Normal operation includes during full power operation and shutdown:
   a) Routine access to controls, gauges, meters, and remote operators for valves
   b) Periodic access to equipment in need of inspection, preventive maintenance, or repair
   c) Refueling and subsequent spent fuel handling
   d) Disposal of radioactive wastes
2. The column "Posting Required" refers exclusively to whether posting is required by 10 CFR 20, not the posting of actual radiation levels.
3. "Limited" in this context means area of access which is controlled by the licensee in accordance with an established health physics program.
4. Posting is required for those areas actually meeting or exceeding the definition of radiation area as defined in 10 CFR 20.
5. Technical Specifications govern access control requirements for high radiation areas.

TABLE 12.3.2-2

RADIATION DOSE CRITERIA FOR DESIGN BASIS ACCIDENT CONDITIONS

Location	Dose*
Control Room	GDC 19
Site Exclusion Boundary	For the Sample Line Break and Small Line Break Outside Containment: 2-hour integrated whole body dose of less than 25 rem or a total radiation dose of less than 300 rem to the thyroid from iodine exposure For all other accidents: As stated in Table 6 of RG 1.183.
Low Population Zone	For the Sample Line Break and Small Line Break Outside Containment: 30-day integrated whole body dose of less than 25 rem or a total radiation dose of less than 300 rem to the thyroid from iodine exposure For all other accidents: As stated in Table 6 of RG 1.183.

* The plant shielding is designed to ensure that the integrated doses from all contained sources contributing direct and air-scattered radiation, together with airborne radiation, are limited to the values defined herein.

TABLE 12.3.2-4

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

NISN	-	Multigroup, Multiregion code solving the Boltzman transport equation for neutrons or gamma-rays in a one-dimension slab, cylindrical, or spherical geometry (Ref. 12.3-6)
QAD	-	Multigroup, Multiregion, three-dimensional, point-kernel code which calculates fast neutron and gamma-ray doses and heat generation rates (Ref. 12.3-4)
G-33	-	A general purpose gamma-ray scattering code which uses the Klein-Nishina approximation to calculate scattering (Ref.12.3-7)
MORSE	-	A three-dimensional Monte Carlo neutron and gamma-ray general transport code (Ref. 12.3-5)

TABLE 12.3.2-5

STPEGS SHIELD COMPOSITIONS

Element	SS 304*	CS**	Concrete	Water***
	(Shield Material Density, g/cm^3)			
H			0.0115	0.078
O			1.1187	0.627
C	0.00442	0.019575	0.0023	
Na			0.0364	
mg			0.005	
Al			0.0773	
Si	0.03614	0.018009	0.7683	
K	0.00096		0.0299	
Ca			0.10	
Fe	5.64549	7.603713	0.0317	
Mn	0.13571	0.104139		
S	0.00048			
Cr	1.46708			
Ni	0.73715	0.043065		
Co	0.00257			
Mo		0.041499		
B	-	-	-	-
	8.03	7.83	2.18	0.705

* ASTM A554, MT-304
** ASTM A533, Grade B
*** At 590°F, 2,250 psia

TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	10^{-1}-10^{4}	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	10^{-1}-10^{4}	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10^{-1}-10^{4}	100

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R_1 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R_1 (4 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R_1 (30 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8097 33 ft S of cols. 28 and 10 ft W of col. N (68 ft-0 in.)	10^2-10^7	1,000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10^{-1}-10^{4}	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8076 col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10^{2}-10^{7}	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location [(1)]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [(2)]
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB (72 ft-0 in.)	10^{2}-10^{7}	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

Tag Number and Location [1]	Range (R/hr)	High Alarm Setpoint (R/hr) [2]
A1RA-RE-8050 RCB (68 ft-0 in.)	10^0-10^8	2100
C1RA-RE-8051 RCB (68 ft-0 in.)	10^0-10^8	2100

CN-3153

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-2

BUILDING VENTILATION MONITORS

Monitor	Service	Sample Location	Detector Number	Detector Type	Analysis Performed	Range(12) (μCi/cm^3)	MDC(1) (μCi/cm^3)	Controlling Isotope	Alert Alarm(11) (μCi/cm^3)	High Alarm(11) (μCi/cm^3)
RT-8014	Mechanical Auxiliary Building (MAB) Ventilation	MAB El. 10 ft See Fig. 9.4.3-2	RE-8014A Particulate (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$2.8x10^{-9}$	$5.5x10^{-9}$
			RE-8014B Iodine (I)	(3)	Gross Gamma	(5)	(8)	I-131	$2.5x10^{-9}$	$5.0x10^{-9}$
			RE-8014C Noble Gas (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8015	MAB Ventilation	MAB El. 10 ft See Fig. 9.4.3-2	RE-8015A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$1.7x10^{-9}$	$3.4x10^{-9}$
			RE-8015B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$1.6x10^{-9}$	$3.1x10^{-9}$
			RE-8015C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8016	MAB Ventilation	MAB El. 10 ft See Fig. 9.4.3-2	RE-8016A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$2.7x10^{-9}$	$5.4x10^{-9}$
			RE-8016B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$2.5x10^{-9}$	$4.8x10^{-9}$
			RE-8016C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8017	MAB Ventilation	MAB El. 60 ft See Fig. 9.4.3-6	RE-8017A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$1.5x10^{-8}$	$3.1x10^{-8}$
			RE-8017B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$1.4x10^{-8}$	$2.8x10^{-8}$
			RE-8017C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8018	MAB Ventilation	MAB El. 41 ft See Fig. 9.4.3-6	RE-8018A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$1.6x10^{-9}$	$3.3x10^{-9}$
			RE-8018B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$1.5x10^{-9}$	$2.9x10^{-9}$
			RE-8018C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8029	MAB Ventilation	MAB El. 41 ft See Fig. 9.4.3-6	RE-8029A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$6.1x10^{-10}$	$1.2x10^{-9}$
			RE-8029B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$5.5x10^{-10}$	$1.0x10^{-9}$
			RE-8029C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$
RT-8030	MAB Ventilation	MAB El. 41 ft See Fig. 9.4.3-6	RE-8030A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	$1.8x10^{-9}$	$3.4x10^{-9}$
			RE-8030B (I)	(3)	Gross Gamma	(5)	(8)	I-131	$1.6x10^{-9}$	$3.1x10^{-9}$
			RE-8030C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	$1.1x10^{-5}$	$2.1x10^{-5}$

1. Minimum Detectable Concentration (calculated)
2. Beta Scintillation Detector
3. Gamma Scintillation Detector
4. $1.6x10^{-11}$ to $1.6x10^{-5}$ μCi/cm^3
5. $3.3x10^{-12}$ to $1.2x10^{-5}$ μCi/cm^3
6. $2.8x10^{-7}$ to $2.8x10^{-1}$ μCi/cm^3
7. $8.6x10^{-12}$ μCi/cm^3 for 0.5 in/hr; $1.7x10^{-11}$ μCi/cm^3 for 1.0 in/hr
8. $4.4x10^{-11}$ μCi/cm^3
9. $2.1x10^{-7}$ μCi/cm^3
10. $3.3x10^{-11}$ μCi/cm^3
11. Setpoints are nominal values and may be adjusted as necessary depending on plant conditions
12. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

12.4 DOSE ASSESSMENT

12.4.1 Estimate of Annual Exposures to STPEGS Station Personnel

The projected annual exposure that could be received by plant personnel during normal operation and anticipated operational occurrences and normal maintenance, with the plant operating continuously under normal radiological conditions, demonstrates that personnel exposures would not exceed 5 rem/yr. Unexpected maintenance and emergency operations were excluded from the annual exposure estimate. To ensure that occupational exposures are as low as is reasonably achievable (ALARA), the design feature guidances given in Regulatory Guide (RG) 8.8 are followed as described in Sections 12.1 and 12.3.

For the purposes of this section, station activities include:

1. Reactor operation and surveillance
2. Routine maintenance
3. In-service inspection
4. Special Maintenance
5. Waste Processing
6. Refueling

Since startup, the South Texas Project has accumulated several years of operational and outage radiation exposure data. In lieu of estimates based on staffing levels, projected occupancy, and anticipated average radiation exposure rate in various areas of the plant, actual personnel exposure data for calendar years 1990 through 1995 has been analyzed. This analysis gives a realistic estimate of expected doses. Normal radiological conditions existed during this time.

Results are tabulated for the South Texas Project operation. Unless otherwise noted, annual doses reported in the tables must be divided by 2 in order to reflect doses on a per-unit basis.

- Table 12.4-1 summarizes the exposed personnel and annual radiation man-rem dose totals by category (licensee or contractor) for calendar years 1990 through 1995.
- Table 12.4-2 summarizes outage doses for refueling outages through 1995.
- Table 12.4-3 summarizes annual doses for licensee and contractor personnel by job category for calendar years 1990 through 1995.

The average annual man-rem dose from direct radiation received during calendar years 1990 through 1995 is 213 man-rem per year with a maximum of 317 man-rem and a minimum of 58.2 man-rem. The average dose to an individual worker for this time period is less than 0.4 rem with no individual dose exceeding 2 rem in a year for this time period. Internal dose is negligible, not meeting the requirement for monitoring for any individual under the current provisions of 10 CFR 20. Doses during future years are expected to be in line with these historical data with internal dose below the threshold requiring monitoring. Exceptions to this projection may occur should unexpected maintenance or emergency operations occur.

1. In Tables 12.4-1 and 12.4-3, the number of personnel are those personnel receiving 100 mrem or more of occupational exposure. The man-rem in these tables is the total man-rem for all personnel in the category, including those with dose below 100 mrem.

2. Man-rem reported in Tables 12.4-1 through 12.4-3 are based primarily on self reading dosimeter results. TLD results are typically lower.

3. At least one refueling outage occurred during all calendar years except 1994.

4. In Table 12.4-3, calendar years 1990 through 1994 show the category Health Physics while 1995 uses the category Generation Support (GenSupt). The Health Physics category includes all Health Physics personnel including decontamination and waste processing personnel. The new 1995 category GenSupt includes not only these, but also chemistry, outage planning and support, and operations support personnel. Other categories are unchanged.

5. It is current practice at South Texas for some individuals to perform different functions during outages than are performed during normal operations. For example, an individual from Nuclear Licensing may act as an RCB coordinator, or a health physics trainer may perform as a health physics technician. Currently, the dose so accrued is accounted in their "home" organization and not included in the dose for the organization they are assisting. This may impact the partitioning of man-rem, as in Table 12.4-3, but it does not affect the man-rem total.

12.4.2 Estimate of Exposure of Unit 2 Construction Workers

DELETED Unit 2 is completed and operational.

12.4.3 Estimate of Restricted Area Doses

Beginning in 1994, doses at the restricted area boundary have been calculated and results summarized in the Annual Radioactive Effluent Release Report. 10 CFR 20.1301 requires that dose to members of the public be kept below 100 mrem total effective dose equivalent in a year with compliance demonstrated according to the provisions of 20.1302. TLD measurements from locations on the protected area fence (corresponding to the restricted area surrounding the units) demonstrates that external exposure is less than 50 mrem in a year. Calculations based on effluent release rates and calculated χ/Q values demonstrate that the internal dose at the TLD locations is less than 50 mrem per year. These measurements and calculations ensure that members of the public not in a restricted area do not receive a dose in excess of the federal limit.

REFERENCES

Section 12.4:

12.4-1 Correspondence, ST-HL-AE-3700, Annual Operating Report, 1990

12.4-2 Correspondence, ST-HL-AE-4020, Annual Operating Report, 1991

12.4-3 Correspondence, ST-HL-AE-4344, Annual Operating Report, 1992

12.4-4 Correspondence, ST-HL-AE-4711, Annual Operating Report, 1993

12.4-5 Correspondence, ST-HL-AE-5013, Annual Operating Report, 1994

12.4-6 Correspondence, ST-HL-AE-5298, Annual Operating Report, 1995

TABLE 12.4-1

TOTAL EXPOSED PERSONNEL AND DOSE FOR 1990 - 1995 BY CATEGORY

Year	Category	Number	Dose
1990	Licensee	257	79.1
	Contractor	367	125.7
	Total	624	204.8
1991	Licensee	253	88.3
	Contractor	449	180.0
	Total	702	268.2
1992	Licensee	181	60.6
	Contractor	298	97.9
	Total	479	158.5
1993	Licensee	171	59.1
	Contractor	535	214.0
	Total	706	273.0
1994	Licensee	54	26.3
	Contractor	92	31.8
	Total	146	58.1
1995	Licensee	250	90.1
	Contractor	557	226.9
	Total	807	317.0

TABLE 12.4-2

DOSES ACCRUED DURING REFUELING OUTAGES THROUGH 1995

Refueling Outage*	Dose (rem)**
1RE01	142.6
1RE02	62.4
2RE01	130.9
1RE03	90.7
2RE02	157.1
1RE04	135.1
2RE03	189.4
1RE05	138.9
2RE04	132.7

* Outage designator UREXX where U is unit, RE is refueling, and XX is consecutive outage number in that unit. Outages are listed in chronological order.

** Dose may be from TLD or direct reading dosimeter results.

TABLE 12.4-3

DOSES FROM 1990 THROUGH 1995 BY JOB CATEGORY

Job	1990		1991		1992		1993		1994		1995	
	Personnel >100 mrem	Dose man-rem	Personnel >100 mrem	Dose man-rem	Personnel >100 mrem	Dose man-rem	Personnel >100 mrem	Dose man-rem	Personnel >100 mrem	Dose man-rem	Personnel >100 mrem	Dose man-rem
Reactor Operations and Surveillance	Ops 9	3.3	Ops 29	6.2	Ops 10	1.6	Ops 15	4.1	Ops 6	3.3	Ops 12	3.8
	Maint 0	0.7	Maint 3	1.3	Maint 0	0.7	Maint 0	1.1	Maint 0	0.6	Maint 3	2.4
	Eng 0	0.3	Eng 2	3.7	Eng 0	0.4	Eng 1	0.5	Eng 0	0.0	Eng 2	1.2
	HlPhys 37	11.1	HlPhys 72	26.3	HlPhys 18	4.3	HlPhys 10	4.3	HlPhys 20	6.0	GenSupt 21	7.0
	NuA&L 0	0.1	NuA&L 0	0.3	NuA&L 0	0.1	NuA&L 0	0.1	NuA&L 0	0.1	NuA&L 2	0.7
	Other 6	2.7	Other 7	4.2	Other 4	3.0	Other 1	3.5	Other 3	2.7	Other 1	6.8
Routine Maintenance	Ops 0	0.4	Ops 1	0.2	Ops 0	0.1	Ops 0	0.1	Ops 2	1.1	Ops 7	1.7
	Maint 74	19.3	Maint 76	25.2	Maint 111	31.7	Maint 87	25.6	Maint 11	4.8	Maint 102	34.2
	Eng 2	0.8	Eng 8	2.7	Eng 28	11.3	Eng 1	0.9	Eng 0	0.0	Eng 6	1.7
	HlPhys 46	13.4	HlPhys 4	3.6	HlPhys 11	3.5	HlPhys 2	1.1	HlPhys 3	1.4	GenSupt 128	47.4
	NuA&L 3	1.2	NuA&L 2	1.2	NuA&L 9	2.1	NuA&L 2	1.3	NuA&L 0	0.4	NuA&L 5	1.1
	Other 29	10.1	Other 45	15.7	Other 52	15.5	Other 29	11.4	Other 23	10.8	Other 2	0.9
In-Service Inspection	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0
	Maint 14	4.2	Maint 1	1.8	Maint 7	3.0	Maint 35	14.4	Maint 5	1.4	Maint 27	8.2
	Eng 9	2.1	Eng 7	2.4	Eng 5	1.6	Eng 12	4.2	Eng 0	0.0	Eng 8	2.5
	HlPhys 2	0.7	HlPhys 0	0.8	HlPhys 0	0.3	HlPhys 0	0.4	HlPhys 1	0.3	GenSupt 10	3.7
	NuA&L 3	1.2	NuA&L 2	0.9	NuA&L 7	2.4	NuA&L 4	1.5	NuA&L 0	0.1	NuA&L 11	3.7
	Other 19	8.2	Other 38	15.2	Other 24	8.0	Other 39	12.4	Other 8	3.1	Other 4	1.4
Special Maintenance	Ops 1	0.3	Ops 0	0.1	Ops 0	0.1	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0
	Maint 40	11.8	Maint 44	14.4	Maint 10	4.0	Maint 79	33.9	Maint 1	0.8	Maint 17	6.7
	Eng 27	9.5	Eng 82	38.2	Eng 0	0.0	Eng 27	13.0	Eng 0	0.0	Eng 4	0.6
	HlPhys 23	8.5	HlPhys 22	6.8	HlPhys 4	1.4	HlPhys 44	10.2	HlPhys 2	1.1	GenSupt 191	100.9
	NuA&L 8	2.0	NuA&L 0	2.5	NuA&L 1	0.7	NuA&L 12	4.3	NuA&L 0	0.3	NuA&L 0	0.0
	Other 121	44.9	Other 63	32.8	Other 5	3.0	Other 118	61.7	Other 21	6.2	Other 1	0.2
Waste Processing	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0	Ops 0	0.0	Ops 0	0.1	Ops 24	7.4
	Maint 0	0.0	Maint 0	0.1	Maint 0	0.1	Maint 5	1.5	Maint 0	0.1	Maint 0	0.0
	Eng 1	0.0	Eng 0	0.3	Eng 0	0.0	Eng 0	0.0	Eng 0	0.0	Eng 0	0.0
	HlPhys 56	16.9	HlPhys 71	24.6	HlPhys 35	10.3	HlPhys 42	12.6	HlPhys 19	5.4	GenSupt 96	35.6
	NuA&L 0	0.0	NuA&L 0	0.1	NuA&L 0	0.1	NuA&L 0	0.0	NuA&L 0	0.0	NuA&L 2	0.5
	Other 3	1.6	Other 2	3.1	Other 2	1.7	Other 7	2.6	Other 5	2.2	Other 4	0.9
Refueling	Ops 9	4.1	Ops 10	3.0	Ops 20	4.4	Ops 0	1.0	Ops 1	0.2	Ops 2	0.4
	Maint 35	11.5	Maint 29	9.5	Maint 28	13.1	Maint 33	11.6	Maint 2	1.0	Maint 12	4.9
	Eng 6	1.6	Eng 15	6.2	Eng 1	0.6	Eng 6	4.5	Eng 0	0.0	Eng 0	0.3
	HlPhys 9	3.5	HlPhys 7	3.7	HlPhys 41	11.0	HlPhys 54	17.5	HlPhys 1	0.4	GenSupt 95	33.6
	NuA&L 2	0.5	NuA&L 2	0.7	NuA&L 0	0.4	NuA&L 0	0.2	NuA&L 0	0.1	NuA&L 0	0.0
	Other 31	8.9	Other 49	13.2	Other 45	15.2	Other 41	11.4	Other 12	3.6	Other 8	2.6
Total Work And Job Function	Ops 19	8.0	Ops 40	9.5	Ops 21	6.3	Ops 15	5.3	Ops 9	4.8	Ops 45	13.3
	Maint 163	47.5	Maint 153	52.2	Maint 166	55.5	Maint 239	88.2	Maint 19	8.8	Maint 161	56.5
	Eng 44	14.3	Eng 114	50.8	Eng 34	13.8	Eng 47	23.0	Eng 0	0.0	Eng 20	6.5
	HlPhys 173	53.6	HlPhys 176	65.8	HlPhys 109	30.9	HlPhys 152	46.1	HlPhys 46	14.8	GenSupt 541	228.1
	NuA&L 16	5.0	NuA&L 15	5.6	NuA&L 17	5.7	NuA&L 18	7.5	NuA&L 0	1.0	NuA&L 20	5.9
	Other 209	76.4	Other 204	84.2	Other 132	46.3	Other 235	103.0	Other 72	28.8	Other 20	6.6

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Organization

12.5.1.1 Program Organization. The Health Physics Division, a part of Nuclear Generation, is responsible for the content of the Health Physics Program at South Texas Project Electric Generating Station (STPEGS). Qualifications of plant personnel are described in Section 13.1.2. The Health Physics Manager is responsible for the overall direction of the health physics program.

The Health Physics Manager delegates various responsibilities to the Health Physics staff composed mainly of supervisors, technicians, health physicists, and specialists. Specific organization of the Health Physics Division and the assignment of responsibilities are left to the discretion of the Health Physics Manager. The Health Physics Manager is responsible for ensuring that all aspects of the overall health physics program are adequately defined and responsibilities assigned commensurate with the assigned duties.

12.5.1.2 Health Physics Objectives. With respect to radiation exposure, the objective of health physics is to ensure that doses from radiation and radioactive materials to personnel onsite is within the limits of 10 CFR 20 and that such exposure is kept ALARA. With respect to radioactive material, the objective is to ensure control of such material consistent with the risk and ensure the material is not inadvertently released from the site.

12.5.1.3 Health Physics Program. The STPEGS health physics program has been initiated and will be in effect continuously until the units are decommissioned. This program consists of policies, practices, and procedures which are used to accomplish the objectives stated above in a practicable and safe manner. The program follows the guidance provided in Regulatory Guide (RG) 8.2. Specifically, the Health Physics Program is contained in the following documents:

- Nuclear Group Policies - Policy statements issued by senior management establishing the expectations of management with respect to the subject of the policy statement.

- Plant General Procedures (PGPs) - Primarily, the PGP03-ZR series of procedures define various components of the Health Physics Program. These procedures contain requirements of the program components and are generically applicable to all site personnel.

- Plant Radiation Protection Procedures (PRPs) - These procedures contain implementation details of the health physics program and are typically applicable to activities performed by Health Physics personnel.

The health physics programs will ensure that the objectives of 12.5.1.2 above are met and the requirements of 10 CFR 20 are implemented.

12.5.2 Equipment, Instrumentation, and Facilities

The health physics facilities, equipment, and instrumentation include:

1. An access control facility, health physics offices, personnel decontamination rooms, laundry, health physics laboratories, counting room, and equipment decontamination room.

2. Protective clothing, respiratory protection equipment, air-sampling equipment, and decontamination equipment.

3. Fixed and portable radiation and airborne radioactivity detection instruments and personnel dosimetry devices.

12.5.2.1 Facilities Related to Health Physics. Unless otherwise identified, the health physics-related facilities discussed below, are located in the Mechanical Electrical Auxiliaries Building (MEAB) of each unit, as shown on the general arrangement drawing listed as Figure 1.2-28 in Table 1.2-1.

The main access control complex in each unit consists of offices and work area, the entrance and exit corridors, change areas, personnel decontamination rooms, toilet and shower facilities, and locker rooms. The offices and work area are located at the entrance and exit corridors to the MEAB and are entered from outside the radiologically controlled area.

Personnel decontamination facilities, located off the exit corridors, are equipped with materials and equipment needed to decontaminate personnel.

Separate locker rooms, showers, and toilet facilities are provided for men and women. Adequate lockers have been provided for routine numbers of permanent and contract personnel.

Donning of protective clothing is performed within the radiologically controlled area. Normally, an area is provided near the entrance corridor for personnel to don protective clothing. During times of heavy use, such as outages, other areas may be designated for this purpose.

The laundry is located near the access control area. Laundry systems are available to clean protective clothing. A laundry monitor is normally used to check the protective clothing for residual radioactivity. Provisions are also made for cleaning and sanitizing respirators. Radiological laundry and respiratory equipment services may be employed from approved vendors. A mobile laundry trailer system may be installed on site to supplement or to be used in lieu of the in plant laundry.

The radiochemistry laboratory, along with the counting room, is designed and equipped so necessary radiochemistry analyses can be performed.

A shielded sampling area containing hoods is adjacent to the radiochemistry laboratory. Sample lines from various radioactive process streams in the unit terminate in the hoods. Samples are collected in the hoods and either analyzed there or taken into the radiochemistry laboratory for analysis, thus reducing possible contamination of the general plant due to accidental spillage of samples during transportation.

The counting room is a shielded room containing the various shielded detectors and radiation detection instruments necessary to analyze the numerous samples. The instruments are discussed in the fixed radiation detection instrumentation portions of Section 12.5.2.2.

The radiation protection instrument calibration facility is located outside the MEAB and is operated under the direction of the Metrology Laboratory Manager. A shielded instrument calibrator that can provide gamma exposure rates from approximately 2 mR/hr to 500 R/hr is used to calibrate most ranges of the portable gamma and beta-gamma survey instruments used on site. The dose rate outside the calibrator is less than 5 mrem/hr. Additional smaller alpha, beta, and gamma sources can be used as necessary to response check the lower ranges of the various portable instruments. A neutron source is used to verify response of the neutron-monitoring instruments with calibration normally performed by an offsite vendor. Dosimeters may be calibrated in this room using the calibrator or other smaller sources, depending upon the range of the dosimeters. The sources used for calibration are traceable to the National Institute of Standards and Technology (NIST) or another standards laboratory.

For major decontamination of tools and equipment, a well-equipped decontamination room is provided at MEAB El. 60.0 ft. The facility contains a double-bowl sink with counter for decontamination of small hand tools. For larger tools and medium-size valves, small tanks are provided. For larger valves and pieces of equipment, 48-inch-deep tanks are provided.

Additional equipment is provided to facilitate decontamination of materials and equipment. For example, an ultrasonic cleaner is provided to obtain the agitation necessary for various chemicals to remove the radioactive corrosion products. The facility is served by a monorail to aid in moving large pieces of equipment. Engineering controls are used to maintain exposures to radiation and radioactive materials ALARA during decontamination.

12.5.2.2 Radiation Protection Instrumentation. The instrumentation used by radiation protection personnel can be divided into five categories: fixed radiation-counting instruments (laboratory type), portable radiation detection instruments, personnel-monitoring instruments, area radiation monitoring instruments, and airborne radiation sampling and monitoring instruments. Most repairs are performed by site personnel. Maintenance may be performed in place (for fixed instruments), in the Metrology Laboratory, or at the vendor facilities, if necessary.

1. Fixed Radiation-Counting Instrumentation

The bulk of the fixed radiation-counting equipment, which can be used for analysis of various air, water and smear samples, is located in the counting room of each unit. Additional equipment may be used elsewhere for such things as counting smears or monitoring hand tools.

The criteria for selection of these various counters were that they could provide the necessary low backgrounds and sensitivities of Nuclear Regulatory Commission (NRC) RG 1.21, the plant operating and environmental monitoring requirements, or good operating practice.

Instruments used to quantify radioactivity levels are calibrated with laboratory standards prepared from materials traceable to NIST or other recognized standard laboratories. The standards are counted in various geometries and/or sample positions that normally are used to count the different types of plant samples.

The laboratory counting equipment available in each unit may change periodically since improvements or new types of equipment may become available or plant requirements may change. The counting equipment in each unit is normally utilized to perform the following types of analysis:

a. Isotopic identification and analysis on air samples and water samples

b. Isotopic identification and analysis on plant smears to determine contamination type and level

c. Gross beta counting for smears and air samples

Equipment is provided for alpha and beta counting and gamma spectroscopy.

2. Portable Radiation Detection Instrumentation

The portable radiation detection instruments include all portable instruments used to perform alpha, beta, gamma, or neutron measurements for radiation or contamination control.

The general criteria for selection of these instruments are that they be rugged, accurate, reliable, and easily serviced, and that they will cover the entire spectrum of radiation measurements expected to be made at the plant during normal operation, shutdowns, and accident conditions.

These instruments are properly calibrated, controlled and operated in accordance with good health physics practice.

Sufficient quantities of each type instrument are available to permit calibration, maintenance, and repair without diminishing the radiation protection supplied. The types of portable survey instruments available include: (1) portable count rate survey meters; (2) ion chambers; (3) high-range gamma dose rate meters; (4) neutron dose rate meters; and (5) alpha survey meters. These may be changed as improvements in instruments occur or plant needs change.

3. Personnel Monitoring Instruments (and Services)

Personnel monitoring instrumentation is provided to measure the radiation dose received by personnel and to determine external and internal contamination levels.

Two types of external dose-measuring devices are utilized: one type (thermoluminescent dosimeter [TLD]) for dose determination (shallow and deep dose equivalent) for required personnel monitoring and one type (direct-reading dosimeter) that is read easily by the individual for "real time" dose monitoring and control. Personnel contamination monitoring equipment is available to determine the location and magnitude of contamination. Both beta and gamma sensitive automated personnel contamination monitors are normally used. A whole body counting system is provided to monitor and quantify personnel internal contamination levels.

The personnel monitoring devices utilized are available in sufficient quantities to permit calibration, maintenance, and repair without diminishing the purpose of supplying the devices. Quantities may fluctuate with plant needs. Typical personnel monitoring devices utilized are as follows:

a. Friskers and automated personnel contamination monitors

b. Direct-reading dosimeters

c. TLD badges

d. Ancillary equipment necessary to support the dosimetry devices in use

e. Whole body counters

f. Excreta containers and associated supplies

4. Area Radiation Monitoring System

The Area Radiation Monitoring System detectors were installed in areas where the radiation level was subject to significant changes and/or which have a high occupancy rate. The system reads out and alarms locally as well as in the control room. For more details see Section 12.3.4.

5. Air Sampling and Monitoring Instrumentation

Air sampling and monitoring instrumentation is used to determine the levels of radioactivity in airborne effluents and to determine the levels of airborne radioactivity in plant areas where personnel are potentially exposed.

Selection of the various types of equipment was based on the following considerations:

- Fixed constant air monitors (CAMs) should be provided on the effluent paths with alarms in the control room to facilitate response to abnormal situations.

- Fixed CAMs are appropriate for in-plant determinations in areas where airborne activity was likely to occur or where airborne activity should be determined during an emergency.

- Portable CAMs are appropriate for in-plant determinations to monitor critical work areas where airborne activity levels are potentially high.

- Portable air samplers should be used for evaluation of airborne activity at some jobsites during maintenance and for routine measurement of air activity levels during normal operation.

The monitors, including the portable CAMs which are used for locations not covered by the Radiation Monitoring System (RMS) air monitors, or to supplement the monitoring by the RMS, are calibrated at lease every 18 months using appropriate sources. Portable air sampler flow rates are also checked and adjusted, as necessary, as prescribed in plant procedures.

12.5.2.3 Radiation Protection Equipment. Equipment is supplied for the protection of personnel as discussed in the following sections.

1. Respiratory Protection Equipment

All respiratory equipment for which a protection factor is assigned has National Institute of Occupational Safety and Health (NIOSH) approval for use in atmospheres containing radioactive materials unless specific exception is granted by the NRC. Respirator types are representative of those commercially available for use in such atmospheres.

2. Protective Clothing

Various types of protective clothing are stocked at the plant to protect against contamination, including liquids. Protective clothing is available for the head, body and extremities.

3. Contamination Control Equipment

Contamination control equipment is used to prevent or limit the spread of radioactive contamination, such as barricade material, postings, and catch containments, and to assist in its removal, such as vacuum cleaners. The equipment is typically stored in or near areas where it is used.

12.5.3 Procedures

Strict adherence to the plant radiation protection procedures ensures that personnel radiation exposures are within the limits of 10CFR20. A discussion of the procedures implemented to keep personnel exposures low is provided in Section 13.5.

Policy and operational considerations for radiation protection are set forth in Sections 12.1.1 and 12.1.3. A general discussion of radiation protection practices to be followed is given in this section.

12.5.3.1 Radiation and Contamination Surveys. Health physics personnel perform radiation and contamination measurements. Areas and frequencies are determined in accordance with plant radiation protection procedures and are based upon accessibility, personnel occupancy, and the potential for contamination and radiation levels changing. Air samples also are taken in normally accessible portions of the Mechanical Auxiliary Building (MAB), RCB, and FHB as prescribed in plant procedures.

This information is supplementary to that continuously obtained by the fixed CAMs, as discussed in Section 12.5.2.2. Additional surveys related to specific operations and maintenance activities are performed. These surveys and measurements, performed prior to, during, or after the activity, or any combination thereof, are based upon the necessity for obtaining information required for protection of personnel. Short-term (high volume) air samplers or portable CAMs are available for evaluation of airborne activity prior to a job or during specific phases.

12.5.3.2 Procedures and Methods to Maintain Exposures ALARA. Procedures and methods to maintain exposures ALARA are not exclusively health physics procedures but include many operating and maintenance procedures. Examples of the various types of procedures and methods that are to maintain exposures ALARA for various operational categories are discussed below.

1. Improved Refueling

Many of the improved refueling features of the Westinghouse Electric Corporation (Westinghouse) design can considerably reduce the time required to perform refueling operations. Rapid refueling is discussed in section 9.1.4.2.

2. Inservice Inspection

Inservice inspection (ISI) is discussed in section 6.6. Use of easily removable insulation and remote inspection techniques provides for decreased time and increased distance from radiation sources, proven techniques for reducing personnel exposure.

3. Radwaste Handling

The handling of radwaste has been minimized by the design of the filters and the relationship of the filters and demineralizers to the radwaste facility. Used filters may be loaded into a shielded container immediately upon removal from the filter housing. Used resins can be sluiced remotely to a shielded storage tank and, after processing, to shipping containers.

4. Spent Fuel Handling, Loading, and Shipping

All spent fuel handling is performed underwater using fuel handling cranes and/or manual tools.

The handling of spent fuel normally requires that a small crew of personnel work in the FHB a few days during the year and usually involves little exposure. Some of the methods used to assure that radiation exposures are ALARA are:

a. An adequate amount of water is maintained above the fuel assembly in the pools to reduce direct radiation.

b. The fuel pool water is filtered to reduce exposure due to water activity and improve clarity.

c. The fuel pool water is cooled to reduce evaporation and potential intakes.

d. Air sampling is performed while moving fuel to evaluate airborne activity.

Written procedures are provided for fuel handling operations.

5. Normal Operation

The plant has been designed so significant radiation sources are separately shielded or located in cubicles. Much of the instrumentation required for normal operation reads out remotely in the control room or outside the cubicles in corridors. Instrumentation which cannot be located remotely, or is read infrequently, is situated where possible so it can be read from the entrance to the cubicle or from a low-radiation area within the cubicle. Survey meters and alarming dosimeters are available for use.

6. Routine Maintenance

Routine maintenance includes preventive maintenance (planned and scheduled maintenance such as lubrication, adjustments, and tests) and corrective maintenance (unscheduled maintenance such as valve packing, pump seal replacement, and stopping leaks). Instructions are available for the usual preventive maintenance jobs and for some recurring corrective maintenance jobs.

Where applicable, the preventive maintenance instructions also list the required lubricants, special tools, and equipment and the acceptance criteria. Additional radiological information, controls, and specific instructions are supplied in work packages or radiation work permits as necessary.

For corrective maintenance jobs in radiation areas, a similar approach is used. Normally, maintenance personnel will determine what has to be done, and plant radiation protection personnel will evaluate the radiological conditions in the work area. Maintenance personnel assemble the required tools and equipment and health physics personnel provide radiological information and instructions. Specific instruction manuals, pictures, or sketches may aid in understanding what is to be accomplished and how it is to be accomplished as safely and quickly as possible. Additional techniques are listed below.

a. Work in areas exceeding an expected threshold deep dose equivalent, stipulated in plant procedures are preplanned. The purpose of the preplanning is to carefully prepare for the job so that it can be performed expeditiously in a proper and safe manner.

b. On complex jobs with potential for considerable personnel exposure, prejob briefings are performed and mockups may be used to familiarize the workers with the exact operations they must perform at the jobsite. These techniques assist in improving worker efficiency and thus minimized the amount of time spent in the radiation field.

c. As much work as practicable is performed outside radiation areas. This includes such tasks as reading instruction manuals or maintenance instructions, adjusting tools or jigs, repairing valve internals, and prefabricating components.

d. For long-term repair jobs, consideration is given by the plant radiation protection personnel to setting up communication systems, or closed-circuit television, so key personnel, e.g., supervisors, can check on work progress remotely.

e. Special tools or jigs are considered when their use would permit the job to be performed more efficiently, would prevent errors, or would increase the distance from the source to the worker. These tools or jigs are used if the total dose, which includes that received during installation and removal, is substantially reduced.

f. Engineering controls are used to minimize the spread of contamination produced during the work.

g. Radiological hazards are posted in accordance with regulatory and procedural requirements. Individuals are trained to stay in the lowest radiation area as much as possible, consistent with performing their assigned jobs.

h. On major maintenance jobs, especially those that involve high radiation levels, post job review is performed to provide guidance at the preplanning stage of future similar operations.

7. Sampling

Most sampling of radioactive systems is performed inside the hoods in the radiochemistry lab. Protective clothing and gloves are used when sampling radioactive systems to prevent contamination of personnel. An instrument is available to check radiation levels while sampling and to determine the level of the sample container. The liquid sampler container is normally washed off with clean water and dried before being taken from the hood for analysis.

8. Calibration

Calibration of most ranges of the portable gamma detection instruments is performed inside a shielded calibrator, thus almost eliminating exposure. Portable sources used to calibrate fixed instruments are transported in shielded containers.

Nonradioactive calibration of instruments may be performed in the shop or in place. Where possible, instruments requiring routine calibration are located where the radiation level is low, and the necessary valves and/or electrical connections are provided so the instrument can be calibrated in place.

12.5.3.3 Controlling Access and Stay Time in Radiologically Controlled Areas. During its construction, Unit 2 was separated by a fence from Unit 1 so construction workers could not enter the protected area of Unit 1. This alleviated the need to badge personnel at Unit 2 during construction. TLD badges situated on the fence or in the work area were used to verify that the radiation exposure of the Unit 2 construction force was minimal.

Personnel entering the protected areas of the plant must receive training or must be escorted. Entrance to the MAB, FHB, and RCB for Unit 1 and Unit 2 is normally through the respective access control points previously discussed in Section 12.5.2. Additional monitoring devices, protective clothing, and respiratory equipment also may be obtained here, if required.

Radiological areas are posted and controlled in accordance with 10CFR20, plant procedures, and the Technical Specifications.

Plant procedures implementing regulations and Technical Specifications specify key controls over entry into radiologically controlled areas. Additional controls are also provided by using RWPs or other written instructions, for example, radiological instructions in craft work packages.

Plant procedures provide for RWPs or other written instructions to describe the work to be performed and the radiological controls necessary to perform the work safely. Approval of radiological controls for performing work in the radiologically controlled area is governed by plant procedures.

12.5.3.4 Contamination Control. Contamination controls for personnel, equipment, and areas are described in the plant procedures. Surveys are performed routinely to determine contamination levels. Additional monitoring may be performed after maintenance work or after an operation which may have increased contamination levels. Any area found contaminated is roped off or otherwise delineated, then posted with appropriate signs, and decontaminated as soon as practicable. In areas where the radiation level is high or where it is considered impracticable to decontaminate the area, entry and egress is via a step-off pad to prevent the spread of contamination. Housekeeping practices that have proved successful at many other plants are also used.

Tools and equipment used in contaminated areas of the radiologically controlled area are controlled to prevent the spread of contamination. All tools and equipment being removed from the RCA are monitored for contamination. If the tools or equipment are contaminated, they are not released for unrestricted use until successfully decontaminated. Decontamination facilities have been discussed previously in Section 12.5.2. Some tools and equipment, because of their design or use, are not considered practicable to decontaminate. These tools and equipment are marked and are used only in a RCA.

The removal of radioactive contamination from surfaces is sometimes necessary to arrest further distribution of the contamination and to reduce the direct occupational radiation exposures from the contaminants. Decontamination is required only when the spread of radioactivity is likely because of operations or if surface contaminants are causing significantly high local radiation levels in the vicinity of a work site. Decontamination of personnel is done as soon as practicable after the contamination is detected.

Control of personnel contamination (external and internal) is provided by using the protective clothing, engineering controls, and respiratory equipment previously discussed in Section 12.5.2. Each individual is responsible for monitoring himself and his clothing when crossing a local control point or the main access control points, which have been discussed previously in Section 12.5.2. If contamination is found, the individual is decontaminated, using one of the facilities previously described in Section 12.5.2, under the direction of health physics personnel.

Area decontamination is expedited by special coatings that are applied to the walls and floors of areas containing radioactive fluids and by a system of floor drains.

In addition, equipment vents and drains are piped directly to sumps or other collection devices to prevent radioactive fluids from flowing across the floor to the drains.

12.5.3.5 Airborne Activity Control. The plant Ventilation Systems (see Sections 9.4 and 12.3.3) provide a means of purging areas of the Reactor Containment to minimize the accumulation of airborne radioactive materials. Air flow is directed from normally occupied or routinely accessible areas of low potential contamination to areas of higher potential contamination. Airborne contamination is minimized by keeping loose contamination levels low and by reducing sources of leakage as much as practicable. The ventilation air flow prevents the buildup of air contamination concentrations.

For jobs that may significantly increase airborne activity, auxiliary ventilation is used where feasible. In some cases, flexible ducts may be connected to the normal exhaust system and then positioned to remove the contaminated air from the work area. In other cases, a mobile HEPA system and flexible ductwork may be used. The mobile unit may be used to recycle the air or to discharge into the normal ventilation system.

If personnel entry is required into areas where the source of airborne radioactivity cannot be removed or controlled, occupancy is restricted and/or respiratory protection equipment is provided consistent with maintaining the total effective dose equivalent ALARA to maintain exposures within the limits of 10CFR20. Entry into these areas is subject to additional written instructions which provides radiation exposure control requirements. Air sampling and job histories are used to select and specify the appropriate device. Whole-body counting, bioassay analysis, and/or nasal smears may be performed to evaluate the protection afforded.

The major portion of the respiratory equipment (previously discussed in Section 12.5.2) is available for entry at the main access control point. Supplementary emergency respiratory equipment is available within the control room envelope and in emergency kits.

The Health Physics Manager is responsible for administration of the respiratory protection program at the plant. To ensure an adequate program, the following controls are incorporated into the program:

1. Procedures are established to ensure adequate personnel training in the correct fitting, use, maintenance, and cleaning of the various types of respiratory equipment.

2. Respirator users are evaluated by medical personnel to ensure that they are able to wear respirators under actual working conditions.

3. Each negative pressure respirator user is fitted during a respirator fit test under simulated conditions to verify that the facepiece will fit properly.

4. Sufficient air samples and surveys are made to identify the nuclides present and to ensure selection of appropriate respiratory equipment.

5. A respiratory protection policy statement has been issued containing all elements required by 10CFR20.

6. Bioassay is performed as necessary to determine intake.

12.5.3.6 Personnel Monitoring. All individuals are required to wear a direct-reading dosimeter and/or a TLD when in an area where direct radiation should be monitored, e.g., areas meeting the minimum requirements for posting a radiation area. However, this requirement may be waived if individuals are escorted by an individual who is wearing a direct-reading dosimeter. TLDs are processed periodically based on TLD characteristics (fade, background accumulation, etc.), or as deemed necessary by Health Physics.

It is not expected that the neutron dose will exceed 300 mrem per quarter; therefore, a calculated neutron dose equivalent measured with portable monitoring instruments and know occupancy times may be used in place of neutron dosimeters. In the event data indicates the neutron dose equivalent does not exceed 30 mrem per quarter, the calculated neutron dose equivalent may be assumed equal to zero. If neutron dosimetry is provided, the dosimetry results are used unless there is reason to suspect its reliability. Bioassay will be performed as necessary to quantify committed dose equivalent and/or committed effective dose equivalent. Gamma sensitive personnel contamination monitors at the main radiologically controlled area exits are capable of detecting an acute intake of 1% of an Annual Limit on Intake for typical plant radionuclides.

Exposure data is collected and recorded on an NRC Form 5 or the equivalent. An attempt will be made to determine prior occupational exposures and, if available, recorded on NRC Form 4 or the equivalent. These records will be maintained and preserved until the NRC authorizes their disposal. Reports of overexposure to radiation workers will be made to the NRC and to the individual involved pursuant to 10CFR20 in accordance with plant procedures.

12.5.3.7 Radioactive Materials Safety Program. The Health Physics Program is designed to control personnnel radiation exposure from the normal by-product, source, and special nuclear material directly associated with the power production aspects of the plant. The various types of sealed and unsealed sources used for calibration are normally used under the direction of personnel who have received training in the safe use and handling of sources.

Controls are prescribed for sources used to calibrate the process and effluent radiation monitors described in Section 11.5, the area radiation monitors described in Section 12.3.4, and the portable and laboratory radiation detection instruments described in Section 12.5.2. Check sources that are integral to the monitors or portable instruments, and that are exempt quantities, do not require special handling, storage, or use procedures for radiation protection. This also applies to exempt quantities of sources used to calibrate or check laboratory instruments.

Recognized methods for the safe handling of radioactive materials will be implemented. External doses are limited by a combination of time, distance, and shielding considerations. Internal doses are controlled by control of loose contamination. The handling of licensed material is addressed in the plant health physics procedures.

Sealed radionuclide sources, except quantities less than those requiring labeling as radioactive materials are subject to material controls for radiological protection. Those controls include:

1. Monitoring of each source for removable surface contamination (leakage testing) at six-month intervals unless excluded by the Technical Requirements Manual.

2. Labeling of each source with the radiation symbol and any information necessary to permit safe handling.

3. Inventory of all sources as prescribed in the Technical Requirements Manual.

Controls are provided for activated and contaminated items consistent with good health physics practice. For example, higher activity radioactive materials are normally stored in locked rooms to preclude inadvertent access. Higher activity waste is segregated from that of lower activity for shielded storage until shipment.

12.5.3.8 Health Physics Training. As part of the general employee training, each member of the permanent operating organization whose duties entail entering restricted areas, or directing the activities of others who enter restricted areas, are instructed in the fundamental aspects of health physics. These personnel are also required to attend a periodic retraining program in health physics. Additional training is provided for personnel who perform work in areas where a typical radiological hazards may be encountered.

An outline of the general employee training in health physics is listed in Section 13.2. The training program includes instructions in applicable provisions of the NRC regulations for the protection of personnel from radiation and radioactive material (10CFR20) and instruction to women concerning prenatal radiation exposure.

In addition to the above training, chemistry and radiation protection technicians receive training in areas that apply to their specific job function such as radiation and contamination surveys, air-sampling techniques, use of portable and laboratory instrumentation, and safe handling of sources. For details, see Section 13.2

TABLE 12.5-1

RADIATION MONITORING INSTRUMENTATION

The approximate quantity, sensitivity, and range of typical survey instruments, TLDs, and direct-reading dosimeters are shown below:

1. SURVEY INSTRUMENTS:

Quantity (Approximate)	Type of Radiation Detected	Range (typical)
20 per unit	Beta, Gamma	1 to 5,000 mrem/hr
5 per unit	Beta, Gamma	10 to 50,000 mrem/hr
2 per unit	Gamma	0.1 to 1000 rem/hr
1	Neutron	0.001 to 5 rem/hr
1 each unit	Alpha	100 dpm to 20,000 dpm

2. TLDs:

1,000	Beta, Gamma	10 mrem to 1000 rem

3. ELECTRONIC DIRECT-READING DOSIMETERS:

200 per unit, normal operation	Gamma	0 – 10 rem
600 per unit, outage	Gamma	0 – 10 rem

4. SPECIAL INSTRUMENATION:

1 per unit to 10,000 rad/hr	Gamma	1