# CHAPTER 12

# RADIATION PROTECTION

This chapter provides information relevant to the radiation protection features of the facility and equipment design, plans and practices to accomplish radiological control, and estimates of occupational radiation exposure to operating and contractual personnel during normal operation and anticipated operational occurrences.

## 12.1. ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

### 12.1.1 Policy Considerations

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It is the policy of River Bend Station to keep all radiation exposure of personnel within the limits established by the federal government as set forth in Title 10 of the Code of Federal Regulations. Administrative procedures and practices maintaining radiation exposure of personnel as low as is reasonable achievable (ALARA) are delineated herein. The specific River Bend Station positions on Division 8 Regulatory Guides, including Regulatory Guides, 8.8 and 8.10, are provided in Section 1.8. The bases for these positions are contained in the discussions found in this section and throughout Chapter 12. The River Bend Station ALARA policy is consistent with the guidelines in Section C.1 of Regulatory Guide 8.8 and Regulatory Guide 8.10 in establishing, organizing, and operating an effective ALARA program.

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The purpose of the ALARA program is to maintain the radiation exposure of plant personnel as far below the regulatory limits as is reasonably achievable.

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The Site Vice President is responsible for implementing the ALARA program by: 1) ensuring that the resources needed to achieve the goals and objectives of the ALARA program are available, and 2) ensuring that the authority for developing the procedures and practices by which the goals and objectives are achieved, measured, and evaluated is delegated to the appropriate personnel (Section 13.1).

The River Bend Station General Manager reports to the Site Vice President and is responsible for the development of plans for review and approval of the radiation protection programs to ensure that radiation exposure of personnel is maintained ALARA.

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The River Bend Station General Manager also assumes responsibility of the overall effectiveness of the ALARA program at the time of fuel loading. This responsibility is carried out through the Manager-Radiation Protection.

The Manager-Radiation Protection is responsible for the development of the required radiation protection programs, including the ALARA program. He reports directly to the General Manager. He reports to management on the effectiveness of the ALARA program. A radiation protection supervisor reports to the Manager-Radiation Protection. This supervisor is responsible for the development of the ALARA program. He provides technical assistance in conducting the ALARA program and in improving the effectiveness of the ALARA program.
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The ALARA Coordinator reports to a radiation protection supervisor and is responsible for coordinating and implementing the established radiation protection ALARA program. He is responsible for maintaing occupational radiation exposure of personnel ALARA. He is responsible for conducting design reviews for plant modifications, plant procedures, and work practices to assure that ALARA objectives are being met.
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Further, it is the responsibility of each supervisor to enforce the requirements for keeping radiation exposure ALARA and the responsibility of each individual to comply with these requirements.

12.1.2 Design Considerations

This section discusses the general methods and features that implement the policy considerations of Section 12.1.1. Detailed provisions for maintaining personnel exposures ALARA are presented in Sections 12.3.2 and 12.5.3.
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The ALARA philosophy was used during the design and construction stages of the River Bend Station by General Electric and SWEC engineers in their continuous reviews of plant design and equipment selection for River Bend Station. Operating experience at BWR's has been incorporated into the River Bend Station design for equipment selection and plant layout to help maintain occupational radiation exposure as low as reasonable achievable.
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During the design and construction stages, River Bend Station performed an independent ALARA design review of River Bend Station. The design review of plant systems and facilities was
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organized and developed within the Radiation Protection Section, then under the direction of the Supervisor - Radiological Programs. An ALARA Coordinator was assigned the responsibility to develop, perform and document the ALARA design reviews. The ALARA Coordinator directed the actual design reviews through the ALARA Review Group, which consisted of GSU Radiation Protection personnel and outside consultants experienced in radiation protection, plant design, operations, and maintenance. The ALARA Review Group performed and documented their reviews in a report to the ALARA Committee. The ALARA Committee was chaired by the Supervisor - Radiological Programs, or as appointed by the Plant Manager, and consists of GSU representatives from the Operations, Maintenance, Chemistry, Radiation Protection , and Engineering Sections. SWEC and General Electric engineers were consulted by the ALARA Committee for design information and for implementing design changes. The ALARA Review Group based on their reports of the plant systems and facility reviews. The actions of the ALARA Committee were approved by the River Bend Station Plant Manager for all design modifications and administrative controls recommended by the Committee to maintain occupational radiation exposure ALARA. The ALARA Committee actions are documented with ALARA Committee Action Items, which provide a history for every ALARA concern identified by the ALARA Review Group.
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12.1.2.1 General Design Considerations for ALARA Exposures

The general design considerations and methods employed to maintain in-plant radiation exposure ALARA have two objectives:

1. Minimizing the amount of time plant personnel spend in radiation areas

2. Minimizing radiation levels in routinely occupied plant areas and in the vicinity of plant equipment expected to require the attention of plant personnel.

Both equipment design and arrangement are considered in maintaining exposures ALARA during plant operation, including: normal operation, radwaste handling, normal maintenance, corrective maintenance, refueling, inservice inspection, other events of moderate frequency, and certain infrequently occurring events.

These considerations are implemented during the development of design drawings in the following manner:

1. Engineering provides Design with general guidance concerning ALARA considerations.

2. Equipment layout and piping drawings are then reviewed and approved by responsible engineering personnel to ensure that ALARA considerations have been incorporated into the design prior to being issued by them.

3. Equipment layout drawings are reviewed and revised as necessary by personnel trained in radiation protection to ensure that layout and shielding provided is adequate to meet established radiation zones.

12.1.2.2 Equipment General Design Considerations for ALARA

Equipment general design considerations to minimize the amount of time plant personnel spend in a radiation area include:

1. Reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventative maintenance.

2. Servicing convenience for anticipated maintenance or potential repair.

3. Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment (including inservice inspection in accordance with ASME Code, Section XI).

4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high.

Equipment general design considerations directed toward minimizing radiation levels in proximity to equipment or components requiring the attention of personnel include:

1. Provisions for draining, flushing, or if necessary remote cleaning of equipment and piping containing radioactive material.

2. Design of equipment, piping, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps.

3. Utilization of welded connections, where feasible, instead of flanged or threaded connections, valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials. In some cases, radwaste equipment is flanged in order to minimize removal time.

4. Provisions for isolating equipment from radioactive process fluids.

12.1.2.3 Plant Layout General Design Considerations for ALARA

Plant general design considerations to minimize the amount of time personnel spend in radiation areas include:

1. Locating equipment, instruments, and sampling stations, which require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation area.

2. Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment.

3. Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area.

Plant general design considerations directed toward minimizing radiation levels in plant areas and in the vicinity of equipment requiring the attention of personnel include:

1. Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high-radioactive fluids not passing through occupied areas).

2. Providing adequate shielding between radiation sources and access and service areas.

3. Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.

4. Providing central control panels to permit remote operation of all essential instrumentation and controls from the lowest radiation zone practicable.

5. Separating highly radioactive equipment, where practicable for package units, from less radioactive equipment, instruments, and controls.

6. Providing means and adequate space for utilizing movable sheilding for sources within the service area when required.

7. Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable.

8. Providing means for decontamination of service areas.

9. Providing remotely operated backflushable filter systems for highly radioactive radwaste and cleanup systems.

10. Providing labyrinth entrances to radioactive pump, equipment, and valve rooms, as required.

11. Providing adequate space in labyrinth entrances for easy access.

12. Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.

12.1.2.4 Decommissioning Design Considerations for ALARA

The radiation protection aspects of the plant design also aid in the ALARA aspects of decommissioning. These include the following:

1. Accessibility for maintenance or removal of equipment.

2. Sheilding to provide protection during maintenance or during storage after termination of plant operations.

3. Provisions for draining, flushing, or decontaminating equipment or piping.

4. Separation of more radioactive equipment from less radioactive equipment.

5. Features to minimize crud buildup.

6. Coatings applied to surfaces likely to become contaminated to facilitate cleanup.

12.1.2.5 Illustrative Examples of ALARA Improvements

Design improvements, which indicate man-rem reductions during operation and maintenance, have been implemented following ALARA reviews. Some examples follow.

Improvement Based on Operational Experience

Operational experience with radwaste filters, sludge processing, and radwaste solidification systems was utilized in the design and selection of components for systems. Use was also made of studies of operating plant experience with similar systems to determine system failure rates, down times, number of personnel and man-hours required to repair the systems, and the man-rem associated with repairs.

Operating experience from other units and the ALARA design features of River Bend Station are utilized in the development and revision of station operating and maintenance procedures and instructions to ensure that occupational radiation exposure is maintained ALARA.

Improvement Based on ALARA Design Review

GSU has performed and documented an ALARA design review of River Bend Station during the design and construction stages as described in section 12.1.2. During the review, the River Bend Station plant facilities and systems were studied in detail to document the design features and any potential design improvements that could minimize the occupational radiation exposure to personnel. The improvements made as a result of the ALARA design review can be categorizied as either administrative controls or design changes.

The administrative controls that were implemented to minimize the occupational radiation exposure at River Bend Station include the following:

1. The identification of potential crud traps to the Radiation Protection Section for incorporation into their radiation surveys prior to maintenance or

2. operation activities in areas near these potential hot spots.

3. The identification of the need to emphasize training of maintenance personnel, including hands-on experience in working in the control rod drive (CRD) maintenance area, prior to handling of radioactive CRD's. This is essential due to the high radiation levels associated with CRD maintenance.

4. The identification of training needs for Radiation Protection technicians in the areas of contamination control and radiological hazards with the CRD changeout, transportation, and rebuild activities.

5. The use of condensate water throughout the plant for a source of flushing and decontaminating water to minimize the amount of radwaste water generated.

6. The identification of training needs for Operations and Maintenance personnel on the component identification system at River Bend Station to minimize unnecessary radiation exposure due to identification errors.

7. Operating procedures for the reactor water cleanup system to minimize dumping of reactor coolant water to the condenser to control buildup of corrosion products in the condenser and the condensate system.

8. The identification of the plant model to the Training Department as a useful ALARA tool for pre-job planning.

9. The use of a dedicated set of operators for radwaste operations will improve the efficiency of radwaste processing and thus minimize the maintenance required.

10. Incorporation of adequate flushing in operating procedures for reactor coolant, radwaste liquid, and resin transfers to minimize radioactive corrosion product buildup.

11. The identification of training needs for Maintenance personnel on the maintenance activities associated with major components in the drywell,

12. such as the main steam isolation valves and recirculation pumps, in order to minimize radiation exposure in potentially high radiation areas.

13. The control of personnel access to areas near the fuel transfer tube during fuel transfer operations, using locked gates, interlocks, and radiation monitors.

The design changes that were implemented to minimize the occupational radiation exposure at River Bend Station included the following:

1. Installation of flush connections on the CRD scram discharge volume headers to minimize radiation fields due to corrosion product buildup.

2. Modification of the condenser sheilding to minimize radiation fields in high traffic areas near the reactor feedwater pumps and instrument racks located in this area.

3. Installation of floor curbs throughout the plant to control the spread of liquid contamination due to leakage from pumps, valves, etc.

4. Installation of flush connections on the liquid radwaste inlet headers to minimize corrosion product buildup.

5. Improved accessibility to plant system components for maintenance and operation activities via platforms, vertical ladders, catwalks, etc.

6. Painting of walls and floors in potentially contaminated areas for ease of decontamination.

7. Use of condensate water for routine sources of flush and decontamination water throughout the plant.

8. Installation of junction boxes for using portable continuous air monitors connected to the digital radiation monitoring system.

9. Design improvements to the reactor water cleanup pump seal design to improve pump reliability.

10. Modification of sheilding around reactor water cleanup backwash tank to minimize radiation exposure in adjacent pump cubicle.

11. Relocation of solenoid operators, instruments, condensate and service air isolation valves outside high radiation areas.

12. Installation of remote monitoring equipment for the condensate demineralizer regeneration process equipment.

### 12.1.3 Operational Considerations

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In accordance with company policy, the radiation exposure of plant personnel is maintained ALARA by means of the radiation protection program presented in Section 12.5.

The radiation protection training provided at River Bend Station is a primary means of instituting the operational ALARA program. The extent of this training provided each person is at least commensurate with his job responsibilities and plant areas frequented. The radiation protection training program is approved by the Manager-Radiation Protection and conducted by either the training department or radiation protection personnel.

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The Radiation Protection Plan is another implementation element of the operational ALARA program. The plan contains the operational ALARA philosophy and is made available to plant personnel. The plan defines management's commitment to ALARA and also designates those individuals who have the responsibility and authority to implement the ALARA program.

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The radiation work permit (RWP) system assists in man-rem tracking and maintaining radiation exposure of personnel ALARA. The RWP system is described in the Radiation Protection Plan and may include the following:

1. Designation of services to be performed on specific components, equipment, or systems.

2. Anticipated radiation, airborne radioactive material, and contamination levels based on current surveys of the work areas and date of survey.

3. Monitoring requirements such as continuous air monitoring, sampling equipment, and radiation surveys.

4. Requirements for adherence to specific radiological precautions, procedures, or practices and special instructions and equipment to minimize exposures to radiation and contamination.

5. Protective clothing and equipment requirements.

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6. Personnel dosimetry requirements including DLRs, or direct-reading dosimeters, or alarming dosimeters, as appropriate.

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7. Estimated exposure time required to complete the tasks and estimated doses anticipated from the exposure.

8. Identification of personnel working on the tasks.

9. Authorization to perform the tasks.

10. Actual exposure time, doses, and other information obtained during the operation to assist with post-job dose evaluation and personnel job and dose tracking. This information can be used to identify deficiencies of problems and may provide the basis for revising procedures, and modifying features or other adjustments that may reduce the man-rems during subsequent similar operations.

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Operations, maintenance, refueling, inservice inspections, and radwaste system operating procedures are reviewed for compliance with ALARA guidelines outlined in Regulatory Guide 8.8, the Radiation Protection Plan, and Section 12 during initial preparation; subsequent revision will be monitored through the pre-job ALARA review and/or information gained from actual post-job evaluations.

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Information gained from actual job exposure evaluations is incorporated into the training programs for the radiation worker, radiation protection technician, and supervisory personnel.

Exposure reviews are performed at least annually by the ALARA Committee and plant management and are used to identify groups with the highest exposure in order to assure that doses are ALARA. Upper management can compare the incremental reduction of doses with the cost of station modifications that could be made.

The radiation work permit system is implemented to help maintain radiation exposure of personnel ALARA by:

1. Requiring specific radiological precautions, procedures, or practices.

2. Providing protective clothing, equipment, and dosimetry requirements for personnel.

3. Setting dose limits as required.

4. Providing for monitoring requirements as needed.

5. Providing a record of doses received by personnel to complete a task.

6. Providing information that can be used in a post-operation review of the task to identify deficiencies or radiological problems. That review may provide the bases for revising procedures, modifying features, or making other adjustments that may reduce exposures during subsequent similar operations.

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In addition to the preceding efforts, all employee are encouraged to submit suggestions on methods of reducing exposure of personnel and/or improving the ALARA program.

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To ensure that personnel comply with established radiological policies, procedures and practices, radiation protection management personnel are charged with the responsibility to promptly advise higher management of any unsafe practices which exceed their authority to correct. They have the authority to halt any operation which in their judgment is unsafe. Radiation Protection Technicians are responsible for notifying the operations shift supervision or radiation protection management immediately in order to stop work on any operation deemed to be radiologically unsafe.

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The radiation protection training, Radiation Protection Plan, radiation work permit system and procedures reviews

all help to ensure that radiation exposure of personnel is maintained ALARA. There are also several work practices or techniques that are used to maintain personnel exposure ALARA. Some of the ALARA techniques include the following:

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1. Personnel required to be monitored for radiation exposure in accordance with 10CFR20.1502 are assigned DLRs to establish exposure history and direct-reading dosimeters to allow determination of accumulated exposure at any time during the job. Dose rate meters are also used, as needed, to provide early evaluation of possible doses to individuals in specific areas.

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2. Work involving high dose rates, high levels of removable contamination, or high levels of airborne radioactivity are preplanned. The purpose of preplanning is to prepare for the job carefully so it can be performed in a proper and safe manner with a minimum exposure of personnel.

3. On jobs involving high dose rates (> 100 mrem/hr), high levels of removable contamination, or high levels of airborne radioactivity, radiation protection coverage is provided as required.

4. On complex jobs or jobs with exceptionally high radiation levels, job briefings and/or dry-run training is utilized and, in some cases, mockups are used to familiarize the workers with the operations that they are to perform. These techniques are beneficial to improving worker efficiency and minimizing the amount of time spent in the radiation field. These efforts are documented, and the experience is used to improve future efforts.

5. On jobs where general area radiation levels are unusually high, stay times are established. This ensures that personnel do not exceed applicable dose limits.

6. On jobs that are known or expected to accumulate significant exposures, the job preplanning includes estimates of the man-rem needed to complete the job. At the completion of the work, a debriefing session is held with the personnel who performed the work (when practical) in an effort to determine how the work could be completed more efficiently, resulting in less accumulated exposure. This

information, together with the procedures used, actual man-rem accumulated, and pertinent radiological information, that may include radiological surveys, contamination reports, ALARA concerns, etc., is compiled and filed for reference to provide guidance during preplanning of similar work situations. This incorporates experience gained in performing these tasks into future work efforts.

7. As practical, entry and exit points for work areas are established in low-level rdiation areas. This is done to minimize dose accumulated while changing protective clothing and respiratory equipment. Access points are also established to minimize the spread of removable contamination from the job site.

8. As much as practicable, jobs are performed outside radiation areas. This includes items such as reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals, and prefabricating components.

9. Individuals working in radiologically controlled areas are trained to be aware of the varying intensities of radiation fields within the general vicinity of their job locations and instructed to remain in the areas of lower radiation levels as much as possible, consistent with performing their assigned tasks. For certain high exposure jobs, maps, postings, and/or detailed instructions are provided to clearly delineate the source of radiation or to alert personnel concerning the location of hot spots and generally higher dose rate areas. Provided with this information and adhering to good radiological work practices, workers will be cognizant of their immediate radiological environment, thus maintaining ALARA exposures.

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10. Protective clothing and respiratory equipment prescribed by radiation protection are commensurate with the radiological hazards involved, and such requirements cannot be decreased by any other personnel. Consideration is given to the discomfort of workers to minimize the effect of protective efforts on efficiency and the time spent in a radiation area.

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11. Contamination containments, i.e., glove bags, plastic bottles, tents, etc., and special ventilation systems are used where practicable when personnel are working on highly contaminated equipment.

12. On some jobs, special tools or jigs are used when their use permits the job to be performed more efficiently or prevents errors, thus reducing the time spent in a radiation area. Special tools are also used if their use increases the distance from the source to the worker, thereby reducing the exposure received.

13. For long-term repair jobs, consideration is given to setting up a communications network, such as sound powered telephones or closed circuit television, to assist supervising personnel in checking on work progress from a lower radiation area.

14. When practical, systems and other pieces of equipment which are subject to crud buildup, such as reactor water cleanup system residual heat removal system, liquid radwaste system, and various pumps, filters, and demineralizers, are equipped with connections which can be used for flushing the system to eliminate potential hot-spot buildup.

    Prior to performing maintenance work, consideration is given to flushing and/or chemically decontaminating the system or piece of equipment in order to reduce the crud levels and hence exposure of personnel.

15. Permanent shielding is used, where possible, to reduce radiation exposure at the work site and in designated "waiting areas" to personnel during periods when they are not actively involved in the work. On some jobs, temporary shielding, such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment, is used. Temporary shielding is used only if the total exposure, which includes exposure received during installation and removal, is reduced. Experience with such operations is used in developing guidelines in this area.

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Operation of the hydrogen water chemistry system results in several changes to the station chemistry balance with impacts to the RBS ALARA program.

1. Injection of excess free hydrogen into reactor feedwater shifts the stoichiometric oxygen concentration in the reactor vessel. The lower oxygen levels can result in free nitrogen combining with the excess hydrogen resulting in generation of volatile $NH_3$ and $NH_4$ compounds versus less volatile NOH compounds. These compounds are removed from the vessel in the main steam and result in increased main steam line dose rates from the additional nitrogen 16. The magnitude of this N-16 shift to steam is based on the feedwater hydrogen concentration.

2. Switching to a hydrogen water chemistry environment from a normal water chemistry environment changes the chemical environment for the BWR from an oxidizing regime to a reducing regime. As such, the iron-based oxide films existing in the primary system are gradually transformed from a hematite structure to a substituted spinel structure. This conversion process increases the transport of insoluble corrosion products, which subsequently deposit on the recirculation system piping and component surfaces, causing increased dose rates and exposures. The intensity of this process appears to be dependant on the amount of hematite-type iron in the plant, the amount of hydrogen injected, and the operation of the hydrogen injection system. The injection of depleted zinc oxide (DZO) into the feedwater is a means of controlling Co-60 transport by retaining Co-60 on the fuel surfaces in a tenacious spinel form. However, high crud loading from the feedwater input will lead to higher dose rates and increase the likelihood of hot spots. Laboratory studies and BWR HWC operating data has shown that shutdown does rates can be 25-50% higher with frequent cycling of the hydrogen system compared to steady-state moderate HWC.

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## 12.2 RADIATION SOURCES

### 12.2.1 Contained Sources

#### 12.2.1.1 General

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with Section 11.1. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average. It should be noted that activation products, principally N-16, control shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transit decay, while at the same time providing for transient increase of the noble gas source, daughter product formation, and energy level of emission. Areas where fission products are significant relative to N-16 include: the condenser off gas system downstream of the steam jet air ejector, liquid and solid radwaste equipment, portions of the reactor water cleanup system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

For application, the design sources are grouped first by location and then by equipment type (e.g., containment, core sources). The following paragraphs represent the source data in various pieces of equipment throughout the plant. General locations of equipment are shown in the general plant arrangement drawings of Section 1.2.

#### 12.2.1.2 Containment

##### 12.2.1.2.1 Reactor Vessel Sources

###### 12.2.1.2.1.1 General

The information in this section defines a reactor vessel model and the associated gamma and neutron radiation sources. This section is designed to provide the data required for calculations beyond the vessel. The data selected were not chosen for any given program, but were chosen to provide information for any of several shield program types. In addition to the source data, calculated radiation dose levels are provided at locations surrounding the vessel. These data are given as potential checkpoints for calculations by shield designers.

12.2.1.2.1.2 Physical Data

Table 12.2-1 presents the physical data required to form the model on Fig. 12.2-1. This model was selected to contain as few separate regions as possible to portray adequately the reactor. Table 12.2-1 provides nominal dimensions and material volume fractions for each boundary and region in the reactor model. To describe the reactor core, Table 12.2-1 provides thermal power, power density, core dimensions, core average material volume fractions, and reactor power distributions. The reactor power distributions are given for both radial and axial distributions. This data contains uncertainties in the volume regions near the edge of the core. The level of uncertainties for these regions is estimated at 20 percent.

12.2.1.2.1.3 Core Boundary Neutron Fluxes

Table 12.2-2 presents peak axial neutron multigroup fluxes at the core equivalent radius. The core equivalent radius is a hypothetical boundary enclosing an area equal to the area of the fuel bundles and the coolant space between them. The peak axial flux occurs adjacent to the portion of the core with the greatest power. As shown by the data in Table 12.2-1, this point is below the core midplane. Since the data are calculated with a core equivalent radius, the flux represents a mean flux in the azimuthal angle around the core. While the flux within any given energy group is not known within a factor of 2, the total calculated core boundary flux is estimated to be within ±50 percent.

12.2.1.2.1.4 Gamma Ray Source Energy Spectra

Core Spectra

Tables 12.2-3a through 12.2-3d present average gamma ray energy spectra per watt of reactor power in both core and noncore regions. In Table 12.2-3a the energy spectra in the core are presented. The energy spectra in the core represent the average gamma ray energy released by energy group per watt of the core thermal power. The energy spectra in MeV per sec per watt can be used with the total core power and power distributions to obtain the source in any part of the core.

The gamma ray energy spectra include the fission gamma rays, the fission product gamma ray, and the gamma rays resulting from inelastic neutron scattering and thermal neutron capture. The total gamma ray energy released in the core is estimated to be accurate to within ±10 percent. The energy release rate above 6 MeV may be in error by as much as a factor of ±2.

Post Operation Gamma Ray Energy Spectra

Table 12.2-3b gives a gamma ray energy spectrum in MeV per sec per watt in spent fuel as a function of time after operation. The data were prepared from tables of fission product decay gamma fitted to integral measurements for operation times of $10^8$ sec or approximately 3.2 yr. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power the element contained during operation.

Noncore Gamma Ray Energy Spectra

In Table 12.2-3c, the gamma ray energy spectra in the cylindrical regions of the reactor from the core through the vessel are given. The energy spectrum is given in terms of MeV per $cm^3$ per sec-watt at the inside surface and outside surfaces of the region. This energy spectrum multiplied by the core thermal power is the gamma ray source. The point on the inside surface of the region is the maximum point with the region. In the radial direction, the variation in source intensity may be approximated by an exponential fit to the data on the inside and outside surfaces of the region. The axial variation in a region can be estimated by using the core axial variation. The uncertainty in the gamma ray energy spectra is due primarily to the uncertainty in the neutron flux in these regions. The uncertainty in the neutron flux is estimated to vary from approximately $\pm$50 percent at the core boundary to a factor of $\pm$3 at the outside of the vessel. The calculations were carried out with voids beyond the vessel. The presence of shield materials beyond the vessel causes an increase in the gamma source on the outside of the vessel.

12.2.1.2.1.5 Gamma Ray and Neutron Fluxes Outside Vessel

Table 12.2-4 presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs on the vessel opposite the portion of the core with the maximum power level. This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation are based on a mean radius core and do not show azimuth angle variations.

12.2.1.2.1.6 Gamma Ray Dose Rates and Fast Neutron Fluxes at the Vessel

Table 12.2-5 presents the calculated fast neutron fluxes with energy greater than 1 MeV and gamma ray dose rates in R/hr at three points outside of the reactor vessel.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam, and Off Gas

The radioactive sources in the reactor water, steam, and off gas are discussed in Chapter 11. This material provides the concentrations during normal operation of the radioisotopes leaving the reactor vessel.

12.2.1.2.3 Reactor Water Cleanup System

The radioactive sources in the reactor water cleanup (RWCU) system are the result of the activity in the reactor water in transit through the system or accumulation of radioisotopes removed from the water. Components for this system include regenerative and nonregenerative heat exchangers, pumps, valves, filter demineralizers, and the backwash receiving tank. Most major components of the cleanup system are located in the containment; however, the RWCU process pumps are located in the auxiliary building. The system is described in Section 5.4.8. The gamma sources for the cleanup system are given in Table 12.2-6. The sources shown for the filters and receiving tank are present during all modes of operation. Therefore, backwashing capability is provided to remove the residual activity for effective radwaste handling.

12.2.1.2.4 Main Steam System

All radioactive materials in the main steam system result from radioactive sources carried over from the reactor during plant operation. In most of the components carrying main steam, the source is dominated by N-16. In components where the N-16 has decayed, the other activities carried by the steam become significant.

12.2.1.2.5 Radiation Sources In The Transversing Incore Probe System

The radiation source for the transversing incore probe (TIP) system is provided in Tables 12.2-13 through 12.2-16. The radiation source is based upon location within the core and residence time. As indicated in the tables, the TIP system is divided into three components for shielding calculations: fissionable material, nonfissionable material, and cable. Sources are provided for each component for 100-sec irradiation and zero-sec decay.

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred underwater while in the cask and loaded into Berylium containers. This is then loaded into the reactor while remaining underwater. The source remains within the reactor for its lifetime. Thus, no unique shielding requirements after reactor operation are required.

12.2.1.3 Auxiliary Building

The basic sources in the engineered safeguard systems during normal plant operation are the result of the radioactive materials in the reactor water or steam being transported to the systems. The design basis source terms for this equipment assume the total activity is the concentration of reactor water or steam decayed for the appropriate time interval.

The design gamma source strengths in the residual heat removal (RHR) system were evaluated for the system operating in the reactor shutdown mode. In this mode, the system recirculates reactor coolant to remove reactor decay heat. The system is operated from approximately 4 hr after shutdown until the end of the shutdown period. The source in the RHR system is the activity in the volume of reactor water contained in the system. The source strengths are shown in Table 12.2-7.

The design gamma source strengths in the reactor core isolation cooling (RCIC) system were evaluated for the system operating under test conditions. During routine testing of the system, the source in the equipment is the activity of the steam driving the system turbine. This activity is dominated by N-16. The radiation source data used in the shield design of this system are shown in Table 12.2-7.

12.2.1.4 Fuel Building

12.2.1.4.1 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in Section 12.2.1.2.1.4, Post-Operation (Table 12.2-3b), in terms of MeV/sec-watt. The design calculation is carried out for a mean element at various times after shutdown.

12.2.1.4.2 Fuel Pool Cooling and Cleanup System

The dominant spent fuel pool liquid source is that which results from crud which may be shaken loose from the fuel rods when they are moved in the pool. Measurements at Nine Mile Point I have been presented in a paper entitled "Corrosion Product Deposits on Fuel at the Nine Mile Point Boiling Water Reactor" by J. Blok, S.G. Sawochka, and D.T. Snyder of General Electric Company at the 19th Annual Meeting of ANS, June 1973. These measurements have been used to estimate the sources expected at River Bend Station. The expected source strengths for the fuel pool heat exchanger, the cleanup filter, and the cleanup demineralizer are listed in Table 12.2-8.

### 12.2.1.5 Turbine Building

#### 12.2.1.5.1 Turbine System

Piping and equipment which contain steam are sources of radiation due to the presence of N-16, the predominant source of activity during operation. Fission product gases (xenon and krypton), water activation products (O-19 and N-17), and the carryover of iodine and other fission products present in steam and condensate are considered as additional activity sources. The carryover is conservatively assumed to be 1.5 percent by weight for halogens and 0.1 percent by weight for other fission products. The N-16 concentrations in equipment are listed in Table 12.2-9. Credit for transit decay for the short-lived N-16 has been factored into the source terms.

#### 12.2.1.5.2 Condensate and Feedwater Systems

The sources in the condensate system are based on decayed main steam activities.

Almost all noncondensables entering the condenser are removed by the air ejectors. Because of this and the relatively long holdup time, the N-16 and other gaseous activity is very low in the hotwell and negligible in the remainder of the condensate system. The activities caused by activated corrosion and fission products are shown in Table 12.2-10 for the condensate and feedwater systems.

#### 12.2.1.5.3 Off Gas System

The off gas system charcoal delay beds located in the turbine building contain sources of radiation based on the holdup of noncondensable gases from the main condenser by the main condenser air removal system. The N-16 activity in the air removal system is shown in Table 12.2-9. The N-16 activity in the off gas system (downstream of the holdup pipe) is negligible due to decay. Therefore, the predominant radiation sources are the fission product gases, xenon, and krypton, and their daughter products. The source terms used for shielding are given in Tables 12.2-11a and 12.2-11b.

### 12.2.1.6 Radwaste Building

The radwaste system sources are radioisotopes, including fission and activation products, present in the reactor coolant. The components of the radwaste systems contain varying degrees of activity depending on the detailed system and equipment design.

The radionuclide sources in the liquid radwaste system such as pipes, tanks, filters, demineralizers, and evaporators used in shielding calculations are listed in Table 12.2-12. Information on the solid radwaste system is provided in Section 11.4.

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### 12.2.1.7 Turbine Rotor Storage Enclosures

The two turbine rotor storage enclosures are located onsite approximately 800 ft southwest of the Turbine Building.

The sources are fixed low level radioactive beta and gamma contamination contained on the two decontaminated stored turbine rotors and associated diaphragms. The contact dose rate for the stored components is a maximum of 3 millirem per hour.

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### 12.2.1.8 Low Level Radwaste Storage Facility (LLRWSF)

The radiation protection design of the LLRWSF, in terms of shielding and dose estimates is based upon the dose rates of the waste containers in the facility when the facility is filled to capacity. Table 11.4-5 includes the approximate waste volumes by type over a four year period. The design basis radiation levels for the wastes to be stored in the facility are based on River Bend Station's historical source terms.

The radiation shielding configuration of the LLRWSF is designed in accordance with the dose rate criteria per 10 CFR 20, USAR and site procedures for both the site boundary and for restricted areas. The nearest site boundary to the LLRWSF is greater than 1350 ft. away. The facility is designed so that the maximum offsite dose received at the site boundary from the waste stored in the LLRWSF will be maintained less than or equal to 1 mRem/yr (see Table 12.4-2) and that the onsite dose due to temporary storage will be maintained within 10 CFR 20. The restricted area is the area surrounding the LLRWSF which will be controlled by the Radiation Protection Group for purposes of protection of individuals from exposure to radiation and radioactive materials. The facility is designed so that the maximum dose when the facility is fully loaded at fence line is less than 0.6 mrem per hour.

The projected waste containers curie content is listed in Table 11.4-5. The location of the LLRWSF is shown in Figure 1.2-2.

### 12.2.1.9 Independent Spent Fuel Storage Installation (ISFSI)

The Holtec International's HI-STORM system radiation protection design, in terms of shielding, is based upon dose rates calculated using the computer code MCNP4A, which is a state of the art Monte Carlo code that offers coupled neutron-gamma transport using continuous energy cross sections in a full three-dimensional geometry. A full analysis was completed and can be referenced with Holtec's Dose to Distance (D2D) document, HI-2043196. As stated in Chapter 9.1.2.5, the ISFSI is located within the protected area of the plant and was designed to contain 40 casks. As stated in D2D, the maximum dose rate at the site boundary would be 3.39E-05 mRem/hr. This dose rate will maintain a less than 1mRem/yr to site boundary and is within 10 CFR 20. Other documents to reference with regard to the ISFSI are the Holtec Certificate of Compliance (CoC), document HI-20443196, Holtec Final Safety Analysis Report (FSAR), NRC Safety Evaluation Report (SER) and the Entergy Nuclear South 10 CFR 72.212 Evaluation Report.

### 12.2.2 Airborne Radioactive Material Sources

Design efforts are directed toward keeping all radioactive material contained. Leaks from process systems, refueling, and decontamination may lead to airborne radioactivity. Equipment cubicles, corridors, and areas routinely occupied by operating personnel do not contain significant airborne radioactivity sources. Radioactive equipment which has the potential for leakage is installed in separate shielded compartments not routinely occupied. In general, the direction of airflow within the building ventilation systems is from areas of low potential for airborne contamination to areas of increasing potential. In this manner, routinely occupied areas are maintained at low levels of airborne radioactivity. Data from operating BWRs corroborate the general lack of airborne activity in corridors and routinely occupied operating areas[2]. Air samples and surface contamination swipe samples are performed to verify the absence of airborne and surface contamination.

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#### 12.2.2.1 Airborne Sources During Normal Operation

The expected airborne radioactivity concentrations in various plant areas for normal plant operation are shown in Table 12.2-17 and are based on the data given in NUREG-0016[1], and EPRI-495[2]. The ventilation system design is evaluated in relation to the location of expected airborne sources as identified in References 1 and 2 to determine the distribution of the radioactivity throughout the plant. These radioactivity concentrations result in the expected annual releases specified in Table 11.3-1, after credit is taken for filtration of radioiodine and particulates. However, determination of the airborne concentrations within the containment is based on plant specific analyses that account for the cleanup provided by the containment purge system.

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The source of airborne radioactivity within the containment during normal plant operation is derived from the assumed discharge of 2,000 pounds per hour of main steam from the safety relief valves to the suppression pool as well as the drywell leakage of radioactivity evolved from the sumps within the drywell. Cleanup of the containment atmosphere is provided by the 7,000 cfm exhaust from the containment purge system.

Corridors and routine access operating areas within the radwaste building are not expected to have significant airborne radioactivity levels. Equipment cubicles are infrequently accessed and may contain low levels of airborne radioactivity, but design provisions are provided to minimize the release of radioactivity.

Radwaste building tanks are filled from the top and as the water splashes into the tanks, dissolved and entrained radioactivity may become airborne. This activity is not released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. Pumps and valves for radioactive systems in the radwaste building are located in separate compartments that are not normally occupied. The radwaste building ventilation design provides airflow from areas of low potential for airborne contamination to areas of increasing potential. This insures that any leakage from radwaste pumps and valves is not directed into normally occupied areas of the building, but is exhausted from the building. An estimate of the airborne levels expected within radwaste building pump rooms is provided in Table 12.2-17. This is based on a distribution of the airborne radioactivity within radwaste building areas as specified in References 1 and 2 and represents a maximum level to be expected within radwaste building equipment cubicles.

The main potential source of airborne radioactivity within the turbine building is leakage from valves on large lines carrying high-pressure steam. The River Bend Station design provides for collection of this leakage and its transport back to the condenser. Therefore, noble gas airborne concentrations are expected to be negligible throughout the turbine building except for within the steam jet air ejector (SJAE) cubicles. These areas normally are not occupied during operation, and the exhaust from these cubicles is exhausted to the environment after filtration to eliminate the possibility of contamination of adjoining areas.

Airborne activities in the auxiliary building are expected to be low except for within the reactor water cleanup (RWCU) pump cubicle. This cubicle is not normally occupied due to radiation levels. However, the exhaust from this cubicle is exhausted to the environment during normal operation and through the standby gas treatment system post accident.

Radioactive equipment in the fuel building which has the potential for leakage is installed in separate shielded compartments which are not normally occupied due to radiation levels. The ventilation in the fuel building directs the air from these cubicles to the environment so that an airborne problem in the normally occupied areas is prevented. The ventilation system is also designed to sweep air from the spent fuel pool surface, thereby removing the major portion of potential airborne contamination. In addition, evaporation from the spent fuel pool is minimized by cooling of the pool.

#### 12.2.2.2 Airborne Sources During Refueling

Experience at operating BWRs has shown that airborne radioactivity can result from the reactor vessel dryer and separator if their surfaces are allowed to dry. Other potential airborne sources could occur during vessel head venting and fuel movement. The airborne radioactive material sources resulting from reactor vessel head removal are minimized by venting prior to removal either to the drywell purge exhaust system or to the main condenser, with vacuum supplied by the mechanical vacuum pump. The contribution to the airborne radioactivity due to the reactor vessel internals is minimized by keeping them wet or submerged. Expected airborne concentrations within the containment during refueling are presented in Table 12.2-17 and are based on data given in NUREG-0016.

#### 12.2.2.3 Airborne Sources for Relief Valve Venting

A special consideration for airborne concentrations within the containment is the radioactivity release via relief valve discharge to the suppression pool. The different modes of this type of discharge and the frequency of occurrence are discussed in Chapter 15. The limiting case is the main steam isolation valve closure. This event results in the release of 834,000 lbm of main steam to the suppression pool over a period of approximately three hours. Using the expected source terms for main steam from Table 11.1-1 and accounting for the partitioning of the radioactivity between the air and water, the airborne radioactivity concentrations within the containment are calculated as shown in Table 12.2-18.

References - 12.2

1. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-Gale Code), NUREG-0016, Rev. 1, January 1979.

2. Sources of Radioiodine at Boiler Water Reactors, EPRI NP-495, Research Project 274-1, Final Report, February 1978.

## 12.3 RADIATION PROTECTION DESIGN FEATURES

### 12.3.1 Facility Design Features

In this section, specific design features to maintain personnel exposures ALARA are discussed. An assessment of compliance of River Bend Station with Regulatory Guide 8.8 is provided in Section 1.8. Applicable discussions can be found in Sections 11.1 and 12.3 through 12.5.

#### 12.3.1.1 Plant Design Description

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of Section 12.1.2 and are outlined in this section for several general classes of equipment (Section 12.3.1.1.1) and several typical layout situations (Section 12.3.1.1.2).

##### 12.3.1.1.1 Common Equipment and Component Designs

This section describes the design features utilized for several general classes of equipment or components. These classes of equipment are common to many of the plant systems; thus, the features employed for each system to maintain minimum exposures are similar and are discussed by equipment class in the following paragraphs.

<u>Filters</u>

Filters which accumulate radioactivity carried in liquids are supplied with the means to remotely backflush the filter or discharge the filter cake.

<u>Demineralizers</u>

Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to the radwaste system for solidification. Fresh resin can be remotely loaded into the demineralizer. Underdrains and downstream strainers are designed for full system pressure drop. The demineralizers and piping are designed with provisions to flush with condensate. Combination vent overflow lines prevent entry of spent resin into the exhaust duct.

<u>Evaporators</u>

Evaporators are provided with chemical addition connections through normal process piping for the use of chemicals for

descaling operations. Space is provided to allow uncomplicated removal of heating tube bundles. The more radioactive components are separated from those that are less radioactive by a shield wall. Wherever practicable, instruments and controls are located on the accessible side of the shield wall. Wherever practicable, valves in radioactive lines are located on the accessible side of the shield wall.

Pumps

Where practicable, pumps are purchased with mechanical seals to reduce seal-servicing time. Pumps and associated piping are arranged to provide adequate spaces for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal, if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease in removal. Pump casings are provided with drain connections for draining the pump for maintenance.

Tanks

Whenever practicable, tanks are provided with sloped or conical bottoms and bottom outlet connections. Overflow lines are directed to the floor and equipment drain system in order to control any contamination within plant structures. Provisions such as compartmentation of tanks and compartment floor drains are made to contain overflows and accidental spills.

Instruments

Instrument devices are located in low-radiation zones and away from radiation sources whenever practical. Primary instrument devices which, for functional reasons, are located in high-radiation zones are designed for easy removal to a lower radiation zone for calibration, if possible. Transmitters and readout devices, whenever practical, are located for servicing in low-radiation zones, such as corridors and the main control room. Some instruments (such as thermocouples) in high-radiation zones are provided in duplicate to reduce the access and service time required.

Instrument sensing lines on process piping which may contain highly radioactive liquids and solids are provided with remote backflushing capability to reduce the servicing time required to keep the lines free of solids. Instrument and sensing line connections are located in such a way as to

avoid corrosion product and radioactive gas buildup. Compression-type fittings are used between instruments and their process lines to allow for easy removal.

Valves

To minimize personnel exposures from valve operations, motor-operated or other remotely actuated valves are used to the maximum extent practicable. Solenoids for air-operated valves in radwaste systems are located on remote racks in low-radiation zones.

Valves are located in valve galleries so that they are shielded separately from major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided so that personnel need not enter the radiation area for valve operation.

When equipment in Zone VI is operated infrequently, only those manual valves associated with safe operation, shutdown, and draining of the equipment are provided with remote-manual operators or reach rods. All other valve operations are performed with equipment in the shutdown mode. Simple reach rods are used to allow operators to retain the feel of whether the valves are tightly closed or not.

For valves located in radiation areas, provisions are made to drain adjacent radioactive components when maintenance is required.

Wherever practicable, valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas.

All manually operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are remotely operated. Personnel do not enter the valve gallery during flushing operations. The valve gallery shield walls are designed for maximum expected filter backflush activities during flushing operations.

Full-ported valves are used in systems expected to contain radioactive solids.

Valve designs with minimum internal crevices are used where crud trapping could become a problem, especially for piping carrying spent resin or evaporator bottoms. Valves are flanged in radwaste systems for ease of removal from piping systems.

Piping

The piping in pipe chases is designed for the lifetime of the unit. There are no valves or instrumentation in the pipe chase. Wherever radioactive piping is routed through areas where routine maintenance is required, pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection requirements. Wherever practicable, piping containing radioactive material is routed to minimize radiation exposure to personnel.

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The majority of the carbon steel Reactor Water Cleanup piping located in the Drywell has been replaced with stainless steel piping to decrease future radioactive crud buildup on the interior surface. In addition, the piping has been electropolished and preoxidized to further enhance the corrosion resistance of the material. Also, the existing drain "Dead Legs", which were crud traps and significant dose contributors, were removed. Snubber reduction was incorporated into this modification to reduce the number of maintenance man-hours spent in the Drywell. These modifications were made to help reduce the area dose rates in the Drywell in accordance with the River Bend Source Term Reduction program.
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Floor Drains

Floor drains and properly sloped floors are provided in each room or cubicle containing serviceable components containing radioactive liquids. When practicable, shielded pipe chases are used for both major and field-run radioactive pipes. If a radioactive drain line must pass through a zone lower than that at which it terminates, proper shielding is provided. Local gas traps or porous seals are not used on radwaste floor drains.

Lighting

Multiple electric lights are used to provide a sufficient margin of illumination for each cell or room containing highly radioactive components so that entry is not required each time a single lamp fails. Incandescent lights, which require less time for servicing than fluorescent lights, are used wherever possible. The fluorescent lights which are

used in some areas do not require frequent service due to the increased life of the tubes. In order to further minimize the exposure of personnel, all burned-out lamps are normally replaced when the system is secured and in the flushed-out condition.

HVAC

The HVAC system design provides for rapid replacement of filter elements and housings.

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Sample Stations

Sample stations for routine sampling of process fluids are located in accessible areas. Shielding is provided at the local sample stations as required to maintain radiation levels. The counting room and laboratory facilities are described in Section 12.5.

Clean Services

Whenever practicable, clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipechases.

12.3.1.1.2 Common Facility and Layout Designs

This section describes the design features utilized for River Bend Station processes and layout arrangements. These features are employed in conjunction with the general equipment designs described in Section 12.3.1.1.1 and are discussed in detail in the following paragraphs.

Valve Galleries

Valve galleries are provided with labyrinths for personnel protection. Floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete floors are covered with a smooth surface coating which allows easy decontamination.

Piping

Pipes carrying radioactive materials are routed through controlled areas properly zoned for that level of activity. Each piping run is individually analyzed to determine the potential radioactivity level and surface dose rate. Where it is necessary that radioactive piping be routed through corridors or other low-radiation zone areas, appropriate shielding is provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Whenever practicable, equipment compartments are used as pipeways only for those pipes associated with equipment in that compartment.

When possible and practicable, radioactive and nonradioactive piping are separated to minimize personnel exposure. Should maintenance be required, provision is made to isolate and drain radioactive piping and associated equipment.

Potentially radioactive piping is always located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (Section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Thermal expansion loops are raised rather than dropped, where possible. In radioactive systems, the use of nonremovable backing rings in the pipe weld joints is prohibited to eliminate a potential crud trap for radioactive materials. Piping carrying resin slurries or evaporator bottoms is run vertically whenever possible, and large radius bends are utilized instead of elbows.

Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Penetrations

To minimize radiation streaming through penetrations, as many penetrations as practicable are located with an offset between the source and the accessible areas. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If these two methods are not used, then alternate means are employed, such as baffle shield walls or grouting the area around the penetration.

Contamination Control

Access control and traffic patterns are considered in basic plant layout to minimize the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. The valves in some radioactive systems are provided with leakoff connections piped directly to the floor and equipment drains.

Decontamination of potentially contaminated areas within the plant is facilitated by the application of suitable smooth surface coatings to the concrete floors and walls.

Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant. In

addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

In uncontrolled access areas where contamination is expected, radiation monitoring equipment is provided (Section 12.3.4). Those systems which become highly radioactive, such as radwaste slurry piping systems, are provided with flush and drain connections. Certain systems have provisions for chemical and mechanical cleaning prior to maintenance.

Concrete shield wall thicknesses are dimensioned for the auxiliary, turbine, fuel, reactor, and radwaste buildings on Fig. 12.3-6 through 12.3-9. These figures also show radiologically controlled areas, personnel and equipment decontamination areas, contamination control areas, location of the health physics facilities, location of area radiation monitors, location of control panels for radwaste equipment and components, location of the onsite laboratory, and location of the counting room.

Equipment Layout

In those systems where process equipment is a major radiation source (e.g., fuel pool cleanup, radwaste, condensate demineralizer), pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. Control panels are located in low-radiation areas.

Major components such as tanks, demineralizers, and filters in radioactive systems are isolated in individual shielded compartments insofar as practicable.

Provision is made on some major plant components for removal of these components to lower radiation zones for maintenance.

Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas.

Exposure from routine inplant inspection is controlled by locating, whenever possible, inspection points in properly shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis

is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required, and permanent shielding is not feasible, sufficient space for portable shielding is provided. In high-radiation areas where routine inspection is required, remote viewing devices are provided as needed. When this is not practicable, written procedures which reduce radiation exposure by reducing the total time exposed to the radiation field are used, and access to high-radiation areas is under direct supervision of radiation protection personnel.

Corrosion Product Control

Control and minimization of the buildup, transport, and deposition of activated corrosion products in the reactor coolant and auxiliary systems are important to maintaining the occupational radiation exposure ALARA. Studies at operating BWR plants have shown that activated corrosion products (primarily Co60) are the most significant contributors to occupational radiation exposures. Several design features are utilized at River Bend Station to minimize the buildup and control the transportation of corrosion products in the reactor coolant and related auxiliary systems. The condensate demineralizers and the reactor water cleanup filter/demineralizers are designed to maintain the required reactor water quality by the removal of corrosion products from the condensate/feedwater and from the reactor recirculation water, respectively. The selection of system components (valves and pumps) and the layout of the system piping are designed to minimize crud trap locations, such as instrument taps off the pipe bottoms. Major components and pipe sections in reactor coolant systems can be isolated and flushed with high quality water to decontaminate the corrosion products built up on internal surfaces prior to maintenance or surveillance operations. Material selection for piping and components in contact with the reactor coolant and feedwater is based on maintaining the amount of cobalt content to a minimum and on minimizing the material erosion and corrosion rates. Reactor vessel internals, feedwater heater tubes, and reactor recirculation piping are all stainless steel material that will not significantly contribute to the activated corrosion products in the reactor coolant. System layup practices and startup operating procedures are established to flush out reactor coolant related auxiliary systems and feedwater system prior to operation so that a minimum amount of corrosion products will enter the reactor water.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled by radiation zoning and access control (Section 12.5.3.3). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures as low as is reasonably achievable and within the standards of 10CFR20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation and shutdown; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Specific zoning for each plant area is shown on Fig. 12.3-1 through 12.3-5b. All frequently accessed areas (e.g., corridors) are shielded for Zone II or III access.

Whenever practicable, the measured radiation level and the location of the source are posted at the entry to any radiation or high-radiation area.

The control of ingress or egress of plant operating personnel to radiologically controlled areas and procedures employed to assure that radiation levels and allowable working time are within the limits prescribed by 10CFR20 are described in Section 12.5.

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Any area having a radiation level which could cause a deep dose equivalent exposure in any 1 hr in excess of 5 mrem is conspicuously posted with a sign or signs bearing the radiation symbol and the words "CAUTION - RADIATION AREA."

In lieu of the control device or alarm signal required by paragraph 20.20.1601 of 10 CFR Part 20, any area in which the intensity of radiation is such that a major portion of the body could receive in any 1 hr a deep dose equivalent greater than
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100 mrem* but less than 1,000 mrem shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

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3. An individual qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area, and who shall perform periodic radiation surveillance at the frequency specified on the RWP by radiation protection.

In addition to the requirements for high radiation areas, accessible areas with radiation levels such that a major portion of the body could receive in 1 hr a deep dose equivalent greater than 1,000 mrem shall be posted at each entrance with a sign bearing the radiation symbol and the words "CAUTION" or "DANGER" and "LOCKED HIGH RADIATION AREA" and provided with locked doors to prevent unauthorized entry. The keys shall be maintained under the administrative control of the Operations Supervision on duty and/or radiation protection. Doors shall remain locked except during periods of access under an approved RWP that specifies the dose rate levels in the immediate work area. For accessible areas that are located within large areas, such as the containment, where

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* Measurement made ≤12 in from the source of radioactivity.

** Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

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no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, and within which radiation levels are such that a major portion of the body could receive in 1 hr a deep dose equivalent in excess of 1,000 mrem*, then that area shall be roped off and conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay-time specification of the RWP, continuous surveillance, direct or remote (such as use of closed-circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.
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Special design features are provided to control access to areas adjacent to the fuel transfer tube to prohibit inadvertant exposure of personnel during fuel transfer. These areas include the containment fuel transfer isolation valve room, the annulus area, and the fuel building midsupport room, areas shown on Fig. 12.3-9 and 12.3-11. A system of administrative controls plus door interlocks and annunciators in the main control room assures that fuel transfer is not conducted when these areas are occupied. The entrance to these areas and the removable access plugs are clearly marked with signs stating that potentially lethal radiation fields are possible during fuel transfer. The removable plugs are locked. These areas are inaccessible once the fuel transfer system is placed in an operational mode. Permanent shielding in these areas is designed to assure that the shield contact radiation levels are no greater than 100 rads per hour. Refer to Figures 12.3-1 through 12.3-9 for the calculated radiation levels.
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## 12.3.2 Shielding

### 12.3.2.1 Design Objectives

The objective of the shielding is to protect the operating personnel and the general public from radiation emanating from the reactor, power-conversion process, and auxiliary systems, including equipment and piping. Shielding requirements in the plant are determined so as to perform the following functions:

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* Measurement made ≤12 in from the source of radioactivity.
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1. Ensure that exposure to radiation of plant personnel, contractors, and visitors is ALARA and within 10CFR20 limits.

2. Limit exposure to radiation of plant personnel in the main control room to within the limits of 10CFR50.67, to ensure that the plant can be maintained in a safe condition under accident conditions. Amendment 132 revised the design basis accident main control room dose limit requirements to incorporate the limits of 10CFR50.67. The limits of 10CFR50 Appendix A, General Design Criteria 19, also remain applicable to the RBS design basis.

3. Limit exposure to radiation of certain components with specified radiation tolerances if they are in high-radiation areas.

4. Limit exposure to persons at the restricted area boundary to a small fraction of 10CFR20 as a result of direct radiation during normal operation.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

In order to meet the design objectives, the following general design guides are used in the shielding analysis of the plant. An assessment of compliance with Regulatory Guides 1.69 and 8.8 in the design of shielding is provided in Section 1.8.

All systems containing radioactivity are identified and shielded, based on the access requirements of adjacent areas. The radiation zone designation for each area and the amounts of shielding needed to ensure these zones are determined. Radiation zones are shown on Fig. 12.3-1 through 12.3-5.

An effort is made to locate process systems in such a manner as to minimize shielding. Use of labyrinths is made in order to eliminate any streaming radiation from equipment. Segregation of radioactive equipment is provided, wherever practicable, such that shielding reduces the dose rate from adjacent components to less than 20 mrem/hr.

Penetrations are placed so that they do not pass through the shield wall in a direct line with the radiation source in order to prevent streaming. If this is not feasible, adequate shielding is provided.

Wherever possible, radioactive piping runs in such a manner as to minimize radiation exposure to plant personnel. This involves minimizing radioactive pipe routing in corridors,

avoiding the running of high-activity pipes through low-radiation zones, using shielded pipe chases where routing of high-activity pipes in low-level areas cannot be avoided, and separating radioactive and nonradioactive pipes for maintenance purposes.

To maintain acceptable levels at valve stations, motor-operated or full-ported valves are used where practical. For valve maintenance, provision is made for drainage of associated equipment so that radiation exposure is minimized. If manual valves are used, provision is made to shield the operator, if practicable, from the valve by the use of temporary shielding or valve stem extensions.

Provisions are made to shield major sources during inservice inspection to allow sufficient access.

The principal shield material is concrete of density 135 lb/cu ft. Alternatively, water, steel, high-density grout, or lead is used as needed.

12.3.2.2.2 Plant Shielding Description

Plant building layouts, which provide locations of equipment containing radioactive fluids and indicate shield wall thickness, radiation zone designations, access control, and radiation monitor locations, are shown on Fig. 12.3-6 through 12.3-9.

The general description of plant shielding in the different plant buildings is as follows.

Reactor Building

Shielding for the reactor building includes the primary shield, drywell, and shield building walls.

The primary shield wall surrounds the reactor and reduces gamma heating in the drywell concrete wall; reduces activation of, and radiation effects on, materials and equipment in the drywell; and provides limited access in the drywell for shutdown periodic inspection and maintenance. The drywell is Zone VI during normal operation.

The drywell wall provides additional shielding in order to permit access to the containment during normal operation.

Within the containment, there are several shielded rooms that enclose reactor water cleanup system equipment and

piping, and safeguard and process equipment. In addition, the main steam lines are within a shielded tunnel.

The shield building wall is a 2 1/2-ft-thick concrete structure that completely surrounds the nuclear steam supply system. This wall attenuates the system radiation to ensure that levels outside the building are less than 0.2 mrem/hr. In addition, in the unlikely event of an accident, the shield building wall shields personnel and the public from radiation sources inside the containment.

Details on the construction of these walls are presented in Section 3.8.

RBS provides shielding designs for the fuel transfer tube and canal commensurate with the guidance of Regulatory Guide 8.8 that results in radiation doses within the limits of 10CFR20.

•→16
Access to areas where contact with the fuel transfer tube may occur is administratively controlled. Further protection against inadvertent personnel exposures is assured through system interlocks that prevent fuel transfer tube operation when these accessible areas are unsecured. Signs are posted stating that potentially lethal radiation fields are possible during fuel transfer.
16←•
Access to the fuel transfer tube area within the reactor building is through a 2 1/2 x 2 1/2 ft opening that is located in the annulus and through an entrance to the isolation valve room in the containment. The access opening in the annulus is plugged with a 5 1/2-ft-thick, solid concrete block to maintain the radiation levels in the area within the zoning requirements. The seismic shake spaces between the steel containment and the structures inside and outside the containment are shielded by steel plates. Fig. 12.3-11 shows the reactor building areas through which the spent fule transfer tube passes.

RBS radiation zone maps identify the maximum expected total radiation levels during operations and refueling with consideration given toward maintaining personnel exposure as low as is reasonably achievable and within the standards of 10CFR20. The zone maps show all the routinely visited areas such as reactor water cleanup (Figure 12.3-4, el 162'-3"), standby liquid control areas (Figure 12.3-3, el 141”-0"), TIP Station (Figure 12.3-2, el 98'-0"), CRD hydraulic control unit (Figure 12.3-4, el 114'-0"), and containment

personnel airlocks (Figure 12.3-4, el 114'-0" and el 162'- 3"). These zone radiation levels include contributions from any potential streaming through the drywell shield wall penetration.

Turbine Building

The anticipated major radiation source in the turbine building is the primary steam containing activation gases, principally N-16, and fission products. Radiation shielding is provided around the following equipment in order to ensure that the required access zone radiation limits are met around the shielded areas:

1. Main condenser
2. Feedwater heaters and drain receiver tanks
3. Steam air ejectors
4. Steam extraction piping
5. Off gas equipment and piping
6. Condensate demineralizers and regeneration facilities
7. Steam seal evaporator
8. Turbines
9. Radwaste steam reboiler
10. Moisture separator/heaters.

Areas within these shields are high-radiation zones and have limited access.

Dose rates to areas north of hydrogen analyzers and off gas sample rack cubicles located at el 123'-6" of the off gas area are estimated to be less than 100 mRem/hr during normal plant operation. Dose rates are gradually decreased to less than 5 mrem/hr to areas north of the glycol cooler/tank/pumps and west of the local panels. A detailed radiation zone description is provided on Fig. 12.3-3.

•→14
In the vicinity of the west side of the condenser there are localized areas where the dose rate is estimated to be greater than 100 mrem/hr.
14←•

•→14
If during operation the dose rate is found to be greater than 100 mrem/hr, then additional shielding will be installed or administrative access control will be implemented.
14←•

Auxiliary and Radwaste Buildings

Concrete walls, removable blocks, labyrinths, and pipe chases are used to shield the safety feature equipment in the auxiliary building and process equipment in the radwaste building, including valves and piping, in accordance with general guides of Section 12.3.2.2.1. Radiation limits are specified for all access zones, and shielding is provided as necessary to maintain the zones within these limits.

Fuel Building

The shielding in the fuel building consists mainly of water and concrete. The water levels are maintained to keep maximum dose rates to 2 mrem/hr when the fuel elements are in the spent fuel racks, as well as from fuel pool cooling and cleanup equipment such as the demineralizers and filters.

•→16
Access to the fuel transfer tube area within the fuel building is through an opening that is 3 ft-0 in by 3 ft-6 in. This opening is blocked with removable solid concrete that is 3 ft-0 in thick to maintain the radiation levels in the area within the zoning requirements. Administrative procedures, which include a locked gate, are employed to control this access. Fig. 12.3-11 shows the fuel building areas through which the spent fuel transfer tube passes.
16←•

Main Control Room

The main control room shielding design is based on the requirements of 10CFR50.67, which requires that personnel can occupy and have access to the main control room following a hypothetical maximum accident, maintain full control, and shut down the plant. Amendment 132 revised the design basis accident main control room dose limit requirements to incorporate the limits of 10CFR50.67. The limits of 10CFR50 Appendix A, General Design Criteria 19, also remain applicable to the RBS design basis.

Direct shielding of the main control room from the fission product inventory in the containment is provided by the concrete walls between them. Emergency air-conditioning and filtration systems are provided for post-LOCA conditions in the main control room, as described in Section 9.4.1.

General Plant Yard Access

Areas adjacent to the off gas treatment facilities, turbine, building, reactor building, and radwaste building that are accessible on an unlimited basis receive less than 0.2 mrem/hr. Protection for these areas is afforded by concrete building walls and by access-controlling fences. Unlimited access areas adjacent to the unshielded condensate storage tanks are protected by fencing and by controlling the level of radioactivity in the stored condensate by passing the stored water through either the condensate demineralizers or the liquid radwaste system before storage.

•→8

Unlimited access areas next to the two stored onsite turbine rotors and diaphragms located approximately 800 feet southwest of the Turbine Building are protected by two modular enclosures that surround the two turbine rotors and diaphragms.

8←•

12.3.2.3 Method of Shielding Design

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The shielding approach and methods are consistent with those described in RP-8A[1]. Where shielded entry ways to compartments containing high-radiation sources are necessary, labyrinths are designed using a general purpose Monte Carlo radiation transport code. The shielding provides a total dose rate contribution at the entry way to below the upper limit of the radiation zone specified for the area.

14←•

12.3.2.4 Post-Accident Access to Vital Areas

NUREG-0737 Item II.B.2 requires that following a design basis accident (DBA) there exist adequate protection of safety-related equipment and the capability for safe and adequate access to vital plant areas consistent with the dose guidelines of GDC 19. At RBS these requirements are met by providing sufficient shielding of components containing post-accident sources. Protection of safety-related equipment is addressed in Section 3.11. This section addresses the capability of safely and adequately reaching and working in designated vital areas following a DBA at RBS. Assessment of this capability is based upon the following considerations:

1. NUREG-0737 source term assumptions.

2. Identification of vital areas and routes to and from these areas.

3. Earliest time post-accident when access to each of these areas is required.

4. Occupancy time required in each of these areas.

5. Walking speeds and driving times along identified access/egress routes.

6. Hoisting times required for moving heavy grab samples.

Based upon the above, dose rates along the designated access/egress routes and at manned locations within the designated vital areas are calculated in order to ensure that anticipated doses to essential personnel from post-accident radiation sources meet the GDC-19 limits of 5 rem whole body, or its equivalent to any part of the body.

River Bend has implemented Alternate Source Term (AST), per Regulatory Guide 1.183, which revised the design basis source term from TID-14844 to NUREG-1465. As allowed per Regulatory Guide 1.183, a scoping/sensitivity evaluation demonstrated that the current vital area access doses using NUREG-0737 bound those using Regulatory Guide 1.183.

12.3.2.4.1 Liquid and Gaseous Source Terms

The post-accident doses within the designated vital areas and enroute to and from these vital areas are determined based upon liquid and gaseous source terms defined in NUREG-0737 as follows:

1. Containment/Drywell Gaseous Source Terms - 100 per-cent of the core equilibrium noble gas inventory and 50 percent of the core equilibrium iodine activity are assumed to be released from the reactor vessel following a DBA LOCA. Of this release, 100 percent of the core noble gases occupy the drywell and containment volumes. Of the 50 percent of the core halogens released, 50 percent plates out in the drywell; the remaining 25 percent of core halogens are distributed in the drywell and containment atmospheres.

   100 percent of the core noble gases and 25 percent of the core halogens are available for release from containment following a DBA LOCA. This source term is the basis for cloud shine doses, ventilation intake doses, shine from ventilation filtration systems, auxiliary building airborne doses, containment shine, and post-accident sampling system (PASS) gaseous piping doses.

   For evaluation of the shielding design and handling of grab samples for plant effluents, an additional source term of 100 uCi/cc of gaseous radioiodines and particulates is assumed for buildup on the filters, in accordance with Regulatory Guide 1.97 and NUREG-0737 Item II.F.1.

   Source Terms for Liquid-Containing Systems - The liquid source term is assumed to consist of

50 percent of the core equilibrium halogen inventory and 1 percent of the remaining core fission products for systems recirculating post-LOCA suppression pool fluids.

For evaluation of the shielding design and handling of PASS liquid samples, an additional source term of 100 percent core noble gases, 50 percent core iodines, and 1 percent remaining core fission products mixed in the reactor pressure vessel coolant volume is assumed. This source term is considered to demonstrate compliance with the Regulatory Guide 1.97 and NUREG-0737 Item II.B.3 requirement for sampling capabilities at 10 Ci/cc.

12.3.2.4.2 Identification of Vital Areas and Access/Egress Routes

The following onsite areas/facilities are identified as requiring access following an accident in order for operators to perform vital accident mitigation and/or recovery tasks:

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1. Operational Support Center (OSC) - The OSC is a staging area for emergency response personnel. The OSC is located in the services building on el 123 ft. Should the OSC become uninhabitable during any portion of the post-accident period, then the Control Room will serve as backup to the OSC. The location of the OSC is shown on Fig. 12.3-15, Sheet 1.

7←•

2. Main Plant Ventilation Exhaust Effluent Monitor Grab Sample Area - Located on the auxiliary building roof (el 171-ft 0 in) near the personnel hatch (as shown on Fig. 12.3-15, Sheet 4), this area is considered vital in order to obtain post-accident grab samples of main plant ventilation exhaust particulate and radioiodine effluents.

3. Post-Accident Sample Station (PASS) Control Panel and Sample Panel - Located on the east side of the auxiliary building on el 114 ft 0-in (as shown on Fig. 12.3-16, Sheet 2, and 12.3-17), this area is considered vital in order to obtain reactor coolant and containment gas samples for radiological and chemical analyses following a LOCA.

4. Chemistry Laboratory - This facility (shown on Fig. 12.3-15, Sheet 7) is located on el 109 ft-3 in of the services building. Following an accident it is the principal laboratory where grab samples are taken to be analyzed. This area is served by a filtered ventilation system. In the event that post-accident radiation levels become restrictive in this laboratory due to large radioactive releases and/or unfavorable meteorology, then the Environmental Services Group Laboratory will serve as a backup facility.

•→11

5. Primary Access Point - The primary access point (PAP), located on the north side of the services building, is the security center for RBS and serves as the facility through which personnel enter and exit the protected area. Should post-accident radiation levels in this facility become prohibitively large, by which time all nonessential personnel will have been evacuated from the site in accordance with the emergency response plan, then the security supervisor will lock all access gates to the protected area. The guard force will move from the PAP to the emergency operations facility (EOF) in the training building and maintain control of unauthorized access to the site. A backup security terminal for protected area access and egress and for personnel accountability is located in the TSC.

11←•

6. Main Control Room - Following an accident, the main control room (located on el 136 ft-1 5/8 in of the control building) serves as the primary emergency response facility. As such, this facility is designed for continuous occupancy. Post-LOCA habitability of the main control room is discussed in Section 6.4, and post-LOCA doses in this facility are presented in Table 15.6-7.

7. Technical Support Center (TSC) - Following an accident, the TSC serves as a backup emergency response facility to the main control room. The TSC is designed for continuous occupancy to serve the needs of emergency operations management and support. The TSC is located on el 123 ft-6 in of the services building and is shown on Fig. 12.3-15, Sheet 1, and Fig. 12.3-16, Sheet 1.

It has been determined that the post-accident access to areas addressed in NUREG-0737, Item II.B.2, which have not

been identified above are not required at RBS. The plant design is such that the accident mitigation functions associated with these additional areas can be performed from the main control room.

Following an accident, essential support plant personnel will be assembled in the OSC and dispatched at appropriate times to the designated vital areas. Since the PAP and the main control room are continuously occupied during normal operations, it is reasonable to expect that essential personnel will be present in these areas when the accident occurs. The main plant exhaust duct effluent monitor grab sample area and the PASS room are vital areas for which access and egress routes have been determined. The egress routes from these two areas lead to the chemistry laboratory in the services building before terminating at the OSC.

Obtaining a post-LOCA iodine grab sample at the main plant exhaust duct monitor is postulated in the RBS vital area dose analysis to be a two-phase effort in order to minimize the dose to any one individual. Phase I personnel start from the OSC and travel to the grab sample area to disconnect the shielded grab sample (weighing approximately 275 lb) and mount it onto a cart. From there, the grab sample is moved by cart to the west side of the auxiliary building roof, where it is left for Phase II personnel to transport to the chemistry laboratory. Phase I personnel proceed down the west stair tower to a waiting vehicle, drive to the services building, and return to the OSC. Phase II personnel start from the OSC and travel by truck to the west side of the auxiliary building, up the west stair tower, and over to the waiting cart and grab sample. Phase II personnel rig the shielded grab sample to a jib crane, lower the grab sample onto the waiting truck, walk down the stairs to the truck, and then drive to the services building. Phase II personnel then transport the grab sample by cart to the chemistry laboratory and finally return to the OSC. The details of these scenarios are presented in Tables 12.3-5 and 12.3-6. Fig. 12.3-15, Sheets 1 through 7, illustrates the routes traveled.

Retrieval of a shielded liquid PASS sample is assumed to be performed in a single phase. The sample retrieval scenario is outlined in Table 12.3-7, and the pathways are illustrated on Fig. 12.3-16, Sheets 1 through 3.

12.3.2.4.3 Major Dose Contributors in the Vital Areas

To determine post-accident doses to personnel going to and working in the previously identified vital areas, the

following radiation contributors are considered in the RBS vital areas analysis:

1. Auxiliary Building

    a. ECCS piping

    b. Containment shine (including shine through personnel hatch at el 170 ft)

    c. Building airborne activity

    d. Overhead cloud shine on auxiliary building roof

    e. Reactor vessel liquid and containment airborne sources in PASS room piping and in the grab samples

    f. Effluent iodine collected on main plant exhaust duct effluent monitor grab sample filters.

2. Services Building - TSC

    a. Containment shine

    b. Overhead cloud shine

    c. Intake of containment releases

    d. Intake of ESF leakage

    e. TSC intake filter shine.

3. Services Building - Chemistry Laboratory

    a. Containment shine

    b. Overhead cloud shine

    c. Intake of containment releases

    d. Intake of ESF leakage

    e. Radiation from grab samples during analyses.

4. OSC, Corridors, and General Areas

    a. Airborne cloud doses

b. Radiation from grab samples along the pathways to the chemistry laboratory

c. Containment shine.

12.3.2.4.4 Vital Areas Access Design Review Results

Results of the RBS vital area access design review, together with major assumptions, are presented in Tables 12.3-5, 12.3-6, 12.3-7, 12.3-8, and 12.3-9. These results show the following:

1. No worker will be exposed to gamma doses in excess of GDC-19 limits if the iodine effluent grab sample retrieval operation is begun as early as 7 hr post-LOCA and this operation is carried out in two phases.

2. No worker will be exposed to gamma doses in excess of GDC-19 limits while retrieving PASS samples if this effort is initiated as early as 1 hr post-LOCA.

3. The TSC design meets the radiological habitability criteria in GDC-19, SRP 6.4, NUREG-0696, and NUREG-0737.

4. The gamma doses to a laboratory technician in the chemistry laboratory will meet the GDC-19 criteria as early as 1 hr post-LOCA. The shielded hot storage room and the unshielded hot laboratory may each be occupied for up to 30 min at 1 hour post-LOCA.

5. The use of a pressure demand self-contained breathing apparatus may be required in order for GDC-19 and SRP 6.4 thyroid dose limits to be met along access/egress routes and in vital areas not serviced by filtered ventilation intakes.

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12.3.2.4.5 Plant Design Features Necessary to Implement Design Review Results

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2•→
The following design changes were implemented in order to support the vital area access scenarios:

1. A jib crane on the west side of the auxiliary building roof to lower the iodine sample onto a truck waiting below.

2. A moveable hoist above the el 123 ft-6 in floor in the east-west tunnel near the auxiliary building stair landing will facilitate the lifting of the shielded PASS sample and cart from el 114 ft-0 in. in the auxiliary building to the east-west tunnel at el 123 ft-6 in.

3. Extension of the shield wall near the main plant exhaust duct radiation monitor.

4. Shielding between PASS system components.

2←•

## 12.3.3 Ventilation

### 12.3.3.1 Design Objectives

The primary function of the plant ventilation systems is to provide suitable environmental conditions that are safe and comfortable for operating personnel and adequate for the functioning of equipment, and to provide effective protection for operating personnel against possible airborne radioactive contamination. The systems are designed to operate such that the in-plant airborne radioactivity levels for normal operation (including anticipated abnormal operational occurrences) for normally occupied areas within plant structures containing radioactive components are within the limits of 10CFR20, Appendix B, Table I. For other buildings, including all administration areas, airborne radioactivity levels are maintained within the limits of 10CFR20, Appendix B, Table II. The maximum values are not expected to be greater than maximum permissible concentration (MPC) during normal operation. Section 12.2.2 discusses airborne radionuclide sources for the fuel building, the reactor building, turbine building, auxiliary building, and radwaste building. Exposures within the plant are discussed in Section 12.4.1.

The main control room ventilation system is designed to provide a suitable environment for equipment and continuous personnel occupancy in the main control room under all modes of plant operation in accordance with 10CFR50, Appendix A, Criterion 19 and 10CFR50.67.

12.3.3.2 Design Guidelines

In order to limit and reduce the airborne radioactivity in accordance with the design objectives, the following general design guidelines are employed to the maximum extent practicable.

Guidelines to Minimize Airborne Radioactivity

1. For radioactive systems, equipment vents and drains are piped directly to a collection device connected to the collection system instead of allowing any contaminated fluid to flow across the floor to a floor drain.

2. All-welded piping systems are employed on contaminated systems to the maximum extent practicable. Gasketed flanged connections are used for equipment where welded connections are impractical.

3. The valves in systems containing radioactive fluids are provided with leakoff connections piped directly to the condenser.

4. Metal diaphragm or bellows seal valves are used for off-gas system (Section 6.2.4).

5. Contaminated equipment is designed to minimize the potential for airborne contamination during maintenance operations. These features may include 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.

6. All sinks and chemical laboratory work areas where radioactive samples or material are handled are provided with exhaust hoods to protect operating personnel from airborne contaminants.

7. Transient airborne contamination may result due to maintenance. Special procedures such as system

isolation and glove boxes are instituted to minimize the contamination on a case-by-case basis.

8. Filters in all systems are changed based upon the airflow and the pressure drop across the filter bank. In the case of prefilters, a pressure drop equivalent to 1 in of water across the bank is cause for changeout. The HEPA filters are changed when the pressure drop reaches 2 in of water equivalent across filter bank. The charcoal adsorbers are changed based on the residual adsorption capacity of the bed as measured by test samples or canisters removed and analyzed at intervals.

9. The majority of the accumulated radioactivity in the filtration units is removed by replacing the contaminated filters.

Guidelines to Control Airborne Radioactivity

1. Air movement patterns are provided from areas of low potential radioactivity to areas of progressively higher potential radioactivity prior to final exhaust.

2. Areas with radioactive contaminants are kept under negative pressure to minimize exfiltrationof contamination. Positive pressure is maintained in the main control room during normal operation and after postulated accidents to minimize infiltration of potential contaminants.

3. HEPA filters and charcoal filters are provided on exhausts from the fuel, auxiliary, containment (including drywell), and portions of radwaste and turbine buildings to remove airborne radioactivity and to reduce onsite and offsite radiation levels. The Engineered Safety Features (ESF) filter units for the fuel, auxiliary, containment, and control buildings are designed and constructed in accordance with Regulatory Guide 1.52 with the exceptions listed in Table 6.5-1. The charcoal filter units for the radwaste, turbine, and containment purge systems are designed and constructed in accordance with the requirements of Regulatory Guide 1.140 with the exceptions listed in Table 9.4-6

4. The fresh air supply to the main control room is from remote air intake and is designed to be operable during loss of offsite power and LOCA. The air is filtered and passed through HEPA filters and charcoal adsorbers to prevent contamination of the control room by excessive radioactivity.

5. Containment isolation valves are installed in various systems in accordance with General Design Criteria 54 and 56, including valve controls, to assure that the containment integrity is maintained (Section 6.2.4).

6. The main control room emergency filtration system, SGTS, and fuel building charcoal filtration systems are provided with redundant Seismic Category I equipment to control the spread of airborne radioactivity. The extent to which redundant components are provided is discussed in Sections 9.4 and 6.5.

7. Atmospheric tanks and condensate demineralizer area tanks which contain radioactive materials are exhausted to the atmosphere through charcoal filters of their respective building ventilation system.

8. The primary drywell/containment purge system reduces airborne radioactivity within the drywell/containment to acceptable levels prior to release to containment. A second system is provided to operate in the event of either mechanical failure of the normal operating system or to purge the containment and drywell at a faster rate for personnel entry. The purge air is exhausted through the standby gas treatment system prior to exhaust to remove airborne iodine and particulates.

Guidelines to Minimize Personnel Exposure from HVAC Equipment

1. The ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel exposure (Fig. 12.3-12 through 12.3-14). The HVAC system is designed to allow rapid replacement of components.

2. Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts to the extent practicable.

3. Air is recirculated in clean areas only.

4. Access and service of ventilation systems in radioactively contaminated areas is expedited by component location to minimize personnel exposure during maintenance, inspection, and testing as follows:

   a. The outside air supply units and building exhaust system components are located in ventilation equipment rooms and in general areas. These units are located in radiation Zone I or II, and are accessible during plant operation. Work space is provided around each unit for anticipated maintenance, testing, and inspection.

   b. Local cooling equipment, servicing the normal building requirements, is generally located in areas of low contamination potential, radiation Zones I, II, III, and IV. The drywell coolers are located in Zone VI. However, fully redundant components are provided, which eliminate the need for servicing during plant operation, thereby allowing servicing to be deferred until the next shutdown period. Several turbine building unit coolers are located in Zone VI; however, these are expected to require a minimum of maintenance which can be scheduled during shutdown.

12.3.3.3 Design Descriptions

The ventilation systems which have been designed in accordance with the guidelines in Section 12.3.3.2 are described below.

The expected gaseous effluents for radiologically significant areas such as the auxiliary building, off gas building, radwaste building, fuel building, reactor building, and turbine building are provided in Table 11.3-1.

12.3.3.3.1 Main Control Room Ventilation

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During normal plant operation, main control room air is recirculated through an air-conditioning unit to maintain main control room design conditions of 80°F and 70 percent maximum relative humidity. Outdoor air for ventilation and pressurization is normally provided from a local air intake located on the roof of the control building at el 161 ft-0 in. A radiation monitoring system and a control room charcoal filter recirculation system have been provided to detect and reduce radiation levels in the main control room in the event of a DBA. Redundant radiation monitors in the local outdoor air intake duct detect high radiation in the outdoor air supply. A high radiation signal alarm in the main control room automatically diverts the outdoor air to charcoal filtration units, shuts down and isolates the utility exhaust fan, and starts the emergency charcoal filtration unit trains. Redundant radiation monitors are also provided in the remote air intake duct. The use of remote air intake is by operator action. When the remote air intake damper is opened, the normal intake isolation damper is closed manually and the air is passed through charcoal filtration units to maintain a positive pressure. An area monitor located in the main control room verifies that radiation levels are kept within acceptable limits.

16←•

There are two 100-percent capacity charcoal filter trains to provide clean, filtered air during DBA. The charcoal filters are activated carbon which have a 98-percent efficiency for elemental iodine removal and 98-percent efficiency for organics at 25°C and 70-percent relative humidity. A complete description of main control room ventilation system and habitability is described in Sections 9.4.1 and 6.4, respectively.

12.3.3.3.2 Fuel Building Ventilation

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The fuel building is ventilated by an air-conditioning unit (with two redundant supply fans) and air distribution ductwork. During normal operation ventilation air is exhausted through an exhaust fan. During a fuel handling accident, the supply air system is shut off and the outside air is supplied through an intake duct. The building air is exhausted through charcoal filtration units. The exhaust system is designed to automatically route the exhaust air through charcoal filter units upon detection of high airborne radioactivity in the building exhaust by redundant radiation monitoring systems.

14←•

Independent unit coolers are provided in various cubicles to maintain a design ambient temperature. The air is recirculated through chilled water cooling coils. A complete description of fuel building ventilation is given in Section 9.4.2.

12.3.3.3.3 Containment

The containment cooling system consists of recirculating air-conditioning units which maintain the design containment temperature and relative humidity.

During normal system operation, the containment coolers recirculate the air through chilled water cooling coils. Vaneaxial fans are provided above the crane rail to avoid stagnation in the dome area and thus prevent accumulation of radioactive gases.

During normal operation, the containment purge system is used intermittently in order to maintain airborne concentrations in the containment less than 25 percent of maximum permissible concentrations as given in 10CFR20, paragraph 20.203.D(1)(ii) and Appendix B, Table 1, Column 1. The drywell is purged only during periods when personnel entry is required. A backup system is provided to operate in the event of either a mechanical failure of the normal operating system or to purge the containment and drywell at a faster rate for personnel entry. During a DBA the containment and drywell purge system are shutdown.

A complete description of the containment ventilation system is found in Section 9.4.6.

12.3.3.3.4 Drywell

The drywell cooling system consists of recirculation type unit coolers to maintain the design drywell temperature and relative humidity. Six unit coolers are provided to distribute cooled air effectively through ductwork. The cooling water for the drywell cooling system is provided by the plant service water system.

The containment continuous purge system is operated by opening the isolation valves to the drywell. The containment purge system can be used for drywell purge, containment purge, or recirculation cleanup. In any mode of operation, the activity of the air is continuously monitored.

A complete description of drywell ventilation is found in Section 9.4.7 and containment purge system in Section 9.4.9.

12.3.3.3.5 Annulus Pressure Control System

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The annulus pressure control system is designed to maintain a negative pressure of at least -3 in W.G. in the annulus space during normal plant operation. Two 100-percent capacity centrifugal exhaust fans are provided to exhaust air from the annulus to the plant exhaust duct. The suction of the two exhaust fans is connected to a common duct which penetrates the shield building to draw the annulus atmosphere.
12←•
Redundant radiation monitors are provided in the exhaust duct and the air is continuously monitored for radioactivity prior to discharge to atmosphere. Following LOCA and/or detection of high airborne radioactivity concentration, the annulus exhaust is diverted to the SGTS automatically. A pressure differential switch is provided to alarm the operator in the main control room when the pressure exceeds -3 in W.G.

A complete description of annulus pressure control system is found in section 9.4.6.

12.3.3.3.6 Auxiliary Building

The auxiliary building ventilation system is designed to maintain design ambient air temperature and relative humidity. Outside air is supplied to various areas of the building through ductwork distribution system. The air is exhausted to the plant main exhaust duct by an exhaust fan and ductwork.

The exhaust air is continuously monitored for radioactivity. Upon detection of high airborne radioactivity, the normal supply and exhaust fans are manually shut off and air is exhausted through the SGTS. In this mode, negative pressure is maintained within the auxiliary building to prevent outleakage of radioactive air.

Unit coolers are provided in various cubicles to maintain design ambient temperature. The auxiliary building electrical tunnel is ventilated by drawing outdoor air through the outdoor air intake duct and discharging the air to the plant exhaust duct.

A complete description of auxiliary building ventilation system is found in Section 9.4.3.

12.3.3.3.7 Radwaste Building

The radwaste building is ventilated by six unit coolers connected to a common intake plenum. Exhaust air from the building is provided by two separate systems. One system serves the general areas and equipment cubicles and the other exhausts air from the area in which tanks containing potentially radioactive fluids are located. The tanks with potentially high levels of radioactivity are vented through iodine filtration units. The radwaste building exhaust is continually monitored for high radiation.

A greater volumetric flow is exhausted from the radwaste building than is supplied to it to maintain a negative pressure. A complete description of the radwaste building ventilation system is found in Section 9.4.3.

12.3.3.3.8 Turbine Building

The turbine building ventilation air system delivers outdoor air to the areas of low airborne contamination of the turbine building. An exhaust air duct system is connected to each of the potentially contaminated cubicles. All the air is exhausted through contaminated cubicles thus maintaining an air flow pattern from areas of low potential radioactivity to areas of high potential radioactivity. In this way, clean area passageways are kept free of radioactive contaminants.

The vault area in the off gas building is served by a low temperature recirculating refrigeration system.

•→16
Radiation monitors are provided in the turbine building and off gas building exhaust ducts. Mechanical vacuum pump discharge is drawn through an iodine filtration unit prior to discharge to the plant exhaust duct.
16←•
A complete description of the turbine building ventilation system is found in Section 9.4.4.

12.3.3.4 Air Cleaning System Description

The air cleaning systems which utilize special filtration equipment to limit airborne radioactive contaminants are:

1. Standby gas treatment system (SGTS) - described in Section 6.5

2. Main control room charcoal filtration system - described in Sections 9.4.1 and 6.4

3. Fuel building charcoal filtration system - described in Section 9.4.2

4. Continuous containment purge system - described in Section 9.4.6

5. Radwaste building exhaust filtration system - described in Section 9.4.3

6. Mechanical vacuum pump system - described in Section 9.4.4.

The guidance and recommendations of Regulatory Guide 1.52 concerning maintenance, in-place testing provisions for atmospheric cleanup systems, and air filtration and adsorption units have been used as a reference in the design of the various safety-related charcoal filter systems. The extent to which Regulatory Guide 1.52 has been followed is discussed in Section 6.5. Nonsafety-related charcoal filters are designed to meet the requirements of Regulatory Guide 1.140 as discussed in Section 9.4.3.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or in-place testing operations are as follows:

1. The filter adsorption of radioactive material during normal plant operation is a slow process; therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity buildup on a scheduled maintenance basis with portable equipment, and the filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard. Filter sections in which the radioactivity level (due to a postulated accident) is such that a change of the filter elements constitutes a personnel hazard are removed and replaced. Therefore, the filters can be maintained as necessary after a DBA, and it is not necessary for workers to handle filter elements until the short-lived isotopes have had sufficient time to decay to minimize exposure during replacement.

2. Active elements of the atmospheric cleanup systems are designed to permit ready removal.

•→8

3. The filter units are located in separate independent rooms. Adequate access to active elements is provided to simplify element handling. Ample space is provided in the filter rooms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling facilities, and during any in-place testing operation. The provision for inservice inspection, testing, and decontamination of charcoal filter units is conducted in accordance with the requirements discussed in the Technical Specification/ Requirements Manual.

8←•

4. Typical layout with minimum distances for access and servicing is shown on Fig. 12.3-12, 12.3-13, and 12.3-14. No filter bank is more than three filter units high; each filter unit is 2 ft by 2 ft.

5. The clear space for doors throughout the plant is a minimum of 2 ft-6 in by 7 ft.

6. The filters are designed with replaceable 2 ft by 2 ft units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with marine bulkhead type doors that are closed against compression gaskets.

The charcoal adsorber cells are gasketless type designed for ease of installation and replacement. The cell module is designed so that charcoal can be added to or removed from the cell as necessary without violating the integrity of the cell. Externally mounted test canisters are provided for laboratory testing of charcoal.

### 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

#### 12.3.4.1 Area Radiation Monitoring Objectives

•→12

The area radiation monitoring system is provided to supplement the personnel and area radiation survey provisions of the health physics program described in Section 12.5 to ensure compliance with the personnel radiation protection guidelines of 10CFR20 and 10CFR50.

12←•

Provisions for post-accident monitoring are also addressed. The specific River Bend Station positions on Regulatory Guides 1.97, 8.2, 8.8, and 8.12 are provided in Section 1.8. Bases for these positions are contained in the discussions found in this section and throughout Chapter 12.

Consistent with this purpose, the area radiation monitors function to:

1. Alert plant personnel entering or working in nonradiation or low-radiation areas of increasing or abnormally high-radiation levels.

2. Inform the main control room operator of the occurrence and the approximate location of an abnormal radiation increase in nonradiation or low-radiation areas.

3. Comply with the requirements of 10CFR50, Appendix A, GDC 63, for monitoring fuel and waste storage and handling areas.

4. Comply with the requirements of 10CFR50, Appendix A, GDC 64, to monitor post-accident radiation levels in the reactor containment and drywell.

5. In general, assist in maintaining personnel exposures ALARA.

The area radiation monitoring system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment.

12.3.4.1.1 Area Radiation Monitoring System Design Criteria

The following design criteria are applicable to the area radiation monitoring system:

1. Range - To cover the various ranges anticipated in the plant five different models of the basic instrument with the following scales are provided:

•→12

$10^{-2}$ to $10^{3}$ mrem/hr
$10^{-1}$ to $10^{4}$ mrem/hr
$10^{0}$ to $10^{5}$ mrem/hr
$10^{0}$ to $10^{8}$ rem/hr (for containment and drywell post accident monitoring)
$10^{-2}$ to $10^{4}$ R/hr (fuel transfer area monitors)

12←•

2. Alarms - Each area radiation monitor is provided with a red beacon to alarm a high radiation condition, an amber beacon to alarm an alert radiation condition, and a horn for an audible alarm on either high or alert radiation condition. A channel failure alarm light, which is on during normal operation, turns off upon detection of channel failure. All alarms are annunciated in the main control room. Alarm set points are adjustable over the range of the detector.

3. Sensitivity - Area monitors are sensitive to gamma radiation of photon energies 100 keV and above, with the exception of the containment post-accident monitors which are sensitive to photon energies of 60 keV and above.

4. Environmental conditions - The area monitors are designed to operate in the normal environmental conditions for the areas in which they are located for the design life of the plant. The post-accident containment area monitors are designed to remain functional during a DBA.

12.3.4.1.2 Criteria for Location of Area Monitors

Generally, area radiation monitors are provided in areas to which personnel normally have access and for which there is a potential for personnel to receive high-radiation doses (e.g., in excess of 10CFR20 limits) in a short period of time. Plant areas which meet one or more of the following criteria are monitored:

1. Areas which during normal plant operations, including refueling, could exceed radiation limits due to system failure or personnel error.

2. Areas which are continuously occupied to perform plant shutdown following an accident.

3. Area monitors are located in accordance with the requirements in GDC 63 of 10CFR50, Appendix A.

•→1

4. Post-accident drywell and containment monitors are located in accordance with the requirements of Regulatory Guide 1.97 and with the requirements in Item II.F.1(3) of NUREG-0737.

1←•

•→1

5. Post-accident vital area monitors are located in accordance with Regulatory Guide 1.97.

1←•

12.3.4.1.3 System Description (Area Radiation Monitoring)

The area radiation monitoring system detects, measures, and records ambient gamma radiation levels at various locations. It also provides audible and visual alarms in monitored areas and annunciation in the main control room if gamma radiation exceeds a specified limit. It provides visual indication in the area monitored and at a main control room annunciator if there is a malfunction in any area monitor.

Each area channel consists of a detector assembly, a check source assembly, a data acquisition unit, indicators, and alarms. All monitors are independent, and failure of one monitor has no effect on any others.

The nonsafety area radiation monitors are powered from 120-V ac regulated buses. The redundant post-accident containment monitors and safety-related monitors are powered by 120-V ac divisional busses.

The location of each area radiation detector is indicated on the radiation zoning and access control drawings, Fig. 12.3-6 through 12.3-10, and is listed in Table 12.3-1. Consistent with the previous criteria, the following general areas are monitored:

1. Areas required for safe shutdown

2. Sampling rooms and chemistry laboratory

3. Containment and drywell

4. Fuel storage and handling areas

5. Waste storage and handling areas

6. Maintenance areas.

12.3.4.1.4 Safety Evaluation

The area radiation monitoring system is not essential for the safe shutdown of the plant, and serves only to warn plant personnel of high-radiation levels in various plant areas. Except for the two high-range containment monitors, the two high-range drywell monitors required for post-accident monitoring, and the two containment purge

isolation monitors, the system serves no active emergency function during operation.

The area radiation monitoring system is designed to operate unattended for extended periods of time. A visual display of ambient radiation dose rate and trend information for any detector is available on demand in the main control room. These monitors provide audible and visual alarms at the detector and annunciate in the main control room if the radiation levels exceed preset limits. A recorder located on the radiation monitor panel in the main control room provides a permanent record of the radiation levels for the four post-accident safety-related containment and drywell monitors and the containment purge isolation monitors. Also, analog indication is provided on the main benchboards in the main control room for quick operator assessment of post-accident monitoring (PAM) conditions.

•→8•→7

12.3.4.1.5 Calibration and Testing

Each of the monitors is calibrated by the instrument manufacturer prior to shipment using sources certified by the National Institute of Standards and Technology (NIST) or traceable to the NIST. In-plant calibration, using a standard radioactive point source traceable to NIST, is done at each refueling or whenever maintenance work is performed on the detectors or in accordance with the Technical Specifications/ Requirements.

8←• 7←•

The proper functioning of each monitor is verified periodically by checking the instrument response to the remotely operated radioactive check source provided with each detector. Proper operation of the monitor's electronics is verified periodically by the use of internal check circuitry provided in each unit.

12.3.4.2 Airborne Radioactivity Monitoring Objectives

Airborne radioactivity monitoring is provided in compliance with 10CFR20 and Regulatory Guide 1.45. The purpose of the airborne radioactivity monitoring system is to monitor the air within an enclosure by direct measurement of either the enclosure atmosphere or the exhaust air from the enclosure. The system indicates and records the levels of airborne radioactivity and, if abnormal levels occur, actuates alarms. Alarms are provided to alert personnel that airborne radioactivity is at or above the selected set point. The system provides a continuous record of airborne radioactivity levels, which aids operating personnel in maintaining airborne radioactivity at the lowest practicable level.

The main objectives of those in-plant airborne radioactivity monitoring systems which are required for safety are to initiate appropriate protective actions to limit the potential release of radioactive materials from the primary and secondary containment or the intake of radioactivity into the main control room if predetermined radiation levels are exceeded in major HVAC streams. Additional objectives are to have those systems available under all operating conditions including accidents, and to provide main control room personnel with an indication of the radiation levels in the major HVAC streams plus alarm annunciation if high-radiation levels are detected.

The safety-related radiation monitoring systems provided to meet these objectives are:

1. Fuel building ventilation exhaust

2. Main control room ventilation.

The main objective of the in-plant airborne radioactivity monitoring systems that are not required for safety is to monitor major building exhaust airborne radioactivity levels and to alarm in the main control room when predetermined levels are exceeded.

The nonsafety-related radiation monitoring systems (RMS) include:

1. Auxiliary building ventilation

2. Turbine building ventilation (including condensate demineralizer area)

3. Off-gas building ventilation

4. Radwaste building ventilation exhaust

5. Main plant exhaust duct monitors

6. Containment and drywell atmosphere monitors.

#### 12.3.4.2.1 Airborne Radioactivity Monitoring System Design Criteria

The criteria for determining the type of airborne radioactivity monitoring systems are based upon the nature and type of radioactive releases expected and the location being monitored.

The guidance of ANSI N13.1 and Regulatory Guide 1.21 is followed for the airborne radioactivity monitoring system design.

In the case of the drywell and containment radioactivity monitoring systems, which are used to detect leakage from the reactor coolant pressure boundary, the guidance of Regulatory Guide 1.45 is followed.

The design criteria for the safety-related in-plant airborne radioactivity monitoring systems are to:

1. Withstand the effect of natural phenomena (e.g., earthquakes) without the loss of capability to perform their functions

2. Perform their intended safety functions under normal and postulated accident conditions

3. Meet the reliability, testability, independence, and failure mode requirements of engineered safety features

4. Provide continuous display on main control room panels

5. Permit the checking of the operational availability of each channel during reactor operation with provision for calibration function and instrument checks

6. Assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The design criteria for the nonsafety-related in-plant airborne radioactivity monitoring systems are to:

1. Provide data output in the main control room on demand

2. Permit checking the operational availability of each channel during reactor operation with provision for calibration function and instrument checks

3. Perform their intended functions under normal operating conditions for the design life of the plant.

Additional design criteria for the main plant exhaust duct extended range monitor and the containment and drywell atmosphere monitors are detailed in the following sections.

12.3.4.2.2 Criteria for Airborne Radioactivity Monitor Locations

The criteria for locating airborne radioactivity monitors are dependent upon the point of leakage, the ability to identify the source of radioactivity so that corrective action may be performed, and the possibility for exposing personnel to airborne radioactivity.

1. Airborne radioactivity monitors sample drywell and containment atmospheres for reactor pressure boundary leak detection.

2. The outside air intake ducts for the main control room area are monitored to measure the possible introduction of radioactive materials into the main control room to ensure habitability of those areas requiring personnel occupancy for safe shutdown.

3. Exhaust ducts servicing an area containing processes which, in the event of a major leakage, could result in concentrations within the plant approaching the limits established by 10CFR20 for plant workers are monitored.

Airborne process and effluent radiation monitor locations and functions are summarized in Table 12.3-2. ANSI N13.1 was used as a guide in locating monitors and sample points. Monitor locations are shown on the shielding arrangement and facilities drawings, Fig. 12.3-6 through 12.3-10.

12.3.4.2.3 System Description (Airborne Radioactivity Monitors)

12.3.4.2.3.1 Drywell and Containment Atmosphere

The drywell and containment atmosphere radiation monitors are designed for early reactor coolant pressure boundary leak detection in accordance with Regulatory Guide 1.45.

A radiation monitor located in the containment is dedicated to sampling the drywell atmosphere. Samples are drawn from the drywell general area and pumped through the monitoring system and then returned to the drywell. Each sample is continuously monitored for particulate and gaseous activities. A removable iodine cartridge filter, which may

be used for laboratory analysis, is provided between the moving particulate filter and the gas sample chamber. Alarms are provided for high and alert radiation levels for each channel. Alarms are also provided for channel or sampling system component failure. All alarms are actuated locally at the monitor and in the main control room. Recorders are provided in the main control room to maintain a permanent record of drywell radiation levels.

Similarly, a radiation monitor is dedicated to sampling the containment atmosphere. This monitor is designed to obtain a representative sample of the radiation level in the containment. The monitor and alarms are identical to those described for the drywell.

The drywell and containment radioactivity monitoring systems are designed to function throughout a design basis seismic event and are powered by 120-V ac divisional buses.

12.3.4.2.3.2 Fuel Building Ventilation Exhaust

The fuel building ventilation exhaust radiation monitors are designed to measure radionuclide effluent emissions to be reported and evaluated in accordance with Regulatory Guide 1.21. The monitoring system is also designed to activate the safety-grade filter systems on detecting high-radiation levels in the effluent and to monitor post-accident effluent gas and particulate levels. During normal operating conditions, the effluent from the fuel building is not filtered and these monitors indicate the airborne radiation levels in the fuel building.

The fuel building ventilation exhaust is monitored by one offline gas monitor and one offline gas and particulate monitor as described in Section 11.5.2.1.2.1. Sampling is performed by an automatic isokinetic sampling system with probes and returns located in the fuel building ventilation exhaust duct. Connections are available for grab and tritium sampling. High and alert radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.

On a high radiation signal from these monitors, the fuel building exhaust air is diverted through the safety-grade filter trains.

The fuel building ventilation exhaust monitors are powered by 120-V ac divisional buses.

12.3.4.2.3.3 Main Plant Exhaust Duct

The main plant exhaust duct radiation monitors are designed to measure radionuclide effluent emissions to be reported and evaluated in accordance wtih Regulatory Guide 1.21. The monitoring system is also designed to monitor post-accident gaseous and particulate effluent levels. During normal operations, these monitors indicate the general unfiltered level of airborne radioactivity that is being released to the environment.

The main stack exhaust radiation monitors are similar to the fuel building ventilation exhaust monitors described in Section 12.3.4.2.3.2. Sampling is performed by an automatic isokinetic sampling system with probes located in the main plant exhaust stack above the auxiliary building. High and alert radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.

The main plant exhaust duct offline gas monitor is powered by the 120-Vac divisional buses. The main plant exhaust duct offline gas/particulate monitor is powered by the 120-Vac regulated bus.

12.3.4.2.3.4 Main Control Room Ventilation

The main control room ventilation radiation monitors are designed to measure the radiation levels in the two inlet ducts of the main control room ventilation system, and automatically divert the air through the emergency filter system on detection of high radiation at the local intake.

Redundant offline gas monitors are provided at both the local and the remote outside air intake ducts. Sampling is performed by sampling systems with probes and returns located near the intakes. The monitors for the local intake are located in the control building. The monitors for the remote intake are located in the standby service water pumphouse.

The main control room radiation monitors provide a single channel for gaseous activity only. Fixed-particulate and iodine filters are located upstream of the gas sample chamber and can be removed for analysis in the health physics laboratory. High and alert radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.

On a high radiation signal from the local outside air intake monitors, the air is diverted through safety-grade filter trains.

The control room ventilation radiation monitoring system is powered by 120-Vac divisional buses.

12.3.4.2.3.5 Nonsafety-Related Process and Effluent Airborne Radiation Monitoring Systems

The nonsafety-related airborne radiation monitoring systems are designed to comply with Regulatory Guide 1.21 for monitoring all effluent streams and major contributing process streams for a more accurate determination of the airborne radiation levels in the final effluent, and for identification of the sources of airborne radiation in the plant. The following nonsafety-related process ventilation systems exhaust through the main plant exhaust duct:

1. Auxiliary building ventilation

2. Turbine building ventilation

3. Condensate/demineralizer and off gas building ventilation.

•→10
The effluent from each of the process streams is monitored by a single gaseous and particulate radiation monitor. The monitors are similar to the drywell and containment radiation monitors described in Section 12.3.4.2.3.1. Sampling is performed by an isokinetic sampling system with probes and returns located in the ductwork. The effluent streams are not filtered except for the auxiliary building exhaust air which is automatically diverted through the SGTS upon a high radiation signal from the reactor building annulus ventilation monitors. High and alert radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.
10←•
Power for these monitors is supplied by 120-Vac buses.

The radwaste building ventilation exhaust effluent radiation monitors are designed to measure radionuclide releases to be reported and evaluated in accordance with Regulatory Guide 1.21. Since the effluent from general areas of the radwaste building is unfiltered, these monitors also indicate airborne levels of radiation in the building.

Sampling is performed by an automatic isokinetic sampling system with probes and returns located in the exhaust duct. The radwaste building ventilation monitors are similar to the fuel building ventilation exhaust monitors described in Section 12.3.4.2.3.2. High and alert radiation levels, channel failure, and sampling system failures are alarmed in the auxiliary and main control rooms.

The radwaste building ventilation exhaust monitors are powered by 120-V ac regulated buses.

12.3.4.2.4 Safety Evaluation

The in-plant HVAC airborne radioactivity monitors have the safety-related functions of isolating their particular ventilation systems and actuating the associated filtered emergency systems, as discussed in Sections 12.3.4.2.1, 12.3.4.2.2, 12.3.4.2.3.2, and 12.3.4.2.3.4. These monitors are redundant, Seismic Category I, and powered from the emergency power system.

The combination of the airborne radioactivity monitoring system in conjunction with the administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns through the plant, and plant equipment layout is sufficient to guarantee the safety of personnel throughout all areas of the plant where access is required.

12.3.4.2.5 Airborne Particulate Detector Ranges and Minimum Detectable Concentrations

Monitor particulate channel ranges are tabulated in Table 12.3-2, and corresponding minimum detectable concentrations (MDCs) are tabulated in Table 12.3-3. Calculated MDC values for monitor particulate channels in the turbine building, fuel building, auxiliary building, radwaste building, and offgas building are sufficiently low that 10 MPC-hr* of airborne iodine and particulate radionuclides in any normally occupied compartment in these five buildings can be detected.

The particulate radionuclide concentrations to be detected by the particulate channels are developed by normalizing the building airborne radionuclide inventories (derived from the building effluent inventories tabulated in Table 11.3-1) to

*Maximum Permissible Concentration, as defined in 10CFR20.

1 MPC and multiplying the resultant radioiodine concentrations by the fraction which is presumed (based upon data in NUREG-0016) to be in particulate form.

A recorder located on the radiation monitor panel in the main control room provides a permanent record of the radiation levels for each safety-related channel. Also, analog indication is provided in the main control room for quick operator assessment of the following:

1. Fuel building ventilation exhaust

2. Main plant exhaust

3. Control room ventilation intake

4. Radwaste building ventilation exhaust.

•→7
The continuous monitoring system consists of radiation monitors calibrated by the manufacturer. Calibration standards are traceable to the NIST. Secondary standards counted in reproducible geometry during the primary calibration are supplied with each continuous monitor. Each continuous monitor is calibrated annually using the secondary radionuclide standards.
7←•
The count rate response of each continuous monitor to remotely positionable check sources supplied with each monitor is recorded by the manufacturer after the primary calibration, again after installation, and together with the instrument background count rate at intervals during reactor operation to ensure proper functioning of the monitors.

Decay corrections are provided for the sources to permit correction for source decay. An electronic circuit check is performed through the use of an internal oscillator or pulse-generating circuit.

12.3.4.3 Accident Consideration

The plant is designed so that only certain vital areas need be occupied during the course of an accident. Should personnel access to specific areas be necessary, portable monitoring equipment described in Section 12.5 is used.

Information on post-accident radiation levels is available from safety-related monitors specifically designed for this purpose. In addition, information may be available from other safety-related and nonsafety-related area and airborne radiation monitors, and no credit is taken for these.

The nonsafety area monitors and nonsafety-related airborne monitors are supplied by 120-V ac buses. The safety-related airborne radioactivity monitors and area monitors are powered by 120-V ac divisional buses.

The ranges of the nonsafety-related monitors include the maximum anticipated operational radiation levels. Radiation or radioactivity levels in excess of the maximum range causes the monitors to continue to read upscale through an anti-saturation circuit. High or upscale radiation or radioactivity is indicated by a local audible and visible alarm, alerting personnel not to enter the affected area until a survey with portable equipment establishes that personnel can not be subjected to doses in excess of established limits. Safety-related post-accident monitors have extended ranges to cover postulated accident radiation levels.

12.3.4.4 Portable Monitors

Portable, moving, and fixed-filter paper continuous air monitors (PCAMs) are provided and are normally operated at fixed locations in the plant to provide trend indication. However, during a period of extensive maintenance in an area with a potential for airborne radioactivity, a PCAM is moved to the maintenance area. The PCAMs have the capability to be connected to the digital radiation monitoring system via plug-in junction boxes. Junction boxes are located throughout the plant, as shown in Table 12.3-4, and provide the capability to remotely monitor airborne activity in the affected area from the control room, auxiliary control room, and Rad Tech clean work area in the services building.

Intermittent sampling using portable air samplers is used to verify and/or supplement continuous air-monitoring equipment. These portable air samplers are also used in areas with a potential for some airborne radioactivity, but not enough to justify installation of a PCAM.

Portable instrumentation is described in Section 12.5.

Reference - 12.3

1. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plant RP-8A. Topical Report, Stone & Webster Engineering Corporation, May 1975.

## 12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily from components and equipment containing radioactive fluids, and to a lesser extent from the presence of airborne radionuclides. In-plant radiation exposures during normal operation and anticipated operational occurrences are discussed in Section 12.4.1. Radiation exposures at other onsite locations outside the plant which arise from onsite radioactive sources, the presence of N-16 in the plant, and radioactive gaseous effluents are discussed in Section 12.4.2.

Dose assessment is the estimation of occupational radiation exposure at River Bend Station to verify that the plant design features and proposed methods of operation ensure that radiation exposures are as low as reasonably achievable (ALARA). The dose assessment involves estimates of occupancy times, dose rates in plant areas, and the number of personnel involved in the following general categories: reactor operation and surveillance; routine maintenance; waste processing; refueling; in-service inspection; and special maintenance.

Radiation exposures to operating personnel are within 10CFR20 limits. Radiation protection design features described in Section 12.3 and health physics program outlined in Section 12.5 assure that the occupational radiation exposures (ORE) to operating and construction personnel during operation and anticipated operational occurrences are as low as is reasonably achievable (ALARA).

The dose assessment evaluation process is aimed at eliminating unnecessary exposures and to consider cost-effective dose reducing methods to minimize the necessary operational exposures.

### 12.4.1 Exposures Within the Plant

The occupational radiation dose assessment for River Bend Station was performed using the guidelines of Regulatory Guide 8.19[3]. The basis for the annual man-rem estimates was operating data from similar BWR plants taking into account the design improvements that impact the occupational radiation exposure at RBS. The projected radiation dose rates throughout the plant facilities are based on assumed radiation conditions after 5 yrs of plant operation and the design radiation dose rates from USAR Section 12.3. Operational data from several BWRs[4] showing the average annual man-rem per unit over several operating years to be

948 man-rem per year is presented in Table 12.4-1. These data indicate that in recent years occupational radiation exposures have been much larger than the radiation exposures reported for operating BWR plants in the mid-1970s. The primary reason for the increase in radiation exposure has been the increase in manpower necessary to support the expanding special maintenance activities.

Distribution of annual occupational radiation exposures suggested in Regulatory Guide 8.19[3] work functions is given in Table 12.4-3 for all BWRs over several years. The average values indicate that operating BWR plants have approximately 76 percent annual occupational exposure attributed to maintenance, 40 percent of which is routine and 36 percent of which is special. In recent years, plant modifications attributed to feedwater sparger repairs, inspection, repair and replacement of recirculation piping, Three Mile Island lesson-learned modifications, and increased snubber and pipe hanger inspections have contributed to the growing amount of occupational radiation exposures associated with special maintenance work functions. Design features described in Sections 12.1 and 12.3 for the RBS BWR/6 plant should minimize the special maintenance work experienced at earlier-designed operating BWR plants.

Design improvements for RBS that are expected to reduce the occupational radiation exposures include the following:

1. Incorporation of flush connections on the CRD scram discharge volume header permits condensate flushing of piping to minimize corrosion product holdup in a high personnel access area.

2. Use of filtered condensate water for CRD hydraulic fluid and for the reactor recirculation pump seal purge provides a clean water source that should extend pump seal life.

3. Installation of permanent hoisting system and access platforms for the recirculation pumps, main steam isolation valves and safety-relief valves minimizes maintenance time in drywell.

4. Improved refueling platform makes fuel handling activities move efficiently so less time is spent on the platform.

5. A multistud tensioner reduces the amount of man-hours necessary to handle the reactor vessel head studs.

6. A new handling tool and platform for the removal of CRDs from beneath the reactor vessel reduces crew size and time spent in the high radiation area.

7. Improved fuel design minimizes the buildup of radiation levels near reactor coolant systems and reduces the amount of fuel assembly shipping activities.

8. Improved piping material for the recirculation system eliminates the special maintenance which was required on older BWR recirculation piping due to stress corrosion cracking.

9. Inservice inspection access is improved by remote equipment development and by access doors and plugs in the biological shield for reactor vessel weld inspection.

10. Installation of a positive pressure leakage control system for the main steam line isolation valves should reduce the surveillance and maintenance activities related to technical specification requirements for leakage.

11. Use of separate shielded cubicles for locating redundant components and highly radioactive components minimizes radiation exposures during maintenance activities.

12. Use of mechanical snubbers should reduce the frequency of inspection necessary in comparison to the hydraulic-operated snubbers.

13. Installation of a CRD flush tank and filter system removes highly radioactive corrosion and fission products from the CRD internals prior to rebuilding.

The occupational radiation exposure for RBS can be determined for each of the Regulatory Guide 8.19[3] work function categories by estimating the occupancy time and the manpower requirements involved for each of the six radiation zones defined in Section 12.3. Table 12.4-4 shows the expected occupancy times and anticipated manpower needs for reactor operations and surveillance, routine maintenance,

waste processing, refueling, and inservice inspection. The annual man-rem for each work function can be estimated by assuming an average dose rate for each radiation zone and assuming that there are 2,080 working hours per year for each person for the operations and surveillance, routine maintenance, and waste processing work activities. Refueling activities are assumed to be based on 160 hr per year for each person and inservice inspection is based on 192 hr per year for each person. These man-hour estimates are consistent with those reported for operating BWR plants and predicted based on GE experience[5]. The average dose rates for each radiation zone are used in Table 12.4-4 as more realistic estimates of the actual radiation environment that workers would experience rather than the maximum dose rates which were determined for shielding design criteria based on very conservative source term assumptions. The estimated occupational radiation doses, summarized for each category in Table 12.4-5, give an overall estimate for RBS as 827.5 man-rem per year. The special maintenance contribution was estimated to be 298.7 man-rem based on previous operational BWR experience (Table 12.4-3). Special maintenance comprises approximately 36 percent of the total annual man-rem. It is anticipated that special maintenance work activities will be significantly reduced for RBS based on the BWR/6 design improvements and the RBS design features. Table 12.4-5 also provides the percentage distribution of the total annual occupational dose for each work function. This is consistent with the average distributions as calculated from operating BWR data shown in Table 12.4-3.

A further estimate of the occupational exposures was made by identifying specific tasks within each of the six work function categories. Various data from operating plants and current publications (References 2 through 10) were used to identify tasks and estimate manpower needs and expected radiation levels. Tables 12.4-6 through 12.4-11 provide the estimates of occupational exposures based on the identification of specific tasks within each of the six work function categories: routine operations and surveillance; routine maintenance; waste processing; refueling; inservice inspection; and special maintenance. Table 12.4-12 summarizes the occupational dose estimates for the six work functions. A comparison between Tables 12.4-12 and 12.4-5 shows that the dose estimates based on the identification of specific tasks are consistent but tend to be less than the dose estimates based on occupancy times for each radiation zone for the general work function classification. The only inconsistent estimate is the special maintenance work function. It is difficult to predict all the possible

special maintenance work at RBS since there is little operating experience associated with BWR/6 plants. Operating experience at older BWRs (Tables 12.4-1, 12.4-3 and 12.4-5) has shown that the special maintenance dose is approximately 36 percent of the total annual average man-rem (948 man-rem/year) which would predict a value of 342 man-rem for special maintenace work only. Design improvements and features previously discussed should reduce the special maintenance man-rem estimate for RBS to a range of 80 to 290 man-rem per year.

RBS also has evaluated personnel exposure resulting from the actuation of SRVs based on Reference 2. The SRV discharge event considered in the analysis is the Type 2 isolation event where the reactor pressure is initially controlled by the cyclic lifting of the SRVs. All SRVs are assumed to open with the low set relief valves remaining open following the closure of the other valves. Design basis radiation sources for normal operation are used in the analysis. Normal ventilation in containment is assumed and airborne concentrations are not corrected for plateout on the walls. This evaluation determines the dose to an operator located in the TIP drive area who leaves this area following the isolation event and exits the containment at the personnel hatch at elevation 114 ft. It is assumed that the operator takes 4 min to exit the containment. A nonhomogeneous distribution of design basis airborne sources within the first 4 min following the event is assumed that is consistent with the reference study. The whole body and lens of eye dose calculated for this event is 140 mrem. The beta skin and thyroid doses are 390 mrem and 0.80 mrem, respectively.

### 12.4.2 Exposures at Locations Outside the Plant Structures

Radiation exposures at locations outside the plant arise from: 1) onsite radioactive sources outside plant buildings, 2) direct and air-scatter (skyshine) contributions due to the presence of N-16 in the plant buildings, and 3) release of gaseous effluents from the plant. The dose due to N-16 is the predominant contributor. Estimated doses for the restricted area boundary are summarized in Table 12.4-2. These estimates meet the dose guidelines of 10CFR20 and 40CFR190.

#### 12.4.2.1 Dose due to Radiation Sources Outside Plant Buildings

The only onsite sources which exist outside of plant buildings with potential for a direct radiation dose

•→8A •→8
contribution to persons outside of plant structures are the condensate storage tank, the temporary dry active waste storage facilities, Radioactive Material Storage Areas and the two turbine rotor modular enclosures. However, the minimal activity within the tank, temporary dry active waste storage facilities, and the two turbine rotor modular enclosures produces a negligible dose rate at the RAB.
8←• 8A←•

12.4.2.2 N-16 Dose Contributions

•→14
Radiation dose rates as measured near plants such as River Bend consist of three parts. The parts are direct radiation, radiation from scattered gammas, and background radiation. Background radiation is a combination of radiation from terrestrial sources and cosmic radiation. Scattering sources comes from a direction other than the original source. Scattering can be accomplished by buildings around and near the source, and by the atmosphere (skyshine).

The Monte Carlo technique was used to evaluate the direct and scatter doses for River Bend Station including the increase N-16 activity as a result of Hydrogen Water Chemistry. The modeling case chosen for River Bend was that for a plant with moisture separators/reheaters (MSRs) on the operating floor, with 3 feet thick perimeter walls, 1.5 feet thick MSR covers, and 1 foot thick inner walls. All walls were assumed to be concrete by the model. While this model is not identical to River Bend design, it is a reasonable match since River Bend's steel inner walls (about six inches thick) have the same stopping power as one foot of concrete. The correlation developed was validated using observed data.

The results indicate that the dose rates drop below 1mR/yr at 1900 feet. The minimum distance from the turbine building to the RAB is 2,760 feet.
14←•

#### 12.4.2.3 Exposures Due to Airborne Activity

Dose rates resulting from airborne activity at the Restricted Area Boundary, based on 2,000 hr/yr occupancy, are listed in Table 12.4-2.

### 12.4.3 Exposure to Construction Workers

RBS is a single-unit plant; therefore, estimated annual doses to workers constructing an additional plant due to radiation from an existing operating plant are not applicable.

References - 12.4

1. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants, RP-8A. Stone & Webster Engineering Corporation, Boston, MA, May 1975.

2. Mark III Containment Dose Reduction Study, GE22A5718, Revision 1, January 29, 1980.

3. Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates, Regulatory Guide 8.19, Revision 1, June 1979.

4. Occupational Radiation Exposure at Commerical Nuclear Power Reactor 1982, NUREG-0713, Volume 4, December 1983.

5. Occupational Radiation Exposure, General Electric Company, NEDO-24606, January 1979.

6. An Assessment of Engineering Techniques for Reducing Occupational Radiation Exposure at Operating Nuclear Power Plants, Atomic Industrial Forum, February 1980.

7. Pelletier, C. A., et al. Compilation and Analysis of Data on Occupational Radiation Exposure Experiment at Operating Nuclear Power Plants. AIF/NESP-005, September 1979.

8. Hall, T. M. Determining Effectiveness of ALARA Design and Operational Features, NUREG/CR-0446, April 1979.

9. Occupational Radiation Exposure Reduction Technology Planning Study, Stone & Webster Corporation, EPRI NP-1842, May 1981.

10. Pettit, P. J. Compendium of Design Features to Reduce Occupational Radiation Exposure at Nuclear Power Plants, AIF/NESP-20, April 1981.

## 12.5 HEALTH PHYSICS (RADIATION PROTECTION) PROGRAM

### 12.5.1 Organization

#### 12.5.1.1 Program and Staff Organization

The objective of the radiation protection program is to control the occupational radiation exposure of personnel within the plant and reduce exposure to the general public and offsite environs from ionizing radiation resulting from operation of the plant. Specifically, the radiation protection program consists of policies, procedures, instructions, rules, and practices to keep individual radiation exposure within the limits set forth in 10CFR20 and beyond that to maintain total radiation exposure of personnel as low as is reasonably achievable (ALARA) (Section 12.1). The program is designed to assure that the plant operations meet the radiation protection and training requirements of 10CFR19, 10CFR20, 10CFR50 Appendix I, and the River Bend Station positions on NRC Regulatory Guides 8.2, 8.8, and 8.10, which are provided in Section 1.8. The program assures that radiation protection training is provided, that personnel and in-plant area radiation monitoring is performed, and that records of training, exposure of personnel, and surveys are maintained. The program provides guidelines covering the handling and monitoring of radioactive materials.

•→13 •→7 •→2

The RBS Plant Staff organization, which includes the radiation protection organization, is shown on Fig. 13.1-2. The General Manager is responsible for the overall effectiveness of the radiation protection program. The General Manager delegates the conduct of the radiation protection program to the Manager - Radiation Protection. Manager - Radiation Protection is equivalent to the Radiation Protection Manager referred to in Regulatory Guide 1.8 (Revision 1-R) and is responsible for the enforcement of the program. It is the responsibility of all plant supervisory personnel to ensure that plant personnel are made aware of the management commitment to keep all occupational radiation exposure ALARA.

The Manager - Radiation Protection is responsible for establishing the radiation protection program and provides technical assistance for conducting the program. He assures that personnel radiation exposure is maintained ALARA.

The Manager - Radiation Protection is required to have a degree in science or engineering or the equivalent with some formal training in health physics and at least 5 yr. responsible professional experience in health physics. A minimum of 3 of the 5 yr. of experience in applied radiation protection is required at a light-water reactor facility.

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Radiation protection supervisors and management personnel report to the Manager - Radiation Protection and ensure that the established radiation protection program is properly implemented.

A radiation protection supervisor implements the operational aspects of the radiation protection and ALARA Programs. This is accomplished through the use of approved procedures, trained and qualified radiation protection personnel, properly calibrated equipment, and appropriate ALARA concepts. A radiation protection supervisor is designated backup for the Manager - Radiation Protection and meets the requirements of Regulatory Guide 1.8 (Revision 1-R).

A radiation protection supervisor implements the Dosimetry and Radiological Health Programs. This is accomplished through the use of approved procedures, an accredited personnel dosimetry program, maintenance of exposure histories, maintenance of properly calibrated equipment, maintenance of whole body, Dosimetry and bioassay counting capabilities, and providing respiratory mask fit tests.
2←• •→8
The organization also includes personnel to provide for the radwaste Process Control Program processing solid radwaste, trending, trend analysis, shield calculations, dose assessment, emergency planning, internal audits of radiological work practices, and computer applications.
8←•
ALARA concepts are implemented by the ALARA coordinator through procedure reviews, Radiation Work Permits, design reviews, system walkdowns, and preplanning.

Corporate support is provided for external audits and assessments.
7←•

12.5.1.2 Radiation Protection Program Objectives

The objectives of the radiation protection program are:

1. To provide administrative control of plant personnel to ensure that radiation exposure is maintained within the limits of 10CFR20 and that such exposure is maintained ALARA.

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2. To provide administrative control over plant effluent releases and to limit these releases below 10CFR20 values and the values given in the plant procedures.

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3. To establish a radiation protection records systemwhich meets the requirements of applicable federaland state regulations.

12.5.1.3 Radiation Protection Program

The radiation protection program becomes effective in phases upon implementation of the preoperational environmental radiological program, 2 years prior to scheduled operation start, and continues throughout operation of the station. The program includes policies, procedures, instructions, rules, and practices to ensure that the program objectives are fulfilled in a safe and reasonable manner. These program facets are reviewed periodically, and recognized improvements are incorporated. The program includes:

1. Necessary measures and guidelines to reduce exposureof personnel and to limit effluent releases.

2. Appropriate radiation protection training of personnel commensurate with their respective responsibilities.

3. Provisions for additional Emergency Plan training of personnel who may be assigned to radiation emergency teams.

4. Respiratory protection training and supplies of equipment where needed to reduce internal exposure to personnel.

5. Appropriate dosimetry of personnel for external dose assessment.

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6. An internal dose assessment program which includes passive monitoring and bioassays.

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7. Requirements for segregating and appropriately posting radiologically controlled areas to control exposure potential.

8. Procedures for access to radiologically controlled areas including radiation work permit authorization.

9. Training in duties to be performed in radiation areas.

10. Radiological instrumentation and equipment to assess exposure accurately.

11. Procedures for properly handling incoming and outgoing shipments of radioactive materials in accordance with NRC, DOT, and other applicable requirements.

12. Proper protective clothing and equipment to limit the potential hazard of transferable contamination.

13. A radiological environmental monitoring program for evaluating the environmental impact of plant operation.

•→12

14. The radiation protection program coordinates with the emergency operating procedures and servere accident procedures for staff and facility activation during an emergency. Emergency planning is discussed in Section 13.3.

12←•

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 Radiation Protection Facilities

An assessment of compliance of radiation protection facilities design with Regulatory Guides 1.97, 8.3, 8.4, 8.8, 8.9, 8.12, 8.14, and 8.15 is provided in Section 1.8.

The radiation protection facilities are located in the services building and contain:

1. Radiation protection offices and work areas

2. Chemistry and radiochemistry laboratories

3. Counting rooms for radioactivity analysis

4. Equipment and supplies storage spaces

5. A first aid facility

6. Personnel and small item decontamination facilities

7. Instrument calibration, maintenance, and storage facilities

8 Locker rooms and toilet facilities

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Radiation survey instruments, air monitoring equipment, dosimeters, counting instruments, and other miscellaneous radiation protection supplies are stored in the services building. The radiation protection offices and work areas are equipped with filing cabinets, desks, and tables to

provide work space for radiation protection personnel and storage space for records.

An adequate supply of clean protective clothing and respiratory equipment is supplied by onsite laundry facilities or vendor supply services. Protective clothing and respiratory protection equipment are available in the services building and at local contamination area access points set up at job sites. Change facilities are provided for these areas consisting of benches and clothes racks.

Radiation protection sample counting equipment is located in the radiation protection workrooms. Equipment used for routine counting of smears/swipes and air samples such as pancake probes or end window G-M counters, alpha and beta scintillation detectors, and/or gas flow proportional counters are located in this area. Samples requiring gamma isotopic analysis can be counted using spectrometric systems located in the chemistry laboratories.

A decontamination facility for personnel is provided in the services building. Decontamination of personnel is conducted under the direction of radiation protection personnel. The decontamination facility consists of a change area, monitoring area, lavatory, and showers. Decontamination of accident victims on stretchers can also be performed.

All personnel exiting contaminated areas are required to monitor/frisk themselves for contamination. In addition to local frisking, a portal monitor or equivalent is also provided at the exit of the radiological controlled areas and the primary access point building to alarm in case of contamination on person or clothing.

### 12.5.2.2 Radiation Protection Instrumentation

#### 12.5.2.2.1 Laboratory Instrumentation

Laboratory instrumentation located in the radiation protection and chemistry work areas allows plant personnel to ascertain the radioactive material present in survey samples. Typical samples would be contamination survey smears, airborne survey filters, and charcoal cartridges. There is also capability to conduct tritium surveys and process various other types of samples. Counting room instrumentation at least equivalent to that listed in Table 12.5-1 is maintained. The criteria for selection of these various counters is to obtain instrumentation that can reliably and efficiently count samples, provide the necessary low backgrounds and sensitivities, and analyze the counting data to provide information in a more easily usable

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form. Each counting system is checked and calibrated at regular intervals when in use and after undergoing repair with standard radioactive sources traceable to the National Institute of Standards and Technology (NIST). They are also calibrated prior to use if the period since the last calibration is greater than the regular interval. Counting efficiency, background count rates, and high voltage settings are checked.
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12.5.2.2.2 Portable Survey Instrumentation

Portable survey instrumentation is located in the radiation protection work area and at inplant control points. Portable equipment allows plant personnel to perform surveys for direct radiation, airborne radioactivity, and surface contamination.

The criteria for selection of these instruments is to obtain accurate and reliable instrumentation that is easily serviced and covers the entire spectrum of radiation measurements expected at the station during normal or accident conditions.
•→8 •→7
Each portable survey instrument is calibrated annually, when in use, and after undergoing repair. They are also calibrated prior to use if the period since the last calibration is greater than the regular interval. Calibrations are performed using calibration sources traceable to NIST, and instruments are source checked to verify proper operation in accordance with plant procedures. The quantities of each type of instrument permit calibration, maintenance, and repair without causing a shortage in operational instrumentation. Portable radiological survey instrumentation is listed in Table 12.5-2.
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12.5.2.2.3 Monitoring Instruments for Personnel

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Monitoring of personnel is provided by the use of a DLR {dosimeter of legal record, which may be an optically stimulated luminescent dosimeter (OSLD) or a thermoluminescent dosimeter (TLD)}, or direct-reading dosimeters, or albedo neutron TLDs or the equivalent, and survey instrumentation. Personnel who are required to be monitored for radiation exposure in accordance with 10CFR20.1502 are issued a DLR. DLRs are analyzed periodically and when special circumstances warrant. DLR readings are normally used as the official record of personnel exposure if required by 10CFR20.1502. Calibrations and quality control performed on the TLDs and the TLD system are in accordance with NRC requirements, by means of a NAVLAP certified facility.
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Direct-reading dosimeters are worn by personnel in the controlled area as specified by plant instructions. These dosimeters provide a day-to-day or job-to-job estimate of personnel exposure. Direct-reading dosimeters are provided as shown in Table 12.5-3.
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If neutron exposure is expected, personnel are issued neutron-sensitive DLRs. These DLRs normally provide the official record of exposure of personnel to neutron radiation.

In the case of a lost or improper reading on a DLR, or entry into the Controlled Area without the properly issued dosimetry, the individual exposure is estimated using appropriate methods and documented in accordance with plant procedures.
9←•
Extremity dosimetry is provided if extremities are expected to receive exposure significantly higher than that to the body. Its use is normally specified on the RWP. The direct-reading dosimeters are calibrated annually, and when damage is suspected, or after repair is performed.
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Survey instrumentation for the measurement of personnel contamination consists of G-M and/or gas flow proportional count rate meters (contamination friskers), Personnel Contamination Monitors (PCM's) and portal monitors. These instruments are calibrated annually or prior to use after undergoing repair. The monitoring instrumentation for personnel listed in Table 12.5-3 is maintained as a minimum.
7←•

12.5.2.2.4 Radiation Protection Equipment

Portable air samplers are used to collect samples for laboratory determination of airborne radioactive material concentrations. Air samplers are calibrated for flow semiannually. Samples may be counted for radioactive particulate, radioiodine, and airborne gaseous activities. Portable continuous air monitors are used to monitor airborne concentrations at specific work or field locations. Local indication is provided as well as trend information. Visual and audible alarms are provided with variable set points.

The portable air samplers shown in Table 12.5-4 may be used to monitor iodine levels in buildings under accident conditions as required in Item III.D.3.3 of NUREG-0737. Sample analysis is performed in the Services Building laboratory facilities (hot lab and counting rooms) on instruments described in Section 12.5.2.2.1.

Respiratory protection equipment is available at specific issue points. Self-contained breathing apparatus for

emergency use is available in the services building and at emergency equipment storage lockers.

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An instrument calibrator is used for calibrating gamma dose rate instrumentation. This is a self-contained, heavily shielded, multiple source calibrator. Neutron, beta, and alpha radiation sources are also available for instrument calibration and/or response checks. Sources are traceable to an NIST source.

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Protective clothing is supplied for personnel working in radiologically controlled areas. The clothing required for a particular instance is prescribed by radiation protection personnel on a radiation work permit based on actual or potential radiological conditions.

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An adequate inventory of protective clothing is maintained on hand to support plant operations and maintenance activities. This clothing typically includes lab coats, coveralls, hoods, caps, plastic suits, gloves (plastic, rubber, cloth), shoe covers, boots, and rubbers as necessary. Tape is provided for sealing.

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Additional contamination control supplies are available. These include vacuum cleaners equipped with HEPA filters, mops, absorbent paper, plastic sheets and bags, barricade ropes, signs, and labels.

Radiation protection equipment includes that given in Table 12.5-4 as a minimum.

12.5.2.2.5 Other Radiation Protection Instrumentation

The area radiation monitoring system is installed in areas where it is desirable to have constant dose rate information. Monitors indicate dose rate locally and/or in the control room and provide local audible and visual alarms upon reaching a preset dose rate. Fixed continuous airborne radioactivity monitors are also provided at strategic locations where the potential for exposure of personnel to airborne contamination is most likely.

12.5.3 Procedures

Section 13.5 provides information on the procedures which cover applicable areas of plant operations and maintenance. In addition, this section describes certain methods embodied in procedures and/or instructions to ensure that personnel radiation exposure is maintained ALARA. Assessments of compliance with Regulatory Guides 1.8, 1.16, 1.33, 1.39, 8.2, 8.7, 8.8, 8.9, 8.10, and 8.13 are provided in Section 1.8, and compliance with 10CFR19 is addressed in

Section 12.5.1.1. The preceding are incorporated into procedures and/or instructions as appropriate.

Strict adherence to the plant radiation protection procedures and instructions ensures that personnel radiation exposures are within the limits of 10CFR20 and ALARA concepts.

12.5.3.1 Radiation and Contamination Surveys

Radiation protection personnel conduct routine surveys throughout the plant for radiation, contamination, and airborne radioactive materials. The techniques used to perform these surveys are specified in radiation protection procedures. The frequency of routine surveys is specified in radiation protection procedures and depends on the location, type of area and how it is used, frequency of use, potential for radiological hazard, and likelihood of changes in radiological conditions. Records of surveys are retained for reference, and appropriate data are posted for information of plant personnel. Availability of current survey information aids in keeping exposure of personnel ALARA. Radiation surveys may be performed for alpha, gamma, beta, and/or neutron exposure rates. Contamination surveys are performed to establish gross beta-gamma contamination levels and may be processed for specific types of radiation (alpha-beta-gamma) or specific radionuclide (via gamma spectroscopy). Air samples are taken to establish airborne concentrations of particulates, noble gases, and/or radioiodine, and specific radionuclide information may be obtained. Special surveys related to specific operations or maintenance activities may be performed prior to, during, or after such activities, based on information required to keep radiation exposure ALARA.

12.5.3.2 Procedures and Methods to Maintain Exposures ALARA

ALARA considerations are incorporated into various types of plant procedures and instructions which cover activities involving exposure. Examples of procedures and methods used to maintain radiation exposure ALARA in the various operational categories are discussed in this section.

12.5.3.2.1 Refueling

Procedures and methods used to maintain radiation exposure ALARA during refueling outages include the following:

1. An RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures and equipment to be used.

3. Before removing the vessel head, the primary system is degassed and sampled to minimize expected airborne activity when the head is removed.

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4. During movement of irradiated fuel assemblies, the active fuel is maintained under a minimum height of water as required by plant procedures.

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5. The refueling cavity water is normally filtered to reduce the activity in the water and to lower exposure rates.

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6. Radiation levels in work areas are monitored and precautions taken as necessary, consistent with ALARA.

7. Filtered or exhaust ventilation is operated as appropriate to minimize airborne radioactive material.

Certain components and surfaces are maintained wet to minimize personnel exposure from potentially high airborne radioactivity concentrations during refueling operation, including the short time the steam dryer and part of the steam separator are out of water. In addition, administrative controls are implemented to minimize personnel exposure using respiratory protection equipment when necessary and containment access control during the transfer operation.

12.5.3.2.2 Inservice Inspection

Protection procedures and methods used during inservice inspections to maintain radiation exposure ALARA are as follows:

1. An RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures, equipment, radiation and contaminationlevels, and protective clothing requirements appropriate to a particular job.

3. Insulation is designed, where practical, for ease of removal and replacement where removal is required for repetitive inspections.

4. Equipment is calibrated and checked for proper operation prior to entry into a radiation area.

5. Temporary shielding is used, where practicable, to reduce radiation exposure.

6. Filtered or exhaust ventilation is provided, as necessary, to minimize airborne radioactive material.

12.5.3.2.3 Radwaste Handling

Procedures and methods used to maintain radiation exposure ALARA during radwaste handling include the following:

1. Handling of radwaste by personnel is minimized by plant design and local procedures.

2. The volume of radwaste which requires handling has been reduced by plant design and operating practices.

3. Special tools are used when practical to minimize time spent close to the radioactive source.

4. An overhead crane is provided to move drums or liners as required.

5. Adequate storage has been provided to minimize multiple handling of drums.

6. The labeling of drums or liners is completed prior to filling.

7. Filtered or exhaust ventilation is provided to minimize the airborne radioactive material from waste handling operations.

12.5.3.2.4 Spent Fuel Handling, Loading, and Shipping

Procedures and methods used during spent fuel handling, loading, and shipping to maintain radiation exposure ALARA include the following:

1. An RWP is used to provide positive radiological controls over work in progress.

2. Training is conducted to familiarize workers with procedures and equipment required to complete assignments.

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3. Movement of irradiated fuel assemblies and loading into shipping casks is accomplished with the active fuel maintained under a minimum height of water as required by plant procedures.

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4. Fuel handling cranes and extension tools are used to handle shipping casks, fuel assemblies, and inserts.

5. The spent fuel pool water is filtered to reduce the radiation exposure of personnel in the area.

6. The spent fuel pool water is cooled, and surface air ventilation is provided to minimize the airborne radioactive material.

7. After a shipping cask is loaded with spent fuel, it is decontaminated using a pressurized water washing device prior to shipment.

12.5.3.2.5 Normal Operation

Procedures and methods used during normal operation to maintain radiation exposure ALARA include the following:

1. By station design and construction, significant radiation sources are minimized and shielded.

2. The RWP system is used to provide positive radiological control over work in progress.

3. Training and retraining in radiation protection is provided to persons assigned tasks in controlled areas.

4. Controlled areas are conspicuously posted and maintained in accordance with 10CFR20.

5. Special access control procedures are used for entry to areas where significant exposure might be received.

6. Protective clothing and equipment are provided.

7. An area radiation monitoring system is installed and provides indication of radiation levels with local alarms, where appropriate.

8. The ventilation system is designed to minimize the spread of airborne contamination.

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9. Shielding effectiveness is verified by an initial startup survey for gamma and neutron radiation.

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12.5.3.2.6 Routine Maintenance

Examples of procedures and methods used during routine maintenance to maintain personnel radiation exposure ALARA are as follows:

1. Maintenance work involving systems that collect, store, contain, or transport radioactive materials must be covered by an approved RWP.

2. Training is provided, as required, to accomplish assigned tasks in controlled areas.

3. Routine maintenance procedures incorporate appropriate precautions and radiological considerations.

4. Equipment is moved to areas with lower radiation and contamination levels for maintenance when practical.

5. Special tools are used when practical.

6. Portable shielding is used as practical.

7. Periodic monitoring by radiation protection technicians, as radiological conditions warrant, is provided.

8. Post-job debriefings are utilized for high exposure jobs to obtain input from personnel actually performing the work. This is utilized in revising procedures and instructions as appropriate for ALARA considerations.

12.5.3.2.7 Sampling

Procedures and methods of maintaining personnel radiation exposure ALARA during sampling include the following:

1. Sampling of radioactive systems is normally performedinside sample hoods.

2. Procedures specify the protective clothing and sampling techniques to be used.

3. Monitoring of radiation levels in the work area during sampling and of the sample container when sample is collected.

4. Training in proper handling of sample containers after samples are collected.

5. Training in proper storage and disposal of radioactive samples.

12.5.3.2.8 Calibration

Procedures and methods of maintaining radiation exposure of personnel ALARA during calibration include the following:

1. Detailed procedures are followed for each calibrator use.

2. The instrument calibrator is properly shielded.

3. An interlock is provided to prevent opening the calibrator door while the source is exposed.

4. Portable sources used to calibrate fixed instruments are transported and maintained in shielded containers as required.

5. The RWP system is used to provide positive radiological controls over calibration.

6. Where possible, fixed instruments are situated in a low radiation area so that necessary test signals can be inserted with the instrument in place.

12.5.3.3 Access Control

One of the means available for ensuring that personnel radiation exposure is maintained ALARA is to control access to areas of potential exposure. Areas of the plant are designated, posted, and controlled according to the degree of radiological hazard or potential hazard in that area.
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For radiological control purposes the restricted area is divided into clean and controlled areas. Controlled areas are posted as required by 10CFR20. Posted areas include the following:

1. Radiation areas

2. High radiation areas

3. Locked high radiation areas

4. Very high radiation areas

5. Contaminated areas
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6. Radioactive material areas

7. Airborne radioactivity areas

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8. Deleted

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9. Radiologically Controlled Area (RCA) (Controlled Access Area (CAA) is an equivalent posting for RCA but RCA is preferred)

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Normal access to the RCA is through an established control point. Regulations and precautions that apply as minimum conditions for entry to a controlled area include:

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1. Personnel monitoring equipment

2. Training as required by 10CFR19

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3. Other requirements posted at the area entrance or imposed by radiation protection personnel.

Positive control is exercised over entries into high radiation areas as specified by 10CFR20 and station procedures. Entrance into radiation areas as well as any work performed in radiologically posted areas requires an RWP.

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12.5.3.4 Contamination Control

Controlling the spread of removable contamination is necessary to ensure that radiation exposure of personnel is maintained ALARA. Contamination limits for personnel, equipment, and surface areas are delineated in radiation protection procedures. Surveys for removable contamination are performed by radiation protection personnel on a routine basis. Special surveys are performed in areas whenever a change in contamination levels is likely and may have radiological importance.

Areas found contaminated are posted, isolated (with ropes, barriers, etc.), and decontaminated as practical.

The level of contamination and number of such areas are minimized. Step-off pads are used at exits from contaminated areas to help control the spread of contamination.

Tools and equipment used in potentially contaminated areas are surveyed for removable contamination and fixed contamination prior to removal from controlled areas. If tools or equipment do not meet the clean area limits, they must be decontaminated before removal from the controlled area unless specifically authorized and controlled by

radiation protection personnel. Some tools and equipment are for use only in a contaminated area. They are normally identified with a conspicuous unique marking and stored in designated areas. These items are also surveyed periodically and decontaminated as appropriate. They are considered to be contaminated and are used only by personnel in appropriate protective clothing.

Personnel are protected from inhalation, ingestion, or absorption of contamination by the use of protective clothing, respiratory protection equipment, and good work practices. Personnel are responsible for surveying themselves for contamination when exiting contaminated areas. If contamination is found, the individual notifies radiation protection personnel and is decontaminated in accordance with approved plant procedures.

12.5.3.5 Respiratory Protection

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When airborne radioactivity is detected in the restricted area at 30 percent of the derived air concentration values in 10CFR20, the area is isolated and posted as an airborne radioactivity area and access is controlled. Entry into these areas requires the issuance of an RWP and monitoring of internal exposure. The RWP system specifies if respiratory protection is required. Air sampling and counting are used to ensure that respiratory protective equipment specified on the RWP remains appropriate. The respiratory protection program conforms to 10CFR20 requirements.

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12.5.3.6 Radiation Work Permit

The purpose of the RWP is to help control exposure of plant personnel to radiological hazards, thus maintaining radiation exposure of personnel ALARA. The RWP maintains exposure of personnel ALARA by providing requirements and instructions to:

1. Control access to radiologically hazardous areas

2. Restrict the spread of radioactive contamination

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3. Assure that proper radiological procedures and precautions are followed

4. Provide a history of how work was completed and exposure received by personnel which can be used for review in efforts to reduce exposure.

5. Document exposures of personnel to ensure no one exceeds the limits of 10CFR20.

RWPs are issued by the radiation protection department and approved in accordance with radiation protection procedures. Typical information contained on an RWP may include:

1. Type and location of job
2. Exposure limits of personnel involved
3. Radiological conditions of work area (radiation, contamination, airborne activity, etc.)
4. Type and number of dosimetry devices to be used
5. Protective clothing and equipment requirements
6. Allowable working time (stay time)
7. Any special precautions or instructions.

Adherence to the RWP requirements ensures that work in posted areas is performed in a radiologically safe manner, and adherence to the permit is mandatory. Repeated or willful violation of RWPs is cause for administrative action. Reporting and documentation is specified in administrative procedures.

12.5.3.7 Dosimetry of Personnel

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Plant employees, contractors, and visitors who are required to be monitored for radiation exposure in accordance with 10CFR20.1502 are required to wear a DLR and a direct reading dosimeter when in a radiologically controlled area, in accordance with 10CFR20. Only those individuals who have completed the required training and qualification program are authorized to enter radiologically controlled areas unescorted. Other personnel must be accompanied at all times by a properly trained escort.

Personnel who are not required to be monitored for radiation exposure in accordance with 10CFR20.1502 are required to wear at least a direct-reading dosimeter when in a radiologically controlled area.

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Neutron sensitive DLRs are issued to those individuals who are expected to be subjected to neutron exposure. RWPs specify the need for neutron dosimetry. DLRs are processed periodically and

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provide the official exposure record of personnel. Direct-reading dosimeters are read at least daily by each individual who has entered a radiologically controlled area to provide a daily estimate of exposure. The DLRs are processed more frequently when an individual's exposure status is in doubt. The results of direct-reading dosimeters and DLRs are recorded and maintained by plant personnel in accordance with approved plant procedures.
4←• 9←• •→10
The internal monitoring program consists of "passive" monitoring and bio-assays. Portal monitors (or the equivalent) are set up at major exits to the radiologically controlled area to provide for passive internal and external monitoring. The whole body counters plus in-vitro analysis (as appropriate) are used in the determination of internal dose. Plant procedures provide guidance with regard to bio-assays.
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Exposure data of all personnel are collected and recorded on NRC Form 5 or the equivalent. Occupational exposures incurred by individuals prior to working at River Bend Station are summarized on NRC Form 4, or the equivalent.
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12.5.3.8 Radiation Protection Training

Plant personnel (permanent and temporary), visitors, and contractors are properly trained in the fundamentals of radiation protection and certified or qualified prior to being allowed to enter the radiologically controlled area unescorted. Training provided is commensurate with the degree of radiological hazard associated with specific work assignments. Permanent plant employees are required to attend a retraining program in radiation protection on a periodic basis. Personnel whose duties do not require entry into radiologically controlled areas are trained to recognize radiation and contamination area signs and are made aware of the reasons for keeping out of such areas. Training must meet the requirements of 10CFR19.

Plant personnel (permanent and temporary) and contractors who enter areas where respiratory protection is needed are trained, certified, and/or qualified in respiratory protection prior to being allowed to enter the area. These

same people are required to attend a retraining program in respiratory protection at least annually. Personnel whose duties do not require entry into controlled areas are trained to recognize respiratory protection area signs and are made aware of the reasons for keeping out of such areas. Training is consistent with NUREG-0041.

12.5.3.9 Radioactive Materials Safety Program

Various types and quantities of radioactive sources are employed to calibrate the process and effluent radiation monitors, the area radiation monitors, and portable and laboratory radiation detectors. Check sources that are integral to the area, process, and effluent monitors consist of small quantities of byproduct material that do not require special instructions for radiation protection purposes. The same consideration applies to solid and liquid radionuclide sources of exempt quantities or concentrations which are used to calibrate or check the portable and laboratory radiation measurement instruments. Recognized methods for the safe handling of radioactive materials are incorporated in procedures and instructions to ensure proper handling of radioactive material. External doses are minimized by a combination of time, distance, and shielding considerations. Internal doses are minimized by the measurement and control of loose contamination.

Sealed radionuclide sources having activities greater than the quantities of radionuclides defined in 10CFR20 and 10CFR30 are subjected to material controls for radiological protection. These controls include:

1. Monitoring of all shipments containing radioactive materials in accordance with 10CFR20.

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2. Inventory and leak testing of sealed sources in accordance with approved plant procedures.

3. Labeling of each source or source holder in accordance with 10CFR20.1904.

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4. Storage in a controlled area of each source that is not installed in an instrument or other equipment.

5. Maintenance of records on the results of inventories, leakage tests, use, location, and the receipt and final disposition dates for all sources.

Additional details of the materials safety program are provided in the plant instructions.

Radioactive sources that are subject to the material controls described herein are used or handled only as authorized by radiation protection personnel. Each individual using these sources is familiar with the radiological restrictions and limitations placed on their use. These limitations protect both the user and the source.