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# 12.0 RADIATION PROTECTION

This chapter provides information on radiation protection features of the plant facilities and equipment, methods employed to achieve such protection, estimated occupational radiation exposures to operating and construction personnel during operation, and anticipated operational occurrences. Information is provided on facility and equipment design, the planning and procedures program, and the techniques and practices employed in meeting the standards for protection against radiation of 10 CFR Part 20, and guidance given in the appropriate NRC regulatory guides.

The "new" 10 CFR 20 regulation mandatory as of January 1, 1994 was implemented by CP&L facilities as of January 1, 1993. The new dose terminology and 10 CFR 20-referenced subsections have been incorporated into this chapter where they are directly interchangeable without impacting the original licensing basis of the plant. However, dose calculation method and resultant data tables (e.g. Tables 12.2.2-2 through 12.2.2-4) which were developed in support of the SHNPP Operating License were based on the "old" 10 CFR 20 in effect at the time. These sections have not been changed in order to preserve the licensing basis data in the FSAR and to remain consistent with corresponding analyses contained in the NRC's Safety Evaluation Report (SER). This licensing basis information is notated by the term "pre-1993 10 CFR 20" to avoid confusion over continued use of the old terms (e.g., MPC, whole body, etc.).

The terms Health Physics (HP), Radiation Control (RC), and Radiation Protection (RP) are used interchangeably throughout this document.

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

### 12.1.1 POLICY CONSIDERATIONS

#### 12.1.1.1 Corporate Health Physics Policy

It is the policy of the Carolina Power & Light Company to develop, implement, and maintain sound health physics programs at each Company facility where radiation producing equipment and/or radioactive materials are used or stored. The health physics programs shall be structured to ensure that radiation doses to Company personnel, contractor personnel, and the general public are maintained at levels which are as low as reasonably achievable (ALARA) and consistent with the United States Nuclear Regulatory Commission Regulations in Title 10 of the United States Code of Federal Regulations and with applicable state regulations.

The line management at each of these Company facilities is responsible and accountable for implementing and enforcing the facility's health physics program. Health physics personnel shall be assigned to these Company facilities to assist line management in carrying out their responsibility to protect workers and the general public. These personnel shall have sufficient independence from the line management, which they assist, to assure that proper health physics practices are not compromised by operational pressures. Furthermore, health physics personnel shall have access to higher levels of management for the resolution of health physics concerns that cannot be resolved at a lower management level.

All Company and contractor employees working in a facility where exposure to radiation might occur are personally responsible for maintaining radiation doses and releases of radioactive materials to unrestricted areas as far below specified limits as reasonably achievable, for minimizing the creation of radwaste, and for supporting the requirements of the health physics programs consistent with the proper discharge of their duties. Personnel who habitually or willfully disregard or violate health physics procedures and practices will be subject to disciplinary action.

A manual shall be developed and maintained as the controlling document for the Company's health physics programs and shall set forth policies and standards for the Company's health physics programs. The health physics programs at the Company's nuclear generating plants shall comply with the manual and meet the intent of the Institute of Nuclear Power Operations' "Guidelines for Radiological Protection at Nuclear Power Stations."

The goal of the Company is to maintain the annual integrated occupational dose at nuclear plants and dose to members of the public from Company activities among the lowest in the country. The design, maintenance, and operation of nuclear facilities shall be consistent with this goal. Modifications to existing nuclear facilities shall be designed and implemented in compliance with the health physics programs to meet the ALARA objective.

To support this goal and to allow management to conduct effective health physics programs, the Company will commit sufficient resources in the form of facilities, equipment, and personnel to the programs. Personnel involved in the conduct of health physics programs, including general employees and contractors, shall be given adequate training and instruction to allow them to contribute to the programs.

At the Corporate level, the Supervisor - Radiological Services provides oversight for administration of the corporate Radiation Control & Protection Manual (RCPM). The requirements of the RCPM help to ensure quality and consistency in the health physics programs at the CP&L nuclear plants. The Supervisor - Radiological Services is also responsible for HP support provided by the corporate Dosimetry Laboratory facilities located offsite. The reporting relationship of the Supervisor - Radiological Services to the Senior Vice President and Chief Nuclear Officer - Nuclear Generation is shown in Figure 12.1.1-1.

### 12.1.1.2 Facility Management Policy

Carolina Power & Light Company has been committed, since the initial design phases of the Shearon Harris Nuclear Power Plant, to a program of keeping occupational radiation exposure as low as reasonably achievable (ALARA). The continuation of this policy during plant operations is another important management commitment. The Operating License, issued by the Nuclear Regulatory Commission, carries with it an obligation to both workers and the general public to maintain exposures as low as is reasonably achievable, considering costs and expected benefits. Carolina Power & Light Company plans to follow the general guidance of Regulatory Guides 1.8, 8.8, and 8.10, and publications that deal with ALARA concepts and practices, including Title 10, Code of Federal Regulations, Part 20. As discussed in Section 12.1.1.1, corporate management has formally committed itself to these concepts by issuing and endorsing the Corporate Health Physics Policy, which ensures compliance with all state and federal regulations that pertain to the safe operation of nuclear power plants.

The Radiation Control and Protection Manual provides the direction necessary for implementing corporate policy.

The Radiation Control and Protection Manual sets forth the basic philosophy and general radiation protection standards that are essential to the safe operation of CP&L's nuclear facilities. The site manager of each nuclear facility is responsible for ensuring that the requirements of this manual are included in the Health Physics Program at that facility.

The primary purposes of the Health Physics Program are to provide personnel with a safe environment in which to work, to protect the general public and the off-site environs, and to establish procedures and a system of records to meet all the requirements of applicable regulations.

Effective control of radiation exposure involves the following major considerations:

1. Management commitment to, and support of, the Health Physics Program,

2. Careful design of facilities and equipment to minimize radiation exposure during operation and maintenance,

3. Good radiation protection practices, including good planning and the proper use of appropriate equipment by qualified, well-trained personnel.

The management of CP&L is firmly committed to performing all reasonable actions for ensuring that radiation exposures are maintained ALARA.

Section 12.1.2 and Section 12.3 discuss the ALARA considerations that have been incorporated into the design of the Shearon Harris Nuclear Power Plant. The facility will be operated and maintained in such a manner as to ensure that occupational radiation exposures are ALARA. Training programs have been established to ensure that personnel understand both why and how occupational radiation exposures will be maintained ALARA.

#### 12.1.1.3 Facility Management Responsibilities

Management's commitment to the Corporate Health Physics Policy is reflected in the design of the plant, the careful preparation of plant operating and maintenance procedures, the provision for review of these procedures and for review of equipment design to incorporate the results of operating experience, and most importantly, the establishment of an on-going training program. Training is provided for personnel, so that each individual will be capable of carrying out his responsibility for maintaining his own radiation exposure, as well as that of others, ALARA consistent with discharging his duties. The development of a proper attitude and an awareness of the potential problems in the area of health physics is accomplished through proper training of all plant personnel.

The responsibility for implementation of the ALARA program resides with the site senior management, with primary support from all site Section and Unit heads. The Superintendent - Radiation Protection (RP) reports to the General Manager - Harris Plant and makes recommendations to plant management concerning the most effective radiation exposure reduction methods. The Superintendent - Radiation Protection includes, as a major portion of his assignment, an analysis of plant operations and maintenance with respect to maintaining an

ALARA approach to personnel radiation exposure. Figure 12.1.1-1 shows the reporting relationships for plant and corporate support personnel in health physics related organizations.

The success of the ALARA program depends upon cooperation between many plant operating groups. The ALARA Specialist acts as a liaison between these groups while maintaining a high degree of organizational freedom. An ALARA committee, composed of an individual from each major plant section/unit, will handle plant worker's suggestions for reducing radiation exposure, ensure interface with various plant groups, and provide a mechanism for the review of outages and maintenance activities.

The Superintendent - Radiation Protection provides for in-plant radiation protection activities on a day-to-day basis. He is assisted by the Health Physics Supervisors, a technician staff, the ALARA Specialist, and other Specialists.

The overall effectiveness of the program is reviewed periodically by appropriate plant and corporate management personnel. Written guidance and procedures have been developed to direct implementation of the program. Included in the formal guidance are the SHNPP Health Physics Program (discussed in Section 12.1.1.2) and a written ALARA program.

The main goals of an ALARA program are to maintain both individual radiation doses and collective radiation exposures ALARA through the use of improved equipment, procedures, and work practices. The first step in achieving this goal is to identify the major radiation exposure areas (such as maintenance, radwaste handling, routine surveillance, in-service inspections, and refueling). For each type of activity, the most beneficial radiation exposure reduction methods are used. For external radiation exposure, the following principles are applied as needed to reduce radiation exposure: reduction of time spent in radiation fields, increasing distance from radiation sources, and provision of adequate shielding from the sources of radiation. The control of internal radiation exposure involves the use of process and engineering controls; a respiratory protection program is in effect in instances where such controls are not practical or adequate.

Proper and timely review of plant procedures and modifications is vital for averting potential, unwarranted personnel radiation exposures. Specific procedures, planning, and practice with mockups may be used to reduce radiation exposure for a particular job. After completion of work on high exposure jobs, actual radiation exposures are evaluated and can be compared to predicted exposures. Radiation exposure can be trended and analyzed for use in planning future work procedures and techniques.

#### 12.1.1.4 Policy Implementation

The ALARA policy is implemented at the Shearon Harris Nuclear Power Plant by the section and unit managers under the direction of the site Vice President. The operational ALARA considerations identified in Sections 12.1.3 and 12.5.3.2 are incorporated in plant procedures.

A training program has been established to give appropriate plant personnel the knowledge necessary to understand why and how they should maintain their occupational radiation exposure ALARA.

12.1.1.5 ALARA Program Implementation

The Plant Management's responsibilities for implementation of corporate policy include:

1. Ensuring that an effective measurement system is established and used to determine the degree of success achieved with regard to the ALARA goals and objectives,

2. Ensuring that the measurement system results are reviewed on a periodic basis, and that corrective action is taken when attainment of the specific objectives appears to be jeopardized,

3. Ensuring that the authority for providing procedures and practices, by which the specific goals and objectives will be achieved, is delegated, and

4. Ensuring that the resources needed to achieve ALARA goals and objectives are made available.

12.1.1.6 Health Physics Program Implementation

The Health Physics Program is based on regulations and personnel experience including or considering the following:

1. Detailed procedures were prepared and approved for radiation protection prior to plant operation. These procedures are revised to incorporate operating experience.

2. Radiological incidents are thoroughly investigated and documented to reduce the chance of recurrence. Reports are made to the NRC, in accordance with 10 CFR 20.2202 and 10 CFR 20.2203 and the Technical Specifications.

3. Periodic radiation, contamination, and airborne activity surveys are performed and recorded to document radiological conditions. Records of the surveys are maintained in accordance with 10 CFR 20.2103.

4. Radiation and high radiation areas are defined and identified in accordance with 10 CFR 20.1003 and 10 CFR 20.1902 and the Technical Specifications. Airborne radioactivity is determined and posted in accordance with 10 CFR 20.1501 and 10 CFR 20.1902. Positive control is exercised for each individual entry into locked high radiation areas.

5. Access control points are established to separate potentially contaminated areas from uncontaminated areas of the plant.

6. Radiation work permits (RWP) are issued for all activities in the Radiologically Controlled Areas (RCAs). Jobs involving significant radiation exposure to personnel are planned to the maximum extent practicable. Emphasis is placed upon the use of mock-ups, special tools, and temporary shielding.

7. Protective clothing is used to help prevent personnel contamination and the spread of contamination from one area to another. Other hazards, such as heat stress, and the impact on total dose, are considered when prescribing the use of protective clothing.

8. Personnel are provided with dosimetry to measure their radiation exposure, in accordance with 10 CFR 20.1502.

9. Process radiation, area radiation, portable radiation, and airborne radioactivity monitoring instrumentation are calibrated periodically.

10. Tools and equipment used in RCAs are surveyed as appropriate for contamination before removal from the RCA. Contaminated tools and equipment that are removed from an RCA will be packaged as necessary to prevent the spread of contamination.

11. A bioassay program that includes whole body counting and/or excreta sampling to evaluate the intake of radioactive material has been established. The program meets the intent of Regulatory Guide 8.9, Revision 1 - 1993.

12. Records of occupational radiation exposure are maintained and reports are made to the NRC, as required by 10 CFR 20.2104 thru 10 CFR 20.2106, and to individuals as required by 10 CFR 19.13.

13. An environmental radiological monitoring program measures any effect of plant radioactive releases to the surrounding environment.

14. All radioactive effluent pathways from the plant are monitored, and records are maintained.

15. Incoming and outgoing shipments which may contain radioactive material are surveyed to assure compliance with 10 CFR 20, 10 CFR 71, and 49 CFR 173, as applicable.

### 12.1.2 DESIGN CONSIDERATIONS

The extensive design effort expended upon SHNPP will contribute greatly in reducing occupational radiation exposures to the plant staff and offsite contractors. Radiation protection design considerations included shielding of radioactive components, ventilation system improvements, equipment design and selection with emphasis upon safety, reliability and maintainability, equipment layout and access, provisions for remote operations and system flushing, facility design to allow ease of component removal, movement to decontamination facilities and repair facilities, as well as numerous innovative design features and equipment additions which have been included in the plant to improve operability and minimize personnel radiation exposure.

The SHNPP radiation protection design considerations are based upon a practical approach for maintaining occupational radiation exposure (ORE) ALARA. This approach, which is based upon the corporate policy previously discussed, is used by establishing and implementing a set of radiation protection design goals. These goals were determined according to conservative and practical criteria in facility and equipment design, experience from past designs and operating plants incorporated to improve the present SHNPP designs and mechanisms and procedures established to ensure design reviews not only by system and component designers, but also by personnel experienced in radiation protection.

The present plant design incorporates the applicable guidance of Regulatory Guide 8.8. This design effort, which preceded this regulatory guide, established a formal design guide for the

Architect/Engineer (A/E) based upon CP&L experience and commitment to keeping ORE ALARA. Numerous dose assessment studies and evaluations were performed in designing plant systems, especially in areas where new technology was utilized. A discussion of specific design features is provided in Section 12.3.1.

Although features have not been incorporated into the design of the SHNPP specifically for ensuring that occupational radiation exposures will be ALARA during decommissioning, many of the inherent design features and policy considerations to ensure that occupational radiation exposures will be ALARA throughout the operating life of the plant are also applicable during the eventual decommissioning of the plant.

12.1.2.1 Radiation Protection Design Goals

The SHNPP radiation protection design goals ensure compliance with the standards for radiation protection specified in 10 CFR 20. General design goals utilized to keep in-plant radiation exposures to ALARA levels include:

a) Minimizing the necessity for and amount of personnel time spent in radiation areas; and,

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

The following methodology for implementing the design goals was used to the extent practicable as a basis for maintaining occupational radiation exposures ALARA:

a) Establish design dose rates and airborne concentration limits for general access areas based upon CP&L experience and 10 CFR 20 regulations,

b) Determine the most severe radiological mode of operation for each piece of equipment and section of pipe,

c) Based upon source terms, determine the source for each piece of equipment, pipe, and general area,

d) Determine shielding and ventilation requirements to maintain design dose rate(s) and airborne concentration limits, respectively,

e) Determine advantages and disadvantages of equipment location, orientation, and segregation,

f) Use predetermined guidelines and criteria for locating piping and penetrations and design of the ventilation system,

g) Implement changes in design, including choice of equipment, wherever practicable to achieve ALARA exposures.

Both equipment and facility designs are considered in keeping exposures ALARA during plant operations, including normal operation, maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste processing, handling and

disposal, and other events of moderate frequency. The actual design features used are described in 12.3.1.

12.1.2.2 General Design Considerations for ALARA Exposures

The SHNPP radiation protection design goals are expanded to the total plant design objectives. These objectives are categorized into several radiation protection concerns, which are described in the following subsections. Plant layout considerations include direct radiation (scattered and direct gamma rays and/or neutrons from non-airborne radiation sources(s)), and ventilation considers airborne radioactivity (see Section 12.2.2).

The design objectives reflect the operating experience at the H. B. Robinson Unit 2 and Brunswick Steam Electric Plant to obtain an improved design.

12.1.2.2.1 Plant Layout and Shielding

The SHNPP layout is based upon a number of considerations including personnel access for ease of maintenance and operations. The location of important features such as decontamination facilities, equipment access hatches, equipment laydown and work areas, and maintenance shops were established with emphasis given to keeping operational radiation exposure ALARA.

Plant facilities' general design considerations to minimize the amount of personnel time spent in a radiation area include:

a) Whenever practicable, locating equipment and instruments (which will require routine maintenance, calibration, or inspection), for ease of access and a minimum of required occupancy time in radiation fields,

b) Where practical, arranging plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment,

c) Providing, where practicable, for transportation of equipment or components requiring maintenance or repair to a lower radiation area,

d) Whenever practicable, provide for removal of equipment from the plant without requiring removal of HVAC duct work, piping and surrounding support structural members

e) Providing rigging and scaffolding insert plates to minimize problems with equipment removal

f) Providing removable block walls or easily removable floor or wall plugs to minimize the radiation exposure in gaining access to highly radioactive components when removal (e.g., tube pulling) is required

Plant general design considerations provided to minimize radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include:

a) Separating radiation sources and occupied areas, where practicable

b) Locating equipment, instruments, and sampling stations, in the lowest practicable radiation zones

c) Providing means and adequate space for using portable shielding

d) Providing means to control contamination and to facilitate decontamination of potentially contaminated areas

In conjunction with the plant and equipment layout, shielding was arranged and designed according to the following objectives:

a) A sufficient quantity of access paths (general access areas) are furnished to allow personnel access to equipment

b) Sufficient shielding is provided to control the levels of radiation present in a general access area.

c) Radiation areas are classified into zones according to expected (maximum) radiation levels.

d) Shielding is provided to accommodate equipment removal and maintenance.

12.1.2.2.2 Ventilation

The plant's ventilation systems are designed to provide heat removal and control of airborne radioactivity. Ventilation systems are designed to direct the airflow from areas of low airborne radioactivity to areas of higher airborne radioactivity. The ventilation systems are described in greater detail in Section 9.4. The radiation protection aspects of the systems are discussed in Section 12.3.3.

12.1.2.2.3 Health Physics

The radiation protection design objectives for health physics are met by the following:

a) The applicable 10 CFR 20 limits are maintained for operating personnel and the general public,

b) The plant's radiation monitoring equipment is designed to detect and annunciate excessive airborne radioactivity and high radiation levels,

c) Personnel radiation monitoring equipment is provided to measure and record personnel radiation exposure,

d) Periodic radiation surveys are performed when required,

e) Access to radioactive contaminated equipment is designed so that, with properly trained personnel, radiation exposures during all modes of plant operation meet the ALARA requirements,

f) Cleaning and decontamination facilities are provided for equipment and protective clothing,

g) Radioactive fluids (liquids and gases) are contained and controlled to keep the release of radioactive materials to general access areas and the environment ALARA,

h) Radioactive effluent release paths to the environment are monitored and facilities for analysis of radioactive samples are furnished, and

i) The 10 CFR 50 Appendix A, Criteria 19, limits for the Control Room are met for a design-basis accident and lesser accidents.

12.1.2.3 Improvements in Facility Design Due to Past Experience and Operation

In addition to SHNPP, Carolina Power & Light Company operates two licensed General Electric BWR's and one other Westinghouse PWR. The operating experience obtained from these plants was incorporated into the design of SHNPP. In addition, published information on radiation problems and radiation protection was used to anticipate and minimize occupational radiation exposure. During plant design, operating reports and data such as that given in WASH 1311, NUREG-75/032, NUREG-109, and AIF paper "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants" September 1974, References 12.1.2-1 through 12.1.2-4 respectively, were reviewed to determine which operations, procedures or types of equipment were most significant in producing personnel exposures. Experienced operating personnel continually reviewed the plant design as the design progressed, and provided recommendations based on their experience.

Prior to the initial promulgation of Regulatory Guide 8.8, several meetings were held between CP&L engineering and operating personnel and Ebasco personnel (Architect/Engineer) responsible for the radiation protection design of SHNPP to discuss ways of improving the design. These meetings culminated in tours of the H. B. Robinson (HBR) Unit 2 operating plant. First-hand knowledge was gained of the design and operating features which could be readily improved in the SHNPP to keep radiation exposures ALARA. Further information related to inspection, maintenance, and repair times and radiological hazards was given to Ebasco so that this information could be factored into the plant layout features, systems design, and estimate of radiation exposures to personnel.

The initial result of these meetings and information exchange was the issuance of a set of guidelines for the design of the SHNPP to minimize personnel radiation exposures. These guidelines were prepared by Ebasco radiation protection engineers and were reviewed and approved by CP&L's cognizant health physics and operating personnel.

The guidelines were then issued to all designers and engineers involved in the SHNPP design effort, and the adherence to the guidelines was monitored by the radiation protection engineers. Additional design feedback continued throughout the project.

It is significant to note that these guidelines, summarized in Section 12.3, paraphrase to a large extent Regulatory Guide 8.8 even though they preceded it. The guidelines were updated to reflect Regulatory Guide 8.8.

Review of the radiation protection design was a continuing process throughout all phases of the design. Ebasco's radiation protection personnel worked side by side with the other disciplines' engineers and designers to ensure that all necessary radiation protection considerations were taken into account.

Listed below are examples of some of the many design changes and improvements that occurred during design and construction:

1. The Waste Processing Building was continually upgraded. In each instance, one of the major considerations for the upgrade was the addition of space for equipment designed to segregate wastes, thereby limiting the high radioactivity to isolated systems, making wastes easier to shield, and providing more space for pipe chases, valve and pump galleries, remote valve operating stations, and sampling stations. Where possible, each tank, valve, pump, and system has been arranged so that work on valves and pumps can be accomplished with a minimum of exposure from other portions of the same system. Even after the last complete upgrade, efforts have continued to improve the radiation protection features.

   In the waste gas compressors and catalytic hydrogen recombiners area, the valve galleries were rearranged and labyrinths added to reduce the exposure doses. In the filter backwash area new walls were added to provide shielding labyrinths. New shielding labyrinths were provided for floor drain pumps. In the same area, roofs were added to provide shielding for the hot lines going from the sumps to the pipe tunnel.

2. Backflushable filters have been installed instead of conventional cartridge filters so that there would be virtually no requirement for filter handling with the concomitant radiation exposures. This was the result of an extensive close assessment of operating PWR experience with cartridge filters. Similar evaluations were performed for other equipment.

3. In the heat exchanger area of the Reactor Auxiliary Building, the piping has been rerouted to arrive at a configuration resulting in less potential personnel exposure. From a design that had most of the piping routed in the valve galleries, the configuration was changed to one in which the piping runs mostly in shielded chases.

4. Shielding walls have been provided for equipment which is expected to be rarely, if ever, radioactive, to ensure protection against such an eventuality. Examples of these shielding walls are those for boric acid tanks.

5. Carolina Power & Light Company developed a state-of-the-art concept for radiation monitoring which allows a computer based system to monitor all remote radiation monitor devices without requiring personnel access. This concept has been adopted by over a dozen other utilities. The bases for this design effort were CP&L's previous experience with such equipment and the desire to improve health physics coverage.

CP&L's engineering procedures require the incorporation of ALARA in plant modification design work performed by CP&L or by design firms under contract to CP&L.

Routine survey data from CP&L's operating plants was reviewed to improve the design of SHNPP. Additional examples of how CP&L's experience has contributed to the SHNPP design can be found in Section 12.3.

### 12.1.2.4 Equipment Design Considerations

Radiation protection general design considerations for equipment include shielding, equipment access, equipment selection, and equipment maintenance. Equipment design objectives deal with access to, and segregation of, radioactive equipment. Equipment design objectives for radiation protection include:

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

2. Servicing convenience including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair,

3. Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment, and

4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when no feasible method is available to reduce radiation levels.

Equipment general design considerations directed toward minimizing radiation levels proximate to equipment or components requiring personnel attention include:

1. Equipment that processes fluids with low radioactivity is located in separate cubicles from equipment that processes highly radioactive fluids,

2. Hatches are provided as needed to allow access to equipment from the top,

3. Equipment is located in accessible parts of cubicles; equipment frequently changed in whole or in part is readily accessible,

4. Hoists or lifting lugs are provided, as needed, for equipment servicing, maintenance, and removal,

5. Provisions for isolating, draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material,

6. Design of equipment layout, piping runs, and location of valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps,

7. Utilization of high quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials, and

8. Provisions for minimizing the spread of contamination into equipment service areas including direct drain connections.

### 12.1.2.5 Equipment Selection

The selection of equipment to handle and process radioactive materials is based upon system requirements and radiation protection requirements. Consideration is given to minimizing

leakage, spillage, and maintenance. Material and coatings are chosen for decontamination properties as well as durability. Reduced occupational radiation exposure is attained by utilizing operating experience and where practical, providing prudent equipment selections such as:

1. Diaphragm seal valves that require no packing,
2. Longer-lived graphite-filled packing, instead of standard packing,
3. Remote systems (or connections) for remote chemical cleaning where practicable,
4. Crossties between redundant equipment and/or related equipment capable of redundant operation to allow removal of contaminated equipment from service,
5. Air connections to tanks containing spargers to allow for air injection to uncake contaminates,
6. Backflushable filters to eliminate handling of spent cartridge filters,
7. Pumps with flanged connections to allow quick removal and installation, and
8. Remote drumhandling equipment for radwaste packaging.

#### 12.1.2.6 Overall Impact of ALARA Exposure Design Considerations

Carolina Power & Light Company has given extensive attention to maintaining occupational radiation exposure ALARA at SHNPP. The design of the plant facilities, equipment, structures, and access areas is based upon a corporate commitment to minimizing radiation exposure and has been implemented, as practicable, in all aspects of the design. Consideration has been given to routine operations, transient operations, operational occurrences, maintenance, refueling, radioactive waste processing and disposal, and abnormal occurrences and accidents.

The SHNPP design takes into account equipment removal, decontamination, ventilation, orientation of equipment, in situ calibration and maintenance, sampling, monitoring, shielding, controlling contaminated fluids, minimizing leakage and spillage, and radiation exposure.

The design philosophy established for SHNPP strives to maintain occupation radiation exposure ALARA and is in compliance with applicable regulations.

### 12.1.3 OPERATIONAL CONSIDERATIONS

Operational considerations at SHNPP that promote the ALARA philosophy include the determination of the origins of radiation exposures, the proper training of personnel, the preparation of radiation protection procedures, the development of conditions for implementing these procedures, and the formation of a review system to assess the effectiveness of the ALARA philosophy.

Efforts were made to factor operational considerations of radiation exposure in the plant layout and system design, by utilizing the guidelines of Section 12.1.2 throughout the design effort. These guidelines incorporated known operational considerations derived from experience.

Information from operating plants was continuously factored into the design as it progressed to reflect new operational considerations.

The Superintendent - Radiation Protection and staff working closely with other departments review and study plant systems such as the NSSS, the radioactive waste management systems, the Residual Heat Removal System, the Spent Fuel Pool Cooling and Cleanup System, and other systems that collect, store, contain, or transport liquid, gaseous, or solid radioactive material. Objectives are to understand the functional aspects of each system, to identify the origins of radiation exposures in the plant, and to know and identify these exposure origins by location, operation, and job category.

Operational radiation protection objectives deal with access to Radiologically Control Areas, exposure to personnel, and decontamination. Working at or near highly radioactive components requires planning, special methods, and criteria directed toward keeping occupational radiation exposure ALARA. Job training, pre job ALARA reviews, and debriefing following selected high exposure jobs contribute toward reduced exposures. Decontamination also helps to reduce exposure. Procedures and techniques are based upon operational criteria and experience that have worked to keep radiation exposure ALARA.

#### 12.1.3.1 Operational Objectives

The operational radiation protection objectives are met by methods that include the following:

1. Knowledge of plant systems,
2. Experienced personnel to direct and train other personnel,
3. Use of periodic radiation surveys,
4. Use of radiation monitoring equipment to detect airborne radioactivity concentrations and high radiation levels and to measure and record personnel radiation exposure,
5. Analysis of radioactive samples to monitor chemistry and check for radiation release,
6. Use of cleaning and decontamination facilities for equipment and protective clothing [Note: These activities may be performed off site],
7. Detailed job planning for high exposure work,
8. Job simulations when appropriate to improve productivity on the job, thereby keeping exposure ALARA,
9. Briefings after selected high-exposure jobs to identify time consuming work and to identify problems, and
10. Improving procedures and techniques for future jobs.

#### 12.1.3.2 Procedure Development

Plant procedures are prepared, reviewed, and approved in accordance with Section 13.5.

#### 12.1.3.2.1 ALARA procedures

To assure adequate emphasis on the necessity to minimize personnel exposures, ALARA procedures are prepared to address all radiological activities at SHNPP. The procedures emphasize acceptable health physics techniques and methods. The ALARA procedures implement considerations of such topics as ALARA training, ALARA review of applicable Global Radiation Work Permits (RWP) and ALARA Tasks, worker feedback, special task training, and evaluation of proposed changes in facilities or equipment. ALARA procedures provide the necessary basis for instruction of plant personnel in the mechanisms available to minimize personnel radiation exposures. ALARA procedures incorporate guidance from Regulatory Guides 8.8 and 8.10, and CP&L guidelines and criteria.

#### 12.1.3.2.2 Plant procedures

Administrative requirements are implemented to assure that procedures developed by other plant disciplines have adequately incorporated the principle of minimizing personnel radiation exposure. Plant administrative documents describe the criteria of selection of those procedures and revisions that are reviewed by health physics personnel. Recommendations made by health physics personnel will be resolved with the appropriate plant discipline prior to submission for final review, approval and implementation.

#### 12.1.3.2.3 ALARA techniques

In order to control radiation exposure to individuals, a detailed Radiation Work Permit (RWP) is required whenever work involving a significant actual or potential radiological hazard is performed.

The requirements for using a Radiation Work Permit (RWP) including associated tasks are established by plant procedures.

Job planning, training, and RWP requirements ensure that the SHNPP's ALARA policy is fulfilled. Techniques that may be used include:

1. Temporary shielding, such as lead sheets or blankets draped or strapped over pipes or pieces of equipment are used; temporary shielding would only be used if total exposure, which includes exposure received during installation and removal of the shielding, will be effectively reduced.

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

3. For long-term repair jobs, consideration is given to setting up communication and closed-circuit television equipment to assist supervising personnel in checking on work progress from a lower radiation area.

4. Entry and exit control points are established in areas with low levels of radiation. This limits the exposure of personnel that are changing protective equipment and generally preparing to work in radiologically controlled areas. The access control points are set up

to limit the spread of contamination from the work areas to as small an area as practicable.

5. Protective clothing and respiratory protection are selected to minimize the discomfort of workers so that efficiency is increased and less time is spent in radiation areas.

6. Personnel are assigned dosimeters that allow determination of accumulated exposure at any time during a work assignment in high radiation areas.

7. On intricate jobs, especially those which involve high or complex radiation levels, the job planning includes estimates of the person-rem needed to complete the job. At the completion of the work, a debriefing session is held (when practical) with the personnel that performed the work in an effort to determine how the work could have been completed more efficiently and with less radiation exposure.

12.1.3.3 Implementation of Procedures and Techniques

The criteria and requirements for developing various procedures and techniques governing radiation exposure related operations are given in the following:

1. Section 12.1, Ensuring that Occupational Radiation Exposures are ALARA

2. Section 12.3, Radiation Protection Design Features

3. Section 12.5, Health Physics Program

4. Section 13.5, Plant Procedures

The implementation of the ALARA philosophy is directed by the ALARA Specialist on a day-to-day basis. Plant procedures that have the potential for radiation exposure are reviewed by Health Physics to ensure that ALARA techniques and practices are being followed and included in procedures. Entrance to the Radiation Areas at SHNPP is controlled by the Health Physics unit, and requires the issuance of a Radiation Work Permit (RWP) including associated tasks. The permit system is discussed in Sub-Section 12.5.3.

The general employee training program will help implement the SHNPP's ALARA policy. The training program assures that workers understand how radiation protection relates to their jobs and how they can minimize their own exposure while performing their jobs. In addition, all workers will have opportunities to discuss radiation safety with the Health Physics personnel when the need arises.

12.1.3.4 Plant Organization

As described in Section 12.5.1, the plant organization provides the Superintendent - Radiation Protection with access to the General Manager - Harris Plant. This organization will allow the General Manager involvement in the review and approval of specific ALARA goals and objectives as well as review of data and dissemination of information related to the ALARA program.

The organization also assigns the ALARA Specialist the responsibility of reviewing and assisting the plant's ALARA program. This individual is primarily responsible for coordination of plant ALARA activities and will routinely interface with appropriate facility supervision in radiation work planning and post job review.

12.1.3.5 Operating Experience

The Radiation Work Permit process described in Subsection 12.5.3 provides a mechanism for collection and evaluation of data relating to personnel radiation exposure. Information collated by work crew and job assists in evaluating design or procedure changes intended to minimize future radiation exposures.

The Superintendent - Radiation Protection is responsible for the review of radiation exposure records, investigating not only individual exposures, but exposures as classified by job description and job location. Information obtained from these reviews will be compared with radiation exposure results from past experience and with data obtained from average exposure results from other plants to assess the effectiveness of the ALARA effort at SHNPP.

12.1.3.6 Exposure Reduction

Specific radiation exposure reduction techniques that are used at SHNPP are described in Section 12.5.3. Procedures assure that applicable plant activities are completed with adequate preparation and planning; work is performed with appropriate health physics recommendations and support; and results of post-job data evaluation are applied to implement improvements.

In addition, the Health Physics staff will, at all times, be vigilant for ways to reduce radiation exposures by soliciting employee suggestions, evaluating origins of plant exposures, investigating unusual exposures, and assuring that adequate supplies and instrumentation are available.

Plant management will periodically review the radiation exposure data in order to identify excessive radiation exposure areas, excessive exposures by job categories, and other exposure trends. They will determine if improvements are needed in plant procedures, health physics procedures, plant equipment or training.

REFERENCES: SECTION 12.1

2.1.2-1 T.D. Murphy, WASH-1311, UC-78, A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants 1969-1973, USNRC Radiological Assessment Branch, May 1974.

12.1.2-2 T.D. Murphy, et al., NUREG-75/032, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974, USNRC Radiological Assessment Branch, June 1975.

12.1.2-3 T.D. Murphy, et al., NUREG-0109, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1975, USNRC Radiological Assessment Branch, August 1976.

12.1.2-4 C. A. Pelletier, et al., National Environmental Studies Project, Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, Atomic Industrial Forum, September 1974.

## 12.2 RADIATION SOURCES

### 12.2.1 CONTAINED SOURCES

The radiation sources used for the design and analysis of the shielding requirements were based on the nominal core power level (2958 MWt). The original plant design and analysis shielding requirements were based on Reference 12.2.1-4 [Rev. 2 of the Radiation Analysis Manual]. As part of the SGR/PUR project, the information in References 12.2.1-4 was reviewed and updated to create Reference 12.2.1-6. The new SGR/PUR analyses in Reference 12.2.1-6 used a core power level of 2958 MWt (2% above the licensed core power) to determine radionuclide inventories and source strengths for the core, spent fuel, and plant equipment. These sources were determined for all phases of plant operation including full power operation, shutdown conditions, refueling operations, and for various postulated accidents. They included the neutron and gamma fluxes outside the reactor vessel, the reactor coolant activation, fission and corrosion product activities, deposited corrosion and fission product sources on reactor coolant equipment surfaces, spent fuel handling sources, and postulated core meltdown sources. In addition, radiation sources for various auxiliary systems are also tabulated.

For SGR/PUR conditions, the original plant design and analysis of shielding requirements were generally determined to be either historical in nature (plant layout evaluations, determinations of required shield wall thickness, etc.) or to be not significantly impacted. While the data tables contained in Sections 11.1 and 12.2 have been updated with post-SGR/PUR information (Reference 12.2.1-6), the design and analysis of shielding requirements remain generally based on the nominal core power level (2958 MWt) information in Reference 12.2.1-4.

#### 12.2.1.1 Reactor Coolant Fission and Corrosion Product Activity

The primary sources of radioactivity during normal full power operation are direct core radiation, coolant activation processes, leakage of fission products from pinhole defects in fuel rod cladding, tramp uranium and activation of reactor coolant corrosion products. The design basis for the shielding source terms for fission products used in shielding design is cladding defects in fuel rods producing one percent of the core thermal power. The design basis for activation and corrosion product activities were derived from measurements at operating plants and are independent of the fuel defect level. The radionuclide activity levels in the reactor coolant at the design basis level are given in Section 11.1. The models and assumptions used in determining the design basis radionuclide activity levels in the reactor coolant for these sources are also given in Section 11.1.

#### 12.2.1.2 Neutron Fluxes at Full Power Outside the Reactor Pressure Vessel

Four group neutron fluxes at the inside surface of the primary shield concrete at the core mid-plane are shown on Figure 12.2.1-1. Values were estimated using a representation of current and future plant operation. Refer to reactor vessel irradiation surveillance program for actual results.

12.2.1.3 Gamma Fluxes at Full Power Operation

Four group gamma ray fluxes at the inside surface of the primary shield concrete at the core mid-plane are shown on Figure 12.2.1-2. Gamma ray dose rates at the inside surface of the primary shield concrete at the core mid-plane are shown on Figure 12.2.1-3.

12.2.1.4 Reactor Coolant N-16 Activity

The N-16 activity of the reactor coolant (shown in Table 11.1.4-1) was the radiation source that determined the shielding requirements for the secondary shield wall.

12.2.1.5 Reactor Coolant System Sources at Shutdown

Following shutdown, residual radiation from the Reactor Coolant System is due to fission product decay gamma radiation emanating from the core and corrosion products. The core average gamma ray source strengths used in the evaluation of radiation levels within and around the shutdown reactor are presented in Table 12.2.1-1 for various times after shutdown. Spent fuel gamma ray source strengths are presented in Table 12.2.1-2 for various times after shutdown. The isotopic fission product inventory at shutdown is listed in Table 12.2.1-3. The gamma ray source strengths of the irradiated control rod are tabulated as a function of time after shutdown as indicated in Table 12.2.1-4. The source values are for an irradiation period of 4 years.

Table 12.2.1-5 lists the maximum gamma ray source strengths for an in-core moveable detector irradiation period of 30 days and a drive cable irradiation period of 400 days. The material used for the control rod cladding is type-304 stainless steel with a maximum cobalt content of 0.12 weight percent. The gamma ray source strengths associated with the activated stainless steel are listed in Table 12.2.1-6 for various times after shutdown. The values are for an irradiation time of 15 years.

The most significant radiation sources encountered during normal maintenance and inspection of most plant equipment (pumps, heat exchangers, tanks, valves, and other out-of-core primary equipment) are deposits from the reactor coolant, such as activated corrosion product and some fission products. The corrosion product deposits are usually a mixture of magnetite and other oxides. The composition and activity of typical out-of-core crud deposits are given in Table 12.2.1-7. Maximum crud radionuclide concentrations in the refueling cavity water with no purification in operation are given in Table 12.2.1-8.

12.2.1.6 Pressurizer Activity

The radioactive source in the pressurizer, steam, and liquid phases, as well as the N-16 sources, are shown in Tables 11.1.1-3 and 12.2.1-9, respectively.

12.2.1.7 Contained Sources in Other Plant Systems

The source intensity in equipment and pipelines handling radioactive fluids is determined from that in the reactor water by considering the processes that the reactor water has undergone prior to entering the equipment and pipe (dilution, filtering, demineralization, delay, change of phase, etc.).

In all cases the process or combination of processes leading to the highest activity is considered for conservatism.

Activities in the volume control tank are shown in Table 11.1.1-4.

The radiation sources utilized for shielding of the demineralizers of the Chemical and Volume Control System (CVCS) are shown in Table 12.2.1-10.

The source terms employed for shielding of the other components of the CVCS and Boron Thermal Regeneration System are given in Tables 12.2.1-11, 12.2.1-12 and 12.2.1-13. Radionuclides that are major contributors to the total source strength eight hours after shutdown to the Residual Heat Removal system are shown in Table 12.2.1-14.

Boron Recycle System source strengths are given in Tables 12.2.1-15, 12.2.1-16, 12.2.1-17 and 12.2.1-18.

Waste Processing System source strengths are given in Sections 11.2, 11.3 and 11.4 and also in Tables 12.2.1-19, 12.2.1-20 and 12.2.1-21.

#### 12.2.1.8 Spent Fuel and Spent Fuel Pool

The design and normal fission and corrosion product activities in the spent fuel pools are given in Table 11.1.7-1.

The source terms employed to determine the minimum water depth above spent fuel and shielding walls around the spent fuel pool, as well as shielding of the spent fuel transfer tube, are given in Table 12.2.1-1.

Spent fuel pool demineralizer specific activity is given in Table 12.2.1-22.

#### 12.2.1.9 Accident Sources

The accident source terms which were employed to determine shielding requirements for emergency accessways, control room and containment shielding, as well as potential doses to equipment inside containment following a loss-of-coolant accident (LOCA) are shown in Table 12.2.1-23. The table assumed a release to containment of the activity stated in TID-14844 (Reference 12.2.1-1), namely 100 percent noble gases, 50 percent halogens, and one percent remaining fission product inventory.

#### 12.2.1.10 Normal Operation Sources

The model for evaluating the expected fission product concentrations in the primary and secondary coolants under normal operating conditions including anticipated operational occurrences, was formulated as ANSI/ANS 18.1-1984 (Reference 12.2.1-2) and is the method recommended by NUREG-0017 (Reference 12.2.1-3).

The pertinent plant parameters and assumptions are listed in Table 11.1.2-2 along with the range of values specified in ANSI/ANS-18.1-1984.

The normal plant operation source terms, based on ANSI/ANS-18.1-1984, are listed in Tables 11.1.2-1 (primary and secondary) and 12.2.1-24 (CVCS demineralizers). Included are the concentrations of the noble gases, halogens, rubidum, cesium, and other isotopes in the various fluid streams. Noble gas distributions in the reactor coolant system and steam generator steam were determined assuming no purge of the volume control tank.

The normal demineralizer nuclide activity is based on the full-power operation with the reactor coolant activities as listed in Table 11.1.2-1. For the mixed-bed demineralizer, a DF of 2 is assumed for all rubidium and cesium isotopes in addition to a DF of 10 for other cation and anion nuclides. For the cation-bed demineralizer, an allowance is made for the activity reduction by the mixed bed upstream of the cation-bed demineralizer. The cation bed is assumed to be in operation 10 percent of the time. The activity concentrations on both demineralizers reflect annual resin replacement.

12.2.1.11 Reactor startup neutron sources

The primary and secondary reactor startup neutron sources (see Section 4.2) remain in the reactor for the duration of their useful life and thus are not considered as a separate radiation shielding problem.

Only the primary reactor startup neutron source was initially radioactive and it was adequately shielded when shipped to, and handled at the site.

12.2.1.12 Post TMI shielding review accident sources

The source terms used in the estimation of radiation doses are based on the Westinghouse Radiation Analysis Manual (WRAM-Reference 12.2.1-4). The plant specific core design parameters are stated in WRAM Table 5-1. Noble gas and halogen inventories are given in WRAM Table 5-10; while solid fission products from the spent fuel fission products inventory at shutdown are given in WRAM Table 5-33. The values are converted to a total core inventory basis. In addition, fission products not mentioned in WRAM but which are described in Reference 12.2.1-5, have been included in the evaluation. Decay daughter dose contributions for all fission products have been included in the inventory.

The model assumes removal by plate-out using a mechanistic model considering elemental, particulate and organic fractions of solid fission products. Plate-out source strengths in containment are given in Table 12.2.1-25. The model also assumes mechanistic spray removal and dilution of the released 50 percent halogen core inventory and 1 percent solid fission products inventory source terms with the combined volumes of the Reactor coolant, accumulators and the refueling water storage tank. The resulting liquid activity in the containment sump as a function of time is given in Table 12.2.1-26.

The gaseous source terms for gamma radiation within the containment atmosphere are based on 100 percent of the core noble gas inventory, 50 percent of the core halogen inventory, and 1 percent solid fission products inventory. The assumptions of removal by plate-out and spray are as stated above. These values are tabulated in Table 12.2.1-27.

## 12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Note: Analysis and terms used in this section are based on "pre-1993 10 CFR 20" (see Section 12.0).

Equipment cubicles, corridors, and areas normally occupied by operating personnel can contain small amounts of airborne radioactivity as a result of equipment leakage. Prior to plant operation, the potential exposure to personnel from this activity was evaluated for the Reactor Auxiliary Building, the Containment, the Fuel Handling Building, the Turbine Building, and the Waste Processing Building. Table 12.2.2-1 includes assumptions, parameters, and sources of airborne radioactivity used in the analysis. The sources were determined for each area assuming that leakage occurred in that area and that the leaking fluid contained a fraction of the reactor coolant activity. This fraction was determined from process consideration of leaking fluid (amount of filtering, degassing, demineralization, etc., prior to leak). The leak rate was based on typical data from operating plants. The equilibrium airborne concentration was then determined by use of the standard equation of buildup and removal, where build up is caused by leakage and removal both by radioactive decay and ventilation.

The isotopic airborne concentrations, as a fraction of maximum permissible concentration, were calculated for those areas normally occupied by operating personnel and where a potential for exposure exists. These values (for the Containment, the Reactor Auxiliary Building, the Fuel Handling Building, and the Turbine Building) as shown in Table 12.2.2-2 indicated that the doses would be well below the maximum allowable limit.

A more detailed room by room analysis was performed for the Reactor Auxiliary Building and Waste Processing Building. In this case, airborne radionuclide concentrations were calculated as well as the whole body dose resulting from inhalation and external exposure. The inhalation and external whole body dose conversion factors were taken from Table E-7 of Regulatory Guide 1.109 (see Section 1.8) for the purpose of evaluating compliance with 10 CFR 50, Appendix I (Revision 1) and Table D-3 of WASH-1258 (Reference 12.2.2-1) respectively. The WASH-1258 values for a semi-infinite cloud model result in highly conservative dose values. The assumptions used in the analysis were essentially the same as those in Table 12.2.2-1. Tables 12.2.2-3 and 12.2.2-4 list the results of this analysis.

A negligible release of radioactivity is expected from the removal of the reactor vessel head or movement of spent fuel. There was no relief valve venting. Therefore, contribution from these sources to airborne activity was not considered.

During plant operation, airborne radioactive material sources are evaluated based on type of work evolution, past experience and air sampling. These activities are described in more detail in FSAR Section 12.5.

The airborne concentration of a radioisotope in an area having a constant leak rate, source strength, and exhaust rate can be calculated by the equation given below. Radionuclide concentrations in other areas such as corridors are calculated assuming the air in the corridor can be contaminated by exhaust from nearby areas.

$$C_i(t)/MPC_i = W\ a_i\ SS\ PF_i\ \frac{(1 - e^{-\lambda_{\tau_1} t})}{V\lambda\tau_i MPC_i}$$

where:

W = leak rate of fluid in $cm^3/hr$

$a_i$ = concentration of ith isotope in the reactor coolant in $\mu ci/cm^3$

SS = source strength defined as fraction of reactor coolant present in the leaking liquid

$PF_i$ = partition factor of ith isotope

$\lambda_{ri}$ = total removal rate constant for ith isotope in $hr^{-1} = \lambda_{di} + \lambda_e$

$\lambda_{di}$ = decay constant for ith isotope in $hr^{-1}$

$\lambda_e$ = removal rate constant due to exhaust in $hr^{-1}$

t = time interval in hr

V = free volume of the area where leak occurs in $cm^3$

$C_i(t)$ = airborne concentration of the ith radioisotope at time t in $\mu ci/cm^3$

For small rooms and other operating areas where $\lambda_e$ is found to be much larger than $\lambda_d$ for most of the radionuclides, the peak or equilibrium activity (Ceq) is given by the following equation:

$$Ceq_i/MPC_i = \frac{W\ a_i\ SS\ PF_i}{V\ (\lambda_{di} + \lambda_e)\ MPC_i}$$

$$= \frac{W_{ai}\ SS\ PF_i}{(CFM) 1.7 \times 10^6\ MPC_i}$$

In order to calculate tritium concentration in the Fuel Handling Building the following equations (Reference 12.2.2-2) were used to calculate evaporation rate from the fuel pools:

$$w_p = \frac{A(95+0.425v)}{y} (p_w - p_a)$$

and ventilation rate:

$$(CFM) = \frac{w_p}{4.5\ (W_i - W_o)}$$

where:

$w_p$ = Evaporation rate of water in lbm/hr

n = Air velocity over surface in ft/min

y = Latent heat at pool surface water temperature in Btu/lbm

$P_a$ = Saturation pressure of vapor at room air dew point temperature in inches of mercury.

$P_w$ = Saturation pressure of vapor at the surface water temperature in inches of mercury.

A = Surface area of pool in $ft^2$

$W_o$ = Moisture content of outdoor air in lbm/lbm of dry air.

$W_i$ = Moisture content of indoor air in lbm/lbm of dry air.

CFM = Ventilation rate in $ft^3$/min

The values presented in Tables 12.2.2-2, 12.2.2-3, and 12.2.2-4 are based on reactor coolant concentrations at 0.12 percent failed fuel. However, to account for anticipated operational occurrences and reactor coolant concentrations at 1.0 percent failed fuel, it is estimated that the values presented in Tables 12.2.2-2, 12.2.2-3, and 12.2.2-4 would be increased by a factor of 8. At that level of airborne concentration, the occupancy in a few areas is expected to be very limited. The ventilation system, as stated in FSAR Section 12.3.3.1, is designed to ensure that the airborne radioactivity concentrations, during normal operation, remain within the limits of 10 CFR Part 20. While it is possible to control airborne radioactivity concentrations to remain within 10 CFR Part 20 limits, it is also possible to exceed the Part 20 occupational dose limit in certain areas as a result of contained sources (as opposed to airborne sources). Ventilation systems will be of limited effectiveness in limiting the dose rates caused primarily by contained sources.

REFERENCES: SECTION 12.2

12.2.1-1 DiNunno, Anderson, Bakes and Anderson, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, Atomic Energy Commission, March 23, 1962.

12.2.1-2 "American National Standard Radioactive Source Term for National Operation of Light Water Reactors. ANSI/ANS-18.1-1984, American Nuclear Society, LaGrange Park, Illinois, Approved December 31, 1984.

12.2.1-3 "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, Revision 1, U. S. Nuclear Regulatory Commission, April, 1985.

12.2.1-4 "Radiation Analysis Manual, Model 312, Carolina Power & Light Company, SHNPP Units 1, 2, 3, and 4, Rev 2," Westinghouse, May 1979.

12.2.1-5 Kolar, Michael J. and Olson, Nolan C., "Calculation of Accident Doses to Equipment Inside Containment of Power Reactors," Nuclear Technology, Volume 36, November 1977, pages 56-64.

12.2.1-6 "Shearon Harris Nuclear Power Plant Unit 1, Radiation Analysis Manual, Rev 3, April 2000 (WCAP-15397).

12.2.2-1 "Numerical Guides for Design Objectives and Limiting Conditions for Operation to meet the Criterion 'As Low as Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," WASH-1258, AEC, July 1973.

12.2.2-2 ASHRAE Guide and Data Book Applications 1968 Heating, Refrigeration, Ventilation, and Air-Conditioning - American Society of Heating, Refrigerating, and Air Conditioning Engineers.

## 12.3 RADIATION PROTECTION DESIGN FEATURES

Note: Except where noted, analysis and terms used in this section are based on "pre-1993 10 CFR 20" (see Section 12.0).

### 12.3.1 FACILITY DESIGN FEATURES

Compliance with the design feature guidance specified in Regulatory Guide 8.8 is discussed in Section 12.1.1 and this section. The layout of plant radiation zones is described in Section 12.3.2.2.

The following design criteria have been used in the analysis of the plant to obtain as low as reasonable achievable personnel exposures.

#### 12.3.1.1 Design Objective

Shielding walls have been designed so that the highest dose rate on the outside surface of the wall is less than or equal to the maximum permissible dose rate as per 10 CFR 20, considering occupancy requirements. For instance, in a controlled area of unrestricted occupancy, the design dose rate at the "hot spot" in the shielding wall would be at most 2.5 mrem/hr. The thickness required to achieve this desired dose rate is maintained throughout the extent of the particular wall (i.e., shielding walls are not contoured) (Regulatory Guide 8.8 Paragraphs C.2.b.2 and C.2.b.3).

#### 12.3.1.2 Separation Criteria

To the extent possible systems and components handling high activity fluids are located in the same general area of the plant, taking into account separation criteria (Regulatory Guide 8.8 Paragraphs C.2.b.2 and C.2.b.3). Such systems and components are separated from low activity systems and components; in turn, the latter are located away from "clean" systems and components. With this arrangement, heavy shielding walls can be shared by various components, thus minimizing space requirements and costs (Regulatory Guide 8.8, Paragraphs C.2.b.2 and C.2.b.3).

To the extent possible, all components and piping that do not normally contain radioactivity, nor can be expected to ever become radioactive, are separated from the radioactive portions of the plant. This simplified division of the plant into controlled and uncontrolled areas, aids in the unimpeded traffic within both portions, reduces the possibility that radioactive piping is run in clean areas, minimizes the need for shielded pipe chases, and helps in controlling contamination spread into clean areas (Regulatory Guide 8.8, Paragraphs C.2.b.2 and C.2.b.3).

The plant ventilation systems are designed so that flow is from lower to higher potential airborne activity areas (Regulatory Guide 8.8, Paragraph C.2.d.1).

12.3.1.3 Equipment Layout

Components belonging to a system that handles radioactivity are generally arranged on the same side, outside of corridors. This is done to minimize the use of shielded pipe chases (Regulatory Guide 8.8, Paragraph C.2.b.6).

Equipment and components that require manual operation, visual inspection or that are expected to need servicing are located in the lowest possible radiation field. For example, a system handling radioactive fluid consisting of a tank, pump, associated valving, sampling lines, and instrumentation is laid out as follows:

The tank, requiring the least servicing, is normally placed in a separate shielded cubicle. Serviceable valving and piping is excluded from this cubicle to the maximum practical extent. The pump and valves, which require maintenance, are placed in a separate cubicle. If practicable or when required, further compartmentalization is achieved by placing the pumps and valves in their own individual cubicles. Sampling lines and instrumentation requiring personnel attendance are brought outside the shield walls to low level radiation zones (II or III), with the exception of inside the Containment. A description of each radiation zone is provided in Section 12.3.2.1. Within the Containment, such instrumentation is located in the lowest possible radiation area (Regulatory Guide 8.8, Paragraphs C.2.b.1, C.2.c.2 and C.2.i.5).

12.3.1.4 Routine Maintenance

Where it is impractical to locate items requiring servicing in low radiation areas, such items are designed so that they can be moved to a low radiation area. Any problems which may occur with rapid removal, local flushing, and decontamination of pieces of equipment (i.e., small pumps, large pumps, etc.) will be reviewed during operation (Regulatory Guide 8.8, Paragraphs C.2.b.9 and C.2.i).

Sufficient clearance is provided within shielding cubicles housing components potentially requiring maintenance and repair (such as valves, pumps, and heat exchangers), so that unimpeded and efficient work on the particular component is allowed. Overly restrictive compartments, while saving space, require longer stays by maintenance or repair personnel. Therefore, overly restrictive compartments are minimized since work would be hampered and inefficient. Furthermore, large compartments ease installation of temporary shielding barriers should they be required. Provisions for a sling, chain and hoist have been made for equipment based on its size, expected frequency of maintenance, and handling problems (Regulatory Guide 8.8, Paragraphs C.2.b.8 and C.2.i).

All instrumentation (flow meters, level gauges) is located in the lowest practical radiation area and is readily accessible to operators. Operators are not required to enter cubicles housing radioactive components to read or activate instruments. The only exception is instrumentation located by necessity within the Containment; and even there, all efforts are made to locate it in the lowest possible radiation field (Regulatory Guide 8.8, Paragraph C.2.c.2).

12.3.1.5 Radioactive Piping

Radioactive piping is either run in shielded pipe chases or within shielded cubicles housing low maintenance equipment, where practical. Radioactive piping is not run in accessways, and the amount of radioactive piping near equipment that requires frequent maintenance has been minimized (Regulatory Guide 8.8, Paragraph C.2.b.6).

To the largest extent possible, but especially for components and piping handling primary coolant, radioactive resins, and concentrated radioactive stream connections are provided for flushing portions of the system. The portion of the system that can be flushed is dictated by the expected frequency of maintenance of the component(s) housed in the shielded cubicle, the size of the cubicle, the number of components housed therein, the geometry of piping, and the valving arrangement (Regulatory Guide 8.8, Paragraphs C.2.f.3 and C.3.h).

The piping used for resin and sludge transfer lines is sized to minimize plugging. Other provisions such as flushing of the lines after resin or sludge transfer also contribute to the minimization of the plugging problem. Oversizing the piping would not significantly decrease the potential for plugging, but would make worse the processing function, and possibly worsen the plugging situation by virtue of the resulting lower flow velocities.

In general, all components and piping within a single cubicle are flushable. Flushing with demineralized water is preferred, since other water or solutions may introduce chemicals which may, if activated, result in further radioactive crud. Flexible hoses from the low drain point to the floor drain are considered acceptable for flushing procedures except for resin carrying lines. These are flushed back to their respective tanks (spent resin and dewatering tanks). If a hose is used to flush, the flushing connection is brought out to a relatively accessible area (Regulatory Guide 8.8, Paragraph C.2.h).

Labyrinths and/or shielding doors are used to eliminate radiation streaming through access openings to the shielded cubicles (Regulatory Guide 8.8, Paragraph C.2.b.4).

12.3.1.6 Pumps and Valves

The extent and degree to which the two following guidelines concerning pumps and valves are used are dependent on the expected radiation levels of any given system. Adherence to these guidelines is more stringent for highly radioactive systems, with progressive relaxation paralleling a lessening in expected activity.

Pumps serving potentially radioactive systems are housed in shielded areas outside cubicles containing other radioactive components. Radioactive piping within the shielded pump cubicle is kept to a minimum (Regulatory Guide 8.8, Paragraph C.2.b.1).

Shielded valve stations for systems handling radioactive fluids are employed, wherever it is feasible, in order to perform valve maintenance without drainage of associated equipment. To further minimize personnel exposure remotely operated valves are utilized where practical and necessary. If manual valves are employed, extension rods through a shield wall to an accessible low radiation area are utilized as necessary. In order to greatly decrease the problems of radiation streaming, the reach rod penetrations are generally offset from the major source of radiation (usually a tank), or are provided with an internal offset (Regulatory Guide 8.8, Paragraphs C.2.c.1 and C.2.i.5).

12.3.1.7 Floor and Equipment Drains

Single traps are not used in floor drainage systems. This subsystem is normally free of any radioactive noble gases; but radioactive halogens are in solution. Transport of radioactive gases from one cubicle to another through floor drainage systems is minimal. A single trap between the drain tanks and the drain header is deemed sufficient to prevent transport of any radioactive volatiles evolving in those tanks (Regulatory Guide 8.8, Paragraph C.2.b.10).

Equipment drains selectively employ a minimum number of traps to prevent transport of radioactive gases from component to component through equipment drainage systems, which could result in noble gas releases when components are maintained or repaired. Typically, the traps are at the equipment sump (Regulatory Guide 8.8, Paragraphs C.2.b.10 and C.2.d.6).

Open sumps are not used as receivers of equipment drains (Regulatory Guide 8.8 Paragraphs C.2.d.6).

12.3.1.8 Crud Control

Equipment and piping that handle radioactive fluids are designed, as described below, in a manner conducive to reducing the retention of radioactive crud and making decontamination easier and more efficient (Regulatory Guide 8.8, Paragraphs C.2.e, C.2.f.3, and C.2.h).

a) Piping runs are sloped whenever possible to prevent accumulation and assist in the removal of radioactive crud deposits.

b) The number of elbows, tees, deadlegs, standpipes, etc., is minimized since these act as crud traps, and also render decontamination difficult.

c) Where elbows are required, large radius elbows are employed if possible with flow moving down through a vertical elbow, rather than up, as the elbow would then act as a crud trap. Flat bending of pipes is used when possible.

d) Orifices are installed in vertical runs as opposed to horizontal ones, when there is an option.

e) Horizontal piping expansion joints are preferentially used. If vertical expansion joints are required, configurations resembling traps (i.e., filled joints) are avoided.

f) For pipes carrying resins, a smooth interior finish is specified, or the pipe is lined with a suitable polymer.

g) Consumable inserts are employed at welds, where this technique is possible. Use of backing rings is minimized.

h) Low leakage valves are employed. ALARA design considerations have included use of low leakage valves with back seats wherever possible. The type of valve and valve arrangement generally used requirements of the system to which the valve belongs.

i) Two and one-half inch and larger valves in the Reactor Coolant System are provided with stem leakoffs to collect leakage and to direct radioactive fluid away from access areas. All valve packing glands have provisions to adjust packing compression to reduce leakage. Valves in highly radioactive systems such as Waste Management Systems are packless diaphragm valves, or are provided with a stem below the seals to reduce leakage (Regulatory Guide 8.8, Paragraph C.2.i.6).

j) Use has been made of radiation resistant seals and gaskets when practical.

k) Vertical valve locations are used wherever possible.

l) Valves are located as close to elbows or bends as possible to avoid the use of valves in the center of long horizontal runs.

Preferential use is made of round-bottomed tanks and vessels to minimize crud buildup in the tank bottom. Effluent process lines are placed as close as possible to the bottom, preferably at the lowest point in these tanks. Drain valves are positioned away from the tank bottom, and preferably located in an accessible area. Operators will not be forced to crouch below tanks to operate any valve, and to receive high exposure from the crud deposited therein (Regulatory Guide 8.8, Paragraph C.2.i.8).

12.3.1.9 Decontamination

Equipment decontamination areas are provided at selected locations in the plant. The choice of these locations is based on:

a) a low radiation area,

b) a location central to equipment which is likely to require decontamination,

c) the capability of routing drain and ventilation lines from the decontamination area to processing systems. Decontamination facilities are provided in the Reactor Auxiliary Building, Fuel Handling Building, and Waste Processing Building. Partial decontamination facilities are also provided in the secondary waste area of the Waste Processing Building.

Each decontamination facility consists of the decontamination area and a storage area, where equipment can be stored prior to decontamination.

Suitable coatings are used on all floor surfaces of areas in the controlled zone. Personnel decontamination facilities and shielded cubicles have surfaces suitably treated for easy decontamination.

The preceding guidelines were derived using experience from prior plant design and from operating plants. In particular, experience was taken from CP&L's H. B. Robinson Plant concerning large spent fuel shipping container decontamination requirements.

12.3.1.10 Access Control

The plant areas are divided into two zones, controlled and uncontrolled. To pass from one of these zones to the other, individuals must go through an access control point (Regulatory Guide 8.8, Paragraph C.2.a).

The controlled zones encompass all areas which either house radioactive equipment, or which can become contaminated during movement of personnel or components. The control point acts as a point at which contamination is detected. All efforts are made to control contamination at its point of origin by posting the extent of contamination and following normal health physics procedures (Regulatory Guide 8.8, Paragraph C.2.a).

Sufficient space is provided at shielded cubicle exits to collect used protective clothing.

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of controls required by paragraph 20.161(a) and (b) of 10 CFR Part 20:

12.3.1.10.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Sources or from any Surface Penetrated by the Radiation

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall:

    1. Possess a radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

    2. Possess a radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

    3. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation

protection personnel responsible for controlling personnel radiation exposure within the area; or

4. Possess a self-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance; or

5. Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device; who is responsible for controlling personnel radiation exposure within the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to entry.

12.3.1.10.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a) Each accessible entryway to such an area shall be conspicuously posted as a locked high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and in addition:

1. All such door and gates keys shall be maintained under the administrative control of the Superintendent-Shift Operations or the Radiation Control Supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

b) Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c) Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d) Each individual or group entering such an area shall:

1. Possess an alarming dosimeter with an appropriate alarm setpoint; or

2. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. Possess a direct-reading dosimeter and be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area and with the means to communicate with and control every individual in the area; or

4. Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device; who is responsible for controlling personnel exposure within the area; or

5. In those cases where the options of sections 12.3.1.10.2.d.2, 12.3.1.10.2.d.3, and 12.3.1.10.2.d.4 above, are impractical or determined to be inconsistent with the "As Low As Reasonably Achievable" principle, possess a radiation monitoring and indicating device.

e) Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f) Such individual areas that are within a larger area where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

Communications outlets are strategically located throughout the plant. Typically, these will be in a low radiation zone near cubicles housing radioactive equipment (especially high radiation areas) or equipment which operates intermittently and which is actuated remotely.

12.3.1.11 Equipment Design Features

Facility design features are incorporated to minimize radiation exposures to personnel who operate, maintain, and inspect systems and components in the Shearon Harris Nuclear Power Plant and to maintain radiation levels ALARA. Examples of additional radiation protection design features are given below:

a) Reactor Vessel - The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the

reinforced areas to assure a smooth transition and pipe branch locations are selected to assure no interference from one branch to the next.

b) Reactor Coolant Pumps - The reactor coolant pump design includes the use of an assembled cartridge seal for the number 3 pump seal, which reduces the time required for replacement. The cartridge seal is also expected to have a useful life double that of previous design. The reactor coolant pump design also includes a spool piece to facilitate separation and replacement of the motor from the pump.

c) Reactor Vessel Insulation - Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal for inspection of the welds. The insulation on the vessel from the bottom of the nozzles down to an elevation approximately 5 ft. below the nozzles is a high efficiency, high temperature insulation. Bonded to this insulation is approximately 3 in. of neutron shielding material. This insulation shielding material is enclosed in stainless steel. The neutron shielding material is used in this area to reduce neutron streaming in compliance with the intent of Regulatory Guide 8.8.

d) Steam Generators - The steam generators incorporate several design features to facilitate maintenance and inspection in reduced radiation fields. The tube ends are designed to be flush with the tube sheet in the steam generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized to facilitate entrance and exit with protective clothing. Handholes to the secondary side are positioned to facilitate maintenance operations. The use of all-volatile treatment (AVT) chemistry on the secondary side serves to increase steam generator reliability and also will reduce occupational radiation exposures.

e) Pressurizer - The pressurizer manways are sized to facilitate entrance and exit with protective clothing.

f) Reactor Coolant Pipe Connections - To minimize crud buildup in branch lines, the piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe whenever possible.

g) Evaporator Package (Boron Recycle System and Liquid Waste Processing System) - The evaporator packages are skid mounted with a 24-in. space for shielding the concentrator tank, gas stripper, and vent condenser from the pumps, valves, and instrument panel, thus reducing the radiation field during operation and maintenance.

h) Heat Exchangers - The heat exchangers are designed such that the shell-to-tube sheet joint need not be broken for inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and/or testing for leaks.

i) Valves - Valves are of the bolted body-to-bonnet forging type. This permits the use of ultrasonic testing in place of radiography for inspection and facilitates assembly and disassembly. This reduces inspection and maintenance time. Additionally, manual valves under 2 in. in diameter are designed for zero stem leakage.

j) Pumps - Pumps (other than the reactor coolant pumps) are designed with flanged connections to facilitate removal for maintenance. Depending on expected conditions, either canned pumps or pumps with high quality mechanical seals are used to reduce leakage and maintenance requirements.

k) Demineralizers - Demineralizer resin screens are constructed for substantially higher than normal pressure differential to accommodate higher than design flows without breakdown.

l) Filters - Filters are designed to be removable from the top with lifting bails in the middle of the head. In addition, they are flushable. Hence, ALARA radiation exposures can be maintained.

m) Materials - Equipment specifications for components exposed to high temperature reactor coolant contain specific limitations on the cobalt impurity content of the base metal as given in Table 12.3.1-1, thereby controlling the potential for production of radioactive Cobalt-60 from the base metal impurity Cobalt-59. The estimated surface area of material in contact with the reactor coolant is given in Table 12.3.1-2. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. (Table 12.3.1-3 shows the estimated total surface area of stellite.) Nickel based alloys (Cobalt-58 is produced from activation of the base metal Nickel-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys is in the Inconel steam generator tubes. The design basis estimate of steam generator surface area in contact with the RCS is given in Table 12.3.1-2 (from Reference 12.3.1-2 [this is the new Radiation Analysis Manual]). This value remains unchanged between the original steam generator estimate and the estimate used for SGR/PUR RCS corrosion evaluations, in accordance with current Westinghouse analysis practices. Neglecting the increase in Inconel surface area associated with the RSG's is not significant relative to maintaining ALARA objectives, since Inconel thermally treated Alloy 690 in the RSG's tubes was selected for, among other reasons, its superior corrosion resistance properties. From Tables 12.3.1-2 and 12.3.1-3, it can be seen that the Inconel surface is the predominant area in contact with the Reactor Coolant System and that the stellite area is minimal. A discussion of material considerations is given in Reference 12.3.1-1.

n) Sampling - The counting room is located in the Waste Processing Building at Elevation 276 ft. It is designed as a radiation zone I with radiation level less than 0.25 mrem/hr. It is shown in Figure 12.3.2-16, along with the laboratory for analysis of chemical and radioactivity samples. Personnel decontamination areas are shown on Figure 12.3.2-15 along with the health physics facilities. These facilities are described in Section 12.5.2. The locations of sampling ports are discussed in Section 9.3.2. In addition, the Radiation Monitoring System (RMS) is described in Section 11.5.2.3. Locations of radiation control panels and instrumentation are given therein. All field mounted process, effluent, and airborne monitors and their associated sample ports are located such that the radiation zone is II or less.

## 12.3.2 SHIELDING[1]

### 12.3.2.1 Design Objectives

The primary design objective of the plant radiation shielding is to protect plant operating personnel and the general public against radiation exposure from the reactor, power conversion, auxiliary, and waste processing systems during normal operation, including anticipated operational occurrences, postulated accident conditions, and maintenance.

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

a) Limit in-plant exposure to radiation of plant personnel, contractors, and authorized site visitors to as far below the limits set forth in 10 CFR 20 as reasonably achievable for normal operation, including anticipated operational occurrences and maintenance, in conformance with Regulatory Guide 8.8 (Revision 3).

b) Shielding was originally designed to limit radiation exposures of control room personnel, in the unlikely event of an accident, to those specified in 10 CFR 50, Appendix A Criterion 19, and to allow habitability of the Control Room. The design also meets the requirements of 10 CFR 50.67 which became applicable as part of the steam generator replacement/power uprate.

c) Limit exposures to the general public offsite from direct and air scattered radiation to a small fraction of the limits set forth in 10 CFR 20 during normal operation and anticipated operational occurrences, and to within the limits specified in 10 CFR 50.67 for postulated accident conditions.

d) Provide barriers for restricting personnel access to high radiation areas and for controlling the spread of contaminants. The plant radiation shielding is also designed to protect certain plant components from excessive radiation damage or activation.

To accomplish this objective, the plant shielding functions to:

1) Reduce neutron activation of equipment, piping, supports, and other materials by the use of suitable shielding around the reactor vessel, designed to minimize neutron streaming into the reactor cavity upper reaches, steam generator subcompartments, and general containment spaces.

2) Limit radiation damage to equipment and materials to below the specific integrated life dose limits.

To meet the above objective, plant shielding is designed to attenuate radiation levels throughout the plant, from direct and scattered neutron and gamma radiation to the dose limits specified in Table 12.3.2-1. The following criteria were used, in part to determine the shielding requirements for pumps and valve galleries:

1. The dose rate in the vicinity (few feet) from the equipment in question.

[1] Further information is contained in the TMI Appendix

2. The annual exposure time anticipated for personnel with respect to the specific equipment. Exposure times were classified according to the modes of activity of the exposed individual relative to the equipment during the exposure period. These were:

   a. function or operation of equipment,

   b. control or surveillance of equipment, and/or

   c. maintenance of equipment.

3. Background radiation dose due to adjacent potentially radioactive equipment.

Average exposure distances were determined for each of the exposure time modes and the resultant dose rate calculated for each distance using the average dose rates prevalent at that distance for that particular exposure mode. Typical exposure times for the various modes of exposures, i.e., operation, maintenance and repair, were obtained from data on similar operations at other nuclear plants, and in particular CP&L's H. B. Robinson 2.

The following guidelines were applicable with respect to shielding requirements of pumps and valve galleries:

1. All pumps and valve galleries involved in the transmission of fluids (liquid and gases) of potential reactor coolant nuclide concentrations were shielded.

2. All pumps and valves involved in the transmission of secondary system fluids, excluding resins and concentrates, were generally not shielded.

3. All pumps and valve galleries involved in the transmission of fluids with the potential for generating significant "crud" build-up because of their function such as spent resin lines, backwash tank lines, concentrator bottom lines, filter backwash transfer lines, and the like were shielded.

4. Penetrations for piping and ducts in shielding walls were designed so as not to be on a direct line with a major radioactive source. Opening for the penetrations were kept as small as possible. Packing of the opening with shielding material was done, when required, to meet exposure criteria (Regulatory Guide 8.8, Paragraph C.2.b.5).

12.3.2.2 Design Description

For shielding design purposes, the plant was divided into radiation access zones, based on the maximum zone dose rate levels listed in Table 12.3.2-1.

A description of each radiation zone chosen for design purposes is given below and shown on Figures 12.3.2-1 through 12.3.2-17. These figures are historical and are maintained in the SAR to document the original design basis. References to 10 CFR 20 listed below refer to "Pre-1993 10 CFR 20" (see Section 12.0). These zone boundaries are not the basis for actual plant posting and occupancy requirements. Health Physics personnel will perform radiological surveys of plant areas on a periodic basis and record the radiation levels for correlation with the various modes of plant operation. Design changes using similar radiation zone descriptions and actual survey results are implemented per engineering procedures.

1. Zone I - This zone has no restriction on occupancy. Such a zone would represent areas in the plant where radiation due to occupancy on a 40 hr/wk, 50 wk/yr basis, will not exceed the whole body dose of 0.5 rem/yr, for personnel in unrestricted areas as specified in Paragraph 20.105 of 10 CFR 20. Most non-employees and visitors to the site will receive considerably less than 0.5 rem/yr because of the relatively short time interval during which they are onsite.

2. Zone II - This zone is a restricted radiation area which can be occupied by plant personnel and authorized visitors on a 40 hr/wk, 50 wk/yr basis without exceeding the allowable whole body dose of 1.25 rem/calendar quarter (10 CFR 20.101).

3. Zone III - This is a restricted radiation area (10 CFR 20.202) that plant personnel can occupy on a limited basis. The average design radiation level in this zone may vary from 2.5 to 5.0 mrem/hr with occasional "hot spots" exceeding the 5 mrem/hr limit. Such areas could be posted with "Caution-Radiation Area" signs.

4. Zone IV - This zone represents a restricted radiation area, which will be posted with "Caution-Radiation Area" signs. The average design radiation level may vary from 5.0 mrem/hr to less than 100 mrem/hr. Occupancy times would be limited. The length of stay in these areas will be determined by the actual radiation level in the area, the past radiation history of the person entering, and the nature of the radiation.

5. Zone V -This zone is a high radiation area. As such it would be posted with "Caution-High Radiation Area" signs and control would be exercised over access to it at all times in accordance with 10 CFR 20.203 and the exception noted in the Technical Specifications. The design radiation level is equal to or in excess of 100 mrem/hr.

12.3.2.3 Methods of Shielding Design

Shield wall thicknesses were determined by using basic shielding data and equations. Data was taken from the table of isotopes, "Reactor Physics Constants, ANL-5800", XDC-59-8-179, and other pertinent texts. Radiation sources were determined as indicated in Section 12.2. The method of calculation normally employed was that of the point kernel integration, outlined in Reference 12.3.2-1 hereafter referred to as Rockwell's method. A computer code, ISOSHLD (Reference 12.3.2-2) was used for some of the actual calculations. This program calculates the decay gamma ray and bremsstrahlung dose rate at the exterior of a shielded radiation source for a number of common geometric arrangements of sources and shields encountered in nuclear power plants. Source geometries that can be used include: point, linear, spherical, truncated conical, disc, cylindrical, and parallel piped sources.

Slab shields were conservatively used for all of the above cases. In addition, special computer codes such as SPAN-4 (Reference 12.3.2-3) and MORSE-CG (Reference 12.3.2-4) were employed.

The correct combination of source and shield was used to approximate the actual configuration in the plant. Tanks and large pipes containing liquid were approximated by cylindrical sources. Gas filled tanks and pipes were simulated by line sources. Where the source shield dose point of geometry was sufficiently complex to preclude use of the Rockwell method or the ISOSHLD program, a point kernel integration, SPAN-4, was utilized. This program calculates the dose

rate at a point from any number of sources having complex geometry and complex shield configurations. The geometry of the sources and shields are described by suitable intersection of quadratic surfaces. ISOSHLD, SPAN-4, and Rockwell's method account for scattering effects in the shields by using appropriate build-up factors. In addition, methods of scattering dose calculations from Radiation Shielding Data (Reference 12.3.2-6) and Schaeffer (Reference 12.3.2-7) were also used.

None of the methods considers the energy degradation (softening) of the radiation as it emerges from the shield, and thus each predicts conservative values of the dose rate at the point of interest. The actual conservatism for selected cases was checked by solving the same problem by a discrete ordinate integration of the transport problem. The computer code used for this purpose is called ANISN (Reference 12.3.2-5).

Whenever scattering effects were expected to be important or dominant such as in the calculation of the neutron and gamma ray streaming inside Containment, the MORSE-CG, Monte Carlo code was used. This code solves the neutron or gamma ray transport problem in arbitrary geometry by following a sufficient number of random "flight paths" of individual particles or rays through the physical system. Importance sampling was used to reduce the number of "histories" which have to be followed, by arbitrarily terminating the "history" of certain rays in regions that were not considered important. Volumetric sources were simulated by a number of point sources.

Comparison of the measured dose rates at operating plants, both BWR and PWR type with corresponding theoretically calculated dose rates, indicated that the models and methods of calculation used predicted higher dose rates than were actually observed. Therefore, shielding calculations based on such models and methods were conservative.

Additional permanent shielding designs are performed using the same general methods described above. Computer codes such as MicroShield, MicroSkyshine, and MCNP may be used to perform these calculations. The use of temporary shielding is described in Section 12.5

12.3.2.4 Compliance with Regulatory Guide 1.69

Regulatory Guide 1.69, Concrete Radiation Shields for Nuclear Power Plants, December, 1973 generally invokes ANSI Standard N101.6-1972, Concrete Radiation Shields, as an acceptable method to the NRC for the design of concrete radiation shields for nuclear power plants. Shearon Harris Nuclear Power Plant complies with the intent of this guide with the following exceptions and clarifications:

| ANSI N101.6-72 Section | Exceptions and Clarifications |
|---|---|
| 4.3.1 | Concrete shielding at SHNPP pertains to gamma and/or neutron shielding only. There are no significant sources of alpha or beta radiation within the plant that could affect concrete shield design. The maximum temperature of the primary shield wall will be 150 F. This wall is designed to afford required shielding at this temperature. |
| 4.3.4 | "The possibility of an explosion in the cell" is not applicable to SHNPP since there are no explosive materials contained within shielded compartments. |
| 4.3.5 | Assumptions and methods used for accident analyses are those given in Chapter 15. These assumptions and methods result in an acceptably conservative design. |

| ANSI N101.6-72 Section | Exceptions and Clarifications |
|---|---|
| 4.3.6 | Regulatory Guide 8.8 is used as guidance in limiting personnel exposure and determining shielding practices. |
| 4.7 | No design drawings will be prepared specifically for formwork. Concrete design drawings are provided in sufficient detail to allow proper design of formwork according to good construction practice. Formwork specifications are provided which require conformance to ACI-347-1968. |
| 4.8 | Not applicable to SHNPP. No heavy aggregates are used. |
| 5.1.2 | Not applicable to SHNPP. No high-density concrete is used. |
| 5.1.3 | Not applicable to SHNPP. No hydrous aggregate is used. |
| 5.1.4 | Not applicable to SHNPP. No boron containing aggregates are used. |
| 5.1.6 | Coatings of clay, silt, gypsum, calcite or caliche on coarse aggregate will total no more than three percent of the total weight of the aggregate. Radiation attentuation calculations take this into account. |
| 5.3.4 | Not applicable to SHNPP. No pozzulans are used. |
| 5.3.5 | Not applicable to SHNPP. No grout fluidifiers are used. |
| 5.4 | For SHNPP, a maximum slump of four inches is permitted for certain applications where less slump is impracticable. |
| 5.4.2 | Not applicable to SHNPP. The preplanned aggregate (PA) method is not used. |
| 5.4.4 | Not applicable to SHNPP. Heavy aggregates are not used. |
| 6.1 | The use of noncombustible or fire retardant formwork for shielding is impractical. Formwork for shielding is consistent with good construction practice and as required by ACI-347-1968. |
| 6.2.1 | ACI-347-1968 is used for the design of formwork. |
| 6.2.2 | Approval of concrete forms prior to construction is per ACI-347-1968. |
| 6.4 | No substitute for detailed thermal stress analysis is made. |
| 6.5 | See position for Section 4.7 with regard to shop drawings. |
| 7.2 | See position on Section 4.7 regarding shop drawings. Any changes in specifications must be reviewed and approved prior to construction activity. The effects of supplemental tracings on shield adequacy are evaluated at that time. |
| 8.1.3 | Not applicable to SHNPP. No high-density concrete is used. |
| 8.1.8 | Aggregate is from one source and is continually sampled throughout the construction phase for conformance to project specifications. Considering these controls bagging and retention of samples is not necessary. |
| 8.2.6 | Vibrators having a speed of 6000 cpm are used. This speed is adequate for producing satisfactory consolidation. Sufficient spare vibrators are maintained but not necessarily one for every two being used. |
| 8.4 | Not applicable to SHNPP. The puddling method is not used. |
| 8.6.1 | The composition and fluidity of the mortar or grout, when used in pressure grouting, is specified in project specifications. |
| 8.6.2 | Filling of forms is done in accordance with good construction practice. Specifications require that no voids will be left in the concrete. |
| 8.7.1 | The only construction joints shown on drawings are those essential to the design of the structure. Therefore, construction joints at other locations do not require approval of the engineer responsible for the design. |

| ANSI N101.6-72 Section | Exceptions and Clarifications |
|---|---|
| | Construction joints are not stepped (i.e., not provided with offsets to prevent radiation streaming). Streaming between joints is not considered to be a problem since sufficient amplitude between joints is provided. |
| 8.7.2 | Concrete is cured for the specified times. The requirements of ACI-347-1968 are not met regarding time limits for removing forms. |
| 8.7.5 | Patching and finishing is performed, as soon as practicable to ensure a quality product; however, not necessarily within the specified times. |
| 8.7.6 | Traffic or other operations is restricted after curing and finishing to prevent damage to the concrete but not necessarily for the time specified. |
| 9.1 | Only certain areas subject to contamination by radioactive substances have a protective coating. ANSI N101.4-72 and ANSI N12-72 are utilized for such applications within SHNPP. |
| 10.1.2 | Dimensional tolerances for hatches and openings as specified in ACI-347-1968 are used rather than those given in Table 1. Minimum practicable joint clearances are specified. |
| 10.1.3 | Not applicable to SHNPP. Service trenches are not used. |
| 10.2.2 | The weight of each block is clearly marked on the block; however, not necessarily by stenciling. |
| 10.2.3 | Blocks are cured according to good construction practice but not necessarily in the absence of direct sunlight or heat. This sunlight or heat, however, does not result in the loss of shielding efficiency. |
| 10.3.1 | There are no present plans for penetrations through shielding plugs. However, if they are required, streaming is prevented by proper design of the penetration. |
| 10.6 | All pre-cast shielding components are fabricated at the site. If, however, offsite fabrication is used, pre-cast shielding components are not necessarily protected from direct sunlight or high temperatures during transit or storage. This exposure is not expected to result in loss or shielding efficiency. |
| 11.5.1 | Preoperational tests of shielding are not performed. Normal post-operational tests identify any areas where additional precautions or shielding are necessary. |
| 11.5.2 | Not applicable to SHNPP. Containment leak testing is performed in accordance with 10 CFR 50, Appendix J as discussed in Section 6.2.6. |

### 12.3.2.5 Description of Plant Shielding

Plant layouts and cross sections of buildings containing process equipment for treatment of radioactive fluids are shown on Figures 1.2.2-3 through 1.2.2-85. These figures indicate appropriate shield walls, floors, ceilings, and hatches. A plot plan is shown on Figure 1.2.2-2.

### 12.3.2.6 Primary Shield

The primary shield consists of reinforced concrete, which surrounds the reactor vessel. The primary shield is designed to meet the following objectives:

1. to attenuate the core neutron flux to limit the activation of component and structural materials,

2. to limit the radiation level after shutdown to permit access to the reactor coolant system equipment and keep radiation exposures to personnel working in the area ALARA,

3. to reduce, in conjunction with the secondary shield and the neutron streaming shield, the radiation level from sources within the reactor vessel to allow limited access to the Containment during normal operation, and

4. to permit access during shutdown for inspections required by ASME, Section XI.

For purposes of primary shielding design, the normal full power operation of the core and resultant neutron and gamma fluxes are the controlling factors.

12.3.2.7 Secondary Shield

The secondary shield surrounds the primary shield and reactor coolant loops and attenuates, to a safe level, the radiation given off by the reactor coolant and steam generators. During full power operation, the major radiation source (which is the design basis of the shield) in the Reactor Coolant System is N-16, which is created by neutron activation of oxygen during passage of reactor coolant through the core.

The secondary shield is designed to permit limited access to certain areas outside the secondary shield wall within the Containment Building during full power operation. The secondary shield wall also serves to reduce the full power radiation levels outside the Containment Building so that normal continuous occupancy outside the Containment Building is afforded. In addition, the secondary shield helps in reducing the radiation intensity outside the Containment Building in the unlikely event of an accidental release of fission products into the Containment Building. This function, is however, primarily accomplished by the Concrete Containment Structure.

After reactor shutdown, the fission and corrosion product activities in the Reactor Coolant System, which are given in Section 11.1, become the dominant radiation sources.

12.3.2.8 Fuel Transfer Shield

The fuel transfer shield protects plant personnel from fission product gamma radiation emitted from the spent fuel assemblies during core refueling operations. The fuel assemblies are removed from the reactor core through a fuel transfer canal to a water-filled spent fuel storage area located in the Fuel Handling Building. After sufficient decay, the spent fuel may be transferred under water to spent fuel shipping containers.

The fuel transfer shielding consists of the water in the refueling cavity, the water over the spent fuel storage area, the walls of the refueling cavity, the fuel transfer canal, the shield box around the transfer tube in the Containment between outside of the refueling cavity wall and the concrete containment structure, and the shield box around the transfer tube in the Reactor Auxiliary Building space between the Containment Building and the Fuel Handling Building. The design basis shielding source terms are given in Table 12.2.1-2.

The fuel transfer tube bellows inspection area inside containment is inaccessible due to being backfilled with mortared concrete block shielding. Removal of these blocks is controlled by the plant modification program which will ensure that they are not removed during fuel movements.

Shielding provided for direct gamma radiation and streaming pathways from the fuel transfer tube in Containment and Fuel Handling Buildings and the space between the buildings is shown

on Figure 12.3.2-18. In addition, procedural controls are put in place during periods of fuel movements to further restrict personnel accessibility to these areas.

12.3.2.9 Containment Building

The Containment Building is a reinforced concrete structure with a cylindrical wall 4.5 ft. thick and a 2.5 ft. thick dome. In conjunction with the primary and secondary shields, it limits the radiation level outside the Containment Building from all sources within the Containment to no more than 0.25 mrem/hr at full power operation.

The Containment Building provides protection to plant personnel from radiation sources inside the Containment following a design basis accident. These radiation sources are discussed in Section 12.2.1.9. The Containment structure will act to greatly attenuate the direct offsite gamma dose following a design basis accident.

12.3.2.10 Reactor Auxiliary Building

Reactor auxiliary building shielding includes concrete walls, covers and removable blocks which shield the sources of radiation originating from the Chemical and Volume Control System, Boron Recycle System, Process Sampling System and portions of the Waste Processing System. Typical components that are shielded include recycle holdup tanks, demineralizers, filters, heat exchangers, and associated piping. For design purposes, radioactivity levels in these systems were based upon normal system operation with clad defects in the fuel rods which generate one percent of rated core thermal power. These levels are specified in Section 12.2.1.7.

12.3.2.11 Waste Processing Building

Waste processing building shielding includes concrete walls, covers, removable blocks, and shield doors which shield sources originating from the Solid, Liquid, and Gaseous Waste Processing Systems assuming clad defects in the fuel rods which generate one percent of rated core thermal power and normal system operations as described in Sections 11.1, 11.2.2, 11.3.2, and 11.4.2. Typical components that are shielded include waste holdup tanks, waste gas compressors, catalytic hydrogen recombiners, gas decay tanks, evaporators, spent resin storage tanks, demineralizers, filters and the associated piping.

Dry low-level wastes are typically shipped to a vendor for processing, but can be processed on site. Remote operation is unnecessary due to low radioactivity levels associated with this waste.

12.3.2.12 Turbine Building and Tank Area

Turbine building and tank building shielding includes concrete walls, covers, and removable blocks that shield sources originating from the Steam Generator Blowdown System and portions of the Waste Processing System. Typical components that are shielded include settling tank, blowdown heat recovery components, condensate polishing demineralizers, and waste monitor tanks. Radioactivity levels in these systems were based upon normal system operation with clad defects in the fuel rods that generate one percent of rated core thermal power and primary to secondary leakage rate of 100 lbm per day.

12.3.2.13 Fuel Handling Building

Shielding is provided for protection during all phases of spent fuel removal and storage. Operations requiring shielding of personnel are spent fuel removal from the reactor, spent fuel transfer through the reactor cavity and transfer tube, spent fuel movement in the transfer canal, fuel placement in the storage pools, spent fuel shipping cask loading and unloading prior to or after transportation, and maintenance and inspection of the Spent Fuel Cooling and Cleanup System.

Since all spent fuel removal and transfer operations are carried out under borated water, a minimum water depth above the top of the fuel assemblies is established to provide radiation shielding. The dose rate at the water surface from spent fuel sources is less than 2.5 mrem/hr. Dose rates at the water surface from other sources, such as activation and fission products suspended in the water, may exceed 2.5 mrem/hr. The concrete walls of the fuel transfer canal and spent fuel pool, supplement the water shielding and limit the maximum continuous radiation dose levels in working areas to less than 2.5 mrem/hr from spent fuel sources.

The refueling water and concrete walls also provide shielding from activated Rod Cluster Control Assemblies (RCCA) and reactor internals that may be removed at refueling times. Although dose rates will generally be less than 2.5 mrem/hr in working areas, certain manipulation of fuel assemblies, RCCAs, or reactor internals may produce areas where dose rates exceed 2.5 mrem/hr for short periods. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.

The spent fuel pool shielding is based upon the following considerations:

1. The controlling factor in the design of the spent fuel pool and fuel transfer canal walls is the irradiated fuel assemblies (see Table 12.2.1-2).

2. The shielding design of the fuel transfer canal is based upon consideration of an average spent fuel assembly with an irradiation time of 3.1 years. Activity decay is taken for the minimum waiting period of three days following shutdown.

3. The shielding design of the fuel pools is based upon the loadings described in Section 9.1.

Shielding for the Spent Fuel Pool Cooling and Cleanup System is based upon source terms derived from normal system operation as described in Sections 9.1.3 and 11.1 in conjunction with approximately two-thirds of a core stored in the pool and fuel clad defects in the fuel rods that generate one percent of rated core thermal power. Refer to Sections 9.1.1 and 9.1.2 for a discussion on the capacity of the fuel pools.

12.3.2.14 Control Room

For the purpose of designing control room shielding, the radioactivity releases from the maximum loss-of-coolant accident (LOCA) are controlling. The two sources considered in designing the shielding were:

1. Direct gamma radiation from the containment atmosphere and from radioactivity collected on emergency filters.

Regulatory Guide 1.183 (Reference 12.3.2-8) source terms, and release fractions are considered. Credit is taken for iodine removal from the atmosphere by sprays. Credit for post-accident decay is considered.

The total direct whole body dose to control room personnel following a LOCA is computed to be less than 1.0 rem in 30 days. Credit for concrete shielding was taken (4.5 ft. for the containment shield wall, 3 ft. for the Control Room shield wall and 2 ft. for the Control Room roof shielding). A minimum shield thickness of 4.5 ft. separates the emergency filters from the Control Room.

In addition to shielding from external exposures, the Control Room is designed to operate in an isolated mode under accident conditions in order to minimize the quantity of airborne radioactivity which enters the Control Room and thereby ensure compliance with GDC 19 of 10 CFR 50 Appendix A and 10 CFR 50.67. A detailed description of the system design is provided in Section 9.4.1. Chapter 15 includes an evaluation of the exposures to control room personnel following the design basis accident.

2. Direct gamma radiation from radiation leakage external to containment

In addition to direct shine exposure from airborne radioactivity in the Containment following a design basis loss-of-coolant accident, control room personnel may receive a small exposure due to external exposure to the passage of the gaseous plume which could result from containment leakage. The integrated exposure (30 days) to control room personnel from this source of exposure is included in the evaluated result reported above.

12.3.2.15 Neutron Streaming Shield

The neutron streaming shield is designed to reduce radiation exposures, due to streaming radiation through the cavity around the reactor pressure vessel, in areas of Containment where occupancy may be required during normal operation. It is divided into two parts. Both parts are removable. One part of the shield is a cylindrical shell placed between Elevations 246.55 ft and 251.2 ft. It consists of 3 one-inch layers of ricorad clad in stainless steel in contact with 1 1/2-inch-thick microthermal insulation, which replaces mirror insulation between these elevations. Ricorad contains 13.3 weight-percent hydrogen and has excellent neutron shielding characteristics. The other part of the shield is placed on the top of billets at Elevation 250.69 ft. It consists of six-inch-thick blocks of silicone elastomer shaped to fit spaces on the top of the billets.

The peak neutron leakage fluxes in the core mid-plane outside the reactor pressure vessel used in MORSE-CG calculations are shown in Table 12.3.2-2.

The potential for radiation streaming (neutron and gamma) through the annulus around the reactor pressure vessel has been analyzed to determine the prevalent radiation fields during normal operation in areas of containment that may require occupancy.

Since operating experience (Reference 12.3.2-18, 12.3.2-19, and 12.3.2-20) indicated that streaming gamma dose rates during normal operation are a relatively small fraction of the

corresponding neutron dose rates, the analysis of the streaming dose rates in containment has been limited to the determination of neutron dose rates. The angular neutron flux as a function of energy emerging from the surface of the reactor vessel has been calculated using the DOT III (Reference 12.3.2-16) discrete ordinates code utilizing an $S_8$ angular quadrature and a $P_3$ Legendre expansion for anisotropic scattering.

The vessel emergent angular fluxes have been used as input to a Monte Carlo analysis of the streaming neutrons. Morse CG (Reference 12.3.2-4), a general purpose Monte Carlo multi-group neutron and gamma ray transport code with combinatorial geometry, has been used to compute the neutron streaming and the resultant dose rates on the operating deck and at various other locations inside containment.

The cross section library used in the calculations is based on DLC-23 or CASK library (Reference 12.3.2-15). This library is a coupled neutron and gamma ray library and the data in the library are obtained by collapsing cross sections over a PWR core spectrum.

The three dimensional geometrical model used for the dose rate calculations includes the reactor vessel steel shell, containment structural concrete, major features of the containment internal structures such as the refueling cavity, the shield walls around the steam generators and pumps, a detailed description of the cavity around the reactor vessel, the primary piping, the missile shield, and the primary shield. In order to determine the effectiveness of the shielding for reducing the dose rates due to neutron streaming on the operating deck, calculations were performed with and without the shield in place. The effective factor of reduction for gamma and neutrons of the installed shield is 5.7.

A check on the accuracy of the modeled geometrical representation of the reactor cavity and its surroundings was performed by comparing the computer generated pictures of the model and general arrangement drawings at various elevations and sections.

The neutron histories started on the surface of the reactor pressure vessel with energies and directions determined by processing the angular fluxes at the outermost mesh of the vessel determined by the DOT III (Reference 12.3.2-16) calculations with the DOMINO code (Reference 12.3.2-17), which is explicitly set up to provide the proper source information for the MORSE Program. Variation in source strength along the circumference has been neglected for conservatism.

The MORSE calculation was performed in two stages using a MORSE to MORSE coupling technique. The first stage stops the random walk either at the vessel flange level or just above the neutron shield. In the second stage particles, which escaped at the flange or above the neutron shield were recorded on tapes that were used as inputs to MORSE runs using the entire three-dimensional geometrical model.

No biasing technique was used for either stage of the calculations. It is known from experience at operating plants that the dose rates in containment are due primarily to fast neutrons. The Monte Carlo calculations of the dose rates in containment were limited to the energy groups of fast neutrons emerging from the reactor vessel with energies above 0.11 MeV. However, in order to be conservative, the response to the energy groups considered was followed down to epithermal energies. A thirty percent uncertainty was assumed in using the Monte Carlo calculational techniques.

Table 12.3.2-9 lists the estimated neutron dose rates at selected locations within containment. While the computed neutron dose rates on the containment operating floor are relatively high, the dose rates in general areas of lower elevations, where personnel may require access during normal operation, are expected to be much lower due to the shielding afforded by various internal components and structures. The computed values of the dose rates at lower elevations are less reliable due to difficulty in modeling the problem, considering the more complex geometry, which causes the large uncertainty in the results.

A reasonable estimate of the expected dose rates was made by comparing the dose rates measured at several operating plants at lower elevations with the corresponding measured dose rates at operating floors. A ratio of these dose rates in conjunction with its predicted dose rates at the operating floor was used to estimate the anticipated dose rates at lower elevations of SHNPP.

Measurements conducted at PWR Plants such as Calvert Cliffs (Reference 12.3.2-18), St. Lucie 1 (Reference 12.3.2-19) and Millstone 2 (Reference 12.3.2-20) scaled to full power indicate that neutron dose rates at locations where predicted SHNPP does rates are approximately 8.7 rem/hr, range from 60 to 65 rem/hr. Dose rates on intermediate elevations in these plants range from 75 to 900 mrem/hr, while dose rates at lower elevations are in the range of 15 to 250 mrem/hr.

Expected neutron dose rates for SHNPP at lower elevations should, therefore, be about one seventh of those of the referenced operating plants and can thus range from 10 to 130 mrem/hr at intermediate floors and from 2 to 36 mrem/hr at lower floors. Streaming gamma dose rates at the above referenced operating plants were measured to be one fourth or less of the neutron dose rates for general containment areas. A similar ratio is expected for Shearon Harris.

These considerations indicate that the neutron streaming shields at Shearon Harris Nuclear Power Plant will provide sufficient shielding to achieve ALARA exposure to occupants in containment while the reactor is at power.

### 12.3.2.16 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In Post-Accident Operations

In accordance with Item II.B.2 of NUREG-0737 (Reference 12.3.2-14), a radiation and shielding design review of areas outside the SHNPP Containment Building that could become highly radioactive following a severe accident has been conducted. The postulated accident used as the basis for calculating the source terms included a loss-of-coolant accident (LOCA) with an uncovering, subsequent damage, and release of fission products from the core. Radionuclide releases from the core inventory used in calculating the source terms were as suggested by Regulatory Guides 1.3, 1.4, 1.7, and SRP 15.6.5.

Using these calculated source terms, an analysis was made of the liquid and air systems that would be operated following the postulated accident to cool down the reactor and bring the plant to a safe condition. The Safety Injection, Containment Spray, Residual Heat Removal and Process Sampling Systems would become highly radioactive. Areas containing equipment and piping from these systems have been identified. Calculated dose rates in these areas indicate plant operators may not be able to enter some of these areas post-accident without additional shielding and/or other radiation protection measures using reduced time in and increased distance from these areas. Areas that do not require additional radiation protection measures,

such as the Control Room, which must be occupied during and after the accident, were also identified.

Alternate methods for providing required radiation protection, using time, and/or distance to minimize operator exposures during and following the accident, have been identified. These methods have been analyzed in detail as part of an ongoing design review.

The overall results of the shielding review, however, were that vital areas of the plant requiring occupancy or access to mitigate the postulated accident are accessible for performing the necessary post-accident operations without overexposing an individual.

The FSAR Section 6.3.2.8 lists all the actions that are necessary for the proper operation of the ECCS. This was taken into account in calculations of the radiation exposures in vital areas of the power plant.

In order to determine the radiation levels in areas where access may be necessary after the accident, a number of calculations were performed. Control Room doses at various times after the accident were calculated using a uniform containment air source and Rockwell Technique (Reference 12.3.2-10).

In the Reactor Auxiliary Building, doses were calculated at all elevations at typical points, taking into account the runs of radioactive pipes such as RHR, SI, and containment spray at each elevation, along with the other radiation sources. In addition, radiation doses due to radioactive piping within the shielded pipe tunnel were also estimated at Elevations 236 ft and 261 ft in the RAB. These calculations were performed using Rockwell Technique and ISOSHLD Code (Reference 12.3.2-11). Contact doses for equipment such as the residual heat removal heat exchangers, the volume control tank, and the charging pumps were determined at various time intervals after the accident. In the case of the residual heat removal heat exchangers, the upper bound was determined using the reactor coolant source terms and the lower bound using the diluted coolant activity source terms.

12.3.2.16.1 Analysis of dose rates at one hour after the accident

In the Reactor Auxiliary Building, at Elevation 190 ft, the radiation levels one hour after the accident are based on the assumption that the recirculation phase of the ECCS begins sometime between 30 minutes and 47 minutes. The RHR pump is located in the vicinity of the containment spray pump. Since these pumps operate continuously, the radiation level in their vicinity is on the order of 500 R/hr at one hour after the accident. The dose rate in stairwell areas at this time is on the order of 5-10 R/hr. The radiation level further away from the pumps is on the order of 10 R/hr. At Elevation 216 ft, the radiation levels are lower because the only source of direct radiation is the Containment Building.

At Elevation 236 ft, the radiation levels are much higher due to the presence of the RHR heat exchangers and their associated piping. Sampling Room 1A shares a common shielding wall with the B RHR heat exchanger and hence is not suitable for post-accident use without additional measures to provide radiation protection. Sampling Room 1B is situated in a better location with respect to high sources of radiation after the accident and its potential use for post-accident sampling was evaluated and found to be inconvenient for carrying samples to the radiochemistry laboratory. The Post-Accident Sampling System as described in Section 9.3.2.2.3 of this FSAR is also situated at the same elevation and is shielded. The control panel

is located outside the shielded cubicle in a radiation field of 20 - 50 mR/hr. It is convenient to carry samples from this location to the radiochemistry laboratory. The Post-Accident Hydrogen Monitoring System as described in Section 6.2.5 is located at the same elevation. The Sample Dilution Panel is located in the vicinity of the Post-Accident Sampling System. Frequent access to these areas is possible after an accident.

At Elevation 261 ft, dose rates in the vicinity of the volume control tank and RHR heat exchangers can be 100-700 R/hr. In the other areas, the radiation levels are much less.

At Elevation 286 ft., the areas directly on the top of the volume control tank can be as high as 70 R/hr with a reduction in the dose rate to about 1-3 R/hr in Switchgear Room B. The auxiliary control room area is accessible after one hour where the dose rate can be 50 mR/hr. In Switchgear Room A, the dose rate can vary between 10 mR/hr to 100 mR/hr. and is accessible after one hour for infrequent use.

At Elevation 305 ft, dose rates in the Control Room are 10-15 mR/hr. at some hot spots after one hour. The guidelines of GDC-19 and 10 CFR 50.67 are satisfied during the course of the accident for continuous occupancy for thirty days.

The radiation levels in other buildings have also been reviewed. The maximum expected dose rate in the Security Building after one hour is on the order of 25 mR/hr. In the Turbine Building, the ground floor and mezzanine floor are well shielded. Hence, the radiation levels are less than 0.25 mR/hr, and access is possible at all times.

The Diesel Generator Building is located away from the containment structure. The only source of radiation is through the dome of the containment structure. The radiation level due to this source is calculated as <20 mR/hr after one hour.

In the Waste Processing Building, the maximum radiation level after one hour in the vicinity of the containment structure is 8 R/hr. The radiochemistry laboratory is situated in this building. Shielding walls and structural walls, amounting to many feet of concrete, separate the laboratory from the containment structure. Hence, continuous access to laboratory facilities is possible after the accident.

The Control Room, the Health Physics offices, the Security Secondary Alarm Station, and the Technical Support Center have been designated as continuous occupancy areas.

12.3.2.16.2 Access to vital areas

The radiation level in areas requiring continuous occupancy are less than 15 mR/hr (averaged over 30 days). These areas have been defined above. Exposures in these areas will contribute to an integrated dose of less than 5 rem for the duration of the accident. The criteria of GDC 19 and 10 CFR 50.67 are satisfied.

Postulated post-accident assumptions and source terms are described in FSAR Section 15.6.5.4.4 Atmospheric dispersion factors used in the Technical Support Center dose calculations were developed using the methodology described by NUREG/CR-6331 (1997) and by Murphy Campe (Reference 12.3.2-21).

Areas requiring possible frequent access have radiation levels between 15 mR/hr and less than 100 mR/hr. Examples of such areas are control panels that are located outside the cubicles housing radioactive sources.

Areas requiring infrequent access are those with radiation levels greater than 100 mR/hr. The direct dose due to airborne activity in the containment atmosphere outside the cylindrical structural wall is estimated to be 8 R/hr. High radiation doses indicate contributions from highly radioactive sources in these areas. Since it is extremely difficult to separate the radiation zones in the absence of the shield walls, a range of radiation levels exist.

The overall result of the analysis is that vital areas required to be accessible to place the plant in a stable shutdown condition following the accident are accessible at the time required for entry. With the exception of closing the Charging Pump Suction and Discharge Header Crossover MOV breakers as discussed in section 6.3.1 and table 6.3.2-6, all operations that are required to mitigate the accident and place the plant in RHR recirculation mode can be performed from the Control Room. Because the Charging Pump Suction and Discharge Header Crossover MOV breakers are closed during the injection mode when area dose rates are not significantly elevated, the operator dose received to complete this action is minimal.

Dose rate zone maps, which identify dose rates in vital areas, are shown on Figures 12.3A-1 through 12.3A-21. Occupancy requirements for various areas following an accident are summarized in Table 12.3.2-3. The dose rates for various time references (one hour, one day and one month) were calculated using source terms discussed in FSAR Section 12.2.1.12 and subsection 12.3.2.16. Sources of radiation were airborne and coolant activity in the containment building and the coolant activity in the RHR, containment spray and safety injection systems outside containment.

12.3.2.16.3 Post-accident equipment doses

Integrated radiation doses have been estimated for equipment in various buildings for a period of one year following an accident. These estimates are based on the source terms given in Tables 12.2.1-25 through 12.2.1-28 and the radiation sources present at different elevations in the buildings.

These integrated doses have been evaluated using the assumptions stated in NUREG-0588 (Reference 12.3.2-12), resolutions stated in NUREG-0588 Rev. 1 (Reference 12.3.2-13), and general guidance given in NUREG-0737 (Reference 12.3.2-14). These are incorporated in Appendix 3.11A and Appendix 3.11B.

12.3.3 VENTILATION

The plant ventilation systems are designed to provide a suitable environment for personnel and equipment during normal plant operation and to provide a safe environment for operating personnel and the public during design basis accident conditions when controlling the plant to a safe shutdown condition.

12.3.3.1 Design Objectives

The plant ventilation systems are designed to accomplish the following objectives:

1. To provide suitable thermal environment and air quality to ensure personnel comfort, health and safety, and proper equipment operation and integrity.

2. To direct air from areas of lower potential radioactivity to areas of higher potential radioactivity.

3. To reduce internal and external radiation exposure from airborne radioactivity to as low as reasonable achievable (ALARA) levels. Guidelines contained in Regulatory Guide 8.8 have been applied.

4. To ensure that maximum airborne radioactivity concentrations in restricted and unrestricted areas, during normal operation (including anticipated operational occurrences) with a design basis fuel defect of one percent, are maintained within the limits of 10 CFR 20, Appendix B, Table 1, and Table 2 respectively. These areas are within the plant structures and the unrestricted and restricted areas on the plant site. The maximum airborne radioactivity levels correspond to those that could result from the design basis reactor coolant leakage. While it is possible to control airborne radioactivity concentrations to remain within 10 CFR 20 limits, it is also possible to exceed the dose limits in certain areas due to contained sources (as opposed to airborne sources).

5. To provide a suitable environment for control room operators to remain in the Control Room to take effective actions to operate the plant safely under normal conditions and maintain the facility in a safe condition following a postulated accident as required by GDC-19 and 10 CFR 50.67.

6. To maintain the site boundary dose within the guidelines of 10 CFR 50.67 following postulated design basis accidents.

To meet the required design objectives for the plant ventilation systems, the following design guidelines were used:

1. Air movement patterns are designed to provide airflow from areas of least radioactive contamination to areas of progressively greater radioactive contamination.

2. Slightly negative pressures are maintained, where applicable, to prevent the uncontrolled escape of contaminated air to the environment. The Control Room is maintained at a slightly positive pressure during all modes of plant operation to prevent infiltration of potential contaminants from adjacent areas. The Control Room Habitability System is discussed in Section 6.4.

3. Isolation dampers are provided for the ventilation systems to allow servicing of redundant equipment without any interruption of system operation.

4. Exhaust air from the ventilation systems is released through the plant vent stack to prevent contamination of ventilation air intakes and to enable an overall assessment of the released radioactivity.

5. High-efficiency particulate air (HEPA) filter and charcoal adsorbers are provided where necessary to reduce airborne radioactivity levels of the exhausted ventilation air.

6. Volumetric flow rate of each filtration unit is 49,333 acfm. This flow rate is required for system balance while maintaining testability.

7. Air handling units and filtration units are provided with adequate space around the units to allow servicing and replacement of sections.

8. The air filtration units are designed to permit in-place testing to confirm their radioactivity removal capabilities. Built-in features are provided for the units to be tested in accordance with ANSI/ASME N510-1980 with the exception that representative samples of the system adsorbent are tested per ASTM D3803-1989.

9. The following provisions are made to minimize the spread of contamination.

    a. Equipment vents and drains are piped directly to the collection device connected to the collection system thus preventing the spread of contamination.

    b. All-welded construction ductwork sections and all-welded piping systems are employed on contaminated systems to reduce system leakage to a minimum acceptable level.

10. Whenever practicable, equipment containing radioactive fluids is located in areas or compartments which are separated from clean areas.

11. For areas having a potential for airborne contamination, airflow rates are selected to ensure that the radioactivity concentration is ALARA and does not exceed the limits indicated in 10 CFR 20 for normally occupied areas.

12.3.3.2 Design Description

The air conditioning, heating, and ventilation systems for all plant buildings are described in Sections 9.4, 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5. The Engineered Safety Feature (ESF) Filter Systems are discussed in 6.5.1. The Control Room Habitability System is discussed in Section 6.4 and the fission product control systems are discussed in Section 6.5.3.

12.3.3.3 Air Cleaning System Design

Air cleaning systems are either safety related, for fission product removal systems that operate following a design basis accident, or non-safety related systems which control airborne radioactivity in normally occupied areas during normal operation. The central exhaust system of the Reactor Auxiliary Building Normal Ventilation System is an illustrative example of a non-safety related air cleaning system that functions during normal operation (refer to Section 9.4.3).

An example layout of the RAB central exhaust system housing showing exhaust fans, filter mountings, access doors, access sections and isolation dampers is provided on Figures 12.3.3-1 and 12.3.3-2.

Periodic inspection/testing for filters and adsorbers will be performed after initial operation. The frequency of changeout of filters and adsorbers will be determined accordingly.

12.3.3.4 Ventilation Systems Compliance to Regulatory Guides

The ESF ventilation systems with atmosphere cleanup features meet the intent of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post-accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 2". SHNPP compliance to Regulatory Guide 1.52 is as described in Sections 1.8 and 6.5.1.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The Radiation Monitoring System (RMS) consists of the following:

1. Area Radiation Monitoring System

2. Airborne Radiation Monitoring System

3. Process and Effluent Radiological Monitoring and Sampling System

The Process and Effluent Radiological Monitoring and Sampling System is discussed in Section 11.5. The (RMS) block diagram is shown on Figure 11.5.2-1 and is discussed in Section 11.5.2.3.

The Area Radiation Monitoring System informs operations personnel, both locally and in the Control Room, the WPB Control Room, the Computer Room, and the Health Physics Office, of radiation levels in areas where area radiation monitoring system detectors are located, provides warning when abnormal radiation levels occur in specific plant areas, and warns of possible equipment malfunctions. Some channels of the Area Radiation Monitoring System are designed to Class 1E requirements and can withstand Loss of Coolant Accident (LOCA) environmental conditions.

The Airborne Radiation Monitoring System provides information, both locally and in the Control Room, the WPB Control Room, the Computer Room, and the Health Physics Office, for the purpose of maintaining low in-plant personnel radiation exposure in accordance with 10 CFR 20, Regulatory Guides 1.21 and 1.97 (see Section 1.8), 8.2 and 8.8 (see Section 12.1), and ANSI N13.1-1969.

The design of the fuel pool racks (see Section 9.1.1) precludes criticality under all postulated normal and accident conditions and meets the requirements of 10 CFR 50.68. Therefore, criticality monitors are not required.

12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design objectives

The objectives of the Area Radiation Monitoring System during normal operating plant conditions and anticipated operational occurrences are:

1. to furnish records of radiation levels in specific areas of the plant,

2. to warn of uncontrolled or inadvertent movement of radioactive material in the plant,

3. to provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area,

4. to annunciate and warn of possible equipment malfunctions and leaks in specific areas of the plant, and

5. to furnish information for making radiation surveys.

By meeting the above objectives, the Area Radiation Monitoring System will aid in keeping radiation exposures ALARA.

The objectives of the Area Radiation Monitoring System during postulated accidents are:

a) provide the capability to alarm and initiate a containment isolation signal in the unlikely event of a LOCA or abnormally high radiation inside the Containment,

b) provide long-term post-LOCA monitoring of conditions inside the Containment and in vital access areas outside Containment, and

c) provide a signal to isolate the Fuel-Handling Building (FHB) and start the FHB emergency ventilation system in the event of a fuel handling accident.

12.3.4.1.2 Criteria for Location of Area Monitors

Considerations for area monitor locations are based on the following:

a) frequency and length of personnel occupancy in specific areas where the potential for radiation exposure exists,

b) potential for personnel unknowingly to receive high radiation doses,

c) potential for equipment malfunction,

d) areas where during normal plant operation including refueling, radiation exposures could exceed the radiation limits due to system failure or personnel error,

e) areas where new and spent fuel is received and stored,

f) containment area for indicating the level of radioactivity and detecting the presence of fission products due to an Reactor Coolant Pressure Boundary leak,

g) normally or potentially radioactive release points, and

h) necessity for post-accident access, into areas where radiation exposures are expected to be significant following a design basis accident.

12.3.4.1.3 General System Description

The Area Radiation Monitoring System contains monitors that are located at strategically selected locations throughout the plant to detect and store information on the radiation levels and, if necessary, annunciate abnormal radiation conditions. These monitors are an integral part of the RMS, which is described in detail in Section 11.5.2.3.

The Area Radiation Monitoring System consists of individual, locally mounted area monitors. Each monitor is composed of the requisite number of channels, with a channel consisting of a detector and check source. The detector assembly consists of either a gamma-sensitive GM tube or an ion chamber. All channels associated with a monitor are served by a local, dedicated microprocessor, and channel information is processed through the microprocessor which is then interrogated by the appropriate RMS computer for further processing, indication on a cathode ray tube (CRT), storage, remote alarming and hard copy production, if so desired.

12.3.4.1.4 Microprocessor

The microprocessor is described in Section 11.5.2.4.5, with the following exception: The microprocessor is designed for 1000 rads integrated dose, therefore, in areas where this dose can exceed 1000 rads during a period of 10-20 years of normal operations, the microprocessor is removed to a more suitable area. For those monitors that serve during and after an accident, the microprocessor is located in an area that is not subjected to the expected high radiation levels.

12.3.4.1.5 Local annunciation

Area monitors have local annunciation, consisting of an audible alarm rated at 80 db at 10 ft and three indicating lights for high radiation, alert radiation and normal operation. Area monitors have a display at the microprocessor indicating the radiation level in mR/hr at the detector. For those monitors which are located in high, or potentially high, normal operation radiation fields, the more sensitive microprocessor is located as described above and a local annunciator is provided near the detector in lieu of the annunciation normally provided by the microprocessor. These displays are a duplicate of the displays available at the microprocessor.

12.3.4.1.6 Power supplies

Each channel is provided with an independent power supply, designed such that a failure in that channel does not affect any other channel. Monitors that are identified as safety related are redundant and are supplied power from the station 120V AC safety related buses. The power supplies for these channels are identified in Table 12.3.4 1. Power to the channels that monitor only normal operations is supplied from the station regulated 120 V AC instrumentation bus.

12.3.4.1.7 Redundancy, diversity, and independence

Area monitors designated as safety related are part of the safety related portion of the RMS (Section 11.5.2.3), and are designed for redundancy, diversity, and independence in accordance with IEEE 308-1974, IEEE 323-1974, IEEE 336-1971, IEEE 344-1975, and IEEE 384-1974. All safety related area monitors except the control room area monitor are designed in accordance with IEEE-279-1971. Area monitors that are Seismic Category I are manufactured and rated to the above standards as applicable to seismic qualification.

12.3.4.1.8 Area radiation monitors

12.3.4.1.8.1 Containment ventilation isolation monitors

The containment ventilation isolation monitors are part of the safety related portion of the RMS (Section 11.5.2.3) and provide an indication to operations personnel of the activity inside the primary shield of the Containment. There are four such monitors around the reactor cavity, at the operating floor inside the Containment. Of these monitors, two are powered by Bus A, and two by Bus B. Each monitor is an ion chamber of the type discussed in Section 11.5.2.5.3.

These monitors provide a high radiation alarm when the dose reaches preset limits. The receipt of these alarms, 2/4 logic, initiates a containment ventilation isolation, and alerts operations personnel to abnormal radiation inside the Containment.

The setpoints are set differently for refueling and normal operational modes.

These monitors also serve as an indication to operations personnel of post LOCA activity and are qualified to survive a post-LOCA environment for 15 minutes plus 1 hour margin.

12.3.4.1.8.2 Post-LOCA monitors

The post-LOCA monitors are part of the safety related portion of the RMS (Section 11.5.2.3) and provide an estimate to operations personnel of the activity inside the Containment following a LOCA. Post-LOCA radiological conditions in containment are monitored by the containment ventilation isolation area monitors and the digital high range monitors as discussed in Sections 11.5.2.7.2.17 and 12.3.4.1.8.1.

12.3.4.1.8.3 Fuel Handling Building (FHB) monitors

The FHB monitors, which are part of the safety related portion of the RMS (Section 11.5.2.3), measure the radiation dose around the spent fuel pools, north and south, and the fuel transfer canal. Using the detectors described in Section 11.5.2.5.2, these monitors shift FHB ventilation to the emergency mode in the event of a fuel handling accident.

The FHB monitors, which are not part of the safety related portion of the RMS, measure the radiation dose around the new fuel pools using the detectors described in Section 11.5.2.5.1

Airflow over the fuel pool is directed from the sides of the pool to the pool center and upwards to the exhaust vents. The monitors located at the side of the fuel pool are area monitors, which will detect radiation emanating from radioactive material in the airflow.

An analysis has been performed which shows that the area monitors located at the side of the fuel pool will be effective in shifting Fuel Handling Building ventilation in the event of a fuel handling accident. It was assumed that the airborne activity released over a period of two hours from a damaged fuel assembly is swept to the center of the pool by the ventilation system.

The activity rises to the exhaust ducts located in the ceiling. Under these assumptions the dose rate at the monitor furthest from the fuel pool center is in excess of an assumed 100 mR/hr isolation setpoint. Detector setpoints <100 mR/hr will be selected to provide prompt alarm and emergency ventilation initiation upon detection of abnormal radiation levels.

Detectors are so arranged that there is always one group of three detectors (one monitor), which cannot be shielded by a crane, or other object, either partially or fully.

These monitors provide a high radiation alarm when the dose reaches preset limits. The receipt of this alarm from any single monitor causes the normal exhaust dampers for the operating floor to close and the normal exhaust fans to stop, while the emergency exhaust dampers open and emergency exhaust fans start up.

There are a total of eight safety related triplet detector systems located on the Fuel Handling Building operating floor. The detectors are located around the appropriate spent fuel pools. Detector alert alarm setpoints will be selected to provide prompt indication of abnormal radiation levels. Personnel in the vicinity of the detectors will be warned of alert levels and high levels of radiation by detector local indicator units.

Because of a potentially high background, the more sensitive microprocessors are removed from the FHB to the RAB.

12.3.4.1.8.4 Post-accident Access Area Monitors

The access area monitors provide an indication to operations personnel of the radiation levels in areas vital for post-accident access, in accordance with Regulatory Guide 1.97. There are five such monitors in the Reactor Auxiliary Building. Each monitor is a High Range Area Monitor of the type discussed in Section 11.5.2.5.12.

A monitor is located at each of the two Hydrogen Sampling Skids, in each of the two Accumulator Vacuum Relief Valve areas and in the Motor Control Center Area.

These monitors provide a high radiation alarm when the dose reaches preset limits.

12.3.4.1.8.5 Other Area Radiation Monitors

There is one non-safety area monitor located in the South fuel pool area that will also alarm to warn personnel about the presence of unacceptably high radiation levels.

Area radiation monitors not listed above use the detector types specified in Section 11.5.2.5.1, provide alarms only, and exert no control action. These monitors, as well as the ones discussed in Sections 12.3.4.1.8.1, 12.3.4.1.8.2, and 12,3.4.1.8.3 are shown in Table 12.3.4-1.

12.3.4.2 In Plant Airborne Radiation Monitoring System (IPARMS)

12.3.4.2.1 Design objectives

The objectives of the IPARMS during normal operating plant conditions and anticipated operational occurrences are:

a) To inform operations personnel of airborne particulate, iodine and gaseous activity trends in the various buildings and structures of the plant,

b) to aid in identifying abnormal increases in the airborne activity levels,

c) to furnish records of gross airborne trends in the various plant areas,

d) to help detect identified or unidentified leaks from the reactor coolant pressure boundary (as recommended in Regulatory Guide 1.45, which is discussed in Section 1.8) and other areas of the plant,

e) to assist personnel in deciding whether or not breathing apparatus is necessary when entering a high airborne activity area,

f) to provide information for evaluation of the performance of all plant systems that function to minimize the release of airborne radioactivity to accessible areas of the plant and to the environment,

g) To provide, during postulated accidents, the capability to alarm and initiate isolation of the normal ventilation systems and actuation of the emergency ventilation systems, and to provide operators with information regarding control of the ventilation systems in an emergency, and

h) to provide information for the purpose of maintaining low in-plant personnel radiation exposure via inhalation of airborne particulates and iodine, in accordance with 10 CFR 20 and Regulatory Guide 8.8 (see Section 12.1).

12.3.4.2.2 Criteria for location of monitors

Considerations for locating the IPARMS monitors are based on the following:

a) areas where the airborne radioactivity can abruptly increase and where personnel normally have access to the areas,

b) in selected ventilation ducts where the monitors can survey the airborne radioactivity level, and

c) inside the Containment for the purpose of monitoring unidentified leaks.

An alarm by a ventilation monitor indicates that the airborne contamination is coming from a limited number of enclosures serviced by the monitored ventilation system. Sampling of the air in each enclosure within this group with portable instruments would be used to identify the source.

12.3.4.2.3 General system description

The IPARMS provides the means to monitor normal airborne radioactivity levels and to detect and annunciate any abnormal radiation conditions occurring throughout the plant. The IPARMS is an integral part of the Radiation Monitoring System, which is described in detail in Section 11.5.2.3.

The IPARMS consists of skid or duct mounted airborne monitors. Each monitor is composed of individual channels, with a channel consisting of a sampling chamber (when required, i.e., off-line, skid mounted monitors), check source, and detector. The detector assembly consists of gamma or beta sensitive scintillation crystal, a photomultiplier tube, and a local amplifier. All

channels associated with a monitor are served by a local, dedicated microprocessor, and all channel information is processed through this microprocessor, which is then interrogated by the appropriate radiation monitoring system computer for further processing, indication on a cathode ray tube (CRT), storage, remote alarming and hard copy production, if so desired. The monitors listed in Table 12.3.4-2 comprise the IPARMS.

Portable CAMS are used in some normally occupied areas of the RCA to detect iodines and particulate airborne concentrations equivalent to 10 DAC- hours or more. Dilution of the sample is not considered because the CAMs are located in the area being monitored and sample directly from the area in question. The CAMs provide local alarms to warn personnel in the area of elevated airborne activity.

12.3.4.2.4 Microprocessor

The microprocessor is described in Section 12.3.4.1.4.

12.3.4.2.5 Local annunciation

All airborne monitors have local annunciation, consisting of an audible alarm and three alarm lights for high radiation, alert radiation and monitor failure. All airborne monitors have a display at the microprocessor indicating the airborne radioactivity concentration in µCi/cc of Iodine 131 or gross particulate or noble gas activity at the detector.

12.3.4.2.6 Power supplies

Each channel is provided with an independent power supply, designed such that a failure in that channel does not affect any other channel. Monitors that are identified as safety related are redundant and are supplied power from the station 120V AC safety related buses. The power supplies for these channels are identified in Table 12.3.4-2 to the channels that monitor only normal operations is supplied from the station regulated 120V AC instrumentation bus.

12.3.4.2.7 Redundancy, diversity, and independence

IPARMS monitors designated as safety related are part of the safety related portion of the RMS (Section 11.5.2.3) and are designed for redundancy, diversity, and independence in accordance with IEEE 308-1974, IEEE 323-1974, IEEE 336-1971, IEEE 344-1975, and IEEE 384-1974. The control room normal and emergency outside air intake monitors are also designed in accordance with IEEE 279-1971. The containment leak detection monitor complies with Paragraph 4.10 of IEEE 279, as required by Reg. Guide 1.45. IPARMS monitors which are Seismic Category I are manufactured and rated to the above standards as applicable to seismic qualification.

12.3.4.2.8 Airborne radiation monitors

12.3.4.2.8.1 Containment atmosphere leak detection monitor

The containment atmosphere leak detection monitor is part of the safety related portion of the RMS (Section 11.5.2.3) and is designed to provide an indication to operations personnel of the particulate and gaseous radioactivity levels inside the Containment. Radioactivity in the containment indicates the presence of fission products due to a reactor coolant pressure

boundary leak, and as such this monitor is a part of the Reactor Coolant Pressure Boundary Leakage Detection System required by Regulatory Guide 1.45 (see Section 1.8). A detailed description of the sampling system associated with this monitor, operation and monitoring requirements is found in Section 5.2.5. The monitor uses the airborne particulate and noble gas detector described in Section 11.5.2.6.5. The monitor is powered by the Emergency A Bus. A containment isolation actuation signal will isolate this monitor from the Containment.

This monitor provides a radiation alarm when concentrations reach preset limits. The receipt of this alarm will alert the operator to the presence of low-level leakage so that additional sampling can be effected in order to locate the leakage source. An interlock is provided to terminate continuous purge operation on high radiation.

12.3.4.2.8.2 Control room normal outside air intake

The control room normal outside air intake plenum has two beta sensitive monitors, one associated with A Bus, and one with B Bus. These monitors are part of the safety related portion of the RMS (Section 11.5.2.3) and use the ambient gas monitors described in Section 11.5.2.6.1.

These monitors provide a high radiation alarm when concentration levels reach preset limits. Upon receipt of the alarm, the monitor closes the normal and emergency outside air intake valves, stops the exhaust fans, closes the exhaust dampers, starts up the emergency filtration fans and opens the required valves and dampers to put the air flow into the recirculatory mode. In addition, the RAB Normal Ventilation System is secured, and the RAB Emergency Exhaust System (RABEES) is started. The RAB Normal Ventilation System must be secured to preclude the possibility of postulated system failures from impacting the ability of the Control Room Envelope (CRE) to maintain a positive pressure of ≥ 1/8 INWG relative to adjacent areas. When the RAB Normal Ventilation System is secured, the RAB Emergency Exhaust System is initiated to maintain the potentially contaminated areas of the RAB at sub-atmospheric pressure in an effort to limit outleakage and to remove radon gas from the RAB. The receipt of these alarms will also alert the operator to check the radiation levels at both emergency outside air intakes, and to open the intake at which the radiation level is lower (Section 12.3.4.2.8.3).

12.3.4.2.8.3 Control room emergency outside air intake

There are two emergency outside air intakes. Each intake has two duct-mounted Beta monitors associated with A Bus (Intake 10 and 11A) and B Bus (Intake 10 and 11A). These monitors are part of the safety related portion of the RMS (Section 11.5.2.3), and use the ambient gas monitors described in Section 11.5.2.6.1.

These monitors provide a high radiation alarm when concentration levels reach preset limits. Upon receipt of the alarm, the monitor closes the normal and emergency outside air intake valves, stops the exhaust fans, closes the exhaust dampers, starts up the emergency filtration fans, and opens the required valves and dampers to put air flow into the recirculatory mode. In addition, the RAB Normal Ventilation System is secured, and the RAB Emergency Exhaust System (RABEES) is started. The RAB Normal Ventilation System must be secured to preclude the possibility of postulated system failures from impacting the ability of the Control Room Envelope (CRE) to maintain a positive pressure of ≥ 1/8 INWG relative to adjacent areas. When the RAB Normal Ventilation System is secured, the RAB Emergency Exhaust System is

initiated to maintain the potentially contaminated areas of the RAB at sub-atmospheric pressure in an effort to limit outleakage and to remove radon gas from the RAB.

These monitors provide indication to the control room personnel of the radioactivity levels at each emergency air intake, thereby allowing the operator to choose which emergency intake to open (see discussion in Section 7.3.1.5.7 and 12.3.4.2.8.2). These monitors also provide a high radioactivity alarm when concentration levels reach preset limits.

There is one outside air intake monitor for the Technical Support Center (TSC). This monitor is also an in duct ambient beta monitor. The radiation levels and any alarms are received on the RMS consoles in the Control Room, the WPB Control Room, the health physics office, and the computer room. They are also monitored by the Emergency Response Facility Information System (ERFIS).

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## 12.4 DOSE ASSESSMENT

Note: Analysis and terms used in this section are based on "pre-1993 10 CFR 20" (see Section 12.0).

### 12.4.1 ANTICIPATED DOSE RATES

The guidance in Regulatory Guide 8.19 has been used in the preparation of estimates of occupational radiation exposures for SHNPP. Prior to plant start-up, external dose rates due to direct radiation which could be experienced by plant personnel during operation, inspection, and anticipated operational occurrences were determined using the dose rates in Figures 12.3.2-2 through 12.3.2-17, time and motion estimates, and pre-operational staffing data. This data is presented in Part I of Table 12.4.1-1.

Radiation exposure within the plant from airborne radionuclides was also calculated. Exposure to this source is usually insignificant in comparison to direct radiation exposure. However,

under certain circumstances of reactor coolant leakage in Containment coupled with high reactor coolant activity, doses from airborne activity could become a major fraction of the allowed limits. Proper purging, however, should maintain airborne exposures so low that their contribution to the total person-rem is insignificant in comparison with exposure from direct radiation.

The majority of the exposure from the plant is due to occupational exposure to direct radiation within plant structures. These doses were calculated based on the area dose rates determined as described in FSAR Section 12.2, and estimated hours spent in each area. These occupancy times were determined using anticipated crew sizes and duration for reactor operation and surveillance, maintenance (routine and special), in-service inspection, waste processing, and refueling activities. The anticipated annual doses received by plant personnel were well below the limits of 10 CFR 20 and compared favorably with doses at operating PWRs.

During plant operation, dose rates are determined by surveys and air samples, personnel are monitored for external and internal exposure, and doses are tracked against established plant goals. Programs are in place to maintain these doses ALARA. These programs are described in FSAR Section 12.5. Annual doses for plant operation are shown in Part II of Table 12.4.1-1.

## 12.4.2 ESTIMATES OF DIRECT EXPOSURES TO PLANT PERSONNEL

12.4.2.1 Deleted by Amendment No. 48

12.4.2.2 Deleted by Amendment No. 48

12.4.2.3 Deleted by Amendment No. 48

12.4.2.4 Deleted by Amendment No. 48

12.4.2.5 Deleted by Amendment No. 48

12.4.2.6 Deleted by Amendment No. 48

12.4.2.7 Deleted by Amendment No. 48

12.4.2.8 Deleted by Amendment No. 15

12.4.2.9 Estimated Annual Dose at the Boundary of Restricted Area and Site Boundary

The maximum exposure rates from the originally planned four Units to an individual due to contained sources was estimated to be $9.7 \times 10^{-2}$ mrem/yr at the boundary of the restricted area (security fence) and $8.8 \times 10^{-3}$ mrem/yr at the site boundary in the direction south-southwest from the plant. The maximum whole body and thyroid dose to an individual due to gaseous effluents has been estimated to be 76.3 and 107.1 mrem/yr respectively at the boundary of restricted area. These numbers were calculated for the originally planned four units and are therefore conservative for one unit operation. The exposures to an individual at the site boundary due to gaseous effluents are discussed in Section 11.3.3.

12.4.2.10 Inhalation Radiation Exposures

As a result of leakage from equipment containing radioactive fluid, some radioactivity is airborne within the compartment housing the equipment. It can contribute to the inhalation radiation exposure if inspection, maintenance, or any similar activity is performed during leakage. Internal thyroid and whole body dose rates calculated from leakage rates are shown in Tables 12.2.2-3 and 12.2.2-4. The occupancy factors are determined from inspection and maintenance activities in the areas. Person-rem estimates based on these assumptions are shown in Table 12.4.2-13.

During plant operation, personnel are monitored for external and internal exposures and doses are tracked against established plant goals. Programs are in place to maintain these doses ALARA. These programs are described in FSAR Section 12.5.

REFERENCES: SECTION 12.4

12.4.2-1 Johnson, L. A., NUREG-1323, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1976, March 1978.

12.4.2-2 National Environmental Studies Project, Compilation and Analyses of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, SAI Services, Sept., 1974.

12.4.2-3 Murphy, T. D., et al., NUREG-75/032, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974, USNRC Radiological Assessment Branch, June 1975.

12.4.2-4 NUREG-0322, Ninth Annual Occupational Radiation Exposure Report, 1976, USNRC Office of Management, Information and Program Control, October 1977.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

12.5.1.1 Introduction

The SHNPP health physics (HP) program is established to provide an effective means of radiation protection for plant personnel, visitors, and the general public.

To provide this radiation protection, the health physics program incorporates a dedicated philosophy from management, qualified personnel to direct and to implement the health physics program, the appropriate equipment and facilities, and written procedures based upon acceptable health physics practices and guidance.

The health physics program at SHNPP is developed and implemented to evaluate and document plant radiological conditions and to ensure that every reasonable effort is made to maintain occupational radiation exposure (ORE) as low as reasonably achievable (ALARA). The organization of the health physics program provides a flexible, responsive and comprehensive structure for attaining these goals. The structural organization is shown on

Figure 12.1.1-1. The qualifications of all plant personnel are provided in Sections 13.1.3 and 1.8.

The development of the health physics program reflects not only the experience of the personnel involved, but also the experience gained at the other CP&L nuclear plants. This experience established the starting point from which the SHNPP health physics program was developed.

12.5.1.2 Responsibilities

The Superintendent - Radiation Protection, who is under the supervision of the General Manager - Harris Plant is responsible for providing the information necessary to establish compliance with regulations pertaining to radiation safety, for uniformly enforcing plant health physics requirements, and for ensuring every reasonable effort to minimize personnel exposures. In addition, he/she is responsible for ensuring that the staff members who implement the health physics program are trained and retrained in operational health physics principles. He/she is assisted by a staff of HP Supervisors and Specialists. HP Specialists are available for special tasks, consultation, and health physics program analysis. The ALARA program is implemented and evaluated under the technical direction of the ALARA Specialist. The ALARA Specialist provides technical direction and expertise to the HP unit and ensures compliance with the corporate commitment to ALARA expressed in the Corporate Health Physics Policy.

The HP unit coordinates with operations, maintenance, work control, and other units to provide health physics coverage for all activities that involve radiation or radioactive material. In doing so, HP provides various services including the following:

1. Implementing health physics procedures for routine and non-routine activities that may be encountered in operating, maintaining, inspecting, and testing the plant.

2. Ensuring that the provisions and standards of 10 CFR Part 20 for permissible dose limits, contamination levels, and potential release levels are not exceeded.

3. Implementing the CP&L personnel radiation dosimetry program at SHNPP and maintaining dosimetry records.

4. Providing radiation surveys of plant areas, maintaining records and posting survey results.

5. Providing, maintaining, and calibrating radiation detection instruments and equipment for assessing the radiation environment at SHNPP.

6. Providing, maintaining, and issuing protective clothing and equipment.

7. Assisting in the shipping and receiving of all radioactive material to ensure compliance with regulatory requirements.

8. Preparing and issuing regulatory and company required reports of personnel dosimetry results and other radiological activities at SHNPP.

9. Assisting in the decontamination of personnel, equipment, and facilities at SHNPP.

10. Implementing and maintaining the SHNPP respiratory protection program.

The responsibilities of the HP Supervisors also include the day-to-day execution of the health physics program through supervision of the routine and special surveys and the programs required by applicable regulations and procedures. The HP Technicians implement the health physics program by performing routine and special surveys and by providing health physics surveillance in accordance with plant health physics procedures.

A major responsibility of the ALARA Specialist is to ensure that every reasonable effort has been made to maintain operational radiation exposure ALARA. The ALARA Specialist develops and coordinates implementation of a plant ALARA program that is responsive to plant conditions. The ALARA Specialist major responsibility is to provide the Superintendent - Radiation Protection with the information needed to ensure that every reasonable effort has been made to minimize personnel exposures. The ALARA Specialist also provides major input to ALARA committee activities.

Specialists reporting to the Superintendent - Radiation Protection or HP Supervisors provide technical support for the health physics program by:

1. Developing and administering health physics procedures which implement regulatory and corporate policies in a manner having sound technical and cost/benefit bases.

2. Tracking and evaluating health physics activities to identify adverse trends and ensure compliance with requirements.

3. Evaluating health physics problems and assisting in the completion of corrective actions.

4. Reviewing plant modifications and procedures that impact the health physics program.

5. Assisting plant training programs by providing technical input and specialized HP training.

6. Directing HP oversight for receipt, offload, and storage of spent nuclear fuel shipped from Brunswick and Robinson plants.

It is the responsibility of each individual to obey all HP procedures and to report to his/her supervisor and HP Supervisors any circumstances where procedures may be incorrect or unsafe activities may be occurring.

Independent oversight of the quality of HNP health physics program activities is provided by periodic assessments by the Nuclear Oversight Section in accordance with Section 17.3.3 of the FSAR. Health Physics program support is also provided at the corporate level by the Supervisor - Radiological Services as discussed in Section 12.1.1.1.

12.5.1.3 Authority

The General Manager - Harris Plant, who is ultimately responsible for all plant activities including radiation safety, receives reports from the Superintendent - Radiation Protection concerning the status of the health physics program. To ensure uniform enforcement of health physics requirements, the General Manager - Harris Plant delegates authority, with respect to radiation safety, to an individual meeting the ANS 3.1 qualifications for Radiation Protection Manager (RPM) as outlined in Regulatory Guides 1.8 and 8.8. The Superintendent - Radiation Protection has the authority to cease any work activity when, in his judgment, worker safety is jeopardized, or in the event of unnecessary personnel radiation exposures.

The Superintendent - Radiation Protection delegates, through the HP Supervisors, the authority to cease any work activity that is not being performed in accordance with Radiation Work Permit including associated task requirements to the responsible HP Technicians. The HP Technicians have the authority to ensure that jobs are conducted in accordance with health physics procedures and Radiation Work Permit (RWP) including associated task requirements.

In the absence of HP supervision during backshifts, weekends and holidays, the authority associated with the above positions may be delegated, in accordance with the plant's health physics procedures, to an ANSI qualified (ANS 3.1-1979 draft) technician on shift. In the event of extended absence of HP supervision for sickness, vacation, or other unforeseen events, the authority for the vacated position will be transferred to a designated relief, who is qualified in accordance with the September 1979 draft of ANS 3.1 as clarified in Section 1.8 of the FSAR.

The ALARA Specialist is responsible for ensuring that jobs can be accomplished with minimal radiation exposures. The ALARA Specialist has the responsibility and authority to conduct informal training and/or discussions with workers and supervisors regarding observed practices and ALARA recommendations.

12.5.1.4 Experience and Qualification

The individual designated as RPM will satisfy the qualifications given in Regulatory Guide 1.8 for the Radiation Protection Manager.

The HP staff, which is responsible for the health physics program at SHNPP, meets minimum experience and qualification requirements as described in FSAR Section 1.8.

12.5.2 FACILITIES, EQUIPMENT AND INSTRUMENTATION

12.5.2.1 Waste Processing Building

The facilities as described below are shown in Figures 1.2.2-49 and 1.2.2-50.

12.5.2.1.1 Health Physics (HP) facilities

Health Physics facilities are located primarily on Elevation 261 ft. of the Waste Processing Building. Space is provided for HP activities which include preparing Radiation Work Permits (RWPs)including associated tasks, performing whole body counts, counting smears and air samples, and issuing dosimeters, survey instruments, and respirators. Facilities are also provided for performing personnel decontamination, cleaning/inspecting/repairing respirators

and sorting/packaging trash and radioactive materials. Space is available for storing HP records and the equipment used for performing the activities described in FSAR Section 12.5.3. The main RCA control point is also located on this elevation. Equipment such as whole body friskers and small article monitors are provided here for performing personnel and small item contamination surveys prior to exiting the RCA.

This building contains the Operational Support Center, which has emergency cabinets containing HP equipment for use during declared emergencies.

The personnel decontamination areas contain the equipment and materials necessary to perform routine personnel decontamination per plant procedures. Sinks and showers drain to the chemical drain tanks for processing through the Liquid Waste Processing System.

Locker rooms and dress out areas, where workers may change from street clothing into protective clothing, are provided near the entrance to the primary RCA. Protective clothing is also available at local access control points within the RCA. Toilets, washrooms, and shower rooms are adjacent to the locker rooms.

12.5.2.1.2 Radiochemistry facilities

Facilities are provided for performing radiochemical analyses on Elevation 276 ft. of the Waste Processing Building. Space is provided for preparing, analyzing, and counting radiochemistry samples.

The radiochemistry laboratory, which is used for sample preparation, and chemical analysis, contains standard equipment for a lab of this type. The fume hoods are exhausted to the Waste Processing Building vent, and the sinks drain to the chemical drain tanks for processing through the Liquid Waste Processing System.

The radiochemistry counting facility is constructed with concrete walls to provide a low background environment for analyzing radiochemistry samples of plant effluents and process streams. Appropriate instrumentation (such as a gas-flow proportional counter, a liquid scintillation counter, alpha and beta scintillators, end window GM, and lithium drifted germanium and/or sodium iodide systems) is used for counting and/or analysis of radiochemistry samples. These instruments can achieve the levels of detection required by procedures and regulations.

12.5.2.1.3 Chemistry laboratory

The chemistry laboratory, which is used for performing analyses on non-radioactive materials, contains chemical fume hood assemblies, and standard laboratory equipment for a lab of this type.

The laboratory chemical fume hoods discharge to the Waste Processing Building vent and the sinks drain to the chemical drain tank for processing through the Liquid Waste Processing System. An emergency shower is accessible from both the radiochemistry and chemistry laboratories.

12.5.2.1.4 Waste packaging facilities

The waste packaging facilities, located on Elevation 261 ft. of the Waste Processing Building, consist of solid waste packaging, decontamination, and monitoring areas.

12.5.2.1.5 Instrumentation facilities

The instrumentation facilities, located on Elevation 276 ft of the Waste Processing Building, consist of an instrument calibration and repair shop and instrument storage area.

The appropriate equipment and radiation sources are provided for calibrating instruments in accordance with plant procedures. Equipment and materials are also available for performing repairs on selected instruments.

12.5.2.1.6 Laundry and respirator decontamination facilities

A laundry facility is available on Elevation 261 ft of the Waste Processing Building for sorting and laundering of contaminated protective clothing. It provides an area for segregating highly contaminated clothing from low level contaminated clothing, and is equipped with washers, and exhausted dryers, a deep sink, miscellaneous tables and carts, and a storage area for laundry supplies. The laundry effluent is discharged to the Laundry and Hot Shower Tanks for sampling prior to processing through the Liquid Waste Processing System.

A respirator cleaning, inspection, and maintenance facility is located on Elevation 261 ft of the Waste Processing Building. Equipment for respirator cleaning includes a respirator washer, sink/container, and drier.

12.5.2.1.7 Health Physics equipment

12.5.2.1.7.1 Protective clothing

Protective clothing is worn in contaminated areas to prevent personnel contamination and to aid in controlling the spread of surface contamination. Protective clothing typically used at SHNPP includes: reusable coveralls and lab coats, disposable coveralls and lab coats, plastic suits, surgeons caps, cloth hoods, plastic hoods, splash shields, cloth gloves, rubber gloves, disposable gloves, rubber shoe covers, cloth shoe covers, and plastic shoe covers.

Protective clothing is stored in the dressout area and in other selected plant areas.

12.5.2.1.7.2 Respiratory protection equipment

Respiratory protection equipment is used to minimize the intake of radioactive material when engineering controls are not practicable and when such use will keep the Total Effective Dose Equivalent ALARA. The respiratory protection program is described in Section 12.5.3.5.

Respiratory protection equipment used at SHNPP meets the approval of National Institute of Occupational Safety and Health/Mine Equipment Safety Administration (NIOSH/MESA) or has been authorized for use per 10 CFR 20.1703(b). Various types of respirators, including: air purifying respirators, self-contained breathing apparatus (pressure/demand), pressure/demand air line respirators, and constant flow air line respirators, hoods, welding masks and plastic

supplied air suits are available as appropriate for the type of work being performed. Different sizes of respiratory devices are available to assure that the differing facial contours of personnel requiring respiratory protection can be properly fitted. Sufficient quantities of respiratory protection equipment are available to allow for the use, decontamination, maintenance, and repair of equipment.

Respiratory protection equipment is available at the RCA checkpoint of the Waste Processing Building. Respiratory protection equipment is also available for emergency use at the Technical Support Center, Operational Support Center, and Control Room.

12.5.2.1.7.3 Air sampling equipment

Airborne activity levels are determined by the use of installed process airborne radiation monitors, continuous air monitors (CAM), and high and low volume portable air samplers. CAMs are equipped with strip chart recorders or local readouts or a microprocessor control with local/remote readouts and data logging capability.

*12.5.2.1.7.3.1 Deleted by Amendment No. 48.*

*12.5.2.1.7.3.2 Deleted by Amendment No. 48.*

*12.5.2.1.7.3.3 Deleted by Amendment No. 48.*

*12.5.2.1.7.3.4 Special air sampling*

Water bubblers, desiccant columns, or cold traps are available for tritium air sampling, and gas sample containers (such as Marinelli containers) are available for sampling airborne gases.

12.5.2.1.7.4 Personnel dosimetry

The personnel dosimetry program is described in Section 12.5.3.6.

Personnel dosimeters such as thermoluminescent dosimeters (TLDs), Pocket Ion Chambers (PICs) and Electronic Dosimeters (EDs) are available for monitoring external radiation exposure.

Electronic dosimeters provide alarm functions for dose, dose rate and stay time.

A whole body counter with sufficient sensitivity to detect, in the thyroid, lungs, or whole body, a fraction of the Annual Limit on Intake for gamma emitting radionuclides of interest is provided to evaluate internally deposited radioactive material. The detectors are used in conjunction with a multi-channel analyzer and associated readout to obtain a permanent record. A whole body counting system off-site may be used as an alternative or supplement to a SHNPP whole body counter.

12.5.2.1.7.5 Miscellaneous equipment

The following miscellaneous radiation protection equipment is stored at various locations in the plant:

Contamination control supplies may include glove bags, containment tents, absorbent wipers, absorbent paper, rags, step-off pads, rope, plastic sheets, plastic bags, tape, contamination area signs, and protective clothing. Appropriate supplies may be assembled into kits and located throughout the plant to aid in the control of a contaminated spill.

Temporary shielding, such as lead bricks, lead sheets, and lead wool blankets, are available to reduce radiation levels.

Portable ventilation units, which include HEPA filters and charcoal filters, are available as engineering controls help control airborne radioactivity.

HEPA-equipped vacuum cleaners are available for use in cleaning contaminated areas. Equipment such as ultrasonic sinks and a bead blaster are provided for performing equipment decontamination.

12.5.2.2 Containment Building

Some shielding materials are stored in the Containment Building during power operations. Additional equipment may be staged in this building to support special evolutions or outages.

12.5.2.3 Deleted by Amendment No. 48

12.5.2.4 Fuel Handling Building

HP equipment in this building includes self-survey personnel monitoring equipment such as portal, hand/foot monitor or hand held frisker equipment.

12.5.2.5 Diesel Generator Building

This building does not normally contain HP equipment.

12.5.2.6 Turbine Generator Building

An air compressor, used to refill air bottles used in self-contained breathing apparatus, is located on the 261 ft. elevation of the Turbine Building.

12.5.2.7 Deleted by Amendment No. 46

12.5.2.8 Deleted by Amendment No. 48

12.5.2.9 Security Building

HP equipment in this building is normally limited to self-survey personnel monitoring equipment such as portal or hand held frisker equipment.

12.5.2.10 Deleted by Amendment No. 44

12.5.2.11 Health Physics Instrumentation

Instruments for detecting and measuring alpha, beta, gamma, and neutron radiation consist of counting room and portable radiation survey/monitoring instruments. All instruments are subjected to operational checks and calibration to ensure the accuracy of measurements of radioactivity and radiation levels. Laboratory standards (utilizing, or prepared from, standards of Sr-90, Am-241, Cs-137, Co-60, H-3, and others, traceable to nationally recognized standards) are used to maintain required accuracies of measurement. Background and operability checks of routinely used health physics counting equipment are performed daily. These instruments will be removed from service whenever their operation is statistically out of limits specified in plant procedures. Routine calibrations will be performed on counting room instrumentation and radiation survey/monitoring instruments on at least an annual basis and after repairs affecting calibration. Sufficient quantities of instrumentation are available to allow for use, calibration, maintenance, and repair.

The instrumentation described in these subsections may be replaced by equipment providing similar or improved capabilities.

12.5.2.11.1 Counting room instrumentation

Counting room instruments for radioactivity measurements include the following:

- A multi-channel analyzer, using a photon sensitive crystal or detector (e.g., NaI, Ge(Li), etc.) for identification and measurement of gamma emitting radionuclides in samples of reactor coolant, process streams liquid and gaseous effluents, and airborne and surface contaminants.

- A computer system which can be interfaced with a pulse height analyzer, equipped with a keyboard for entering instructions and a printer for hard copies of results.

- A low background proportional counter used for gross alpha and gross beta measurements of prepared samples.

- A liquid scintillation counter used for measurement of tritium in reactor coolant, and liquid and gaseous wastes.

12.5.2.11.2 Health Physics office and workroom instrumentation

Health Physics instrumentation normally located near the RCA checkpoint and in the instrument storage room include the following instruments, or equivalent:

- GM beta-gamma survey meters equipped with either an internal or external probe with ranges from 0.2 mrem/hr to 2,000 mrem/hr used for detection of radioactive contamination on surfaces and for low-level exposure rate measurements.

- Ionization chamber beta-gamma survey meters 0-5 rem/hr (0-5 mrem/hr most sensitive range) used to cover the general range of dose rate measurements necessary for radiation protection evaluations.

- Wide range ionization chamber beta-gamma survey meters (0-5 mR/hr most sensitive range, maximum range 0-50 R/hr) used for exposure rate measurements.

- Remote monitoring (telescoping probe) GM beta gamma survey meters, 0-1,000 R/hr, (0-2 mR/hr most sensitive range) used for exposure rate measurements.

- Neutron Rem Counters 0-5 rem/hr (0-5 mrem/hr most sensitive range). The instrument is used to measure the dose equivalent rate due to thermal, intermediate, and fast neutron fluxes.

- Alpha scintillation survey meters, 0-500K cpm used for measurement of alpha surface contamination.

- Radiation exposure ratemeters with range settings to 20,000 R/hr

12.5.2.11.3 Personnel contamination monitoring instrumentation

Personnel monitoring instruments consisting of hand-held friskers, hand/foot monitors, personnel contamination monitors, or portal monitors described below, are used at the locations specified in Section 12.5.2.1:

- Beta-gamma count rate meters (hand-held friskers), 0-50,000 cpm range, adjustable audio and/or visual alarms, used to detect contamination on personnel, materials, protective clothing, and equipment.

- Personnel contamination monitors consisting of audio and/or visual alarmed detectors to provide head to foot beta-gamma detection capability. The count rate alarm and the counting time are adjustable.

- Personnel hand/foot contamination monitors consisting of audio and/or visual alarmed detectors provide hand and foot beta-gamma detection capability. The count rate alarm and the counting time are adjustable.

- Portal monitors consisting of audio and/or visual alarmed detectors to provide head to foot gamma detection capability. The count rate alarm and the counting time are adjustable.

Personnel contamination monitoring instrumentation will be calibrated on at least an annual basis or following repair, in addition to periodic source checks, to determine proper response and alarm operability.

12.5.2.11.4 Miscellaneous HP instrumentation

A condenser R-meter, electrometer, or equivalent instrument is used to accurately measure radiation dose levels over several ranges.

A pulse generator is used for calibrating pulse counting instruments up to $10^6$ cpm.

An air flow standard (e.g. anemometer), a face piece test bench, and respirator fit test equipment are examples of other HP instruments available.

Instruments are available for testing HEPA filters.

12.5.2.12 Areas Outside of Plant Structures

Radiation sources in the form of Dry Active Waste, resins, spent and new fuel, radiography sources, contaminated tools and equipment, irradiated components, etc. may be present in areas outside of plant structures.

Temporary facilities, such as trailers, tents, Sea-Land containers, etc. may also be located in areas outside of plant structures for use in performing decontamination, maintenance on contaminated components, radiography, waste processing, or other activities.

Administrative controls and plant procedures will be used to maintain the doses from these sources and facilities during normal operations within regulatory limits and ALARA. Based on the type of materials and facilities involved, administrative limits on source activity will be established to keep the dose from accidental releases below allowable limits.

Effluent monitoring will be performed as deemed necessary.

12.5.3 PROCEDURES

Health Physics procedures developed for SHNPP are an integral part of the ALARA policy as discussed in Section 12.1. These procedures, developed through careful planning and preparation and used by well-trained and qualified personnel, should contribute significantly to the overall reduction of the occupational radiation exposures. These procedures reflect the operating experience that CP&L has gained at its other nuclear facilities. Carolina Power & Light Company commitments to Regulatory Guides, as well as the provisions and suggestions of Regulatory Guides 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 1.8, 1.16, 1.33 and 1.39, have been used as guidance in the development of the HP procedures. The HP procedures cover the appropriate administrative, operating and ALARA-related operations and conditions at SHNPP.

12.5.3.1 Access Control

As specified in Section 12.1, physical and administrative controls are instituted at SHNPP to ensure that the philosophy of maintaining personnel exposures as low as reasonably achievable (ALARA) is implemented.

12.5.3.1.1 Physical controls

12.5.3.1.1.1 Security check point and access control

The plant's primary security checkpoint(s) are continuously manned. A Biometric System (Hand Geometry) is employed at the Access Entry point to verify identity of personnel who possess security badges affording unescorted access to the protected area.[2] The security force ensures that all personnel who enter the plant possess appropriate badges in accordance with plant procedures. A restricted area access list is maintained at the security entrance. Any individual

[2] On December 20, 1994, the NRC issued CP&L an exemption from the requirements of 10CFR73.55(d) permitting the use of an alternative identity verification system and the removal of security badges/access control devices from the site by non-licensee personnel.

not authorized access must be accompanied by a person who is an authorized escort for restricted areas. The training, retraining and testing requirements for unescorted access are described in the SHNPP Plant Access Training Procedures.

While not specifically intended to control access to radiation areas, the security interlocked door system ensures that only authorized individuals are able to open security entrances to various plant areas and facilities. Security entrances are locked or provided with continual surveillance. Details of security access control are contained in the SHNPP Security Plan.

12.5.3.1.1.2 Door and area posting and locking

Physical control is provided by posting and use of barricades or locked doors as appropriate, for radiation and high radiation areas. Radiation areas as defined in 10 CFR 20.1003, are posted in accordance with 10 CFR 20.1902(a).

High radiation areas, as defined in 10 CFR 20.1003, are posted in accordance with 10 CFR 20.1902(b).1902(c) and plant Technical Specification 6.12. Entries to high radiation areas are equipped with audible and/or visible alarms in accordance with 10 CFR 20.1601(a)(2) or are controlled in accordance with 10 CFR 20.1601(a)(3) or plant Technical Specification 6.12.

Very high radiation areas are defined in 10 CFR 20.1003 and posted in accordance with 10 CFR 20.1902(c). The methods described above and additional administrative controls are used to control access to these areas as required by 10 CFR 20.1602.

Entrances to radiation areas, high radiation areas, and very high radiation areas are posted in accordance with limits specified in plant procedures, to reflect any special conditions required for entry. Postings are updated as necessary for changes in radiological conditions.

12.5.3.1.1.3 Health physics surveillance

When appropriate, health physics surveillance of work activities is provided to assure a positive control of access and stay time in radiation areas. Surveillance may also be provided for tasks in areas where conditions may warrant timely instructions to workers (e.g., on jobs where the radiological conditions can fluctuate greatly).

12.5.3.1.1.3 Health physics surveillance

When appropriate, health physics surveillance of work activities is provided to assure a positive control of access and stay time in radiation areas. Surveillance may also be provided for tasks in areas where conditions may warrant timely instructions to workers (e.g., on jobs where the radiological conditions can fluctuate greatly).

12.5.3.1.2 Administrative controls

12.5.3.1.2.1 Training

As specified in Section 12.5.3.7, personnel allowed unescorted restricted area access receive health physics and related training in accordance with 10 CFR 19.12. During this training, the individual's responsibility of using proper health physics procedures in Radiologically Controlled

Areas is emphasized. The methods used at SHNPP to physically and administratively control access are reviewed.

12.5.3.1.2.2 Radiation Work Permits

The Radiation Work Permit (RWP) system is implemented to administratively control access and stay time in radiation areas. The system consists of Radiation Work Permits (RWP) including associated tasks. Work planning is normally done at the task level and work is normally performed a task associated with the RWP. The amount of planning required is specified in plant procedures and is commensurate with the radiological hazards associated with the job. Radiation Work Permits (RWP) including associated tasks may be referred to collectively as "RWPs". All entries into an RCA require an RWP. Approved RWPs specify access requirements and special instructions. An approved RWP is in effect until conditions warrant a change, or as specified in plant health physic procedures, and is subject to cancellation by the HP Supervisor. RWPs are reviewed on a regular basis.

12.5.3.1.2.3 Procedure review

Health physics procedures related to control of access and stay time in radiation areas are at all times subject to review to ensure that every reasonable administrative effort has been made to minimize personnel exposure. Recommended changes are evaluated and, if necessary, a proposed change is forwarded through appropriate review and approval channels.

12.5.3.2 ALARA

To effectively implement the corporate ALARA commitment as discussed in Section 12.1.1, a plant ALARA Program is utilized at SHNPP to assure that activities are performed with ALARA personnel exposure. Carolina Power & Light Company considers it necessary to apply the basic concepts of ALARA to both internal and external exposure to assure proper emphasis on both modes of potential radiation exposure. Procedures employed to implement the program described are subject to review and revision to ensure that the ALARA program is responsive to plant needs and conditions.

12.5.3.2.1 ALARA procedures common to external and internal exposure

12.5.3.2.1.1 Training

Individuals allowed unescorted restricted area access receive health physics training as described in Section 12.5.3.7. The individual's responsibility to avoid unnecessary exposure is emphasized during these training sessions.

As appropriate, individuals involved in potentially high exposure jobs receive pre-job training in exposure reduction techniques and controls applicable to the specific job. Post-job reviews are held, as appropriate, to provide positive feedback on improved job performance.

12.5.3.2.1.2 Radiation Work Permit (RWP)

A detailed RWP is generally required whenever work involves a significant actual or potential radiological hazard. The requirements for using an RWP are described in plant procedures.

Health physics personnel evaluate the radiological conditions associated with the work to be performed and specify appropriate protective clothing/devices, respiratory protective equipment, dosimetry, special samples, surveys, procedures, precautions to be taken, and expiration date.

The RWP is evaluated to ensure that the work will be performed using good health physics practices and an ALARA approach. The evaluation includes, as appropriate, review of proposed special tools, mock-ups and special training, communications needs, manpower requirements, and health physics coverage. Potential incidents such as fires, spills, equipment failure, and other unusual conditions are evaluated and proper response action discussed with the HP personnel and the workers' supervisory personnel, when applicable. For high exposure work, job planning includes person-rem estimates, comparison with similar jobs, establishing person-rem exposure goals, and simulated mock-up operations, as appropriate, to increase job efficiency and keep radiation exposures ALARA.

The RWP is approved and signed by an HP Supervisor or designated alternate prior to commencement of work. The RWP implementation process is detailed in plant procedures.

The ALARA Specialist selectively reviews completed and returned RWPs. Arrangements are made, when necessary, to hold a de-briefing meeting with the responsible supervisor and workers. Debriefing and RWP review is conducted when unexpected airborne concentrations, high person-rem exposures or high individual exposures are encountered. These reviews emphasize and analyze problems or difficulties encountered during performance of work. Alternative work methods are discussed and, where improvements are practicable, the responsible supervisor initiates the review, approval and implementation process.

12.5.3.2.1.3 Work scheduling

Use of the Radiation Work Permit system establishes a data base from which supervisory staff is able to efficiently schedule workers. Health physics personnel can provide reports to supervisors that indicate current individual radiation exposure status to assist in work scheduling and assure that individual exposures are minimized.

12.5.3.2.1.4 Job planning and exposure goals

Filling out an RWP is a form of job planning. The responsible supervisor ensures that individuals selected to perform the task are familiar with the appropriate procedures to be employed. When applicable, a tool list to include special tools that reduce exposures is completed and reviewed. When practicable, the responsible supervisor observes dry-run procedure performance. Special emphasis is placed on job planning for work in high radiation areas to maximize the use of temporary shielding and distance and minimize the work time.

On major dose accumulating job functions, total person-rem exposure goals are established prior to commencement of scheduled work. A general goal is based on the lowest dose commitment recorded on jobs of similar nature. A general goal of equaling or bettering the lowest total work time expended on jobs of similar nature may be used when airborne concentrations or dose rates are unpredictable or subject to variations. These general goals may be modified if work tasks are not identical, or estimated if there are no available historical data. Significant deviations above established goals will be investigated by HP personnel. Methods to improve performance on future jobs will be investigated and implemented, if appropriate.

12.5.3.2.1.5 ALARA program reviews

In an effort to provide more efficient methods of control, evaluation, and reporting, the ALARA Specialist and HP supervisory personnel periodically conduct reviews of the RWP program and procedures used to minimize personnel radiation exposure. Results of internal reviews are reported to appropriate levels of plant management. In addition, the HP group performs special reviews or studies requested by corporate committees to assist management in assuring that all aspects of the ALARA program are implemented.

12.5.3.2.1.6 Worker's recommendations

An informal mechanism of soliciting worker's recommendations for improvement of job efficiency is used to evaluate alternative work methods. Supervisors encourage workers to present alternatives that will reduce work time in radiation areas and airborne concentrations. Responsible supervisors may consult with the HP group during or following evaluation of a recommended change to ensure that individual and group radiation exposures are not adversely affected. Changes in methods or equipment that are anticipated to improve efficiency and reduce radiation exposure are reviewed, approved and implemented in accordance with plant procedures.

12.5.3.2.2 External ALARA

12.5.3.2.2.1 Administrative limits

Administrative limits are implemented by plant procedures to maintain personnel exposures ALARA with respect to federal regulations. A tier of plant radiation exposure limits is established requiring approval of successively higher management levels in order to receive successively higher exposure. HP personnel will investigate unapproved radiation exposures exceeding plant limits to identify causes and establish methods to prevent recurrence.

12.5.3.2.2.2 Dosimeter evaluations

Each entry to the RCA requires each worker to wear at least one dosimeter that provides a real-time integrated dose reading to the wearer (i.e., is direct-reading). Upon completion of an RCA entry, or at least once per shift if multiple RCA entries are made, the net dosimeter reading, entry time(s), exit time(s), worker identification, and job identification are recorded. This is normally accomplished using the login/logout features of an exposure monitoring system.

Recording entry and exit times allows total person-hours spent on particular tasks to be tabulated. A comparison of radiation exposure rate multiplied by person-hours expended and measured individual or group dose totals may be made to assure proper data entry and verify that no significant exposure rate changes occurred. The person-hours expended are also used as a data base to assist the plant staff in planning work of similar nature.

12.5.3.2.2.3 Special alarms and instruments

The use of special alarms and instruments will be evaluated. Electronic dosimeter alarms are used to warn workers that they are approaching the maximum allowable dose or work time, or that they are exceeding the dose rate expected in the work area. Remote radiation monitors may be installed in the general work area to allow readouts in lower radiation areas. Portable

survey instruments may be placed in high radiation work areas to allow workers to monitor changes in exposure rate. Radiation rate meters with audible pre-set alarms may be used to warn workers of unexpected radiation levels.

12.5.3.2.2.4 Temporary shielding and special tools

During the planning phase of RWP work, qualified HP personnel will evaluate the use of temporary shielding. Care is taken to ensure that installation and removal of shielding does not cause larger person rem total exposures than expected without its use. Every reasonable effort is made to use temporary shielding (such as lead blankets), that can be quickly installed on initial entry and easily removed upon exit.

Every reasonable effort is expended to ensure that any necessary, special or modified tools are available for specific tasks. Tools known to be available that significantly reduce stay time in radiation areas and maximize distance from radioactive sources are included on job procedure tool lists. Appropriate personnel review tasks to identify procedures that may be improved by modifications or replacement of tools and/or apparatus.

12.5.3.2.2.5 Non-RWP work review

Health physics personnel review radiation surveys to identify areas not normally meeting RWP criteria. These areas are studied to locate those of the highest occupancy frequency and/or duration of stay time. Health physics may make recommendations pertaining to shielding or occupancy limits to comply with changing plant conditions. These recommendations are implemented whenever practicable to assure that the exposures incurred in low dose rate areas are as low as reasonably achievable.

12.5.3.2.3 Internal ALARA

To minimize potential intake of radioactive material in excess of federal limits, plant limits are established. Airborne radioactivity concentrations in excess of these limits may require engineering controls, work restrictions, use of respiratory protection, and/or special *in vivo* or bioassay studies.

12.5.3.2.3.1 Engineering controls

Minimizing airborne radioactivity concentrations by utilizing practicable engineering or physical controls ensures that occupational exposures are as low as reasonably achievable. Airborne concentrations are minimized by the appropriate use of containment techniques, temporary exhaust mechanisms, and the review of air flow patterns and velocities. Control and evaluation of airborne radioactivity is described in Section 12.5.3.5.

12.5.3.2.3.2 Respiratory protection

When engineering controls are not practicable or sufficient, the use of respiratory protection is evaluated. Respiratory protection may be used to minimize the intake of radioactive material. The respiratory protection fitting and training program is described in Section 12.5.3.5.

12.5.3.2.3.3 Pre-Work air surveys

When RWP requests indicate that work is required in areas containing potential airborne radioactive material, appropriate air samples are taken. Any area that is posted as an airborne radioactivity area is sampled and analyzed prior to commencement of scheduled work. Whenever practicable, surveyors use respiratory protection and/or remote air sampling techniques to minimize their exposures. When existing airborne radioactive materials are not specifically identified, the DAC (Derived Air Concentration) for unidentified alpha and/or beta-gamma materials as specified in 10 CFR 20, Appendix B, is used for scheduling work, calculating anticipated DAC-hours of exposure, and determining the need for respiratory protection.

When applicable, samples of breathing zone air are taken with portable air samplers equipped with appropriate filter media during work in actual or potential airborne radioactivity areas. The data from analyses of these air samples is used to assist in future job planning and to demonstrate that exposures to airborne material are as low as reasonably achievable.

Continuous air monitors are placed in representative areas to sample those locations where airborne concentrations may be generated. These samplers are checked periodically to verify proper function and assure that unexpected airborne concentrations are detected at the earliest possible time. The air sampling program is described in Section 12.5.3.5.

12.5.3.2.3.4 Control of absorption and ingestion

When work is scheduled on equipment or systems that contained or may contain radioactive liquids, every reasonable effort to prevent skin contact with radioactive solutions is made. Items such as plastic suits, rubber gloves, rubber boots, face shields and hoods may be used as appropriate.

The ingestion of radioactive materials is minimized by ensuring that adequate protective equipment is properly worn, removed, stored, laundered and surveyed. These physical controls in conjunction with administrative requirements and training in the areas of self-survey, prohibition of eating and smoking in contaminated areas and decontamination techniques ensures that potential ingestion of radioactive material is minimized.

12.5.3.2.3.5 Control of area and equipment contamination levels

Contaminated areas and equipment are decontaminated to as low a level as reasonably achievable in accordance with plant procedures. Each worker is responsible for ensuring that work areas are maintained in a neat and orderly manner. The housekeeping practices employed facilitate clean-up and decontamination efforts and thus minimize personnel stay time in radiation/contamination areas, and reduce the likelihood of an airborne radioactivity hazard.

12.5.3.2.3.6 Airborne exposure evaluation

Exposure to airborne radioactive material is evaluated in accordance with 10 CFR 20.1202 thru 20.1204. When respiratory protection is employed, appropriate reductions of intake are based upon recommended protection factors. Section 12.5.3.5 describes the respiratory protection program.

12.5.3.2.4 Operational considerations and procedures

Methods to maintain exposures ALARA are in some cases included in RWPs, and are also contained in applicable operating and maintenance procedures. Some examples of methods that are used to maintain exposures ALARA are discussed for the following operations:

12.5.3.2.4.1 Refueling

After the RCS is depressurized, it is degassed and sampled to verify that the gaseous activity is low prior to removing the reactor head. After flooding the reactor well and the refueling cavity, purification of the pool water is continued to maintain minimal radioactivity in the water and, therefore, radiation exposures ALARA.

Some examples of procedural methods of maintaining exposures ALARA during refueling are:

1. Refueling cavity water is maintained less than 140 F, and surface ventilation is provided to minimize airborne radioactive material,

2. Prior to removing the vessel head, the RCS is degassed to minimize expected airborne levels when the head is removed,

3. Movement of irradiated fuel assemblies is accomplished with the assembly maintained under water,

4. Work performed in the controlled area is staged, i.e., workers are briefed on assignments and familiarized with procedures and equipment needed to complete assignments,

5. Current radiological survey information is used,

6. Ventilation is provided to minimize airborne radioactive material, and

7. The radiation work permit system is used to maintain positive radiological control over work in progress.

12.5.3.2.4.2 In-Service inspection

Some examples of procedural methods of maintaining exposures ALARA during in service inspection are:

1. Equipment is calibrated and checked prior to entry into the radiation area,

2. Portable shielding is used where practicable,

3. Workers are briefed on assignments and are familiar with procedures and equipment needed to complete assignments,

4. Current radiological survey information is used,

5. Ventilation is provided to minimize airborne radioactive material, and

6. The radiation work permit system is used to maintain positive radiological control over work in progress.

12.5.3.2.4.3 Radwaste handling

Some examples of procedural methods of maintaining exposures ALARA during radwaste handling are:

1. The volume of radwaste handling has been minimized by plant design,
2. Radwaste systems are heavily shielded and remotely located so that operator and other personnel exposure is minimized,
3. Extension tools are used when practical,
4. Portable shielding is available for use as necessary,
5. Ventilation is provided to minimize airborne radioactive material from waste handling operations, and
6. Current radiological survey information is used.

12.5.3.2.4.4 Spent fuel handling, loading and shipping

Some examples of procedural methods of maintaining exposures ALARA during spent fuel handling are:

1. The spent fuel pool water is filtered to remove radioactive material,
2. The spent fuel pool water is cooled and surface air ventilation is provided to minimize airborne radioactive material,
3. Loading of the shipping cask is performed under water,
4. Fuel handling cranes and extension tools are used to handle shipping casks, fuel assemblies, and inserts,
5. Movement of irradiated fuel assemblies is accomplished with the assemblies maintained under water,
6. Workers are briefed on assignments and are familiar with procedures and equipment needed to complete assignments,
7. Current radiological survey information is used,
8. The radiation work permit system is used to maintain positive radiological control over work in progress,
9. Ventilation is provided to minimize airborne radioactive material, and

10. After the shipping cask is loaded, it is decontaminated using a pressurized water washing device.

12.5.3.2.4.5 Normal operation

Some examples of procedural methods of maintaining exposures ALARA during normal operation are:

1. The plant is designed so that significant radiation sources are minimized and shielded,

2. An area radiation monitoring system is available and provides indication of radiation levels and, as applicable, local and/or remote alarms,

3. Workers are briefed on assignments and are familiar with procedures and equipment needed to complete assignments,

4. Current radiological survey information is used,

5. Ventilation is provided to minimize airborne radioactive material,

6. The radiation work permit system is used to maintain positive radiological control over work in progress,

7. During initial start-up, neutron and gamma dose rate surveys were performed to verify shielding adequacy, and

8. Areas are conspicuously posted in accordance with 10 CFR 20 and plant procedures

12.5.3.2.4.6 Routine maintenance

All maintenance work at SHNPP that involves systems that contain, collect, store, or transport radioactive materials and may cause radiation exposure requires a radiation work permit (RWP). The RWP specifies radiological hazards associated with a job and the safety precautions required for performing the job.

When applicable, procedures specify portions of radioactive systems or components that are to be isolated, flushed, and/or drained. This reduces the radiation levels in the maintenance area.

Some examples of procedural methods of maintaining exposures ALARA during maintenance are:

1. Equipment is moved to areas with lower radiation and contamination levels for maintenance when practicable,

2. Extension tools are used when practical,

3. Portable shielding is used as practical,

4. Workers are briefed on assignments and familiarized with procedures and equipment needed to complete assignments,

5. Current radiological survey information is used,

6. The radiation work permit system is used to maintain positive radiological control over work in progress,

7. Routine maintenance is proceduralized and precautions specified, and

8. Required tools are specifically listed in procedures where practical.

12.5.3.2.4.7 Sampling

Most of the sampling of radioactive systems is performed in chemical fume hoods. The fume hoods provide negative air pressure and prevent the spread of possible contamination. Special protective clothing and equipment are specified in the RWP. Where applicable, survey meters are used to monitor radiation levels at the fume hood and on the sample container.

The possibility of radioactive spills and radiation exposure is maintained ALARA during sample transport by the use of multiple containers and sample shields as appropriate based on sample dose rates.

12.5.3.2.4.8 Calibration

Some examples of procedural methods of maintaining exposures ALARA during calibration are:

1. Visual indication that the calibration room source is in use,

2. Use of the calibration room requires a radiation work permit.

3. The calibration source has a locking device used when the source is not attended.

4. The calibration source has a microprocessor control which requires a user name and password to actuate the source.

12.5.3.3 Radiation Surveys

The health physics program uses radiation surveys to document plant radiological conditions and identify sources of radiation that contribute to occupational radiation exposure. Health physics personnel normally perform radiation and contamination surveys of all accessible areas in the plant. The surveys are performed on an appropriate frequency, depending on the probability of radiation and contamination levels changing, and the frequency with which the areas are visited. Surveys related to specific operations and maintenance activities may be performed prior to, during, and/or after the activity, based on information required to keep radiation exposures ALARA.

Radiation level surveys may be performed for alpha, gamma, beta, and/or neutron exposure rates. Availability of current survey information aids in keeping exposures ALARA.

The radiation survey program is subject to evaluation by HP supervision to ensure that necessary data are collected while exposures to surveyors are as low as reasonably achievable.

12.5.3.3.1 Radiation survey program controls

12.5.3.3.1.1 Record review

A member of HP supervision or his/her designee reviews the record(s) completed by survey technicians to assure proper data entry. If a need for additional data is noted, supervision ensures that such readings or supplemental surveys are taken and recorded. In addition, supervision reviews data to ensure that unwarranted readings that contribute to time spent in radiation areas are not taken. If appropriate, HP supervision ensures that proper corrective measures are taken.

12.5.3.3.1.2 HP technician dose evaluation

Every reasonable effort is expended to assure that occupational radiation exposure to HP technicians is maintained ALARA. HP technicians' radiation exposure is tabulated in accordance with the ALARA program described in Subsection 12.5.3.2. Health physics personnel are issued appropriate dosimetry. Beginning and ending dosimeter readings are recorded. This dosimeter data is updated to reflect group person-rem exposures incurred during radiation survey and other HP technician duties. Analyses of exposures allow investigation and implementation of methods to control and minimize the radiation exposure of HP personnel.

Every reasonable effort is made to use work assignment scheduling and rotation of HP personnel to assure that HP technician exposure is evenly distributed. This rotation allows comparison of surveyor performance, minimization of individual exposures, and assurance that familiarity with all areas of the plant is maintained.

Deviations from approved health physics procedures or discrepancies in radiation measurements are investigated and appropriate corrective actions are taken to prevent recurrence.

12.5.3.3.1.3 Training

Continuing training is provided for HP technicians. This training includes high radiation area survey techniques, data evaluation and special instrument operation. Training sessions emphasize the importance of collecting necessary data while exercising the factors of time, distance and shielding to minimize occupational exposures.

Training sessions will be held as needed to assure state-of-the-art understanding or improved performance in areas where reviews have indicated the need for additional training.

12.5.3.3.2 Radiation survey program

12.5.3.3.2.1 Instrument selection

Plant health physics procedures describe the instrument type(s) to be used during radiation survey work. The survey technician is required to enter instrument descriptions(s) and identification number(s) on survey forms. Prior to performing a radiation survey, the technician checks the calibration status of the portable instrument(s) selected for use. The instrument selected is checked for battery strength, if applicable, and, at least one scale's response to

known check sources(s) is verified. Personnel are instructed to report instrumentation suspected to be malfunctioning.

12.5.3.3.2.2 Radiation area surveys

Routine surveys within radiation areas are performed in a manner that minimizes HP technician dose. Every reasonable effort is made to use readings from the Area Radiation Monitoring System (ARMS) to identify changes of radiation levels. In addition, radiation surveys are taken at the entrances to high radiation areas on a frequency dependent upon occupancy in the vicinity variation in radiation levels and plant experience. If surveys at entrances or ARMS readings show significant change, additional surveys may be performed to update the readings within the area. In order to minimize occupational exposure of surveyors, high radiation area survey frequency may be reduced when operating conditions are stable.

The frequency of routine surveys is based on plant experience. Areas subject to variations in radiation levels or increased time of occupancy may be surveyed on a more frequent basis, as appropriate. When reactor conditions are operationally stable, survey frequency in radiation areas may be reduced to spot checks at the boundaries to minimize HP personnel exposures.

Areas in and around the RCA not considered potential radiation areas are periodically surveyed to establish that every reasonable effort has been made to keep measurable radiation levels as low as reasonably achievable. Portable instrument surveys are performed to assure a representative number of non-radiation areas are surveyed periodically or as conditions require. Areas subject to significant change or variation are surveyed on a more frequent basis as appropriate. Any area, not previously noted, that is found to be a radiation area is promptly posted with a "Caution Radiation Area" sign and reported to HP supervision.

12.5.3.3.2.3 Radiation Work Permit (RWP) surveys

A member of the HP staff screens incoming RWP requests to assure inclusion of special measurements or considerations.

Special radiation surveys are performed as requested by operating groups, regulatory agencies, or corporate committees. These survey requests are coordinated by HP supervision to assure the need for the survey justifies occupational exposure of the survey technician. A member of HP staff may draft special instructions for performance of the survey. Emphasis is placed on ensuring that necessary data are collected with a minimum of exposure.

12.5.3.3.2.4 Deleted by Amendment No. 15

12.5.3.3.2.5 Survey records

Radiation surveys are documented, reviewed, and retained in accordance with approved plant procedures. Records retention complies with Section 17.3.4.3 of the FSAR.

12.5.3.4 Contamination Survey Procedures

Periodic contamination evaluation surveys are utilized to minimize the spread of radioactive material. Surveys of personnel, equipment and surface contamination are also made to demonstrate the efficiency of engineering and procedural controls. Contamination surveys are

normally performed to establish gross beta-gamma contamination levels, but may be processed for specific types of radiation (alpha, beta, gamma) or specific nuclides. In addition, the contamination survey program is evaluated to assure that survey personnel exposures are ALARA.

12.5.3.4.1 Personnel contamination surveys

Evaluation of exposures due to personnel contamination is conducted in accordance with Section 12.5.3.6.

Personnel friskers or hand/foot monitors are placed in strategic locations within the RCA. Every effort is made to locate these instruments in as low a radiation background area as possible in order to maximize sensitivity. Personnel are trained in the use of the instrument(s) and interpretation of the readings.

Whole body friskers are installed at exits from the primary RCA and personnel are required to frisk themselves prior to exiting.

In the event of frisker malfunction, personnel are requested to notify HP personnel. Audible and/or visible alarms are present at a suitable point above background to minimize spurious alarms and maximize sensitivity.

Personnel contamination causing frisker alarm requires notification of HP personnel. HP personnel take appropriate actions to minimize further spread of contamination, and direct appropriate decontamination of affected areas and personnel. When personnel contamination is noted, a health physics investigation appropriate to the incident will be performed. A contamination incident found to have caused an ingestion of radioactive material will be promptly reported to appropriate supervision.

If contamination in excess of plant limits is detected in or on the mouth, a shower and a whole body count will be performed. Fecal and/or urine collection may be initiated to more accurately determine ingested amounts. All intakes resulting in greater than ten percent ALI (Annual Limit for Intake) will be investigated, evaluated, documented and reported to appropriate supervision and appropriate corrective measures will be taken.

To control inadvertent entry of radioactive material in wounds, cuts or abrasions, individuals will be responsible for bringing such matters to the attention of supervisors and/or HP personnel prior to beginning work. Supervisory personnel will assure that reported skin breaks are brought to the attention of the HP group during job planning or when requesting an RWP. HP personnel will be responsible for assuring that skin breaks are properly protected prior to beginning work. Open wounds that cannot be adequately sealed will be sufficient grounds to restrict the worker from work involving contamination.

Any injury that may have caused contamination of a wound requires the worker to immediately exit the work area and report the incident to radiation protection supervision, specifically the HP Supervisor. The wound will be flushed, then surveyed with portable instrumentation. If contamination is detected in the wound, the HP Supervisor will initiate action in accordance with a personnel decontamination procedure. If injury is sufficient to prevent the worker from moving or exiting the area, the Superintendent - Shift Operations will be immediately notified and the

Emergency Plan will be initiated, if appropriate. Whole body counts and/or bioassays will be performed, as appropriate, following any needed medical treatment.

12.5.3.4.2 Equipment contamination surveys

12.5.3.4.2.1 RCA equipment surveys

Movement of equipment from the RCA requires notification of HP personnel. Fixed and removable contamination levels are evaluated as appropriate and a clearance for removal is issued in accordance with plant procedures.

Routinely used tools may be permanently marked to indicate they are contaminated and are normally stored inside the RCA. Repair or use outside of the RCA requires HP approval. Permanently marked tools are surveyed by HP personnel as necessary and at the request of the appropriate supervisor.

12.5.3.4.2.2 Protective clothing surveys

Reusable protective clothing and shoe covers used in contamination areas are collected in receptacles at step-off areas and sent for laundering/decontamination on site or by an off-site vendor. If clothing is cleaned at the plant's laundry facilities, it is removed from containers, sorted, and scanned with a GM detector or other appropriate instrument to locate highly contaminated items that may require separate decontamination or disposal. Following washing and drying (or dry cleaning), clothing is re surveyed to assure that items are within plant limits. Items returned from vendors are spot checked with survey instruments to ensure that residual contamination levels are less than the applicable plant limits. Every reasonable effort is made to assure that clothing is maintained at as low a contamination level as practicable.

Protective clothing that is shipped off-site for laundering or dry cleaning is prepared for shipment and labeled in accordance with applicable U.S. Department of Transportation and NRC regulations. Records of survey results are maintained for each outgoing shipment per Section 17.3.4.3 of the FSAR.

12.5.3.4.2.3 Respiratory protection device surveys

Respiratory protection device surveys are described in FSAR Sections 12.5.3.5.6.5 and 12.5.3.5.6.6.

Exterior surfaces of other protective devices, such as supplied air hoods and suits, self-contained breathing apparatus and hoses, may be checked for contamination levels following job completion. Items other than face pieces that are routinely reused in contamination zones may be bagged and labeled to reflect the latest survey findings.

12.5.3.4.2.4 Personal item surveys

Change-out procedures require that individuals leaving contamination areas and the RCA perform the appropriate surveys of personal items that may have become contaminated during work. Items such as dosimeters, TLD or badge holders, pens and pencils, are monitored for contamination prior to release. Contamination noted on such items is reported to HP personnel.

Additional surveys are performed and the items decontaminated or discarded as radioactive waste as appropriate.

12.5.3.4.2.5 Surveys involving receipt/shipment of radioactive material

Security and storeroom personnel are instructed to notify HP personnel upon the arrival of radioactive shipments at the site. Shipping containers are monitored for radiation and/or contamination in accordance with 10 CFR 20.1906. Whenever practicable, the container is monitored prior to removal from the vehicle. If removable contamination or radiation levels are found to exceed the limits of 10 CFR 20.1906, HP Supervision will ensure that the final delivering carrier and the Nuclear Regulatory Commission (NRC) Inspection and Enforcement Regional Office are notified as required by plant Technical Specifications.

When applicable, HP supervision ensures that prior to leaving the site, transport vehicle surface contamination and radiation levels are within limits specified in plant procedures.

Plant procedures specify special procedures and precautions to be taken when opening packages containing licensed material, including instructions pertaining to specific types of shipments normally received at SHNPP. Radioactive material is shipped in accordance with U.S. Department of Transportation and NRC Regulations. Plant procedures implement the applicable regulations with regard to proper packaging and labeling requirements. Appropriate removable contamination and dose rate surveys are taken, records completed, and shipments labeled accordingly.

Contaminated equipment to be shipped from the site for repair and return is surveyed for removable and fixed contamination and radiation levels in accordance with plant procedures. The item(s) will be packaged in appropriate containers. The containers are surveyed, labeled, and shipped in accordance with U.S. Department of Transportation and NRC requirements.

12.5.3.4.3 Surface contamination surveys

12.5.3.4.3.1 Radiologially Controlled Areas (RCAs)

A smear survey program is utilized to assure that a representative number of routinely accessible surface areas within the RCA are checked for removable contamination. Special emphasis is placed on survey of the clean side of established contamination area step-off areas. Smears are analyzed on appropriate counting equipment and records of results are maintained. If results indicate removable contamination exceeds plant limits, the area will be posted as a contamination area. The area will be decontaminated and re-surveyed as soon as practicable. Area signs and barriers will be removed when surveys indicate that removable contamination is below appropriate plant limits.

In representative areas where gamma background permits, surveys are performed with portable detectors to establish the level of fixed surface contamination in normally occupied RCAs. A fixed contamination survey is performed prior to any sanding, chipping, welding, grinding and sawing, of potentially contaminated surfaces.

12.5.3.4.3.2 Non-RCA zone areas

Occupied plant areas outside the RCA are surveyed to assure that a representative number of floor surfaces are checked for removable contamination. The exit areas from the RCA receive special emphasis to minimize the spread of contamination. Smear survey, analyses and record keeping techniques are as described above. Non-RCAs found to have removable contamination levels exceeding plant limits are decontaminated and resurveyed.

Lunch room facilities and vending machine areas frequented by RCA workers are checked for removable contamination. Stoves, benches, table tops, and floor surfaces are representatively smeared to assure minimal contamination in eating areas. Removable contamination in excess of release limits are reported to HP or shift supervision and the area is restricted from further use until decontaminated. Special emphasis is placed on eating or cooking surfaces to assure that these items are as far below release limits as reasonably achievable.

Other specific areas are checked for removable contamination to demonstrate the effectiveness of the contamination controls exercised within RCAs. These areas include:

1. The Control Room
2. The Security Building
3. General floor areas of shower and locker room facilities
4. Service Building
5. Turbine Building
6. Administration Building

Floor surfaces in areas that have the potential to be repeatedly contaminated may be maintained as contamination areas to assure positive contamination control. In addition to the routine check outside step-off areas, a general survey of contamination levels inside these areas is performed whenever practicable. Dose rates within the areas, frequency of occupancy, past survey results, and actual need for such surveys are evaluated by HP supervision when selecting established contamination areas to be surveyed. When area dose rates permit, every reasonable effort is made to minimize contamination levels.

12.5.3.4.3.3 Implementation, review, and reporting practices

Contamination limits, general survey locations and survey frequencies are specified in plant health physics procedures. Procedures are subject to review by HP supervision periodically to assure contamination survey implementation is responsive to plant status.

A member of HP supervision or his/her designee reviews the records completed by survey technicians to assure proper data entry. In the event of contamination in excess of station limits, a member of HP supervision is responsible for ensuring that corrective measures are implemented and that further reports through appropriate channels are initiated if required. Records are retained per plant procedures.

12.5.3.5 Airborne Radioactive Material

Every reasonable effort is made to assure that airborne radioactive material within the plant is minimized. A sampling and analysis program is used to determine airborne concentrations in a representative number of routinely occupied areas. These routine measurements as well as special surveys, respiratory protection procedures and administrative procedures are implemented to minimize airborne contamination and the potential intake of radioactive material.

12.5.3.5 Physical controls

12.5.3.5.1.1 Air flow patterns

A survey prior to fuel load was performed to demonstrate that air flow patterns within the RCA are toward areas of higher actual, or expected, airborne contamination. Affected areas are re-surveyed following ventilation modifications to assure proper air movement. Appropriate measures will be taken if flow patterns are found to be unacceptable.

12.5.3.5.1.2 Contamination confinement

Contaminated items are properly confined to prevent inadvertent airborne contamination. Such items are sealed in appropriate material or stored in ventilated areas whenever practicable. When necessary, alternatives such as temporary tents or enclosures, storage in rooms or areas where air movement is away from occupied areas, or wetting or greasing of the item may be utilized to minimize airborne radioactivity. Contaminated trash is sealed prior to disposal whenever practicable. Every reasonable effort is made to assure that contaminated trash receptacles are closed when not in use.

12.5.3.5.1.3 Air exhaust

Exhaust of areas or items where airborne radioactivity may be generated is employed whenever practicable. Contaminated laundry sorting areas, fume hoods, and sampling stations are typical locations where air exhaust is utilized. Exhaust flow rates or face velocities on such equipment is verified after major ventilation modifications to assure proper function. Items that may contain highly contaminated materials such as high level fume hoods can be equipped with a visual indicator or alarm to warn individuals upon loss of exhaust flow. Portable exhaust fans are directly discharged to building exhaust whenever practicable. When discharge to building exhaust is not practicable, the portable exhaust fan is filtered to minimize airborne radioactivity.

12.5.3.5.1.4 Posting

Accessible areas containing concentrations of airborne radioactivity exceeding the limits specified in 10 CFR 20.1003 are posted with a "Caution - Airborne Radioactivity Area" sign.

12.5.3.5.2 Administrative controls

12.5.3.5.2.1 Health physics review

A member of HP supervision, or his/her designee, periodically reviews all posted airborne radioactivity areas. Methods to reduce existing airborne radioactivity are forwarded through appropriate channels for review, approval, and implementation. During the review, HP

supervision ensures that every reasonable effort has been made to reduce the risk of inadvertent entry in airborne radioactivity areas.

12.5.3.5.2.2 Health physics investigation

When an occurrence produces unusually high airborne radioactivity in occupied areas, HP supervision ensures that an investigation appropriate to the incident is completed. The first priority is evaluation and follow-up of personnel intake of radioactive material if applicable. The second portion of the investigation emphasizes determination of the events leading to the occurrence. Recommendations to prevent recurrence are forwarded through appropriate plant channels for implementation.

12.5.3.5.2.3 RWP procedures

Radiation Work Permit procedures, as described in Section 12.5.3.2, are a primary administrative control of exposure to airborne radioactive material. HP review prior to approval ensures that every reasonable effort is made to minimize the production of, or reduce existing, airborne radioactivity before work commencement.

12.5.3.5.3 Deleted by Amendment No. 48

12.5.3.5.4 Airborne concentration sampling

Air samples are normally taken to establish airborne concentrations of noble gases, particulates, and/or radioiodine, but specific radionuclide information may also be obtained.

12.5.3.5.4.1 Routine sampling

Routine sampling in selected areas of potential airborne radioactivity is accomplished with continuous air monitors (CAM) or portable air monitors. CAM sampling media and detectors are selected as appropriate to the intended use of the device. CAMs are routinely checked for proper operation. Abnormal readings or equipment malfunctions are reported through appropriate channels for investigation and/or repair. Alarms, if applicable, are checked for operability during source check and calibration procedures. Fixed filter devices are changed on a frequency specified by health physics procedures to assure optimum sampling time, meaningful results, and proper equipment operation.

12.5.3.5.4.2 Special air sampling

Records are maintained to reflect the reason for the special surveys, device(s) used and final results. The majority of special air samples are taken as result of RWP requests and pertinent results are recorded thereon.

12.5.3.5.5 Air sample evaluation

12.5.3.5.5.1 Particulate initial evaluation

A data sheet is completed to reflect sample location, date, starting flow rate, starting time, and sampler used. At completion of sampling, the date, time, and ending flow rate are recorded. Air sample filters are counted as soon as practicable following collection. Results are recorded on

an analysis form to reflect counter used, counting time, background count rate, gross sample count rate, net sample count rate, and sample activity (beta, and/or beta-gamma, and/or alpha). Manufacturer's certification of collection efficiency is used in calculating airborne concentration.

12.5.3.5.5.2 Subsequent particulate evaluations

In instances where time delay before analysis in conjunction with suspected short-lived isotopes is significant, repeated counts may be performed to obtain a decay curve. Extrapolation and subtraction techniques may be used to determine initial amounts and half-lives of component isotopes.

When necessary, fixed filter samples may be gamma scanned with a NaI, Ge(Li), or similar detector to identify gamma-emitting isotopes. When this or other specific analyses are not practicable, the DAC for unidentified beta-gamma emitters is used for exposure evaluation and procedural controls.

Other evaluations that may be used are beta absorption counting, radiochemical separations and analysis, and liquid scintillation counting.

12.5.3.5.5.3 Gaseous evaluations

Airborne radioiodine samples are normally collected on charcoal canister or cartridges, and analyzed on a NaI, Ge(Li), or similar detector. Appropriate standard sources in reproducible geometries are used to obtain efficiency curves for analysis equipment. Photopeak areas, counting efficiency and branching ratios for the identified isotope are used to calculate the amount of deposited activity. Collection efficiency and total volume of sampled air are incorporated to calculate the concentration of airborne radioactivity. If conditions warrant, special materials such as silver zeolite may be used for radioiodine sampling.

Airborne tritium samples are normally collected in water bubblers or desiccant columns. Collection and counting efficiencies and total air volume is verified and used to calculate airborne concentrations.

If analyses of air for noble gases are required, sample chambers may be analyzed with NaI, Ge(Li), or similar detectors to identify isotopes.

12.5.3.5.6 Respiratory protection

The respiratory protection program assures that personnel intake of radioactive material is minimized. The respiratory protection program is not used in place of practical engineering controls and prudent radiation safety practices. Every reasonable effort is made to prevent potential, and minimize existing, airborne concentrations. When controls are not practicable, protective devices may be used to minimize potential intake of airborne radioactive material. Such usage is evaluated to ensure total effective dose equivalent is ALARA.

The SHNPP Respiratory Protection Program ensures that the following minimum criteria are met: written standard operating procedures; proper selection of equipment, based on the hazard; proper training and instruction of users; proper fitting, use, cleaning, storage, inspection, quality assurance, and maintenance of equipment; appropriate surveillance of work area conditions, consideration of the degree of employee exposure to stress; regular inspection and

evaluation to determine the continued program effectiveness; program responsibility vested in one qualified individual and an adequate medical surveillance program for respirator users.

12.5.3.5.6.1 Training and fitting

The training and fitting program is described in Section 12.5.3.7.

12.5.3.5.6.2 Written procedures

The Respiratory Protection Program and program responsibility is implemented by health physics procedures. Applicable health physics procedures include as a minimum: description of equipment; information regarding issuance, maintenance, selection, use, and return of equipment; and training techniques.

12.5.3.5.6.3 Selection of equipment

Health Physics personnel determine the need for respiratory protection after evaluation of any appropriate engineering controls. Airborne concentrations are determined by air sampling methods described in this section. The hazard is evaluated and applicable respiratory protection prescribed in accordance with the RWP evaluation, review, approval and implementation process as described in Section 12.5.3.2.

12.5.3.5.6.4 Issue and use

For normal work situations, respirators are issued after approval of an RWP. Individuals' I.D. cards, electronic database or qualification list are used to assure only the specific models approved for the worker are issued. After issuance, the worker is responsible for proper use of the device. Approved health physics procedures for use, storage and return of respirators are reviewed during qualification training sessions.

12.5.3.5.6.5 Contamination surveys

If desirable, respirators are scanned with a GM detector during final change out procedures upon completion of assigned work. Based upon findings and suspected isotopes, further evaluations may be required in accordance with Section 12.5.3.6.

12.5.3.5.6.6 Cleaning, decontamination, inspection, maintenance, disinfection and storage

Plant procedures specify cleaning, decontamination, survey, inspection, maintenance, disinfection and storage requirements. Respirators are normally used no more than one day (shift) prior to return for cleaning, survey, inspection, maintenance, if needed, and disinfection. In no case is a respirator issued to another individual prior to cleaning, survey, inspection and disinfection. Respiratory face pieces are washed, dried, surveyed for removable and fixed contamination levels, inspected, disinfected and stored in accordance with approved health physics procedures. Inspection of masks emphasizes defects at critical points, proper function of attached fittings and valves, and proper shape of face-piece. Simple maintenance and repair is performed as necessary. Maintenance and repair of regulators are performed only by specially trained and qualified individuals. Every effort is made to assure proper storage of masks to prevent deformation of face-piece parts. Survey results are recorded.

12.5.3.5.6.7 Quality controls

Respiratory protection devices are inspected and tested prior to initial use, after use, and when in storage, at frequencies specified in plant procedures to ensure that they are ready to use. Procedures specify the components of each type of device to be inspected and the acceptance criteria when applicable.

12.5.3.5.6.8 Surveillance of work area conditions

For work conditions involving respiratory protection, air sampling surveillance provides an estimate of the potential intake of airborne radioactive materials and resulting exposure of the individual worker, indicates the continuing effectiveness of existing controls, and warns of the deterioration of control equipment or operating procedures.

The periods of time respirators are worn continuously and the overall durations of use are kept to a minimum by procedural controls and work surveillance. Workers are instructed on provisions for leaving areas where respirator use is required for relief in case of equipment malfunction, undue physical or psychological distress, procedural or communication failure, significant deterioration of operational conditions, or any other conditions that might require such relief.

12.5.3.5.6.9 Evaluation of program effectiveness

HP personnel investigate respirator failures, evidence of respirator leakage, and other equipment problems. Proposed changes to prevent recurrence or improve efficiency of the program are forwarded through appropriate channels for review, approval and implementation.

Bioassays, urinalysis and/or whole body counts are performed in accordance with plant procedures and, as required to evaluate individual uptakes of radionuclides.

Respiratory protection program effectiveness is evaluated by bioassay results correlated with air sampling results as described in Section 12.5.3.6. Evidence of a rise in exposure levels attributable to inhalation is investigated.

12.5.3.5.6.10 Medical surveillance

Prior to wearing respiratory protection equipment in a potentially hazardous atmosphere, individuals are evaluated by competent medical personnel to ensure that they are physically and mentally able to wear respirators under anticipated working conditions.

Individuals involved in the respiratory protection program are also re-evaluated at the required intervals with respect to physiological and psychological factors affecting respirator use.

Details of the medical surveillance program are specified in plant procedures.

12.5.3.5.7 Handling of radioactive material

Recognized methods for the safe handling of radioactive materials, such as those recommended by the National Council of Radiation Protection and Measurement, are

incorporated into procedures to ensure proper usage. Procedures specify handling techniques, storage, and other safety considerations, including:

1. Minimizing the distances over which large radioactive sources are transported,
2. Using shielded transporters,
3. Storing sources in appropriately shielded containers,
4. Labeling all radioactive material properly (per 10 CFR 20),
5. Inventorying all radioactive sources in accordance with plant procedures,
6. Testing sources for leaks as described in Section 12.5.3.5.7.2.
7. Monitoring all packages received containing radioactive material in accordance with 10 CFR 20.1906.

12.5.3.5.7.1 Unsealed material

Radioactive material in liquid form is stored in sealed or vented/exhausted containers whenever practicable. When containers are opened to atmosphere and generation of airborne concentrations is possible, they are opened in fume hoods, exhausted areas, or in locations where air movement is away from workers' breathing zones. Whenever practicable, liquid radioactive material is transported in unbreakable containers or in a secondary container to collect material in case of breakage.

Gaseous radioactive material is similarly stored and opened. Transport of gaseous samples is done in sealed, gas tight containers.

Solid articles that are sufficiently contaminated with particulate and/or volatile material so as to pose a potential airborne hazard are handled and stored as described in Section 12.5.3.5.1.2.

Protective clothing, respiratory protection, and special precautions are specified by health physics procedures and/or RWPs for handling unsealed material.

12.5.3.5.7.2 Sealed materials

Sources are controlled in accordance with plant procedures. When sources produce a whole body or contact radiation dose rate greater than limits established by plant procedure, an RWP is completed and approved prior to use. Remote devices such as forceps, tongs, or manipulators are used whenever practicable or required by the RWP.

Sealed sources are monitored for leakage to assure that storage or use is not causing the spread of contamination or airborne radioactive material. When monitoring of the source capsule is not practicable, removable contamination surveys are performed at places on the container or source holder where contamination might be expected to accumulate if the source were leaking. Samples are analyzed on counting equipment appropriate to the source material, and records of the results are maintained. Test frequency, materials to be tested, and record-keeping requirements of the site's NRC license and plant Technical Specifications are

implemented by plant health physics procedures. Sealed sources found to be leaking are sealed from atmosphere whenever practicable and/or stored in ventilated areas until disposal or repair.

12.5.3.6 Personnel Monitoring

12.5.3.6.1 External personnel monitoring

Personnel monitoring devices are used at SHNPP to evaluate external occupational exposure to radiation sources. Exposure information is used for work function exposure evaluation, job planning, reporting requirements, incident analysis, and an indication of the effectiveness of ALARA practices.

12.5.3.6.1.1 Personnel dosimetry evaluation

Individuals requiring personnel dosimeters are instructed in the purpose and use of the devices, plant administrative exposure limits, and the interpretation of direct reading dosimeter displays. Appropriate dosimetry devices are issued in accordance with plant procedures implementing 10 CFR 20.1502.

A dosimeter is normally worn on the front of the body between the neck and the waist in a clearly visible location but the dosimeter may be worn elsewhere as directed by HP personnel. When appropriate, dosimeters are issued and worn on the extremities and/or other body locations. Dosimeters may be packaged to prevent contamination of these devices when entering contaminated areas.

As described in Section 12.5.3.2, self-reading dosimeter results are used for specific ALARA job exposure evaluation as well as to indicate current individual exposure status. Dosimeters of appropriate ranges are available for use during work in radiation and high radiation areas. Radiation workers are responsible for checking their dosimeter readings when working in RWP areas. The frequency of dosimeter checking depends upon the nature of the job and whole body dose rates, and may be discussed with the radiation workers during RWP pre-job planning. Off-scale or malfunctioning dosimeters are reported to Health Physics. HP personnel evaluate the occurrence, issue a replacement dosimeter and test the suspect dosimeter. Dosimeters are removed from service if the test results exceed acceptance criteria specified in the plant HP procedures.

Self-reading dosimeters are used to monitor gamma exposure only. Methods of calculating neutron dose equivalent using dose rate information are specified in plant HP procedures.

TLDs or electronic dosimeters are used as the dosimeters of record as specified in plant procedures. Personnel dosimeter readings are evaluated as determined by HP supervision. The data obtained from TLDs is evaluated to determine shallow and deep dose equivalents. For the purpose of comparison to applicable dose limits, the shallow and deep dose equivalents are attributed to the skin of the whole body and the whole body, respectively. The dose equivalents from extremity monitoring devices are normally reported as the extremity dose. However, under some situations, the dose equivalent from extremity monitoring devices is added to the whole body dose equivalent for comparison to the extremity dose limits. TLDs are read and processed by a NVLAP-accredited dosimeter processor.

12.5.3.6.1.2 Administrative exposure control

Administrative exposure limits are established and implemented by HP procedures to assure the limits of 10 CFR 20.1201 are not exceeded and personnel occupational exposures are maintained ALARA.

12.5.3.6.1.3 Methods of recording and reporting

Designated supervisors receive reports of their employees' accumulated exposures for use in RWP job planning and scheduling. Updates of exposure totals are compiled from self-reading dosimeter readings. Unapproved exposures exceeding station limits will be reported to the facility Vice President and appropriate supervision, and investigated by HP personnel to identify causes and establish methods to prevent recurrence.

Occupational radiation exposure received during previous employment is used in preparation of individuals' Forms NRC-4, or equivalent. When an individual's occupational exposure history cannot be obtained, the values specified in 10 CFR 20.2104(e) are used to establish the individual's administrative limits. Records used in preparing Form NRC-4, or equivalent, are retained and preserved until the NRC authorizes disposition.

Records of the radiation exposure of all individuals issued personnel dosimetry in accordance with 10 CFR 20.1502 are maintained on Form NRC-5, or equivalent. Exposures are tabulated for periods not exceeding one calendar year. Records of radiation exposure received during employment at SHNPP are maintained until NRC authorizes disposal.

Reports of exposure to radiation or radioactive materials are made to individuals as specified in 10 CFR 19.13. When reports of individual exposure to radiation or radioactive material are made to the NRC, the individual concerned is also notified. This notice is forwarded to the individual at a time no later than the transmittal to the NRC and complies with 10 CFR 19.13.

A report of each individual's exposure to radiation or radioactive material incurred while employed or working at SHNPP is furnished to the NRC in accordance with 10 CFR 20.2206.

In the event of an exposure in excess of 10 CFR 20.1201 limits,

HP supervision will investigate the event and document the description of the occurrence; conditions under which the exposure occurred; names of personnel involved and amount of exposure received; action taken at time of occurrence; recommendations for corrective measures and means of implementation to prevent a similar occurrence.

In the event of an unauthorized exposure in excess of plant administrative limits, HP supervision will investigate the event to determine the cause(s). Recommendations for corrective measures will be forwarded for review, approval, and implementation in accordance with plant procedures.

Reports of overexposures at SHNPP will be submitted to the NRC and the individual(s) involved in accordance with 10 CFR 19.13 and 10 CFR 20.2203. Reports will also be forwarded to appropriate committees for review and recommendation for follow-up action.

12.5.3.6.2 Internal radiation exposure assessment

When engineering controls are impracticable, airborne concentrations exceed plant limits, and the use of respirators will maintain the Total Effective Dose Equivalent ALARA, trained individuals are equipped with properly fitted respirators. Internal exposure evaluations are used to determine the effectiveness of the Respiratory Protection Program and to evaluate suspected intakes of radioactive material. The Respiratory Protection Program is described in Section 12.5.3.5. Whole body counting and/or bioassay techniques are used to compare the quantity of radioactive material present in the body to that quantity which would result from inhalation for a working year of 2000 hours at uniform airborne concentrations specified in 10 CFR 20 Appendix B, Table 1, Column 3, or the Annual Limit on Intake (ALI) specified in Appendix B, Table 1, Column 1 (for ingestion) or Column 2 (inhalation).

12.5.3.6.2.1 Bioassay methods

Whole body counting is used to qualitatively and quantitatively identify radionuclides deposited in the body that emit penetrating radiations. Depending upon the physical construction and geometry of the whole body counter, sensitivity of the detector(s), and biological factors, concentrations of radionuclides may be detected in the whole body, thyroid, lung, or wounds.

Urine analysis may be conducted to identify the presence of alpha or beta emitters in extracellular body fluids. Under favorable circumstances, with a full 24-hour sample and further analyses, the amount of radionuclides may be qualitatively and quantitatively determined. Results may be used to substantiate in vivo analyses findings.

Fecal analysis is normally used to evaluate intake of non-transportable (i.e., insoluble) material and provide evidence of the clearance of such material from the lungs. When it is suspected that a non-transportable radionuclide has been inhaled, the total amount excreted in feces during the succeeding few days may be used to estimate the amount initially deposited in the lungs. Standard lung models recommended by International Commission on Radiological Protection (ICRP) may then be used to evaluate the amount inhaled.

The dose commitment for internal deposits may be estimated by calculating the amount of airborne radioactive material inhaled, based on airborne radioactive material measurements, exposure times, standard lung models and breathing rates.

12.5.3.6.2.2 Administrative controls

Records, approved plant procedures, program reviews, and investigations assure proper administrative control over the internal personnel monitoring program. Reviews of the internal personnel monitoring program and investigations of individual cases of suspected or known intakes are performed and documented by HP supervision and reported to appropriate committees.

12.5.3.6.2.3 Criteria for participation or selection

Selection of personnel, frequency of routine whole body counting, and bioassay analyses are implemented by health physics procedures.

The following are guidelines for participation in special whole body counting and/or bioassay analyses:

1. Personnel suspected of having contamination in the nasal passages in excess of limits specified in health physics procedures.

2. Personnel suspected to have ingested a detectable level of radioactive material, or absorbed a detectable level of radioactive material through a wound or break in the skin.

The following are guidelines for selection of personnel for special, non-routine urine analysis:

1. When there is suspicion of an intake of a pure beta or pure alpha emitter.

2. In conjunction with non-routine fecal analysis.

In addition to the above criteria, personnel may be required to submit urine samples to evaluate clearance rates of radioactive material identified by special or routine whole body counts, or as directed by HP supervision.

Fecal sampling and analysis is normally done on a non-routine basis as designated by HP supervision. Fecal analysis may be done as a follow-up on whole body or lung counts.

12.5.3.6.2.4 Evaluation and reporting

Identifiable deposits are evaluated against the criteria of 10 CFR 20.1204 assuming conservative conditions and time frames with respect to the time of intake. Reports are generated as required by 10 CFR 20.2203. The reports are reviewed by appropriate supervision and maintained on file subject to NRC inspection.

Specific organ counting may be performed if appropriate. Organ content may be assigned using whole body measurements and ICRP-2 recommended fractions and clearance times, when organ counting is not possible. Dose commitment to blood forming organs, gonads, whole body or eyes resulting from deposits in other organs may be calculated using ICRP 30 data.

12.5.3.7 Health Physics Training Programs

Health physics training programs assure that personnel, who have unescorted access to the restricted area, possess an adequate understanding of HP to maintain occupational radiation exposures ALARA. Special training/retraining is administered upon recommendation of the Training Supervisor or HP Supervisor. Record-keeping and training scheduling are performed by the training supervisor or designated alternate. This program covers the following:

1. General employee health physics,

2. General employee respiratory protection,

3. Contractor health physics,

4. Contractor respiratory protection,
5. General employee retraining,
6. HP technician training,
7. HP technician retraining.

Training and testing may be performed using computer-based training.

12.5.3.7.1 Health physics training

All persons allowed unescorted access to the restricted area will demonstrate proficiency in the following areas as evidenced by passing an examination:

1. Requirements of 10 CFR 19.12,
2. Radiation/Contamination (examples and control),
3. ALARA (Corporate commitments, meaning and individual responsibility),
4. Personnel Monitoring and Self-Survey Requirements,
5. Radiological Control Signs and Posting Requirements,
6. Radiation Exposure Control and Limits,
7. Radiation Emergency Plan and Applicable Procedures,
8. Prenatal Radiation Exposure.

Additional health physics training is normally administered to individuals who require access to the RCA. Additional training is provided for personnel who enter contamination areas. The need for such training is evaluated and scheduled by the Training Supervisor, or designated alternate. Training is administered to provide radiation workers with an adequate knowledge to effectively cope with job situations while maintaining radiation exposures as low as reasonably achievable. The individual demonstrates proficiency as evidenced by passing an examination. Depending on the type of areas the individual will enter, this training will include one or more of the following:

1. ALARA (applicable procedures),
2. Contamination Control and Self-Survey Requirements,
3. Fundamentals of Radioactivity,
4. Radiation Dose Units and Biological Effects,
5. Radiation and High Radiation Area Survey Techniques,

6. Principles of Radiation Safety (Time, Distance and Shielding),

7. Radiation Work Permits (RWP),

8. Use of protective clothing/devices.

To assure individual proficiency in HP practices, periodic retraining and retesting will be performed. Scheduling, records, and test results are maintained by the Training Supervisor or designated alternate. Individuals changing job classification receive training of the level required by their new job classification.

Training/retraining is administered, under the direction of the Training Supervisor or designated alternate, to candidates for Nuclear Regulatory Commission (NRC) operating licenses and those holding NRC licenses.

12.5.3.7.2 Respiratory protection training program

Individuals requiring access to areas where respiratory protection will be used, and their supervisors, are required to complete the Respiratory Protection Training Program. The instructor is a qualified individual with a thorough knowledge and considerable experience regarding the application and use of respiratory protective equipment and the hazards associated with radioactive airborne contaminants.

Training includes lectures, demonstrations, discussions of pertinent plant procedures, and actual wearing of respirators to become familiar with the various devices utilized at SHNPP. The program includes as a minimum: discussion of the airborne contaminants against which the wearer is to be protected, including their physical properties, DACs, physiological action, toxicity, and means of detection; discussion of the construction, operating principles, and limitations of the respirator and the reasons the respirator is the proper type for the particular purpose; discussion of the reasons for using the respirators and an explanation of why more positive control is not immediately feasible, including recognition that every reasonable effort is being made to reduce or eliminate the need for respirators, and that use of respirators may not keep Total Effective Dose Equivalent ALARA; instruction in procedures for ensuring that the respirator is in proper working condition; instruction in fitting the respirator properly and checking adequacy of fit; instruction in the proper use and maintenance of the respirator; discussion of the application of various cartridges and canisters available for air purifying respirators; instruction in emergency action to be taken in the event of malfunction of the respiratory protective devices; review of radiation and contamination hazards, including the use of other protective equipment that may be used with respirators; classroom and field training to recognize and cope with emergency situations; and other special training as needed for special use.

Individuals are required to don the device(s) that may be used, perform appropriate pressure tests for leak detection, and be exposed to a challenge atmosphere. If excessive leakage is detected, the individual is not approved to use the device. After successful completion of training and fitting programs, appropriate records are maintained to assure individuals are issued only the approved type and model of protective device(s). These records reflect expiration dates. Individuals will receive retraining and reconfirmation of respirator fit on an annual basis or prior to use. Fit test records are maintained in accordance with plant procedures.

12.5.3.8 Calibration/Periodic Testing

Instruments are subjected to operational checks and calibration to ensure the accuracy of measurements of radioactivity and radiation levels. Laboratory standards and sources traceable to nationally recognized standards are used to perform calibration and testing.

Instrument calibration may be performed by a qualified vendor.

Records of calibration and repair for portable radiation survey instruments are reviewed and retained in accordance with plant procedures.

Sources are controlled in accordance with plant procedures.

12.5.3.8.1 Survey instruments

Portable radiation survey instruments are normally calibrated in the calibration room using a calibration apparatus and appropriate radiation sources. The calibration apparatus requires the use of sources of varying strength and energy and/or varying thicknesses of shielding to provide a radiation field of known strength for use in calibrating portable radiation survey instruments. Provisions are available for calibrating instruments in reproducible geometries. A calibrated condenser R-meter, electrometer, or equivalent instrument, is used to accurately measure radiation levels to determine source-to-detector distances for desired instrument calibration radiation levels.

12.5.3.8.2 Contamination monitoring equipment

Personnel contamination monitoring instrumentation will be calibrated on at least an annual basis or following repair, in addition to periodic source checks, to determine proper response and alarm operability.

12.5.3.8.3 Air sampling equipment

Continuous Air Monitors (CAMs) sampling rates will be checked against calibrated air flow standards (e.g., anemometer) on at least an annual basis and after repairs which could affect the flow rate calibration. Calibrations will be done routinely on at least an annual basis and whenever an instrument is returned to service after any repair or maintenance that could affect instrument calibration. Calibration will include determining detector efficiency using a source of known strength traceable to standards. A detector response check will be performed, as a minimum, on a quarterly basis to assure instrument reliability. If applicable, operation of local alarms will also be verified on at least an annual basis. Personnel performing instrument calibrations will be trained in the operation and calibration of each instrument and will be familiar with the theoretical aspects of the instrumentation before being assigned calibration responsibilities.

Each portable air sampler will be tested at least annually for flow- rate. Devices utilizing flow meters are checked against calibrated air flow standards. The manufacturer's certification of flow-rate is used when physical flow measurements are not possible due to equipment configurations.

12.5.3.8.4 Dosimeters

All personnel dosimeters are calibrated prior to initial use, at intervals thereafter, and after any repairs that affect calibration.

12.5.3.8.5 Whole body counter

The whole body counter is calibrated using phantoms and standard sources of various radionuclides such as Co-60, Cs-137, and Ba-133.

12.5.3.8.6 Counting equipment

Routine calibrations will be performed on counting room instrumentation and radiation survey/monitoring instruments on at least an annual basis and after repairs affecting calibration. Background and operability checks of routinely used health physics counting equipment are performed daily. These instruments will be removed from service whenever their operation is statistically out of limits specified in plant procedures.

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TABLE 12.2.1-1

CORE AVERAGE GAMMA RAY SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

| Energy Group | Source Strengths at Time After Releases (MeV/W-s) | | | | |
|---|---|---|---|---|---|
| MeV/gamma | 12 Hours | 24 Hours | 100 Hours | 1 Week | 1 Month |
| 0.00 - 0.15 | $2.0 \times 10^{9}$ | $1.8 \times 10^{9}$ | $8.7 \times 10^{8}$ | $5.1 \times 10^{8}$ | $1.1 \times 10^{8}$ |
| 0.15 - 0.30 | $2.0 \times 10^{9}$ | $1.6 \times 10^{9}$ | $6.3 \times 10^{8}$ | $3.1 \times 10^{8}$ | $1.1 \times 10^{7}$ |
| 0.30 - 0.45 | $8.6 \times 10^{8}$ | $7.7 \times 10^{8}$ | $4.9 \times 10^{8}$ | $3.6 \times 10^{8}$ | $6.7 \times 10^{7}$ |
| 0.45 - 0.70 | $4.9 \times 10^{9}$ | $4.0 \times 10^{9}$ | $2.3 \times 10^{9}$ | $1.8 \times 10^{9}$ | $8.5 \times 10^{8}$ |
| 0.70 - 1.00 | $6.4 \times 10^{9}$ | $5.6 \times 10^{9}$ | $4.1 \times 10^{9}$ | $3.6 \times 10^{9}$ | $2.4 \times 10^{9}$ |
| 1.00 - 1.50 | $1.8 \times 10^{9}$ | $1.1 \times 10^{9}$ | $4.4 \times 10^{8}$ | $3.0 \times 10^{8}$ | $7.9 \times 10^{7}$ |
| 1.50 - 2.00 | $3.3 \times 10^{9}$ | $3.0 \times 10^{9}$ | $2.5 \times 10^{9}$ | $2.2 \times 10^{9}$ | $5.9 \times 10^{8}$ |
| 2.00 - 2.50 | $2.8 \times 10^{8}$ | $1.9 \times 10^{8}$ | $1.3 \times 10^{8}$ | $1.1 \times 10^{8}$ | $4.5 \times 10^{7}$ |
| 2.50 - 3.00 | $1.8 \times 10^{8}$ | $1.7 \times 10^{8}$ | $1.5 \times 10^{8}$ | $1.3 \times 10^{8}$ | $3.5 \times 10^{7}$ |
| 3.00 - 4.00 | $5.0 \times 10^{6}$ | $1.8 \times 10^{6}$ | $1.6 \times 10^{6}$ | $1.4 \times 10^{6}$ | $4.0 \times 10^{5}$ |
| 4.00 - 6.00 | $3.4 \times 10^{5}$ | $1.8 \times 10^{4}$ | 3.5 | 3.5 | 3.4 |
| 6.00 - 11.00 | $6.6 \times 10^{-1}$ | $6.6 \times 10^{-1}$ | $6.6 \times 10^{-1}$ | $6.6 \times 10^{-1}$ | $6.3 \times 10^{-1}$ |
| | | | | | |
| MeV/gamma | 3 Months | 6 Months | 1 Year | 5 Years | |
| 0.00 - 0.15 | $4.2 \times 10^{7}$ | $2.1 \times 10^{7}$ | $1.2 \times 10^{7}$ | $1.3 \times 10^{6}$ | |
| 0.15 - 0.30 | $1.4 \times 10^{6}$ | $5.8 \times 10^{5}$ | $3.4 \times 10^{5}$ | $1.8 \times 10^{5}$ | |
| 0.30 - 0.45 | $5.1 \times 10^{6}$ | $2.2 \times 10^{6}$ | $1.7 \times 10^{6}$ | $5.5 \times 10^{5}$ | |
| 0.45 - 0.70 | $4.2 \times 10^{8}$ | $2.7 \times 10^{8}$ | $2.0 \times 10^{8}$ | $8.0 \times 10^{7}$ | |
| 0.70 - 1.00 | $1.4 \times 10^{9}$ | $6.6 \times 10^{8}$ | $1.7 \times 10^{8}$ | $2.6 \times 10^{7}$ | |
| 1.00 - 1.50 | $3.8 \times 10^{7}$ | $3.0 \times 10^{7}$ | $2.3 \times 10^{7}$ | $7.3 \times 10^{6}$ | |
| 1.50 - 2.00 | $2.7 \times 10^{7}$ | $2.5 \times 10^{6}$ | $1.6 \times 10^{6}$ | $2.7 \times 10^{5}$ | |
| 2.00 - 2.50 | $1.9 \times 10^{7}$ | $1.4 \times 10^{7}$ | $9.2 \times 10^{6}$ | $2.8 \times 10^{5}$ | |
| 2.50 - 3.00 | $1.6 \times 10^{6}$ | $1.5 \times 10^{5}$ | $9.5 \times 10^{4}$ | $6.0 \times 10^{3}$ | |
| 3.00 - 4.00 | $4.4 \times 10^{4}$ | $2.5 \times 10^{4}$ | $1.8 \times 10^{4}$ | $1.1 \times 10^{3}$ | |
| 4.00 - 6.00 | 3.1 | 2.7 | 2.3 | 1.7 | |
| 6.00 - 11.00 | $5.8 \times 10^{-1}$ | $5.1 \times 10^{-1}$ | $4.4 \times 10^{-1}$ | $3.2 \times 10^{-1}$ | |

TABLE 12.2.1-2

SPENT FUEL GAMMA RAY SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

| Energy Group | Source Strengths at Time After Releases (MeV/W-s) | | | | |
|---|---|---|---|---|---|
| **MeV/gamma** | **12 Hours** | **24 Hours** | **100 Hours** | **1 Week** | **1 Month** |
| 0.00 - 0.15 | $2.3 \times 10^{9}$ | $2.0 \times 10^{9}$ | $9.9 \times 10^{8}$ | $5.8 \times 10^{8}$ | $1.2 \times 10^{8}$ |
| 0.15 - 0.30 | $2.2 \times 10^{9}$ | $1.8 \times 10^{9}$ | $7.2 \times 10^{8}$ | $3.6 \times 10^{8}$ | $1.4 \times 10^{7}$ |
| 0.30 - 0.45 | $9.4 \times 10^{8}$ | $8.3 \times 10^{8}$ | $5.2 \times 10^{8}$ | $3.8 \times 10^{8}$ | $7.3 \times 10^{7}$ |
| 0.45 - 0.70 | $5.6 \times 10^{9}$ | $4.7 \times 10^{9}$ | $2.9 \times 10^{9}$ | $2.4 \times 10^{9}$ | $1.4 \times 10^{9}$ |
| 0.70 - 1.00 | $6.6 \times 10^{9}$ | $5.8 \times 10^{9}$ | $4.2 \times 10^{9}$ | $3.7 \times 10^{9}$ | $2.5 \times 10^{9}$ |
| 1.00 - 1.50 | $2.1 \times 10^{9}$ | $1.3 \times 10^{9}$ | $6.6 \times 10^{8}$ | $4.9 \times 10^{8}$ | $1.8 \times 10^{8}$ |
| 1.50 - 2.00 | $3.3 \times 10^{9}$ | $3.0 \times 10^{9}$ | $2.6 \times 10^{9}$ | $2.2 \times 10^{9}$ | $6.0 \times 10^{8}$ |
| 2.00 - 2.50 | $3.5 \times 10^{8}$ | $2.7 \times 10^{8}$ | $2.0 \times 10^{8}$ | $1.7 \times 10^{8}$ | $7.1 \times 10^{7}$ |
| 2.50 - 3.00 | $1.9 \times 10^{8}$ | $1.7 \times 10^{8}$ | $1.5 \times 10^{8}$ | $1.3 \times 10^{8}$ | $3.6 \times 10^{7}$ |
| 3.00 - 4.00 | $5.3 \times 10^{6}$ | $1.9 \times 10^{6}$ | $1.6 \times 10^{6}$ | $1.4 \times 10^{6}$ | $4.3 \times 10^{5}$ |
| 4.00 - 6.00 | $4.0 \times 10^{5}$ | $2.1 \times 10^{4}$ | $1.8 \times 10^{1}$ | $1.8 \times 10^{1}$ | $1.8 \times 10^{1}$ |
| 6.00 - 11.00 | 3.5 | 3.5 | 3.4 | 3.4 | 3.3 |
| | | | | | |
| **MeV/gamma** | **3 Months** | **6 Months** | **1 Year** | **5 Years** | |
| 0.00 - 0.15 | $5.4 \times 10^{7}$ | $3.1 \times 10^{7}$ | $2.0 \times 10^{7}$ | $3.8 \times 10^{6}$ | |
| 0.15 - 0.30 | $2.4 \times 10^{6}$ | $1.4 \times 10^{6}$ | $1.0 \times 10^{6}$ | $5.9 \times 10^{5}$ | |
| 0.30 - 0.45 | $8.9 \times 10^{6}$ | $5.3 \times 10^{6}$ | $4.2 \times 10^{6}$ | $1.4 \times 10^{6}$ | |
| 0.45 - 0.70 | $9.1 \times 10^{8}$ | $6.9 \times 10^{8}$ | $5.6 \times 10^{8}$ | $2.4 \times 10^{8}$ | |
| 0.70 - 1.00 | $1.6 \times 10^{9}$ | $8.7 \times 10^{8}$ | $3.8 \times 10^{8}$ | $8.4 \times 10^{7}$ | |
| 1.00 - 1.50 | $9.8 \times 10^{7}$ | $8.3 \times 10^{7}$ | $6.8 \times 10^{7}$ | $2.5 \times 10^{7}$ | |
| 1.50 - 2.00 | $3.2 \times 10^{7}$ | $6.2 \times 10^{6}$ | $4.2 \times 10^{6}$ | $8.8 \times 10^{5}$ | |
| 2.00 - 2.50 | $2.9 \times 10^{7}$ | $2.1 \times 10^{7}$ | $1.3 \times 10^{7}$ | $4.3 \times 10^{5}$ | |
| 2.50 - 3.00 | $1.7 \times 10^{6}$ | $3.2 \times 10^{5}$ | $2.2 \times 10^{5}$ | $1.4 \times 10^{4}$ | |
| 3.00 - 4.00 | $8.3 \times 10^{4}$ | $5.8 \times 10^{4}$ | $4.1 \times 10^{4}$ | $2.6 \times 10^{3}$ | |
| 4.00 - 6.00 | $1.6 \times 10^{1}$ | $1.5 \times 10^{1}$ | $1.3 \times 10^{1}$ | $1.0 \times 10^{1}$ | |
| 6.00 - 11.00 | 3.1 | 2.8 | 2.4 | 1.9 | |

TABLE 12.2.1-3

SPENT FUEL FISSION PRODUCT INVENTORY AT SHUTDOWN

| Nuclide | Inventory (curies/kg of Uranium) |
|---|---|
| | |
| Kr-85 | $1.35 \times 10^{1}$ |
| Rb-86 | 3.63 |
| Sr-89 | $1.24 \times 10^{3}$ |
| Sr-90 | $1.05 \times 10^{2}$ |
| Y-90 | $1.12 \times 10^{2}$ |
| Y-91 | $1.62 \times 10^{3}$ |
| Zr-95 | $2.27 \times 10^{3}$ |
| Nb-95 | $2.29 \times 10^{3}$ |
| Nb-95m | $1.66 \times 10^{1}$ |
| Zr-97 | $2.28 \times 10^{3}$ |
| Nb-97 | $2.30 \times 10^{3}$ |
| Nb-97m | $2.16 \times 10^{3}$ |
| Mo-99 | $2.50 \times 10^{3}$ |
| Tc-99m | $2.19 \times 10^{3}$ |
| Ru-103 | $2.18 \times 10^{3}$ |
| Rh-103m | $1.97 \times 10^{3}$ |
| Rh-105 | $1.40 \times 10^{3}$ |
| Ru-106 | $7.35 \times 10^{2}$ |
| Rh-106 | $8.50 \times 10^{2}$ |
| Ag-110m | 6.24 |
| Ag-111 | $1.06 \times 10^{2}$ |
| Cd-115 | $2.66 \times 10^{1}$ |
| In-115m | $2.67 \times 10^{1}$ |
| Sn-123 | $5.03 \times 10^{0}$ |
| Sn-125 | $1.87 \times 10^{1}$ |
| Sb-125 | $1.86 \times 10^{1}$ |
| Te-125m | $3.89 \times 10^{0}$ |
| Sb-127 | $1.54 \times 10^{2}$ |
| Te-127 | $1.52 \times 10^{2}$ |
| Te-127m | $1.98 \times 10^{1}$ |
| Te-129 | $4.46 \times 10^{2}$ |
| Te-129m | $6.61 \times 10^{1}$ |
| Te-131 | $1.23 \times 10^{3}$ |

| Nuclide | Inventory (curies/kg of Uranium) |
|---|---|
| | |
| Te-131m | $2.02 \times 10^{2}$ |
| I-131 | $1.39 \times 10^{3}$ |
| Xe-131m | $1.55 \times 10^{1}$ |
| Te-132 | $1.97 \times 10^{3}$ |
| I-132 | $2.00 \times 10^{3}$ |
| I-133 | $2.79 \times 10^{3}$ |
| Xe-133 | $2.73 \times 10^{3}$ |
| Xe-133m | $8.77 \times 10^{1}$ |
| Cs-134 | $2.62 \times 10^{2}$ |
| Xe-135 | $5.10 \times 10^{2}$ |
| Cs-136 | $8.38 \times 10^{1}$ |
| Ba-136m | $1.38 \times 10^{1}$ |
| Cs-137 | $1.45 \times 10^{2}$ |
| Ba-137m | $1.37 \times 10^{2}$ |
| Ba-140 | $2.39 \times 10^{3}$ |
| La-140 | $2.52 \times 10^{3}$ |
| Ce-141 | $2.27 \times 10^{3}$ |
| Pr-142 | $1.43 \times 10^{2}$ |
| Ce-143 | $2.07 \times 10^{3}$ |
| Pr-143 | $2.02 \times 10^{3}$ |
| Ce-144 | $1.72 \times 10^{3}$ |
| Pr-144 | $1.74 \times 10^{3}$ |
| Pr-144m | $2.07 \times 10^{1}$ |
| Nd-147 | $9.11 \times 10^{2}$ |
| Pm-147 | $1.51 \times 10^{2}$ |
| Pm-148 | $4.12 \times 10^{2}$ |
| Pm-148m | $3.72 \times 10^{1}$ |
| Pm-149 | $9.13 \times 10^{2}$ |
| Pm-151 | $2.82 \times 10^{2}$ |
| Sm-153 | $7.92 \times 10^{2}$ |
| Eu-154 | $1.59 \times 10^{1}$ |
| Eu-155 | $1.02 \times 10^{1}$ |
| Eu-156 | $3.55 \times 10^{2}$ |

TABLE 12.2.1-4

IRRADIATED Ag-In-Cd CONTROL ROD SOURCE STRENGTHS

| | Source Strength at Time After Shutdown (MeV/cm$^3$-s) | | | | | |
|---|---|---|---|---|---|---|
| Energy Group (MeV/gamma) | 1 Day | 1 Week | 1 Month | 6 Months | 1 Year | 5 Years |
| 0.20 - 0.40 | $2.3 \times 10^{8}$ | $2.3 \times 10^{8}$ | $2.2 \times 10^{8}$ | $1.4 \times 10^{8}$ | $8.5 \times 10^{7}$ | $1.5 \times 10^{6}$ |
| 0.40 - 0.90 | $1.1 \times 10^{12}$ | $1.1 \times 10^{12}$ | $1.0 \times 10^{12}$ | $6.6 \times 10^{11}$ | $4.0 \times 10^{11}$ | $7.1 \times 10^{9}$ |
| 0.90 - 1.35 | $2.0 \times 10^{11}$ | $1.9 \times 10^{11}$ | $1.8 \times 10^{11}$ | $1.2 \times 10^{11}$ | $7.2 \times 10^{10}$ | $1.3 \times 10^{9}$ |
| 1.35 - 1.80 | $3.7 \times 10^{11}$ | $3.7 \times 10^{11}$ | $3.4 \times 10^{11}$ | $2.3 \times 10^{11}$ | $1.4 \times 10^{11}$ | $2.5 \times 10^{9}$ |

The absorber cross-sectional area is 0.589 square centimeters per rod.

The absorber material density is 10.17 grams per cubic centimeter.

TABLE 12.2.1-5

IRRADIATED IN-CORE DETECTOR AND DRIVE CABLE MAXIMUM WITHDRAWAL SOURCE STRENGTHS

| Energy Group (MeV/gamma) | Incore Detector (MeV/$cm^3$-s) | Drive Cable (MeV/$cm^3$-s) |
|---|---|---|
| 0.20 - 0.40 | $3.8 \times 10^{10}$ | $6.0 \times 10^{8}$ |
| 0.40 - 0.90 | $1.6 \times 10^{11}$ | $5.1 \times 10^{10}$ |
| 0.90 - 1.35 | $1.1 \times 10^{11}$ | $1.6 \times 10^{10}$ |
| 1.35 - 1.80 | $1.1 \times 10^{11}$ | $3.1 \times 10^{8}$ |
| 1.80 - 2.20 | $2.9 \times 10^{10}$ | $3.8 \times 10^{10}$ |
| 2.20 - 2.60 | $3.1 \times 10^{10}$ | $1.3 \times 10^{9}$ |
| 2.60 - 3.00 | $1.6 \times 10^{10}$ | $1.3 \times 10^{9}$ |
| 3.00 - 4.00 | $2.1 \times 10^{10}$ | $3.1 \times 10^{8}$ |
| 4.00 - 5.00 | $1.5 \times 10^{10}$ | 0 |
| 5.00 - 6.00 | $1.4 \times 10^{9}$ | 0 |

The effective diameter and length of the in-core detector are 0.48 centimeter and 5.33 centimeters, respectively.

The effective cross-sectional area of the drive cable is 0.095 square centimeters.

TABLE 12.2.1-6

IRRADIATED TYPE-304 STAINLESS STEEL (0.12 WEIGHT PERCENT Co)
CLADDING SOURCE STRENGTHS

| | Source Strength at Time After Shutdown (MeV/cm$^3$-s) | | | | | |
|---|---|---|---|---|---|---|
| Energy Group (MeV/gamma) | 1 Day | 1 Week | 1 Month | 6 Months | 1 Year | 5 Years |
| 0.20 - 0.40 | $7.1 \times 10^9$ | $6.1 \times 10^9$ | $3.4 \times 10^9$ | $8.3 \times 10^7$ | $9.9 \times 10^5$ | 0 |
| 0.40 - 0.90 | $3.1 \times 10^{10}$ | $2.9 \times 10^{10}$ | $2.6 \times 10^{10}$ | $1.2 \times 10^{10}$ | $6.4 \times 10^9$ | $2.3 \times 10^8$ |
| 0.90 - 1.35 | $2.4 \times 10^{11}$ | $2.3 \times 10^{11}$ | $2.3 \times 10^{11}$ | $2.1 \times 10^{11}$ | $2.0 \times 10^{11}$ | $1.2 \times 10^{11}$ |
| 1.35 - 1.80 | $1.9 \times 10^8$ | $1.8 \times 10^8$ | $1.4 \times 10^8$ | $3.3 \times 10^7$ | $5.4 \times 10^6$ | 0 |

The various cladding cross-section areas per rod are:

Ag-In-Cd control rod - 0.136 square centimeters

Sb-Be secondary source rod - 0.136 square centimeters

Burnable poison rod (including inner sheath) -0.159 square centimeters

TABLE 12.2.1-7

ISOTOPIC COMPOSITION AND SPECIFIC ACTIVITY OF TYPICAL OUT-OF-CORE CRUD DEPOSITS

| | Activity (microcuries per milligram) of Deposited Crud for Effective Full-Power Years of Plant Operation | | | |
|---|---|---|---|---|
| Composition (Nuclide) | 1 Year | 2 Years | 5 Years | 10 Years |
| Mn-54 | 1.0 | 1.1 | 1.3 | 1.4 |
| Fe-59 | 0.5 | 0.5 | 0.5 | 0.5 |
| Co-58 | 12.0 | 12.0 | 12.0 | 12.0 |
| Co-60 | 1.5 | 2.3 | 4.0 | 6.0 |

In addition to corrosion products, about 1.0 microgram of mixed actinides and fission products may be present for each 1 gram of deposited crud.

TABLE 12.2.1-8

MAXIMUM REFUELING CAVITY CRUD SPECIFIC ACTIVITY
ASSUMING NO PURIFICATION

| Nuclide | Activity (μCi/g) |
|---|---|
| Cr-51 | $1 \times 10^{-2}$ |
| Mn-54 | $2 \times 10^{-3}$ |
| Fe-59 | $1 \times 10^{-3}$ |
| Co-58 | $2 \times 10^{-2}$ |
| Co-60 | $3 \times 10^{-3}$ |
| Zr-95 | $1 \times 10^{-3}$ |
| Ru-103 | $1 \times 10^{-3}$ |
| Ce-141 | $6 \times 10^{-5}$ |
| Ce-144 | $3 \times 10^{-5}$ |

TABLE 12.2.1-9

PRESSURIZER N-16 SOURCE STRENGTHS

| Discrete Energy (MeV/gamma) | Energy Group (MeV/gamma) | Source Strength (MeV/gamma-s) |
|---|---|---|
| 1.75 | 1.35-1.80 | $6.5 \times 10^{2}$ |
| 2.74 | 2.6-3.0 | $5.9 \times 10^{3}$ |
| 6.13 | 6.0-7.0 | $1.2 \times 10^{6}$ |
| 7.12 | 7.0-7.5 | $1.0 \times 10^{5}$ |

TABLE 12.2.1-10

SPECIFIC ACTIVITIES IN DEMINERALIZERS ($\mu Ci/cm^3$)

| Nuclide | Mixed-Bed (for 30 $ft^3$ of resin) | Cation-Bed (for 20 $ft^3$ of resin) | Boron Thermal Regeneration (for 150 $ft^3$ of resin) |
|---|---|---|---|
| Co-60 | $1.3 \times 10^{2}$ | | |
| Br-64 | $8.6 \times 10^{-1}$ | | $1.6 \times 10^{-2}$ |
| I-131 | $1.2 \times 10^{4}$ | | 3.8 |
| I-132 | $2.3 \times 10^{2}$ | | $2.0 \times 10^{2}$ |
| I-133 | | | $8.9 \times 10^{1}$ |
| I-135 | $5.4 \times 10^{2}$ | | 8.2 |
| Rb-88 | $4.8 \times 10^{1}$ | $7.2 \times 10^{1}$ | |
| Cs-134 | $2.8 \times 10^{4}$ | $4.2 \times 10^{3}$ | |
| Cs-136 | $6.9 \times 10^{2}$ | $1.0 \times 10^{3}$ | |
| Ba-137m | $2.7 \times 10^{4}$ | $4.0 \times 10^{3}$ | |
| Cs-138 | $1.8 \times 10^{1}$ | $2.6 \times 10^{1}$ | |

The nuclides listed by activity are significant contributors to the source strengths

TABLE 12.2.1-11

REGENERATIVE HEAT EXCHANGER SPECIFIC ACTIVITY

| | Activity (μCi/g) | |
|---|---|---|
| Nuclide | Tube Side | Shell Side |
| Kr-88 | 1.7 | 3.2 |
| Kr-89 | | $9.3 \times 10^{-2}$ |
| Xe-133 | $2.7 \times 10^{2}$ | $2.8 \times 10^{2}$ |
| Xe-135 | 7.2 | 8.5 |
| I-132 | | 2.5 |
| I-133 | | 7.2 |
| I-135 | | 1.8 |
| Rb-88 | 7.9 | 7.9 |
| Cs-134 | 1.5 | 1.5 |
| Cs-136 | 3.2 | 3.2 |
| Ba-137m | 1.5 | |
| Cs-138 | $9.8 \times 10^{-1}$ | $9.8 \times 10^{-1}$ |
| N-16 | | $1.2 \times 10^{2}$ |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-12

SPECIFIC ACTIVITIES IN EXCESS LETDOWN, LETDOWN, AND SEAL WATER HEAT EXCHANGERS (µCi/g)

| Nuclide | Excess Letdown Tube Side | Letdown and Seal Water Tube Side | Letdown Reheat Tube Side | Letdown Reheat Shell Side |
|---|---|---|---|---|
| Kr-88 | 3.2 | 3.2 | 3.2 | 3.2 |
| Kr-89 | $9.3 \times 10^{-2}$ | $9.3 \times 10^{-2}$ | $9.3 \times 10^{-2}$ | $9.3 \times 10^{-2}$ |
| Xe-133 | $2.8 \times 10^{2}$ | $2.8 \times 10^{2}$ | $2.8 \times 10^{2}$ | $2.8 \times 10^{2}$ |
| Xe-135 | 8.5 | 8.5 | 8.5 | 8.5 |
| I-132 | 2.5 | 2.5 | 2.5 | |
| I-133 | 7.2 | 7.2 | 7.2 | |
| I-135 | 1.8 | 1.8 | 1.8 | |
| Rb-88 | 7.9 | 7.9 | 7.9 | 7.9 |
| Cs-134 | 1.5 | 1.5 | 1.5 | 1.5 |
| Cs-136 | 3.2 | 3.2 | 3.2 | 3.2 |
| Cs-138 | $9.8 \times 10^{-1}$ | $9.8 \times 10^{-1}$ | $9.8 \times 10^{-1}$ | $9.8 \times 10^{-1}$ |
| N-16 | $1.2 \times 10^{2}$ | | | |

The nuclides listed by activity are significant contributors to the source strengths.

TABLE 12.2.1-13

LETDOWN CHILLER HEAT EXCHANGER (TUBE SIDE) AND MODERATING HEAT EXCHANGER SPECIFIC ACTIVITY

| Nuclide | Activity ($\mu$Ci/g) |
|---|---|
| Kr-88 | 3.2 |
| Kr-89 | $9.3 \times 10^{-2}$ |
| Xe-133 | $2.8 \times 10^{2}$ |
| Xe-135 | 8.5 |
| Rb-88 | 7.9 |
| Cs-134 | 1.5 |
| Cs-136 | 3.2 |
| Cs-138 | $9.8 \times 10^{-1}$ |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-14

RESIDUAL HEAT REMOVAL LOOP SPECIFIC ACTIVITY - 8 HOURS AFTER SHUTDOWN

| Nuclide | Activity ($\mu$Ci/g) |
|---|---|
| Kr-88 | $2.5 \times 10^{-1}$ |
| Xe-133 | $2.6 \times 10^{2}$ |
| I-132 | $2.1 \times 10^{-1}$ |
| I-133 | 1.9 |
| I-135 | $2.8 \times 10^{-1}$ |
| Rb-88 | $2.7 \times 10^{-1}$ |
| Cs-134 | $5.4 \times 10^{-1}$ |
| Cs-136 | 1.1 |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-15

SPECIFIC ACTIVITIES IN BORON RECYCLE SYSTEM DEMINERALIZERS ($\mu Ci/cm^3$)

| Nuclide | Recycle Evaporator Feed (for 30 $ft^3$ of resin) | Recycle Evaporator Condensate (for 30 $ft^3$ of resin) |
|---|---|---|
| Mn-56 | $1.5 \times 10^{-2}$ | |
| Br-84 | $1.6 \times 10^{-7}$ | $1.3 \times 10^{-5}$ |
| I-131 | | 1.8 |
| I-132 | 1.2 | $3.3 \times 10^{-2}$ |
| I-133 | | $8.7 \times 10^{-1}$ |
| I-135 | 8.8 | $7.0 \times 10^{-2}$ |
| Cs-134 | $1.3 \times 10^{3}$ | |
| Cs-136 | $6.1 \times 10^{2}$ | |
| Ba-137m | $1.6 \times 10^{3}$ | |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-16

RECYCLE HOLDUP TANK SPECIFIC ACTIVITIES

| Nuclide | Liquid Phase (for 84,000 gallons of liquid) (μCi/g) | Vapor Phase (for 500 $ft^3$ of gas) (μCi/g) |
|---|---|---|
| Kr-87 | 1.1 | 1.2 |
| Kr-88 | 3.2 | 3.6 |
| Kr-89 | $9.3 \times 10^{-2}$ | $1.0 \times 10^{-1}$ |
| Xe-133 | $2.8 \times 10^{2}$ | $3.1 \times 10^{2}$ |
| Xe-135m | | $4.8 \times 10^{-1}$ |
| Xe-135 | 8.5 | 9.5 |
| Xe-138 | $6.3 \times 10^{-1}$ | $7.0 \times 10^{-1}$ |
| I-132 | $2.5 \times 10^{-1}$ | |
| I-133 | $7.2 \times 10^{-1}$ | |
| I-135 | $1.8 \times 10^{-1}$ | |
| Rb-88 | $7.9 \times 10^{-1}$ | |
| Cs-136 | $3.2 \times 10^{-1}$ | |
| Cs-138 | $9.8 \times 10^{-2}$ | |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-17

RECYCLE EVAPORATOR VENT CONDENSER SPECIFIC ACTIVITY

| Nuclide | Activity ($\mu Ci/cm^3$) |
|---|---|
| Kr-87 | $3.7 \times 10^1$ |
| Kr-88 | $1.1 \times 10^2$ |
| Kr-89 | 3.1 |
| Xe-133 | $9.2 \times 10^3$ |
| Xe-135m | $1.4 \times 10^1$ |
| Xe-135 | $2.8 \times 10^2$ |
| Xe-138 | $2.1 \times 10^1$ |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-18

RECYCLE EVAPORATOR CONCENTRATES SPECIFIC ACTIVITY

| Nuclide | Activity ($\mu$Ci/g) |
|---|---|
| Br-84 | $2.2 \times 10^{-7}$ |
| I-131 | 3.9 |
| I-132 | $2.8 \times 10^{-1}$ |
| I-135 | 1.5 |
| Cs-134 | $1.8 \times 10^{1}$ |
| Cs-136 | $3.7 \times 10^{1}$ |
| Ba-137m | $1.8 \times 10^{1}$ |
| Cs-138 | $2.8 \times 10^{-5}$ |
| La-140 | $2.6 \times 10^{-3}$ |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-19

SPECIFIC ACTIVITIES IN LIQUID WASTE PROCESSING SYSTEM DEMINERALIZERS (μ Ci/cm$^3$)

| Nuclide | Waste Evaporator Condensate (for 30 ft$^3$ of resin) | Waste Monitor Tank (for 30 ft$^3$ of resin) |
|---|---|---|
| Br-84 | $1.2 \times 10^{-4}$ | |
| I-131 | $1.5 \times 10^{1}$ | $6.7 \times 10^{1}$ |
| I-132 | $2.9 \times 10^{-1}$ | 5.1 |
| I-133 | 7.8 | $1.2 \times 10^{2}$ |
| I-135 | $6.3 \times 10^{-1}$ | $1.1 \times 10^{1}$ |
| Rb-88 | $1.2 \times 10^{-2}$ | |
| Cs-134 | 3.1 | $7.4 \times 10^{1}$ |
| Cs-136 | 3.7 | $1.3 \times 10^{2}$ |
| Ba-137m | 3.0 | $7.3 \times 10^{1}$ |
| Mn-56 | | $5.1 \times 10^{-2}$ |
| Cs-138 | $2.7 \times 10^{-3}$ | |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-20

REACTOR COOLANT DRAIN TANK SPECIFIC ACTIVITIES

| Nuclide | Liquid Phase (fr 175 gallons of liquid) ($\mu$Ci/g) | Vapor Phase (for 23.4 $ft^3$ of vapor) ($\mu Ci/cm^3$) |
|---|---|---|
| Kr-85 | | $2.3 \times 10^{2}$ |
| Kr-87 | | $1.4 \times 10^{-1}$ |
| Kr-88 | 3.2 | $8.7 \times 10^{-1}$ |
| Kr-89 | $9.3 \times 10^{-2}$ | $4.8 \times 10^{-4}$ |
| Xe-133 | $2.8 \times 10^{2}$ | $1.8 \times 10^{3}$ |
| Xe-135 | 8.5 | 7.1 |
| I-132 | 2.5 | |
| I-133 | 7.2 | |
| I-135 | 1.8 | |
| Rb-88 | 7.9 | |
| Cs-134 | 1.5 | |
| Cs-136 | 3.2 | |
| Cs-138 | $9.8 \times 10^{-1}$ | |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-21

HYDROGEN RECOMBINER, WASTE GAS COMPRESSOR, AND GAS DECAY TANK SPECIFIC ACTIVITY

| Nuclide | Activity ($\mu Ci/cm^3$) |
|---|---|
| Kr-85m | 6.5 |
| Kr-85 | $3.7 \times 10^{-1}$ |
| Kr-87 | 1.1 |
| Kr-88 | 8.9 |
| Xe-131m | 2.3 |
| Xe-133m | $1.2 \times 10^{1}$ |
| Xe-133 | $3.7 \times 10^{2}$ |
| Xe-135 | $3.6 \times 10^{1}$ |

The nuclides listed by activity are the significant contributors to the source strengths.

TABLE 12.2.1-22

SPENT FUEL POOL DEMINERALIZER SPECIFIC ACTIVITY

| Nuclide | Activity ($\mu Ci/cm^3$) |
|---|---|
| CR-51 | 3.41 E2 |
| Mn-54 | 6.94 E2 |
| Co-58 | 1.50 E2 |
| Co-60 | 3.32 E2 |
| CS-134 | 2.54 E2 |
| Cs-137 | 7.63 E2 |

The nuclides listed by activity are the significant contributors to the source strengths and represent greater than one percent of gamma source strengths.

TABLE 12.2.1-23

INSTANTANEOUS GAMMA RAY AND BETA SOURCE STRENGTHS AT VARIOUS TIMES FOLLOWING A MAXIMUM CREDIBLE ACCIDENT (TID-14844 RELEASE FRACTIONS)

| Energy Group | Source Strengths at Times After Releases (MeV/W-s) | | | | |
|---|---|---|---|---|---|
| MeV/gamma | 0 Hours | 0.5 Hours | 1 Hour | 2 Hours | 8 Hours |
| 0.00 - 0.15 | $1.8 \times 10^{8}$ | $1.1 \times 10^{8}$ | $1.1 \times 10^{8}$ | $1.0 \times 10^{8}$ | $9.8 \times 10^{7}$ |
| 0.15 - 0.30 | $8.9 \times 10^{8}$ | $2.7 \times 10^{8}$ | $2.3 \times 10^{8}$ | $2.1 \times 10^{8}$ | $1.6 \times 10^{8}$ |
| 0.30 - 0.45 | $8.4 \times 10^{8}$ | $4.0 \times 10^{8}$ | $3.1 \times 10^{8}$ | $2.5 \times 10^{8}$ | $1.7 \times 10^{8}$ |
| 0.45 - 0.70 | $2.8 \times 10^{9}$ | $1.6 \times 10^{9}$ | $1.4 \times 10^{9}$ | $1.1 \times 10^{9}$ | $5.8 \times 10^{8}$ |
| 0.70 - 1.00 | $3.9 \times 10^{9}$ | $2.3 \times 10^{9}$ | $1.8 \times 10^{9}$ | $1.1 \times 10^{9}$ | $2.6 \times 10^{8}$ |
| 1.00 - 1.50 | $4.0 \times 10^{9}$ | $2.2 \times 10^{9}$ | $1.8 \times 10^{9}$ | $1.2 \times 10^{9}$ | $4.8 \times 10^{8}$ |
| 1.50 - 2.00 | $2.6 \times 10^{9}$ | $1.2 \times 10^{9}$ | $9.8 \times 10^{8}$ | $7.6 \times 10^{8}$ | $2.8 \times 10^{8}$ |
| 2.00 - 2.50 | $2.6 \times 10^{9}$ | $1.4 \times 10^{9}$ | $1.1 \times 10^{9}$ | $7.6 \times 10^{8}$ | $1.8 \times 10^{8}$ |
| 2.50 - 3.00 | $7.8 \times 10^{8}$ | $3.1 \times 10^{8}$ | $2.4 \times 10^{8}$ | $1.3 \times 10^{8}$ | $1.4 \times 10^{7}$ |
| 3.00 - 4.00 | $6.6 \times 10^{8}$ | $6.2 \times 10^{7}$ | $4.2 \times 10^{7}$ | $2.1 \times 10^{7}$ | $2.5 \times 10^{6}$ |
| 4.00 - 6.00 | $4.0 \times 10^{8}$ | $5.0 \times 10^{6}$ | $5.0 \times 10^{6}$ | $4.0 \times 10^{6}$ | $9.0 \times 10^{5}$ |
| 6.00 - 11.00 | $1.8 \times 10^{5}$ | 0 | 0 | 0 | 0 |
| Beta | $3.0 \times 10^{10}$ | $5.2 \times 10^{9}$ | $4.5 \times 10^{9}$ | $3.4 \times 10^{9}$ | $1.5 \times 10^{9}$ |
| | | | | | |
| MeV/gamma | 1 Day | 1 Week | 1 Month | 6 Months | 1 year |
| 0.00 - 0.15 | $9.1 \times 10^{7}$ | $4.3 \times 10^{7}$ | $2.9 \times 10^{6}$ | $2.1 \times 10^{5}$ | $1.2 \times 10^{5}$ |
| 0.15 - 0.30 | $8.0 \times 10^{7}$ | $6.2 \times 10^{6}$ | $7.1 \times 10^{5}$ | $5.8 \times 10^{3}$ | $3.3 \times 10^{3}$ |
| 0.30 - 0.45 | $1.4 \times 10^{8}$ | $8.3 \times 10^{7}$ | $1.1 \times 10^{7}$ | $2.2 \times 10^{4}$ | $1.7 \times 10^{4}$ |
| 0.45 - 0.70 | $3.0 \times 10^{8}$ | $3.3 \times 10^{7}$ | $1.0 \times 10^{7}$ | $2.7 \times 10^{6}$ | $2.0 \times 10^{6}$ |
| 0.70 - 1.00 | $1.0 \times 10^{8}$ | $3.9 \times 10^{7}$ | $2.5 \times 10^{7}$ | $6.6 \times 10^{6}$ | $1.7 \times 10^{6}$ |
| 1.00 - 1.50 | $1.1 \times 10^{8}$ | $3.2 \times 10^{6}$ | $7.9 \times 10^{5}$ | $3.0 \times 10^{5}$ | $2.3 \times 10^{5}$ |
| 1.50 - 2.00 | $6.3 \times 10^{7}$ | $2.2 \times 10^{7}$ | $5.9 \times 10^{6}$ | $2.5 \times 10^{4}$ | $1.6 \times 10^{4}$ |
| 2.00 - 2.50 | $9.3 \times 10^{6}$ | $1.1 \times 10^{6}$ | $4.5 \times 10^{5}$ | $1.4 \times 10^{5}$ | $9.2 \times 10^{4}$ |
| 2.50 - 3.00 | $1.9 \times 10^{6}$ | $1.3 \times 10^{6}$ | $3.5 \times 10^{5}$ | $1.5 \times 10^{3}$ | $9.5 \times 10^{2}$ |
| 3.00 - 4.00 | $6.3 \times 10^{4}$ | $1.4 \times 10^{4}$ | $4.0 \times 10^{3}$ | $2.5 \times 10^{2}$ | $1.8 \times 10^{2}$ |
| 4.00 - 6.00 | $1.8 \times 10^{4}$ | 0 | 0 | 0 | 0 |
| 6.00 - 11.00 | 0 | 0 | 0 | 0 | 0 |
| Beta | $7.7 \times 10^{8}$ | $2.4 \times 10^{8}$ | $5.6 \times 10^{7}$ | $2.1 \times 10^{7}$ | $1.4 \times 10^{7}$ |

TABLE 12.2.1-24

NORMAL PLANT OPERATION SOURCE TERMS
CVCS DEMINERALIZER ACTIVITY CONCENTRATIONS
(Based on ANSI/ANS-18.1-1984)

| Nuclide | Mixed Bed Activity (μCi/g) | Nuclide | Cation Bed Activity (μCi/g) |
|---|---|---|---|
| Br-84 | $3.7 \times 10^{-1}$ | Rb-88 | 1.7 |
| I-131 | $2.2 \times 10^{2}$ | Cs-134 | $5.9 \times 10^{1}$ |
| I-132 | $1.9 \times 10^{1}$ | Cs-136 | 4.5 |
| I-133 | $8.5 \times 10^{1}$ | Cs-137 | $9.1 \times 10^{1}$ |
| I-134 | $1.3 \times 10^{1}$ | Sr-89 | $1.1 \times 10^{-1}$ |
| I-135 | $5.9 \times 10^{1}$ | Sr-90 | $4.6 \times 10^{-3}$ |
| Rb-88 | 1.3 | Sr-91 | $7.9 \times 10^{-3}$ |
| Cs-134 | $4.4 \times 10^{2}$ | Ba-137m | $8.6 \times 10^{1}$ |
| Cs-136 | 3.4 | Ba-140 | 2.6 |
| Cs-137 | $6.7 \times 10^{2}$ | | |
| Na-24 | $2.1 \times 10^{1}$ | | |
| Cr-51 | $4.9 \times 10^{1}$ | | |
| Mn-54 | $1.6 \times 10^{2}$ | | |
| Fe-55 | $1.5 \times 10^{2}$ | | |
| Fe-59 | 7.5 | | |
| Co-58 | $1.8 \times 10^{2}$ | | |
| Co-60 | $7.1 \times 10^{1}$ | | |
| Zn-65 | $4.5 \times 10^{1}$ | | |
| Sr-89 | 4.0 | | |
| Sr-90 | 1.7 | | |
| Sr-91 | $2.9 \times 10^{-1}$ | | |
| Ba-137m | $6.4 \times 10^{2}$ | | |
| Ba-140 | $9.5 \times 10^{1}$ | | |

TABLE 12.2.1-25

CONTAINMENT PLATE-OUT SOURCE STRENGTHS

| Energy Group | Source Strength at Time After Release ($s/cm^2$-s) | | | | | | |
|---|---|---|---|---|---|---|---|
| MeV/gamma | 1 Hour | 1 Day | 3 Days | 7 Days | 14 Days | 1 Month | 1 Year |
| 0.00-0.106 | 2.51 (+7) | 3.16 (+7) | 2.31 (+7) | 1.51 (+7) | 9.03 (+6) | 3.84 (+6) | 7.05 (+5) |
| 0.106-0.440 | 8.05 (+8) | 5.62 (+8) | 4.65 (+8) | 3.27 (+8) | 1.81 (+8) | 4.97 (+7) | 6.32 (+5) |
| 0.440-0.865 | 3.92 (+9) | 7.97 (+8) | 2.42 (+8) | 9.68 (+7) | 6.97 (+7) | 4.59 (+7) | 5.65 (+6) |
| 0.865-1.332 | 1.82 (+9) | 1.76 (+8) | 5.95 (+7) | 4.51 (+7) | 3.92 (+7) | 3.00 (+7) | 1.65 (+5) |
| 1.332-1.720 | 1.53 (+8) | 1.26 (+8) | 8.65 (+7) | 4.81 (+7) | 1.93 (+7) | 2.54 (+6) | 6.35 (+4) |
| 1.720-2.210 | 4.81 (+8) | 3.24 (+7) | 4.86 (+5) | 6.51 (+4) | 5.46 (+3) | 9.97 (+1) | 2.95 (-9) |
| 2.210-2.754 | 3.65 (+7) | 3.57 (+6) | 2.12 (+5) | 3.65 (+4) | 2.02 (+3) | 2.69 (+0) | - 0 - |
| 2.754-3.930 | 4.60 (+6) | - 0 - | - 0 - | - 0 - | - 0 - | - 0 - | - 0 - |

Note: Numbers in parentheses indicate power of ten.
This table contains data that was originally used in the Post-TMI shielding review, and is now considered historical. It will not be updated for actual plant conditions.

TABLE 12.2.1-26

SUMP ACTIVITY

| Energy Group | Source Strengths at Time after Release (gammas-s/cm$^3$-s) | | | | | | |
|---|---|---|---|---|---|---|---|
| MeV/gamma | 1 Hour | 1 Day | 3 Days | 7 Days | 14 Days | 1 Month | 1 Year |
| 0.00 - 0.106 | 4.12(+7) | 5.43(+7) | 3.89(+7) | 2.51(+7) | 1.52(+7) | 6.78(+6) | 1.65(+6)* |
| 0.106 - 0.440 | 1.21(+9) | 8.58(+8) | 7.06(+8) | 4.95(+8) | 2.74(+8) | 7.72(+7) | 1.55(+6) |
| 0.440 - 0.865 | 5.81(+9) | 1.23(+9) | 3.91(+8) | 1.69(+8) | 1.26(+8) | 8.51(+7) | 1.29(+7) |
| 0.865 - 1.332 | 2.67(+9) | 1.95(+8) | 2.12(+7) | 3.37(+6) | 1.61(+6) | 9.42(+5) | 1.24(+5) |
| 1.332 - 1.720 | 1.65(+8) | 4.29(+7) | 1.48(+7) | 2.70(+6) | 4.43(+5) | 1.86(+5) | 1.31(+5) |
| 1.720 - 2.210 | 7.10(+8) | 4.86(+7) | 8.38(+5) | 1.23(+5) | 1.04(+4) | 1.90(+2) | ----- |
| 2.210 - 2.754 | 5.42(+7) | 5.49(+6) | 3.95(+5) | 6.98(+4) | 3.85(+3) | 5.13 | ----- |
| 2.754 - 3.930 | 6.83(+6) | ----- | ----- | ----- | ----- | ----- | ----- |

* Power of ten

This table contains data that was originally used in the Post-TMI shielding review, and is now considered historical. It will not be updated for actual plant conditions.

TABLE 12.2.1-27

CONTAINMENT AIRBORNE ACTIVITY

| Energy Group | Source Strength at Time After Release (gammas/cm$^3$-s.) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| MeV/gamma | Zero | 1 Hour | 1 Day | 3 Days | 7 Days | 14 Days | 1 Month | 1 Year |
| 0.00-0.106 | 4.43 (+7) | 3.95 (+7) | 3.24 (+7) | 2.45 (+7) | 1.42 (+7) | 5.65 (+6) | 7.11 (+5) | 2.01 (-1) |
| 0.106-0.440 | 2.33 (+8) | 5.57 (+7) | 4.78 (+6) | 1.13 (+6) | 5.70 (+5) | 2.60 (+5) | 6.27 (+4) | 1.94 (-1) |
| 0.440-0.865 | 2.97 (+8) | 1.71 (+7) | 1.08 (+6) | 2.44 (+5) | 5.05 (+4) | 2.42 (+4) | 7.08 (+3) | 1.24 (+3) |
| 0.865-1.332 | 1.49 (+8) | 1.06 (+7) | 1.91 (+5) | 1.47 (+4) | 5.73 (+2) | 2.14 (+0) | 7.95 (-3) | 1.51 (-4) |
| 1.332-1.720 | 3.95 (+7) | 7.49 (+6) | 2.95 (+4) | 3.51 (+3) | 1.44 (+2) | 5.49 (-1) | --- | --- |
| 1.720-2.210 | 6.43 (+7) | 1.08 (+7) | 6.92 (+4) | 2.92 (+2) | 3.19 (-2) | 1.65 (-3) | --- | --- |
| 2.210-2.754 | 2.29 (+7) | 1.21 (+7) | 3.76 (+4) | 2.89 (+1) | 1.29 (-2) | 6.51 (-4) | --- | --- |
| 2.754-3.930 | 1.34 (+7) | 1.28 (+4) | --- | --- | --- | --- | --- | --- |

Note: Numbers in parentheses indicate power of ten.

This table contains data that was originally used in the Post-TMI shielding review, and is now considered historical. It will not be updated for actual plant conditions.

# TABLE 12.2.2-1

## ASSUMPTIONS AND PARAMETERS USED TO CALCULATE AIRBORNE CONCENTRATIONS (a)

| | |
|---|---|
| Leak Rates: | |
| Containment* | 1 percent of the noble gas inventory/d 0.001 percent of the iodine inventory/d |
| Reactor Auxiliary Building | 160 lb/d |
| Cubicles and other areas | Tables 12.2.2-3 and 12.2.2-4 |
| Partition Factors: | |
| Reactor Auxiliary Building | |
| Cubicles and Other Areas | 0.0075 for iodines, 1 for noble gases |
| Letdown heat exchanger** | 0.1 for iodines, 1 for noble gases |
| Ventilation Rates (cfm): | |
| Containment | Isolated case |
| Fuel Handling Building | 174,000 |
| Reactor Auxiliary Building | 135,000 |
| Volumes (ft3): | |
| Containment | $2.4 \times 10^6$ |
| Fuel Handling Building | $1.3 \times 10^6$ |
| Reactor Auxiliary Building | $1.8 \times 10^6$ |
| Other Factors: | |
| Failed fuel fraction | 0.12 percent |
| Plant load | 80 percent |
| Outside air conditioning (winter) | 24F; 48.0 percent relative humidity |
| Fuel Pool Parameters: | |
| Surface temperature | 123F |
| Surface area | 5000 $ft^2$ |
| Air velocity over surface | 10 ft/min |

Notes:

* NUREG 017

**WASH 1258

(a) This table is for historical purposes only.

TABLE 12.2.2-2

AVERAGE AIRBORNE C/MPCa(c) IN THE REACTOR AUXILIARY BUILDING TURBINE BUILDING, CONTAINMENT AND FUEL HANDLING BUILDING (d)

| ISOTOPE | CONTAINMENT (a) | TURBINE BUILDING (b) | FUEL HANDLING BUILDING (b) | REACTOR AUXILIARY BUILDING (a) |
|---|---|---|---|---|
| Kr-83M | 8.8(-2) | - | - | 3.2(-4) |
| Kr-85m | 1.4(-1) | - | - | 2.7(-4) |
| Kr-85 | 1.2(+2) | - | - | 1.1(-5) |
| Kr-87 | 1.5(-1) | - | - | 8.3(-4) |
| Kr-88 | 9.7(-1) | - | - | 3.0(-3) |
| Xe-131m | 4.6(-1) | - | - | 1.3(-5) |
| Xe-133m | 9.6(-1) | - | - | 1.5(-4) |
| Xe-133 | 1.1(+2) | - | - | 7.2(-3) |
| Xe-135m | 6.6(-3) | - | - | 1.4(-4) |
| Xe-135 | 4.9(+0) | - | - | 1.1(-3) |
| Xe-137 | 1.1(-3) | - | - | 4.2(-5) |
| Xe-138 | 2.0(-2) | - | - | 4.2(-4) |
| I-131 | 9.9(+0) | - | - | 3.1(-3) |
| I-133 | 4.7(-1) | - | - | 1.4(-3) |
| H-3 | 6.9(+0) | - | 5.0(-1) | 1.1(-4) |

Notes:

a) Represents power of 10
b) (-) Indicates negligible C/$MPC_a$
c) Pre-1993 10 CFR 20
d) This table is for historical purposes only

TABLE 12.2.2-3

WASTE PROCESSING BUILDING
ROOM BY ROOM Σ C/MPCa AND WHOLE BODY DOSE COMMITMENT VALUES (c) (d)

| LOCATION AND/OR COMPONENT | ELEVATION (FT.) | LEAK RATE (GPD)[b] | ∑ C/MPCa | DOSE COMMITMENT (mrem/hr.) OCCUPANCY | | |
|---|---|---|---|---|---|---|
| | | | | INTERNAL WHOLE BODY | EXTERNAL WHOLE BODY | INTERNAL THYROID |
| | | (a) | | | | |
| Waste Gas Compressor - 3 & 4A | 211 | 1.8(-6) CFM | 9.1(-2) | -0- | 6.8(-4) | -0- |
| Waste Gas Compressor - 1 & 2A | 211 | 1.8(-6) CFM | 9.9(-2) | -0- | 7.4(-4) | -0- |
| Waste Hold Up Tank - 1 & 2X | 211 | 7.0(-2) | 3.4(-3) | 4.6(-4) | 1.1(-6) | 3.9(-2) |
| Equipment Drain Sump Pumps - 1-4A,B | 211 | 6.3(-2) | 1.2(-2) | 6.4(-4) | 4.2(-6) | 1.5(-1) |
| Waste Evap. Cond. Pumps - 1 & 2A | 211 | 6.3(-2) | 8.4(-5) | 8.2(-5) | 3.2(-10) | 1.1(-5) |
| Filter Backwash Stg Tank Pump 1-4A | 211 | 6.3(-2) | 5.4(-4) | 2.0(-4) | 3.5(-6) | 1.2(-3) |
| Filter Backwash Stg Tank Pump 1-4B | 211 | 6.3(-2) | 6.8(-4) | 2.5(-4) | 4.4(-6) | 1.5(-3) |
| Spent Resin Trans Pumps 1-4A | 236 | 6.3(-2) | 5.9(-3) | 4.0(-4) | 2.1(-6) | 7.4(-2) |
| Low Cond. Holding Tank Pumps 1 & 2A | 216 | 6.3(-2) | 9.4(-5) | 8.9(-5) | 1.3(-9) | 4.6(-5) |
| Low Cond. Holding Tank 1 & 2B | 216 | 7.0(-2) | 4.7(-5) | 4.5(-5) | 6.6(-10) | 2.3(-5) |
| Gas Decay Tank | 236 | 1.8(-6) | 2.4(-1) | -0- | 1.3(-3) | -0- |
| Valve Gallery | 211 | 1.2(-1) | 1.2(-2) | 1.6(-3) | 3.9(-6) | 1.3(-1) |
| Valve Gallery | 236 | 2.4(-1) | 9.4(-5) | 9.2(-5) | 3.7(-10) | 1.3(-5) |
| Waste Evaporator 1 & 2A | 236 | 9.5(-2) | 3.0(-4) | 2.9(-4) | 1.4(-9) | 4.7(-5) |
| Valve Gallery | 236 | 1.2(-1) | 4.7(-5) | 4.6(-5) | 1.9(-10) | 6.5(-6) |
| Waste Evaporator | 236 | 9.5(-2) | 8.5(-5) | 8.3(-5) | 3.3(-10) | 1.2(-5) |
| Valve Gallery | 211 | 4.8(-2) | 3.2(-3) | 8.2(-4) | 8.9(-7) | 3.1(-2) |
| Floor Drain Sump Pumps 1-4A,B | 211 | -0- | 1.2(-2) | 6.4(-4) | 4.2(-6) | 1.5(-1) |
| Valve Gallery | 211 | 7.1(-2) | 8.1(-5) | 7.9(-5) | 3.1(-10) | 1.1(-5) |
| Waste Evaporator Conc. Tank 1-2X | 211 | 7.0(-2) | 3.7(-4) | 3.6(-4) | 1.4(-9) | 5.0(-5) |
| Waste Evaporator Conds. Tank 1 & 2B | 211 | 1.4(-1) | 1.4(-4) | 1.4(-4) | 5.4(-10) | 1.9(-5) |
| Catalytic Recombiner 1 & 2A | 211 | 9.5(-2) | 1.9(-2) | -0- | 2.6(-4) | -0- |
| Catalytic Recombiner 1 & 2B | 211 | 9.5(-2) | 1.7(-2) | -0- | 2.4(-4) | -0- |
| Waste Evaporator Conds. Tank 1 & 2A | 211 | 7.1(-2) | 1.6(-4) | 1.6(-4) | 6.2(-10) | 2.2(-5) |
| Pipe Tunnel 1-5 & H-Gx | 211 | -0- | 1.8(-2) | -0- | 2.5(-4) | -0- |
| Pipe Tunnel L & My-G | 211 | -0- | 3.3(-3) | 1.4(-4) | 4.3(-5) | 1.9(-5) |
| High Cond. Holding Tank 1 & 2X | 211 | 7.0(-2) | 2.0(-3) | 7.1(-5) | 3.7(-7) | 1.3(-2) |

NOTES:

a) Represents powers of 10
b) Leakage rates do not include controlled leakages from leak-off connections.
c) Pre-1993 10 CFR 20This table is for historical purposes only.

TABLE 12.2.2-4

REACTOR AUXILIARY BUILDING
ROOM BY ROOM Σ C/MPCa AND WHOLE BODY DOSE COMMITMENT VALUES (c) (d)

| LOCATION AND/OR COMPONENT | ELEVATION (FT.) | LEAK RATE (GPD)[b] | ∑ C/MPCa | DOSE COMMITMENT (mrem/hr.) OCCUPANCY | | |
|---|---|---|---|---|---|---|
| | | | | INTERNAL WHOLE BODY | EXTERNAL WHOLE BODY | INTERNAL THYROID |
| Cont. Spray & R.H.R. Pump 1A-SA | 190 | 3.0(-1) | 4.3(-2) | 1.7(-3) | 1.1(-4) | 4.6(-1) |
| Floor Drain Transfer Tk & Pumps | 190 | 1.6(-1) | 5.0(-2) | 1.8(-3) | 1.8(-4) | 4.8(-1) |
| Cont. Spray & R.H.R. Pump 1B-SB | 190 | 2.4(-1) | 3.4(-2) | 1.4(-3) | 8.9(-5) | 3.6(-1) |
| Equip. Drain Transfer Tk Pumps | 190 | 1.4(+0) | 8.7(-2) | 2.1(-3) | 6.2(-4) | 5.4(-1) |
| R.H.R. Heat Exch. 1B-SB | 236 | 3.8(-1) | 1.5(-1) | 5.9(-3) | 3.9(-4) | 1.6(+0) |
| Charging Pump 1B-SB | 236 | 6.3(-2) | 6.2(-3) | 9.2(-4) | 2.3(-5) | 4.9(-2) |
| Valve Gallery 31-D-E | 236 | 3.3(-1) | 3.3(-2) | 4.9(-3) | 1.2(-4) | 2.6(-1) |
| Recycle Evaporator Package | 236 | 2.5(-1) | 2.8(-2) | 2.2(-3) | 1.0(-5) | 3.4(-1) |
| Valve Gallery Jx-Jz-42 | 236 | 3.1(-1) | 3.0(-2) | 1.9(-3) | 1.7(-4) | 2.2(-1) |
| Recycle Evap. Feed Pump | 236 | -0- | 3.0(-2) | 1.9(-3) | 1.7(-4) | 2.2(-1) |
| Recycle Hold-Up Tank | 236 | 7.0(-2) | 1.1(-3) | 1.6(-4) | 4.1(-6) | 8.7(-3) |
| Valve Gallery 41-42-G | 236 | 3.1(-1) | 1.7(-1) | 6.5(-3) | 4.3(-4) | 1.8(+0) |
| Letdown Heat Exch. | 236 | 3.8(-1) | 6.1(-1) | 2.4(-2) | 1.6(-3) | 6.5(+0) |
| Letdown Reheat Exch. | 236 | 3.8(-1) | 5.4(-1) | 2.1(-2) | 1.4(-3) | 5.7(+0) |
| Valve Gallery 41-42-F | 236 | 9.5(-2) | 4.2(-3) | 6.2(-4) | 1.5(-5) | 3.2(-2) |
| Letdown Chiller Heat Exch. | 236 | 3.8(-1) | 7.9(-2) | 1.2(-1) | 2.9(-4) | 6.3(-1) |
| Moderating Heat Exch. | 236 | 3.8(-1) | 3.4(-2) | 5.1(-3) | 1.3(-4) | 2.7(-1) |
| Entrance To Valve Gallery 42-43-F | 236 | -0- | 4.2(-3) | 6.2(-4) | 1.5(-5) | 3.3(-2) |
| CVCS Chillers & Pumps 1A & 1B | 236 | 9.5(-2) | 3.1(-3) | 1.1(-4) | 1.3(-5) | 2.8(-2) |
| Seal Water Heat Exch. | 236 | 3.8(-1) | 6.6(-1) | 2.4(-2) | 2.8(-3) | 6.0(+0) |
| Boric Acid Transfer Pumps | 236 | -0- | 3.1(-3) | 1.1(-4) | 1.3(-5) | 2.8(-2) |
| Valve Gallery D-E-42 | 236 | -0- | 3.1(-3) | 1.1(-4) | 1.3(-5) | 2.8(-2) |
| Service Water Booster Pump 1B-SB | 236 | -0- | 3.1(-3) | 1.1(-4) | 1.3(-5) | 2.8(-2) |
| Valve Gallery Fw-H-42 | 261 | 4.7(-2) | 2.0(-3) | 9.3(-5) | 2.2(-5) | 5.0(-3) |
| Boric Acid Tank Area | 261 | 1.4(-1) | 8.2(-4) | 1.1(-4) | 9.7(-6) | 3.4(-5) |
| Volume Control Tank | 261 | 4.8(-2) | 3.3(-2) | 1.2(-3) | 1.4(-4) | 3.1(-1) |
| Mechanical Penetration | 236 | 9.3(-1) | 3.9(-3) | 1.7(-4) | 2.7(-5) | 2.4(-2) |

NOTES:

a) Represents powers of 10
b) Leakage rates do not include controlled leakages from leak-off connections.
c) Pre-1993 10 CFR 20
d) This Table is for historical purposes only.

TABLE 12.3.1-1

EQUIPMENT SPECIFICATION LIMITS FOR COBALT IMPURITY LEVELS

| Component | Material | Maximum Weight Percent Cobalt |
|---|---|---|
| Reactor internals (non-active region) | SS* | 0.20 |
| Reactor internals (active region) | SS | 0.12 |
| Reactor vessel clad | SS | 0.20 |
| Reactor coolant piping | SS | 0.20 |
| Reactor internal bolting material | SS | 0.25 |
| Reactor coolant pumps | SS | 0.20 |
| Pressurizer | SS | 0.20 |
| Steam Generators | Inconel | 0.10** |
| Fuel (non-active region) | SS | 0.12 |
| Fuel (active region) | SS | 0.08 |
| Fuel | Inconel | 0.10 |
| Fuel | Zircaloy | 0.002 |

*SS = Stainless Steel

**Heat transfer tubes - Inconel 690 (ASME SB-163, Alloy UNS N06690) (Cobalt 0.016% Maximum)

TABLE 12.3.1-2

APPROXIMATE REACTOR COOLANT SYSTEM WETTED SURFACE AREAS

| Component | Material | Surface Area ($ft.^2$) |
|---|---|---|
| Reactor internals | SS* | 4236 |
| Reactor vessel clad | SS | 2190 |
| Reactor coolant piping | SS | 2758 |
| Reactor internal bolting material | SS | Negligible |
| Reactor coolant pumps | SS | Negligible |
| Steam generators | Inconel | $1.46 \times 10^5$ |
| Fuel (non-active region) | SS | 2000 |
| Fuel (active region) | SS | 3600 |
| Fuel | Inconel | $7.80 \times 10^3$ |
| Fuel | Zircaloy | $7.78 \times 10^4$ |

*SS - Stainless Steel

TABLE 12.3.1-3

APPROXIMATE REACTOR COOLANT SYSTEM WETTED SURFACE AREA OF STELLITE

| Component | Surface Area ($ft.^2$) |
|---|---|
| Reactor internals | 3.2 |
| Reactor coolant pump journals | 17.2 |
| Control rod drive mechanisms | 10.8 |
| Reactor coolant system valves | 2.6 |

TABLE 12.3.2-1

ALLOWABLE DOSE RATES

| Location (Area) | Dose Rate mrem/hr (max.) |
|---|---|
| Site Boundary | 0.05 |
| Service Building | 0.05 |
| Control Room, Reactor Auxiliary Building, Fuel Handling Building, Waste Processing Building, Tank Building and Turbine Building Areas outside controlled access areas | 0.25 |
| Reactor Auxiliary Building, Fuel Handling Building, Waste Processing Building, Tank Building and turbine building areas inside controlled access areas: areas of 40 hr/wk occupancy | 2.5 |
| Reactor Auxiliary Building Fuel Handling Building, Waste Processing Building, Tank Building, and Turbine Building: areas of 20 hr/wk occupancy | 5.0 |
| Reactor Auxiliary Building Fuel Handling Building, Waste Processing Building, and Containment: areas where occupancy is determined by health physics staff | 100 |
| High Radiation Areas | >100 |
| Control Room following maximum hypothetical accident | 1 rem over 30 days following the accident |

TABLE 12.3.2-2

NEUTRON LEAKAGE FLUXES (a)

| Group Number | Lower Energy Bound (MeV) | Leakage Flux Neutrons/$cm^2$-s |
|---|---|---|
| 1 | 7.79 | 4.56 (+6)* |
| 2 | 6.07 | 1.41 (+7) |
| 3 | 4.72 | 1.91 (+7) |
| 4 | 3.68 | 1.28 (+7) |
| 5 | 2.87 | 3.2 (+7) |
| 6 | 2.23 | 6.54 (+7) |
| 7 | 1.74 | 1.17 (+8) |
| 8 | 1.35 | 1.66 (+8) |
| 9 | 1.05 | 3.23 (+8) |
| 10 | 8.21 | 5.14 (+7) |
| 11 | 3.88 (-1) | 1.35 (+9) |
| 12 | 1.11 (-1) | 1.97 (+9) |
| 13 | 4.09 (-2) | 5.17 (+8) |
| 14 | 1.5 (-2) | 2.69 (+8) |
| 15 | 5.53 (-3) | 1.22 (+8) |
| 16 | 5.83 (-3) | 1.77 (+8) |
| 17 | 7.89 (-5) | 6.10 (+7) |
| 18 | 1.07 (-5) | 6.09 (+9) |
| 19 | 1.86 (-6) | 3.76 (+7) |
| 20 | 3.00 (-7) | 2.51 (+7) |
| 21 | 0.0 | 1.44 (+5) |

Notes:
(a)This table is for historical purposes only.
* Power of ten

TABLE 12.3.2 3

AREAS REQUIRING ACCESS FOLLOWING AN ACCIDENT

| AREA | LOCATION | OCCUPANCY REQUIREMENTS | MAXIMUM RADIATION LEVEL (MREM/HR) | REMARKS |
|---|---|---|---|---|
| 1. Control Room | RAB EL 305.00' | Continuous | 10 | |
| 2. Technical Support Center | FHB EL 324.00' | Continuous | 15 | |
| 3. Post-Accident Sampling Facility | RAB EL 236.00' | Infrequent Access | 50 | |
| 4. Radiochemistry Lab | WPB EL 276.00' | Infrequent Access | 0.25 | |
| 5. Switchgear & Battery Rooms | RAB EL 286.00' | Infrequent Access | 100 | |
| 6. Diesel Gen | Diesel Gen Bldg | Frequent Access | 10 | |
| 7. Health Physics Offices | WPB EL 261.00' | Continuous | 0.25 | |
| 8. Hydrogen Analyzer Sample Dilution Panel | RAB 236.00' | Infrequent Access | 50 | |
| 9. RHR Pump Room | RAB EL 190.00' | Access during long term recirculation phase | 104 | 20-30 minutes stay is possible after one day |
| 10. Cont Spray Pump Room | RAB EL 190.00' | Access during long term recirculation phase | 104 | 20-30 minutes stay is possible after one day |
| 11. Security Secondary Alarm Station (SAS) | RAB EL 305.00 | Continuous | 15 | |

TABLE 12.3.2-9

NEUTRON STREAMING DOSE RATES IN CONTAINMENT
DURING NORMAL OPERATION (a)

| LOCATION | NEUTRON DOSE RATE (REM/HR) |
|---|---|
| 1) Manipulator Crane Parked along Refueling Cavity El. 286.00' | 8.7 |
| 2) Fuel Transfer System Control Panel El. 286.00' | 7.6 |
| 3) Edge of Refueling Cavity near Steam Generator 1C - SN El. 286.00' | 11.9 |
| 4) Missile Shield on Reactor Vessel Integrated Head El. 286.00' | 13.5 |
| 5) Emergency Escape Lock El. 261.00' | 0.12 |
| 6) Grating Floor El. 261.00' | 0.01 - 0.13 |
| 7) Personnel Lock El. 236.00' | 0.034 |
| 8) Grating Floor El. 236.00' | 0.002 - 0.036 |

Notes:

(a)This table is for historical purposes only.

## TABLE 12.3.4-1 AREA RADIATION MONITORS

| Description | Tag # | Subsection of 12.3.4.1.8 Described In | Detector Type | Range (mR/hr) | Typical Sensitivity (CPM/(mR/hr) | Accuracy (%) | Typical High Alarm Setpoints*(mR/hr) | Location | Power |
|---|---|---|---|---|---|---|---|---|---|
| Control Room Area | RM-21RR-3560 SA | .5 | GM Tube (11.5.2.5.1) | $10^{-2}$ - $10^{3}$ | $7 \times 10^{2}$ | ±20 | 2.5 | Control room | Instrumentation AC Safety Bus |
| Containment Isolation A | RM-1CR-3561 SA | .1 | Ion chamber (11.5.2.5.3) | $10^{1}$ - $10^{7}$ | $10^{-9}$ A/R/hr | ±15 | $2 \times 10^{3}$ | Around Reactor Cavity | † |
| Containment Isolation B | RM-1CR-3561B SB | .1 | " | " | " | " | " | " | " |
| Containment Isolation C | RM-1CR-3561C SA | .1 | Ion chamber (11.5.2.5.3) | $10^{1}$ -$10^{7}$ | $10^{-9}$ A/R/hr | ±15 | $2 \times 10^{3}$ | Around Reactor Cavity | Instrumentation AC Safety Bus |
| Containment Isolation D | RM-1CR-3561D SB | .1 | " | " | " | " | " | " | " |
| Fuel Handling Bldg. South | RM-*1FR-3564A SA<br>RM-*1FR-3564B SB<br>RM-*1FR-3564A SA<br>RM-*1FR-3565B SB | .3<br>.3<br>.3<br>.3 | GM Tube (11.5.2.5.2)<br>"<br>" | $10^{2}$ - $10^{3}$<br>"<br>" | $7 \times 10^{2}$<br>"<br>" | ±15<br>"<br>" | $10^{2}$<br>"<br>" | Around FHB<br>"<br>" | †<br>†<br>†<br>† |
| Fuel Handling Bldg. North | RM-*1FR-3566A SA<br>RM-*1FR-3566B SB<br>RM-*1FR-3567A SA<br>RM-*1FR-3567B SB | .3<br>.3<br>.3<br>.3 | "<br>"<br>"<br>" | "<br>"<br>"<br>" | "<br>"<br>"<br>" | "<br>"<br>"<br>" | "<br>"<br>"<br>" | "<br>"<br>"<br>" | †<br>†<br>†<br>† |
| Containment Bldg. In-Core Inst. Controls | RM-1CR-3577 | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $.2 \times 10^{2}$ | ± 15 | 200 | Plant | Instrumentation AC bus |
| Access Aisle Between Valve Galleries | RM-21CR-3581A<br>RM-21CR-3581B | .5<br>.5 | GM Tube (11.5.2.5.1)" | $10^{-1}$-$10^{4}$<br>" | $1.2 \times 10^{2}$<br>" | ± 15<br>" | 5<br>5 | Plant<br>" | *<br>* |
| In-Containment Post-Accident | RM-1CR-3589SA | | Ion Chamber (11.5.2.5.3) | $10^{0}$-$10^{8}$ R/hr | Later | ± 20 | $10^{4}$ | Containment | ‡ |
| In-Containment Post-Accident | RM-1CR-3590SB | | Ion Chamber (11.5.2.5.3) | $10^{0}$-$10^{8}$ R/hr | Later | ± 20 | $10^{4}$ | Containment | ‡ |
| Volume Control Tank | RM-1RR-3595 | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $10^{2}$ | ± 15 | 500 | Plant | * |
| RHR Pump 1B | RM-1RR-3597 | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $1.2 \times 10^{2}$ | ± 15 | 500 | Plant | * |

**TABLE 12.3.4-1 AREA RADIATION MONITORS**

| Description | Tag # | Subsection of 12.3.4.1.8 Described In | Detector Type | Range (mR/hr) | Typical Sensitivity (CPM//(mR/hr) | Accuracy (%) | Typical High Alarm Setpoints*(mR/hr) | Location | Power |
|---|---|---|---|---|---|---|---|---|---|
| RHR Pump 1A | RM-1RR-3598 | .5 | " | $10^{-1}$-$10^{4}$ | " | " | 500 | " | * |
| Charging Pump 1A | RM-1RR-3599A | .5 | " | " | " | " | 500 | " | * |
| Charging Pump 1B | RM-1RR-3599B | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $1.2 \times 10^{2}$ | ± 15 | 500 | Plant | * |
| Charging Pump 1C | RM-1RR-3599C | .5 | " | " | " | " | 500 | : | * |
| Recycle Evaporator Valve Gallery | RM-1RR-3600 | .5 | " | " | " | " | 10 | " | * |
| Letdown Hx Valve Gallery | RM-1RR-3601 | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $1.2 \times 10^{2}$ | ± 15 | 15 | Plant | * |
| Moderating Hx Valve Gallery | RM-1RR-3602 | .5 | " | " | " | " | 15 | " | * |
| Boric Acid Pump Valve Gallery | RM-1RR-3603 | .5 | " | " | " | " | 10 | " | * |
| Sample Room 1A | RM-1RR-3605A | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $1.2 \times 10^{2}$ | ± 15 | 10 | Plant | * |
| Sample Room 1B | RM-1RR-3605B | .5 | " | " | " | " | 10 | " | * |
| RHR Hx 1A | RM-1RR-3606A | .5 | " | " | " | " | 150 | " | * |
| RHR Hx 1B | RM-1RR-3606B | .5 | " | " | " | " | 150 | " | * |
| Spent Fuel Pool South | RM-1FR-3620 | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{4}$ | $1.2 \times 10^{2}$ | ± 15 | 5 | Plant | * |
| Access Aisle Between Spent Fuel Hx | RM-1FR-3625 | .5 | " | " | " | " | 5 | " | * |
| Spent Resin Pumps<br>1-2A<br>1-2B | <br>RM-*1WR-3644A<br>RM-*1WR-3644B | <br>.5<br>.5 | <br>GM Tube (11.5.2.5.1)<br>" | <br>$10^{-1}$-$10^{4}$<br>" | <br>$1.2 \times 10^{2}$<br>" | <br>± 15<br>" | <br>15<br>" | <br>Plant<br>" | <br>*<br>* |
| Filter Area | RM-1WR-3647A | .5 | " | " | " | " | 200 | " | * |

**TABLE 12.3.4-1 AREA RADIATION MONITORS**

| Description | Tag # | Subsection of 12.3.4.1.8 Described In | Detector Type | Range (mR/hr) | Typical Sensitivity (CPM/(mR/hr) | Accuracy (%) | Typical High Alarm Setpoints*(mR/hr) | Location | Power |
|---|---|---|---|---|---|---|---|---|---|
| Gas Decay TK Valve Gallery 1&2 | RM-21WR-3647A | .5 | GM Tube (11.5.2.5.1) | $10^{-1}$-$10^{-4}$ | $1.2 \times 10^{2}$ | ± 15 | 200 | Plant | * |
| 1&2 | RM-21WR-3647B | .5 | " | " | " | " | 100 | " | * |
| TSC Command Room | RM-*1TS-3653B | .5 | " | " | " | " | 10 | " | * |
| TSC Hallway | RM-*1TS-3653A | .5 | " | " | " | " | 10 | " | * |

# TABLE 12.3.4-2

## AIRBORNE RADIATION MONITORS

| Description | Tag # | Subsection of 12.3.4.1.8 Described In | Detector Type | Range (mR/hr) | Typical Sensitivity (CPM/(mR/hr) | Accuracy (%) | Typical High Alarm Setpoints* (mR/hr) | Location | Power |
|---|---|---|---|---|---|---|---|---|---|
| Containment Atmosphere Leak Detection A | RM-1LT-3502A SA | .1 | Particulate - (11.5.2.5.8) & Noble Gas - (11.5.2.5.6) | $8 \times 10^{-12}$<br>$8 \times 10^{8}$<br>$4.6 \times 10^{-7}$<br>$4.6 \times 10^{-1}$ | $1.08 \times 10^{5}$ CPM/µCi CS-137<br>$1.37 \times 10^{6}$ CPM/µCi/cc Xe-133 | ±10 | $4 \times 10^{-8}$<br>&<br>$10^{-3}$ | Outside Containment | * |
| Main Control Rooms Normal Outside Air Intake | RM-1CZ-3504A SA | .2 | Beta Sensitive (11.5.2.5.4) | $3.5 \times 10^{-8}$<br>$3.5 \times 10^{-2}$ | $2.8 \times 10^{8}$ CPM/µCi/cc Kr-85 | ±20 | $1.0 \times 10^{-5}$ (Kr-85) | Duct | * |
| TSC Outside Air Intake | RM-*1TS-3653C | .3 | Beta Sensitive (11.5.2.5.4) | $3.5 \times 10^{-8}$<br>$3.5 \times 10^{-2}$ | $2.8 \times 10^{8}$ CPM/µCi/cc Kr-85 | ±20 | $1.0 \times 10^{-5}$ (Kr-85) | Duct | * |
| Main Control Rooms Normal Outside Air Intake | RM-1CZ-3504B SB | .2 | Beta Sensitive (11.5.2.5.4) | $3.5 \times 10^{-8}$<br>$3.5 \times 10^{-2}$ | $2.8 \times 10^{8}$ CPM/µCi/cc Kr-85 | ±20 | $1.0 \times 10^{-5}$ (Kr-85) | Duct | * |
| Main Control Room Emerg. Outside Air Intake | RM-1CZ3505A SA | .3 | " | " | " | " | $1.0 \times 10^{-5}$ (Kr-85) | " | * |
| Main Control Room Emerg. Outside Air Intake | RM-1CZ3505B SB | .3 | " | " | " | " | " | " | * |
| Main Control Room Emerg. Outside Air Intake | RM-1CZ3505A2 SA | .3 | Beta Sensitive (11.5.2.5.4) | $3.5 \times 10^{-8}$<br>$3.5 \times 10^{-2}$ | $2.8 \times 10^{8}$ CPM/µCi/cc Kr-85 | ±20 | $1.0 \times 10^{-5}$ (Kr-85) | Duct | * |
| Main Control Room Emerg. Outside Air Intake | RM-1CZ3505B2 SB | .3 | " | " | " | " | " | " | * |

* Instrumentation AC Safety Bus

TABLE 12.4.1-1

Distribution of Personnel Exposures by Work Function

Part I - Pre-operational Dose Estimates

| | Operating PWRs | SHEARON HARRIS PLANT | |
|---|---|---|---|
| | Percentage of Annual Person-rem | Annual Person-rem | Percentage of Annual Person-rem |
| Reactor Operation and Surveillance | 9.6 | 42.1 | 12.2 |
| Maintenance (Routine + Special) | 67.7 | 205.7 | 59.4 |
| In-service Inspection | 8.6 | 30.6 | 8.8 |
| Waste Processing | 3.3 | 13.4 | 3.9 |
| Refueling | 10.8 | 54.6 | 15.7 |
| Total | | 346.4 | |

Part II - Annual Doses for Plant Operations (person-rem)

| Year | Normal Operations* | Refueling | Total |
|---|---|---|---|
| 1987 | 33.489 | -- | 33.489 |
| 1988 | 15.860 | 153.500 | 169.360 |
| 1989 | 18.180 | 137.400 | 155.580 |
| 1990 | 84.910 | -- | 84.910 |
| 1991 | 37.340 | 189.100 | 226.440 |
| 1992 | 39.270 | 173.730 | 213.000 |
| 1993 | 30.789 | -- | 30.789 |
| 1994 | 26.592 | 195.435 | 222.027 |
| 1995 | 30.419 | 143.624 | 174.043 |
| 1996 | 17.238 | -- | 17.238 |
| 1997 | 15.082 | 133.866 | 148.948 |
| 1998 | 15.834 | 117.663 | 133.497 |
| 1999 | 15.538 | -- | 15.538 |
| 2000 | 12.146 | 88.836 | 100.982 |
| 2001 | 10.982 | 241.259 | 252.241 |

*Includes non-refueling outages, spent fuel shipping program, and material upgrade doses.

TABLE 12.4.2-13

ANNUAL INHALATION RADIATION EXPOSURE ESTIMATES[a] [b]

| System or Component | Occupancy Man-Hours | Annual Dose Thyroid | (Man-Rem) Whole Body |
|---|---|---|---|
| Cont. Spray Pump | 2 | 9.2(-4) | 3.4(-6)* |
| RHR Pump | 3 | 1.4(-3) | 5.1(-6) |
| Floor Drain Transfer Tank & Pump | 1.5 | 7.2(-4) | 2.7(-6) |
| Equipment Drain | 1.5 | 8.1(-4) | 3.2(-6) |
| RHR Heat Exchanger | 4 | 6.4(-3) | 2.4(-5) |
| Charging Pumps | 10 | 4.9(-4) | 9.2(-6) |
| Recycle Evaporator Package | 11 | 3.7(-3) | 2.4(-5) |
| Valve Gallery | 4 | 8.8(-4) | 7.6(-6) |
| Recycle Evap. Feed Pump | 4 | 8.8(-4) | 7.6(-6) |
| Recycle Holdup Tank | 1 | 8.7(-6) | 1.6(-7) |
| Letdown Heat Exch. | 2 | 1.3(-2) | 4.8(-5) |
| Valve Gallery | 2 | 3.6(-3) | 1.3(-5) |
| Letdown Reheat Exch. | 2 | 1.1(-2) | 4.2(-5) |
| Valve Gallery | 2 | 3.1(-3) | 1.1(-5) |
| Moderating Heat Exch. | 2 | 5.4(-4) | 1.0(-5) |
| Valve Gallery | 2 | 1.7(-3) | 3.2(-5) |
| Letdown Chiller Heat Exch. | 2 | 1.3(-3) | 2.4(-5) |
| Valve Gallery | 2 | 1.3(-3) | 2.4(-5) |
| Seal Water Heat Exch. | 2 | 1.2(-2) | 4.8(-5) |
| Valve Gallery | 2 | 3.3(-3) | 1.3(-5) |
| Boric Acid Trans. Pump | 2 | 5.6(-5) | 2.2(-7) |
| Volume Control Tank | 1 | 3.1(-4) | 1.2(-5) |
| Valve Gallery | 2 | 6.2(-4) | 2.4(-5) |
| Waste Holdup Tank | 2.5 | 9.8(-5) | 1.2(-6) |
| Valve Gallery | 2.5 | 9.8(-5) | 1.2(-6) |
| Equipment Drain Sump Pumps | 3 | 4.5(-4) | 1.9(-6) |
| Filter Backwash Storage Tank Pump | 3 | 3.6(-6) | 6.0(-7) |
| Spent Resin Transfer Pump | 3 | 2.2(-4) | 1.2(-6) |
| Low Conductivity Holding Tank | 1 | 4.6(-8) | 9.0(-8) |
| High Conductivity Holding Tank Pump | 2 | 5.2(-5) | 1.4(-7) |
| High Conductivity Holding Tank | 1 | 1.3(-5) | 7.1(-8) |
| Waste Evaporator | 11 | 5.2(-7) | 3.2(-7) |
| Valve Gallery | 5 | 3.3(-7) | 2.3(-7) |
| Floor Drain Sump Pumps | 2 | 3.0(-4) | 1.3(-6) |
| Valve Gallery | 5 | 5.5(-8) | 4.0(-7) |
| Waste Evaporator Conc. Tank | 2.5 | 1.3(-7) | 9.0(-7) |
| Valve Gallery | 2.5 | 2.8(-8) | 2.0(-7) |
| Waste Evaporator Condensate Tank | 2.5 | 4.8(-8) | 3.5(-7) |
| Waste Evaporator Cond. Tank Pump | 3 | 5.5(-8) | 2.5(-7) |

*Power of ten

(a)Thyroid and whole body doses are pre-1993 10 CFR 20 (see Section 12.0).

(b)This table is for historical purposes only.

| FIGURE | TITLE |
|---|---|
| 12.1.1-1 | SHNPP RADIATION CONTROL ORGANIZATION AND CORPORATE HEALTH PHYSICS SUPPORT ORGANIZATIONS |
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| 12.2.1-2 | AZIMUTHAL DISTRIBUTION OF GAMMA RAY ENERGY FLUX INCIDENT ON THE PRIMARY SHIELD CONCRETE AT THE REACTOR CORE MIDPLANE |
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| 12.3.2-14 | RADIATION ZONES - WASTE PROCESSING BLDG. - PLAN EL. 236.00' |
| 12.3.2-15 | RADIATION ZONES - WASTE PROCESSING BLDG. - PLAN EL. 261.00' |
| 12.3.2-16 | RADIATION ZONES - WASTE PROCESSING BLDG. - PLAN EL. 276.00' |
| 12.3.2-17 | RADIATION ZONES - WASTE PROCESSING BLDG. - PLAN AT EL. 286.00' AND 291.00' |
| 12.3.2-18 | REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE |
| 12.3.3-1 | REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE |
| 12.3.3-2 | REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE |
| 12.3A-1 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS - LEGEND |

| FIGURE | TITLE |
|---|---|
| 12.3A-2 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-3 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-4 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-5 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-6 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-7 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-8 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-9 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
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| 12.3A-17 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-18 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-19 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-20 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |
| 12.3A-21 | POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS |

FIGURE 12.1.1-1

SHNPP RADIATION CONTROL ORGANIZATION AND CORPORATE HEALTH PHYSICS SUPPORT ORGANIZATION

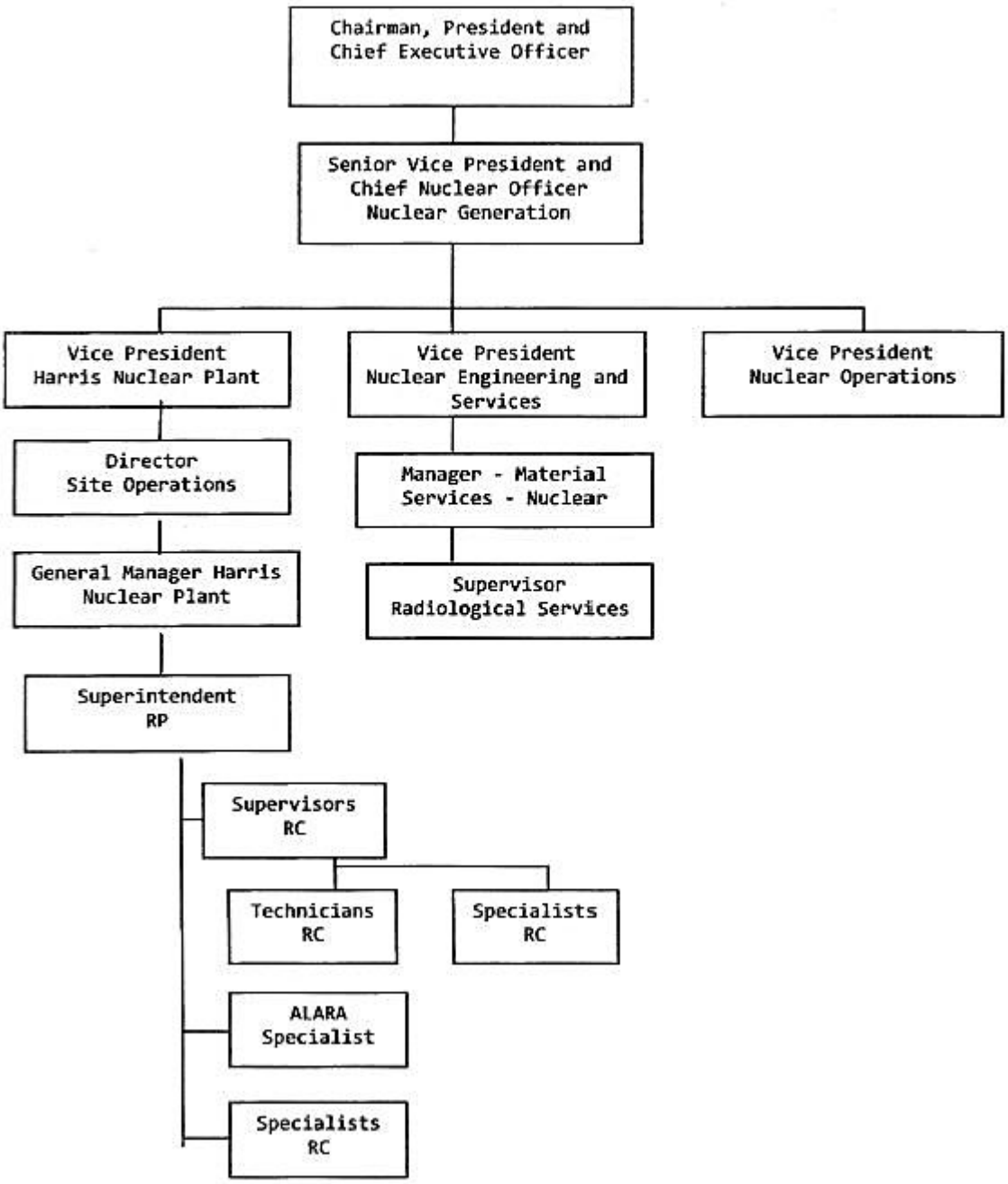

FIGURE 12.2.1-1

AZIMUTHAL DISTRIBUTION OF NEUTRON FLUX INCIDENT ON THE PRIMARY SHIELD CONCRETE AT THE REACTOR CORE MIDPLANE

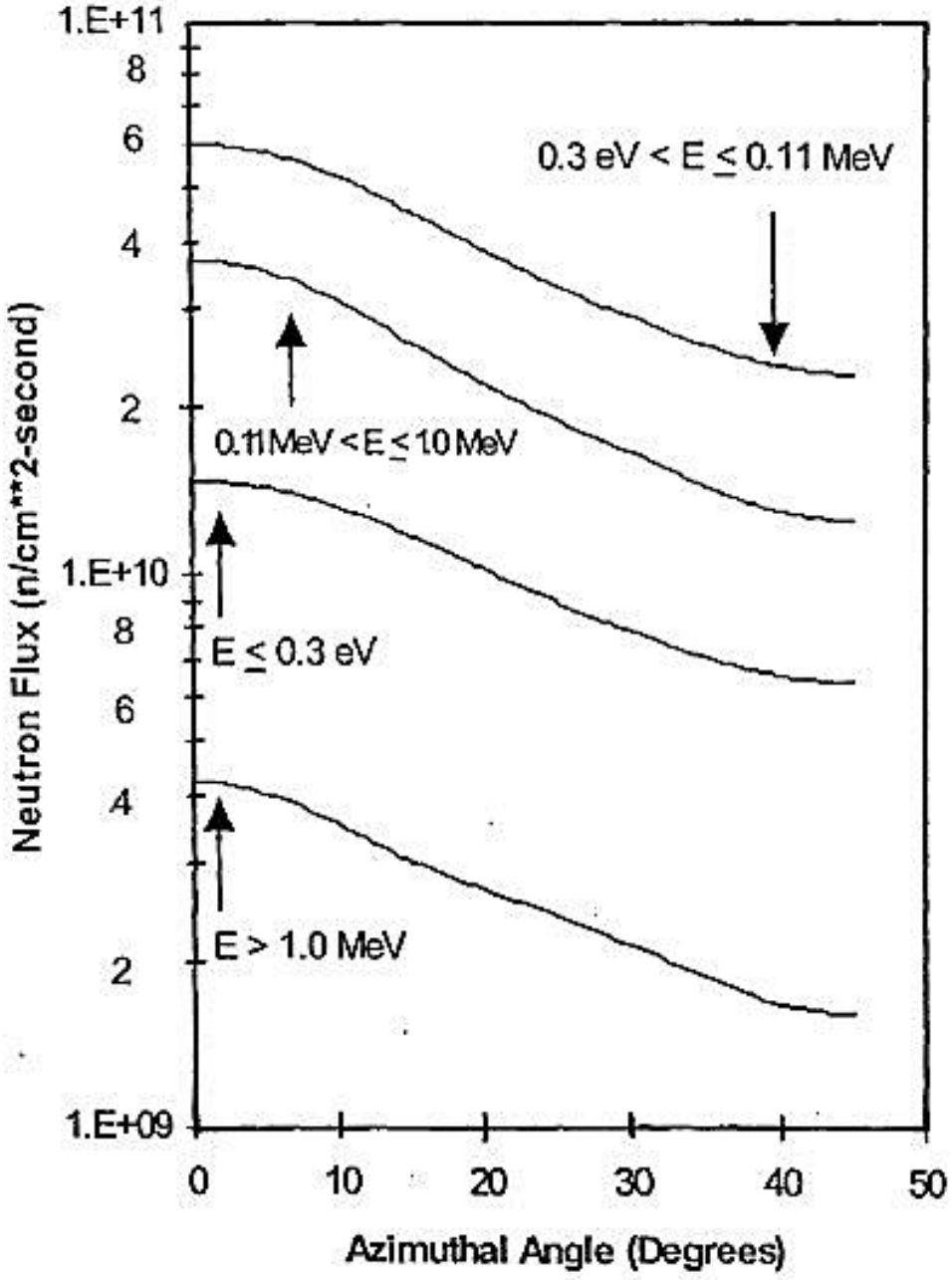

FIGURE 12.2.1-2

AZIMUTHAL DISTRIBUTION OF GAMMA RAY FLUX INCIDENT ON THE PRIMARY SHIELD CONCRETE AT THE REACTOR CORE MIDPLANE

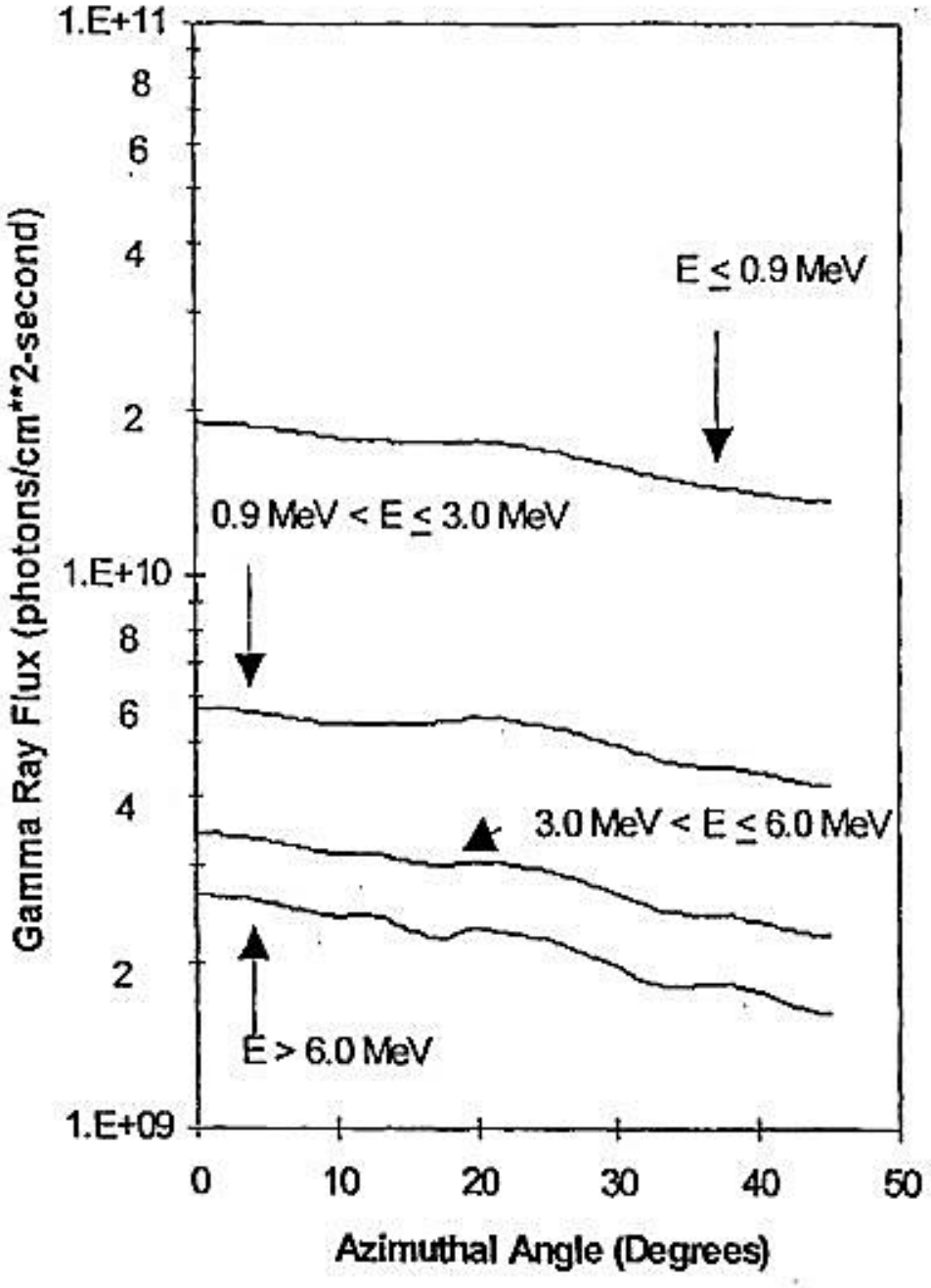

FIGURE 12.2.1-3

AZIMUTHAL DISTRIBUTION OF GAMMA RAY DOSE RATE INCIDENT ON THE PRIMARY SHIELD CONCRETE AT THE REACTOR CORE MIDPLANE

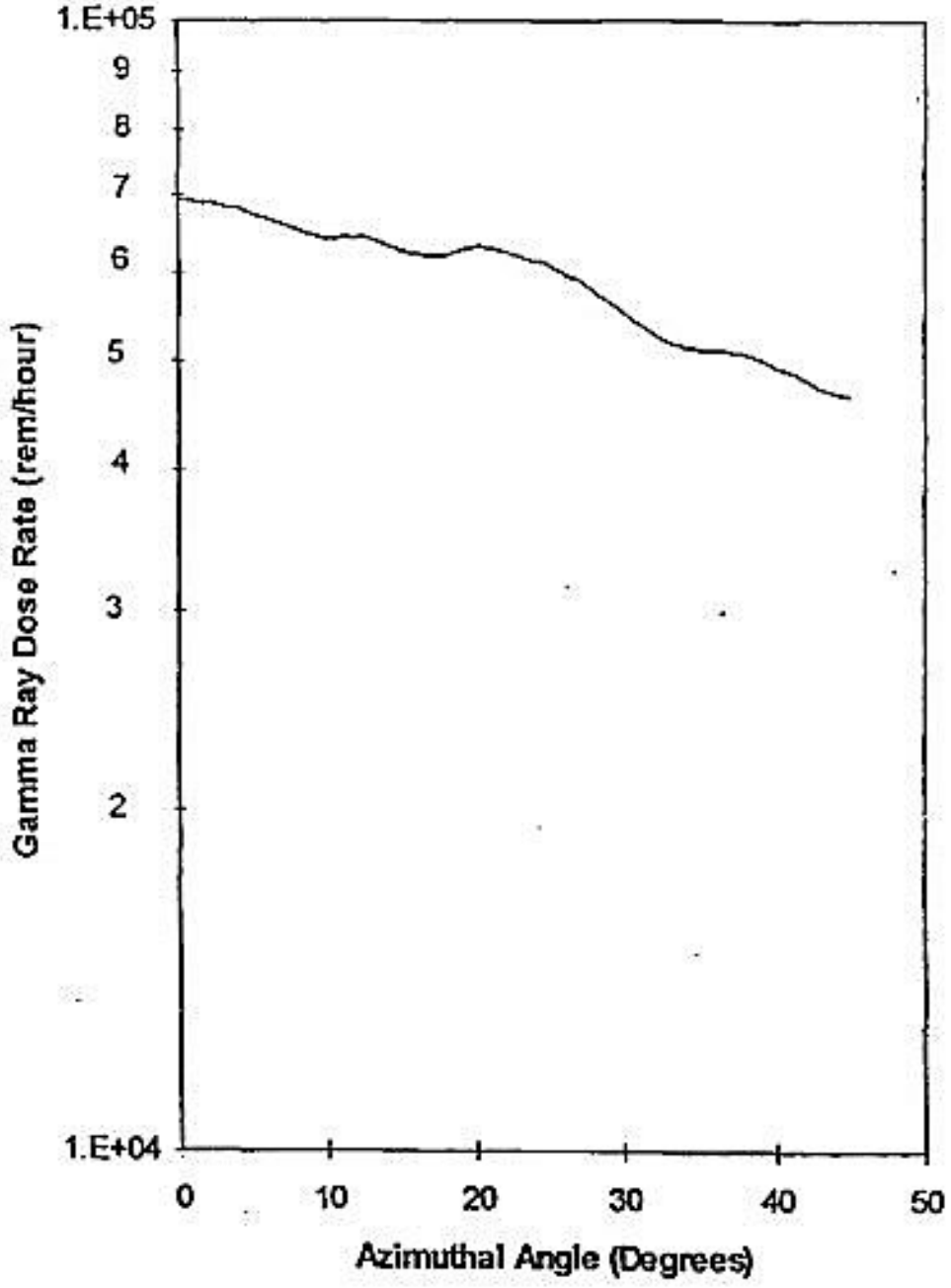

FIGURE 12.3.2-1

RADIATION ZONES – LEGEND

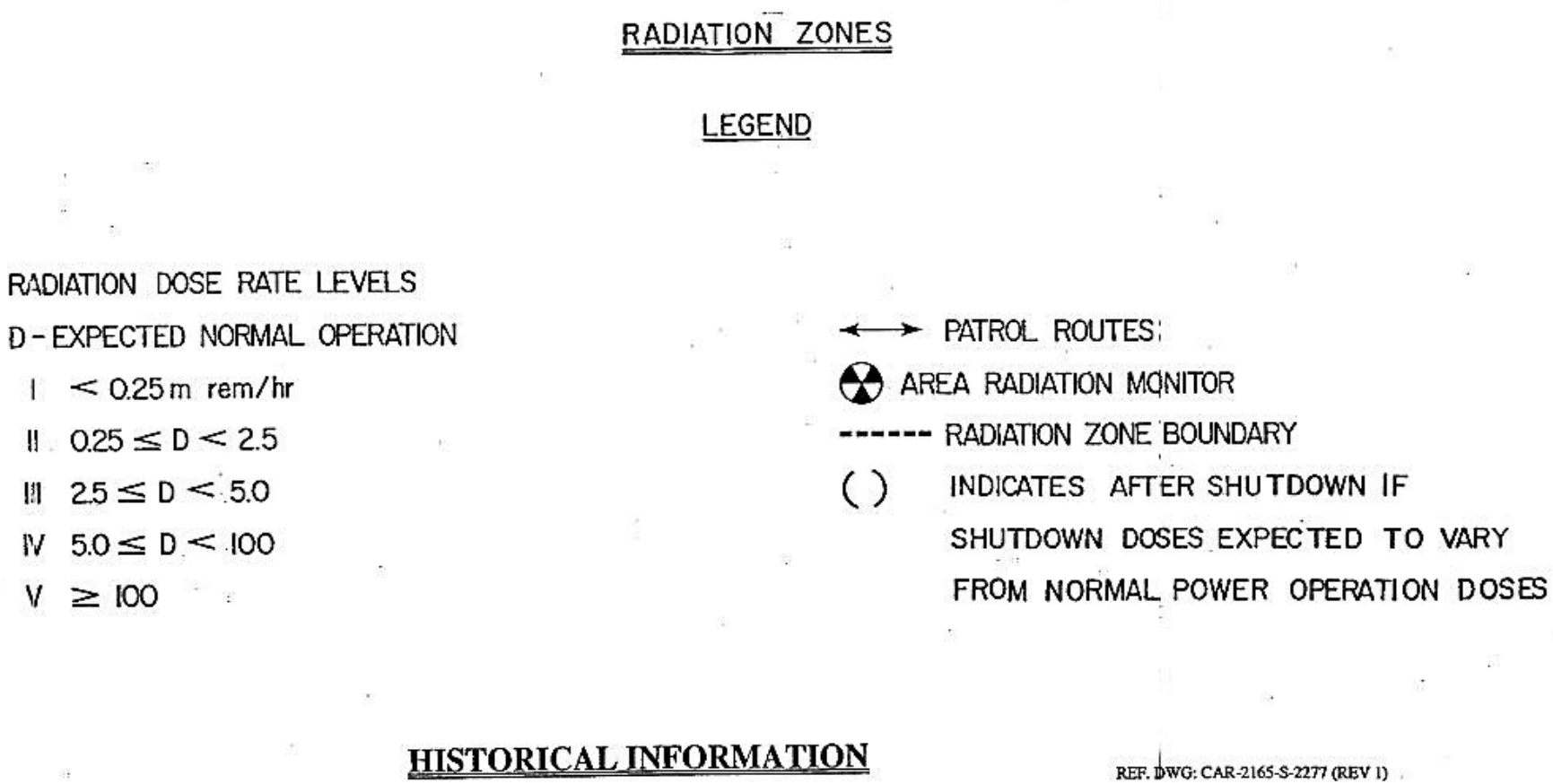

FIGURE 12.3.2-02

RADIATION ZONES CONTAINMENT BUILDING, PLAN EL. 221.00' AND 236.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-03

RADIATION ZONES CONTAINMENT BUILDING, PLAN EL. 261.00' AND 286.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-04

RADIATION ZONES REACTOR AUXILIARY BUILDING, PLAN ELEVATION 190.00’ AND 216.00’

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-05

RADIATION ZONES REACTOR AUXILIARY BUILDING, PLAN ELEVATION 236.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-06

RADIATION ZONES REACTOR AUXILIARY BUILDING, PLAN ELEVATION 261.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-07

RADIATION ZONES REACTOR AUXILIARY BUILDING, PLAN ELEVATION 286.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-8

RADIATION ZONES REACTOR AUXILIARY BUILDING PLAN EL 305.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-9

RADIATION ZONES FUEL HANDLING BUILDING PLANS (SHEET 1)

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-10

RADIATION ZONES FUEL HANDLING BUILDING PLANS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-11

RADIATION ZONES TURBINE BUILDING PLAN EL. 240.00" UNIT 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-12

RADIATION ZONES TANK AREA PLANS AND SECTIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-13

RADIATION ZONES WASTE PROCESSING BUILDING PLAN, AT EL. 211.00' AND 216.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-14

RADIATION ZONES WASTE PROCESSING BUILDING PLAN EL. 236.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-15

RADIATION ZONES WASTE PROCESSING BUILDING PLAN EL. 261.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-16

RADIATION ZONES WASTE PROCESSING BUILDING PLAN EL. 276.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3.2-17

RADIATION ZONES WASTE PROCESSING BUILDING PLAN AT EL. 286.00' AND 291.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-1

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS – UNIT 1, LEGEND

| DOSE RATE | ACCESSIBILITY | SYMBOL |
|---|---|---|
| < 15 MREM/HR | CONTINUOUS ACCESS | |
| 15 – 100 MREM/HR | POSSIBLE FREQUENT ACCESS | |
| 100 MREM/HR – 50 R/HR | POSSIBLE INFREQUENT ACCESS | |
| >50 R/HR | NO ACCESS | |

| | |
|---|---|
| 1 HA | ONE HOUR AFTER THE ACCIDENT |
| 1 DA | ONE DAY AFTER THE ACCIDENT |
| 1 MA | ONE MONTH AFTER THE ACCIDENT |
| ◄► | POST-ACCIDENT SAMPLING ANALYSIS ROUTE |
| ▭ | ZONE BOUNDARY |

FIGURE 12.3A-2

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-3

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-4

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-5

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS – UNIT 1

ecurity-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-6

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS – UNIT 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-7

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS – UNIT 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-8

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-9

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-10

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-11

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-12

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-13

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-14

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-15

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-16

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-17

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-18

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-19

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-20

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 12.3A-21

POST-ACCIDENT DOSE RATES AND ACCESSIBILITY ANALYSIS – UNIT 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390