

12.0 RADIATION PROTECTION

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

Quantitative standards of radiation protection are established and maintained by the International Commission on Radiation Protection (ICRP), the National Committee on Radiation Protection (NCRP), and the former Federal Radiation Council (FRC) which is now part of the Environmental Protection Agency (EPA). The recommendations of these bodies are reflected in 10 CFR Part 20, "Standards for Protection Against Radiation." Specifically, this regulation establishes maximum allowable levels for occupational radiation exposure. In addition to complying with the exposure limits, the regulation specifies that every reasonable effort should be made to maintain exposures as low as is reasonably achievable (ALARA).

12.1.1 Policy Considerations

Refer to the applicant's safety analysis report for a discussion of the applicant's management policy and organizational structure related to ensuring that occupational radiation exposures are ALARA.

The applicant should provide information describing the implementation of policy, organization, training, and design review guidance provided in Regulatory Guides 1.8, 8.8, and 8.10 and the training requirements in 10 CFR 19.12.

Administrative programs and procedures, in conjunction with facility design, should ensure that occupational radiation exposures, both individual and collectively, will be kept below regulatory limits and as low as reasonably achievable as required by 10 CFR 20.1(c).

The applicant's policy should also include a "Radiation Protection Plan" as defined in Section 5 of NUREG 0761.

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Westinghouse is committed to helping ensure that occupational radiation exposures are ALARA in pressurized water reactors by providing systems and components whose designs take into consideration the radiation exposures associated with operation, inspection, and maintenance. In keeping with this commitment, Westinghouse has defined the responsibilities of the radiation protection groups and has provided an environment in which the radiation protection functions can properly perform their duties.

While Westinghouse does not have responsibility for policy considerations related to the operation of systems and components designed and supplied by Westinghouse, a commitment has been made to gather operational information related to radiation protection aspects of Westinghouse systems and components. This operational information is then utilized by the radiation protection staff in working with the designers thereby assuring that operational aspects related to the ALARA philosophy are considered during the design stage. This information is utilized in the development of Westinghouse recommended operation, maintenance, and inspection procedures for the Westinghouse equipment to reflect ALARA practice.

The policy considerations outlined above ensure that occupational radiation exposures are ALARA in compliance with Regulatory Guides 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable," and 8.10, Revision 1, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," and 10 CFR Part 20 (Section 20.1(c)).

12.1.2 Design Considerations

The basic philosophy embodied in Westinghouse pressurized water reactor design considerations to ensure that occupational radiation exposures are ALARA can be expressed as:

1. Design of systems and components to ensure increased reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components.
2. Design of systems and components to reduce the radiation fields to ensure that operation, maintenance, and inspection activities are performed in the minimum radiation field feasible.
3. Design of systems and components to reduce the time spent in radiation fields during operation, maintenance, and inspection.
4. Design of systems and components to accommodate remote and semi-remote operation, maintenance, and inspection procedures.

For the WAPWR, Westinghouse has incorporated many design improvements aimed at reducing operational radiation exposure, including:

1. Simpler/faster refueling operation via integrated head package (these features are discussed further in Section 12.3).
2. Corrosion resistant steam generators.
3. Steam generator maintenance features.
4. 18-24 month fuel cycles.
5. Increased CVCS filtered flow.
6. Shielding/layout improvements.
7. Simplification of major fluid systems (e.g., reduced valve count).
8. Reduction in CVCS safety class (allows much less inspection → less exposure).

In translating this design philosophy into practice, Westinghouse calls upon experience from past designs operating in the field and upon other relevant field experience as well as laboratory tests. Many diverse sources of relevant field experience and data are used in implementing this design philosophy including: NRC publications, the Atomic Industrial Forum (AIF) NESP studies, Electric Power Research Institute studies, internal Westinghouse programs, and personal communications with plant operators. Internal Westinghouse programs, which involve measuring and recording exposure and radiation level data in operating plants during operation, maintenance, and inspection, as well as personal communications with operating plant staffs, have been an invaluable source of feedback from operating plants. Important information is also obtained from Radiation Exposure Management (REM) seminars. During 1984 Westinghouse hosted its sixth annual REM seminar for utility health physics and radiation personnel with representation from nearly all of the Westinghouse reactor sites in the United States and attendees from several foreign countries. The information from all of these sources is assimilated and evaluated by the staff responsible for radiation protection functions. The feedback and radiation data from operating plants is used to construct models to predict occupational radiation exposure patterns for various operations, maintenance, and inspection activities on systems and components of the Westinghouse scope of supply. These models and exposure patterns are further described in Section 12.4. From these models, the potential for improvements in areas such as reliability, repair time, and operational techniques related to occupational radiation exposures can be identified for further study.

Recommended design practice and design considerations are communicated to the system and component designers in three ways within Westinghouse Water Reactor Divisions.

1. Consultation and personal communication.
2. ALARA training programs
3. Design reviews.

The first of these communications techniques is an informal process employing open communication and sound engineering judgment. The second communication method (ALARA training programs) is a formal training session administered by the cognizant radiation protection personnel. The program is given to engineers within the various Westinghouse divisions whose responsibility includes design of systems and components associated with nuclear plants. The aim of the program is to provide design engineers with criteria, design features, operational guidelines, and operating plant experience relevant to radiation protection and the minimizing of occupational radiation exposures. Thus the designer has an awareness of the field conditions and problems imposed by a radiation environment on the operation, maintenance, and inspection of systems and components. The final communication method (design reviews) is the final check by the radiation protection specialists of the system and component designs to ensure that occupational radiation exposures will be ALARA. These design reviews are conducted coincident with the safety design review on systems and components required by 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

As an aid to the designer in ensuring that ALARA features have been addressed, the use of an ALARA design checklist is encouraged. The recommended checklist includes various ALARA considerations grouped by the basic approaches to exposure reduction, such as time, distance, shielding, and source reduction.

Many of the inherent design features and policy considerations to ensure that occupational radiation exposures will be ALARA throughout the operating life are also applicable during the eventual decommissioning of the plant. These features include equipment design for ease of accessibility and maintenance, provisions for remote flushing of equipment, ability to use remote handling equipment, decontamination techniques, and component design features to minimize crud buildup. Specifications and limitations on cobalt content in equipment components will serve to limit radiation doses from crud buildup during both operation and subsequent decommissioning. Westinghouse has been active in studies by AIF and NRC to define the impact of decommissioning by

providing relevant information to the principle investigators. Westinghouse will continue to be aware of decommissioning impacts in the design of systems and components.

The design considerations outlined above ensure that occupational radiation exposures will be maintained ALARA in accordance with Section C.2 of Regulatory Guide 8.8 and in compliance with 10 CFR Part 20 (Section 20.1(c)).

12.1.3 Operational Considerations

Refer to the applicant's safety analysis report for a discussion of the applicant's operational plans and procedures for ensuring that occupational radiation exposures are ALARA.

While Westinghouse does not have ultimate responsibility for policy considerations related to the operation of systems and components designed and supplied by Westinghouse, recommendations and limitations on the operation of the systems and components are supplied to the utility applicant. These recommendations and limitations are developed taking into account the ALARA philosophy. The field experience feedback mechanism described in Section 12.1.2 is also used to develop recommended operation, maintenance, and inspection procedures for systems and components described in RESAR-SP/90 which may contain radioactive materials. These operational considerations are also factored into system and component designs as described in Section 12.1.2.

The operational considerations outlined above ensure that occupational radiation exposures are ALARA in compliance with Regulatory Guides 8.8 and 8.10 and 10 CFR Part 20 (Section 20.1(c)).

12.2 RADIATION SOURCES

This section discusses and identifies the sources of radiation that form the basis for shield design calculations and the sources of airborne radioactivity used for the design of personnel protection measures and dose assessment.

12.2.1 Contained Sources

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

12.2.1.1 Sources for Full Power Operation

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, and activation of reactor coolant corrosion products. The design basis for the shielding source terms for fission products in this section is cladding defects in fuel rods producing 0.25 percent of the core thermal power. The design basis for activation and corrosion product activities is derived from measurements at operating plants and is independent of fuel defect level. The radionuclide activity levels in the reactor coolant at the design basis level are given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management", as are the models and assumptions used in determining these sources.

Westinghouse provides, to the applicant, numerous reactor radiation source values for the at-power condition, including:

1. Neutron particle fluxes at the inside surface of the primary shield concrete at the core midplane.
2. Gamma ray energy fluxes at the inside surface of the primary shield concrete at the core midplane.

3. Gamma ray dose rates at the inside surface of the primary shield concrete.
4. Detailed angular distributions of radiation leakage (neutron and gamma ray) from the reactor pressure for streaming analyses.

The nitrogen-16 activity of the coolant (produced from oxygen activation) is the controlling radiation source in the design of the secondary shield and is tabulated in Table 12.2-1, in microcuries per gram of coolant, as a function of transport time in a reactor coolant loop. The nitrogen-16 source in the pressurizer is given in Table 12.2-2.

Fission and corrosion product activities circulating in the reactor coolant and out-of-core crud deposits comprise the remaining significant radiation sources during full power operation. The fission and corrosion product activities circulating in the reactor coolant are given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management". The fission and corrosion product source strengths in the reactor coolant pressurizer liquid and vapor phases are given in Table 12.2-2. The isotopic composition and specific activity of typical out-of-core crud deposits are given in Table 12.2-3. Typically, 1 milligram of deposited crud material is found in one square centimeter of a relatively smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high-affinity-material areas, and possibly thin-boundary-layer regions.

Systems which process or contain reactor coolant also contain radiation sources during full power operation. These systems include the chemical and volume control system (CVCS) and the boron recycle system (BRS), as described in RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Table 12.2-4 gives the radiation sources for the CVCS, specifically delineating the sources for:

1. CVCS letdown stream.
2. Mixed bed demineralizers.
3. Cation bed demineralizer.

4. Volume control tank (liquid and vapor phases).
5. Reactor coolant, seal water injection, seal water return, boric acid, and boric acid polishing filters.
6. Regenerative, letdown, seal injection, and excess letdown heat exchangers.

Radiation sources in the CVCS consist of radionuclides carried in the reactor coolant. The design of the CVCS ensures that most of the N-16 has decayed before the letdown stream leaves the containment by placing a delay mechanism in the letdown flowpath. All CVCS system heat exchangers other than the seal injection heat exchanger are located in the reactor containment building.

The shielding design is based on the maximum activity in each component:

A. CVCS Ion Exchangers

The mixed bed demineralizer is in continuous use and removes fission products in cation and anion form. It also is highly effective in removing corrosion products. The cation bed demineralizer is used intermittently to remove lithium for pH control. It also is highly effective in removing the monovalent cations, cesium and rubidium. The short-lived isotopes are assumed to build up to saturation activities on both beds. For the long-lived isotopes, the activity retained is assumed to be evenly distributed between the two demineralizers.

B. Volume Control Tank

The radiation sources in the volume control tank are based on a nominal operating level in the tank of [] (a,c)
] in the vapor phase and on the stripping fractions given in Table 11.1-1 of RESAR-SP/90 PDA Module 12, "Waste Management", assuming no volume control tank purge.

C. CVCS Filters

The design criterion for CVCS filter shielding is based primarily on operating experience.

The source strengths for the reactor coolant filter correspond to an exposure rate of 500 rem/hr at contact; the source strengths for the remaining filters correspond to an exposure rate of 100 rem/hr at contact. These dose rates are arrived at assuming the filters are homogeneous sources with the dimensions and composition given in Table 12.2-4.

D. CVCS Heat Exchangers

The regenerative, letdown, and excess letdown heat exchangers are located in the containment building. They provide the cooling for the reactor coolant letdown. Their radiation sources include N-16.

The magnitude of the N-16 source strength is highly sensitive to the location of these heat exchangers with respect to the RCS loop piping. Therefore, the N-16 source strengths for these heat exchangers are based on the average value in the crossover leg between the steam generator and the reactor coolant pump. The shielding design takes into account the N-16 decay from the crossover leg to each heat exchanger.

Table 12.2-5 gives the radiation sources for the BRS, specifically delineating the sources for:

1. Recycle evaporator feed and condensate demineralizers.
2. Recycle holdup tanks (liquid and vapor phases).
3. Recycle evaporator feed and condensate filters.
4. Recycle evaporator vent condenser vapor.
5. Recycle evaporator concentrates.

The shielding design is based on the maximum activity in each component:

A. BRS Ion Exchangers

The recycle evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed-bed resins that remove anion and cation activity from the reactor coolant entering the holdup tanks, along with corrosion products not retained by the CVCS mixed bed demineralizers.

The recycle evaporator condensate demineralizer is charged with anion resin to remove any boron and iodine that may be carried over with the evaporator condensate.

B. Recycle Holdup Tank

The recycle holdup tanks are each equipped with a diaphragm. Gases that flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until about 600 cubic feet of gas has accumulated. The gases are then removed to the gaseous waste system. The radiation sources in the holdup tanks are based on the unit letting down to a single holdup tank at the maximum letdown flow rate, assuming 50 percent of the gases flash into the vapor phase. The liquid phase is assumed to contain reactor coolant that has been processed through the CVCS mixed bed and recycle evaporator feed demineralizers.

C. BRS Filters

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

The source strengths for the feed filter correspond to a radiation level of 100 rem per hour at contact with the filter housing. The condensate filter source strengths result in radiation levels that are less than 1 rem per hour at contact with the filter housing.

The filters are assumed to be drained of process fluid and are considered to be homogeneous sources with the dimensions and compositions shown in Table 12.2-5.

D. Recycle Evaporator

The maximum gaseous activity in the evaporator occurs while processing reactor coolant. The gases are concentrated in the vent condenser portion of the evaporator.

The source strength of the evaporator concentrates is based on an evaporator process limit of 40 microcuries of long-lived radionuclides per gram of concentrates.

The nitrogen-16 activity is not a factor in the radiation sources for systems and components located outside containment such as the CVCS and BRS due to its short 7.11 second half-life and the greater than 1 minute transport time criteria.

Other auxiliary systems containing radiation sources which require shielding include the liquid waste processing system (LWPS), the gaseous waste processing system (GWPS), the solid waste processing system, the spent fuel pit cooling system (SFPCS), and the steam generator blowdown processing system (SGBPS).

The radiation sources for the LWPS are presented in Table 12.2-6.

The liquid waste processing system (LWPS) is considered as several subsystems, based on its intended use during normal operation. The equipment items normally associated with processing reactor-grade water are the reactor

coolant drain tank, waste holdup tank, waste evaporator feed filter, and waste evaporator. The evaporate distillate is directed to the waste condensate tank and may be further processed through the waste evaporator condensate demineralizer and filter. The waste evaporator concentrates are sent to the drumming room for packaging.

Low activity, nonreactor-grade water is directed to the floor drain or laundry and hot shower subsystems. Normally this water is analyzed, then discharged. If activity levels prevent this, the water can be processed by a demineralizer and filter, or the waste evaporator. The equipment included in the subsystem is the floor drain tank and filter, laundry and hot shower tank and filter, waste monitor tank demineralizer and filter, and two waste monitor tanks. The source strengths for the floor drain tank filter are the same as those for the waste evaporator feed filter, since both filters perform similar functions. The source strengths for the waste monitor tank demineralizer and filter are based on processing reactor coolant through these components.

Radioactive spent resins discharged from the various demineralizers are retained in the spent resin storage tank. The short-lived radionuclides are allowed to decay before the resin is directed to the drumming station for packaging. The associated equipment includes the spent resin storage tank and the resin sluice pump and filter.

Radiation sources in the various pumps in this system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

Radiation sources in the laundry and hot shower tank and filter, and in the waste condensate tank are negligible; these items do not require shielding.

The waste evaporator condensate filter and waste monitor tank filter are located downstream of their respective demineralizers. The source strengths for the condensate filter result in radiation levels of less than 5 rem per hour at contact with the filter housing. The maximum radiation level for the waste evaporator feed, spent resin sluice, floor drain tank, and waste monitor tank filters is 100 rem per hour at contact with the filter housing.

The filters are assumed to be drained of process fluid and are considered to be homogeneous sources with the dimensions and compositions shown in Table 12.2-6.

The source strength of the evaporator concentrates is based on an evaporator process limit of 40 microcuries of long-lived radionuclides per gram of concentrates. For shielding design purposes, the concentrates activity should be assumed in the recirculation pump, concentrates heater, flash tank, concentrates cooler and interconnecting piping.

The radiation sources for the GWPS are presented in Table 12.2-7. Radiation sources for each component of the GWPS are based on operation with the maximum activity conditions as given in Subsections 11.1 and 11.3 of RESAR-SP/90 PDA Module 12, "Waste Management". The major equipment items in the GWPS are the refrigerated waste gas dryer, charcoal guard and adsorption beds, gas surge tank, and gas compressor.

The radioactive gases removed from the reactor coolant system at the volume control tank are directed through the refrigerated waste gas dryer, charcoal guard bed and a series of six charcoal adsorption beds. The effluent is then normally discharged via the plant vent although provisions are made for recirculating back to the volume control tank. The gamma ray source strengths for this equipment are derived from plant operation during which the radioactive gases are stripped from the reactor coolant system.

The gamma ray source strengths for the gas compressor and gas surge tank are based on the maximum sources associated with components which can be routed to the surge tank, i.e., the evaporators, RCS drain tank, pressurizer relief tank and BRS holdup tank.

The radiation sources for the solid waste processing system are presented in Table 12.2-8.

The spent resin and evaporator concentrates are packaged at the solid waste drumming station for shipment to a burial or long-term storage facility, which is generally located offsite. Prior to shipment, the packaged waste is stored in a drum storage area.

The initial gamma ray source strength in the spent resin storage tank is assumed to be the same as that in the CVCS mixed-bed demineralizer. After a 30-day decay period, only the cesium and cobalt isotopes are significant contributors. The source strength for the evaporator concentrates is based on an evaporator process limit of 40 microcuries of long-lived radionuclides per gram of concentrates. The initial source strength is based on degassed reactor coolant, less short-lived radionuclides, concentrated by a factor that will yield 40 microcuries per gram at evaporator shutdown. Table 12.2-8 includes the specific gamma ray source strengths, by energy group, at the time of processing and following a 30 day decay period.

The radiation sources for the SFPCS are presented in Table 12.2-9.

The spent fuel pit cooling system is capable of accomplishing simultaneous cleanup of both spent fuel pit and refueling water. The major equipment items considered for shielding design in this system are the demineralizers and filters.

The spent fuel pit demineralizer is charged with 75 cubic feet of mixed-bed resin and is used to remove particulate radionuclides that may be present in the spent fuel pit or refueling water. The specific gamma ray source strengths for the demineralizer are based on purification of a refueling cavity with a water volume of 500,000 gallons.

The spent fuel pit filters are located downstream of the spent fuel pit demineralizer and serve to retain particulates and any resin fines which may escape from the demineralizers.

The filter source strengths correspond to an exposure rate of 100 rem per hour at contact with the filter housing. The filter is assumed to be drained of process fluid and is considered to be a homogeneous source with the dimensions and compositions shown in Table 12.2-9.

The radiation sources for the SGBPS are presented in Table 12.2-10.

Although only those portions of the SGBPS which are safety-related are within the scope of the WAPWR NPB and the non-safety-related portions which process the blowdown are the responsibility of the plant specific applicant (see Section 10.4.8 of RESAR-SP/90 PDA Module 6/8), source strengths for a typical Westinghouse system are presented here for shielding design purposes.

The steam generator blowdown processing system maintains the water effluent from the steam generators at a chemical and radiological specification suitable for its recycle into the main condenser or for its discharge. The major equipment items considered for shielding design in this system are the demineralizers, the SGBPS spent resin storage tank, and the filters.

The blowdown demineralizers contain mixed-bed resins that remove anion and cation activity from the blowdown fluid.

Spent resins from the blowdown demineralizers are transferred to the spent resin storage tank for temporary storage prior to drumming. The spent resin sluice pump takes suction from the liquid contained in the spent resin storage tank.

The blowdown prefilter is located upstream of the blowdown demineralizers and is provided to prevent plugging of the resin beds. The blowdown outlet filter is located downstream of the blowdown demineralizer and serves to retain particulates and any resin fines which may escape from the demineralizers. The spent resin sluice filter serves to retain any resin fines contained in the sluice water.

The filters are assumed to be drained of process fluid and are considered to be homogeneous sources with the dimensions and compositions shown in Table 12-2.10.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only significant sources requiring permanent shielding consideration are the spent reactor fuel, the residual heat removal system, and the incore detector system. Individual components may require shielding during shutdown due to deposited crud material. Estimates of accumulated crud are given in Subsection 12.2.1.1 while dose rates may be obtained from Subsection 12.4.1. The radiation sources in the reactor coolant system, CVCS, and BRS are bounded by the sources given for full power operation with the exception of a short time period (i.e., less than 24 hours) following shutdown during which the fission product spiking phenomena and crud bursts can result in increased radiation sources. The spiking phenomena involves the release of a portion of the accumulated water soluble salts from the interior cladding surface (e.g., iodine and cesium) and gases (e.g., xenon and krypton) of defected fuel rods during the shutdown and coolant depressurization (Reference 1). Crud bursts are the resuspension or solubilization of a portion of the accumulated deposited corrosion products into the reactor coolant system during shutdown such as during oxygenation of the reactor coolant. However, special shielding considerations to accommodate these increases should be unnecessary due to several factors including:

1. The spike or crud burst release is of short duration (generally less than 6 hours).
2. The CVCS purification loop (i.e., letdown through the demineralizer and filters) is generally in operation at full capability during the shutdown [].

(a,c)

The maximum gamma ray source strengths in the residual heat removal system loop are given in Table 12.2-11 for 4 and 8 hours after reactor shutdown. The residual heat removal system is placed into operation at 4 hours following a

shutdown at the maximum shutdown rate. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are maximum values with credit for 4 and 8 hours of fission and corrosion product decay and purification.

Core average gamma ray source strengths are given in Table 12.2-12. These source strengths are used in the evaluation of radiation levels within and around the shutdown reactor. These sources are based on a three region core with the regions operated at 435, 870 and 1306 effective full power days, respectively. Spent fuel gamma ray source strengths are given in Table 12.2-13. These source strengths are used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These sources are based on one core region operated for 1306 effective full power days. All of these sources may be put on a per unit volume of homogenized core basis by multiplying by the power density of 86.6 watts per cubic centimeter for the reactor described in RESAR-SP/90. Core average and spent fuel neutron source strengths are given in Table 12.2-14.

The only source material, byproduct material, or special nuclear material requiring shielding consideration for the Westinghouse nuclear power block described in the RESAR-SP/90 is the photoneutron source material used in the secondary source rods. The gamma ray and neutron source strengths for the secondary source rods, irradiated for 400 days, are given in Table 12.2-15.

Irradiated hybrid incore detector/thermocouple and drive cable maximum gamma ray source strengths are given in Table 12.2-16. These source strengths are used in determining shielding requirements when detectors are being moved during or following a flux mapping of the reactor core. These source strengths are given for a detector irradiation period of 30 days and a drive cable irradiation period of 400 days and are given in terms of per cubic centimeter of detector and drive cable. Irradiated incore detector, drive cable average gamma ray source strengths are given in Table 12.2-17. These source strengths are used in determining shielding requirements when the detectors are not in use and for shipment when the detectors have failed. The values are given in terms of per cubic centimeter of drive cable after an

irradiation period of 400 days. Irradiated incore flux thimble gamma ray source strengths are given in Table 12.2-18. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per cubic centimeter of Inconel-600 for an irradiation period of 15 years. The flux thimbles are made of Inconel-600 with a maximum cobalt impurity content of 0.10 weight percent.

12.2.1.3 Sources for Design Basis Accident

The radiation sources of importance for the design basis accident are the containment source and the residual heat removal system source.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in TID-14844 (Reference 2). These assumptions are consistent with those provided in Regulatory Guide 1.4 and Section II.B.2 of NUREG-0737. The integrated gamma ray and beta particle source strengths for various time periods following the postulated accident are given in Table 12.2-19.

The post-accident recirculation system and shielding should be designed to allow limited access to the high head safety injection (HHSI) and the residual heat removal pumps following a maximum credible accident. The sources are based on the assumptions in TID-14844 with only the nongaseous activity being retained in the sump water, which flows in the residual heat removal loop. Noble gases formed by the decay of halogens in the sump water are assumed to be released to the containment and not retained in the water. Gamma ray source strengths for radiation sources circulating in the residual heat removal loop and associated equipment are given in Table 12.2-20.

Isotopic fission product sources from the maximum credible accident, based on the assumptions in TID-14844, are given in Chapter 15 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System".

12.2.2 Airborne Radioactive Material Sources

Sources of airborne radioactive material in equipment cubicles, corridors, or operating areas normally occupied by operating personnel from systems and components described in RESAR-SP/90 may be obtained from the reactor coolant activities given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management".

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance, etc.) and operating procedures followed during and after shutdown. The permissible coolant activity levels following depressurization should be based on the noble gases evolved from the reactor coolant system water upon removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification should be established based on maximum permissible concentration considerations and containment ventilation system capabilities of the plant. Operating plant experience has indicated that coolant xenon-133 concentrations of less than 0.05 microcuries per gram have posed no problem to the containment atmosphere during vessel head removal.

The exposure rates at the surface of the reactor cavity and spent fuel pool water are dependent on the purification capabilities of the reactor vessel cavity and spent fuel pool cleanup systems. A water activity level of less than 0.005 microcuries per gram for the dominant gamma emitting isotopes at the time of refueling has been shown in operating experience to maintain the dose rate at the water surface to less than 2.5 millirem per hour.

12.2.2.1 Model for Calculating Airborne Concentrations

For those regions which are characterized by a constant leakrate of the radioactive source at constant source strength and a constant exhaust rate of

the contaminant, the peak or equilibrium airborne concentration of the radioisotope in the regions can be calculated, using the following equation:

$$C_i(t) = \frac{(LR)_i A_i (PF)_i (1 - e^{-\lambda_{Ti} t})}{V \lambda_{Ti}}$$

where

$(LR)_i$ = Leak or evaporation rate of the i^{th} radioisotope in gm/sec, in the applicable region

and

A_i = activity concentration of the i^{th} leaking or evaporating radioisotope in $\mu\text{Ci/gm}$

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the i^{th} radioisotope

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

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

$\lambda_{di} + \lambda_e$ are the removal rate constants in sec^{-1} due to radioactive decay for the i^{th} radioisotope and the exhaust from the applicable region

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

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

TABLE 12.2-2 (Sheet 1 of 2)

RADIATION SOURCES - PRESSURIZER

Liquid Phase (1500 ft³)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	9.5×10^4 (a)
0.4 - 0.9	1.6×10^5
0.9 - 1.35	6.8×10^4
1.35 - 1.8	3.8×10^4
1.8 - 2.2	3.5×10^4
2.2 - 2.6	3.4×10^4
2.6 - 3.0	4.7×10^3
3.0 - 4.0	1.9×10^3
4.0 - 5.0	2.2×10^2

Liquid Phase (1000 ft³)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.6×10^5 (a)
0.4 - 0.9	1.1×10^4
0.9 - 1.35	2.9×10^3
1.35 - 1.8	8.0×10^3
1.8 - 2.2	1.3×10^4
2.2 - 2.6	2.8×10^4
2.6 - 3.0	1.9×10^2
3.0 - 4.0	1.5×10^2

TABLE 12.2-2 (Sheet 2 of 2)

RADIATION SOURCES - PRESSURIZER

Nitrogen-16 Sources

Discrete Energy (Mev/gamma)	<u>Specific Source Strength</u>	
	Liquid Phase (Mev/gm-sec)	Vapor Phase (Mev/cm ³ -sec)
1.75	4.3×10^1	1.2×10^0
2.74	3.9×10^2	1.1×10^1
6.13	7.9×10^4	2.2×10^3
7.12	6.7×10^3	1.8×10^2

(a) Includes 80 kev xenon-133.

TABLE 12.2-3

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

<u>Composition (Nuclide)</u>	<u>1 Year</u>	<u>2 Years</u>	<u>5 Years</u>	<u>10 Years</u>
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 12.2-4 (Sheet 1 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEM

Letdown Coolant Sources^(a)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	9.5×10^4 (b)
0.4 - 0.9	1.6×10^5
0.9 - 1.35	6.8×10^4
1.35 - 1.8	3.8×10^4
1.8 - 2.2	3.5×10^4
2.2 - 2.6	3.4×10^4
2.6 - 3.0	4.7×10^3
3.0 - 4.0	1.9×10^3
4.0 - 5.0	2.2×10^2

Mixed Bed Demineralizer
(75 ft³ of Resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	2.2×10^7
0.4 - 0.9	2.0×10^8
0.9 - 1.35	2.3×10^7
1.35 - 1.8	5.8×10^6
1.8 - 2.2	1.7×10^5
2.2 - 2.6	1.0×10^5
2.6 - 3.0	2.3×10^4
3.0 - 4.0	6.7×10^3
4.0 - 5.0	7.8×10^2

TABLE 12.2-4 (Sheet 2 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEMCation Bed Demineralizer
(75 ft³ of Resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.3×10^6
0.4 - 0.9	1.8×10^8
0.9 - 1.35	8.3×10^6
1.35 - 1.8	4.3×10^6
1.8 - 2.2	6.5×10^4
2.2 - 2.6	2.2×10^4
2.6 - 3.0	2.2×10^4
3.0 - 4.0	5.2×10^3
4.0 - 5.0	7.5×10^2

TABLE 12.2-4 (Sheet 3 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEM

Volume Control Tank^(c)

Vapor Phase

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.2×10^6 (b)
0.4 - 0.9	8.3×10^4
0.9 - 1.35	1.6×10^4
1.35 - 1.8	4.8×10^4
1.8 - 2.2	8.4×10^4
2.2 - 2.6	1.8×10^5
2.6 - 3.0	1.0×10^3
3.0 - 4.0	5.9×10^2

Liquid Phase

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	7.7×10^4 (b)
0.4 - 0.9	8.7×10^4
0.9 - 1.35	3.9×10^4
1.35 - 1.8	1.5×10^4
1.8 - 2.2	2.0×10^4
2.2 - 2.6	1.1×10^4
2.6 - 3.0	4.3×10^3
3.0 - 4.0	1.3×10^3
4.0 - 5.0	1.8×10^2

TABLE 12.2-4 (Sheet 4 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEMReactor Coolant Filter^(d)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.4 - 0.9	5.7×10^7
0.9 - 1.35	1.5×10^7

Seal Water Injection Filter^(d)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.4 - 0.9	1.4×10^8
0.9 - 1.35	2.8×10^7

Seal Water Return Filter^(d)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6

Boric Acid Filter^(d)Boric Acid Polishing Filter^(d)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.5×10^5
0.4 - 0.9	8.9×10^5
0.9 - 1.35	3.6×10^5
1.35 - 1.8	4.6×10^4
1.8 - 2.2	3.3×10^3
2.2 - 2.6	2.0×10^3

TABLE 12.2-4 (Sheet 5 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEM

Regenerative, Letdown, and Excess Letdown Heat Exchangers

Energy Group (Mev/gamma)	Specific Source Strength (Mev/gm-sec)	
	Regenerative (Shell Side), Letdown (Tube Side), and Excess Letdown (Tube Side)	Regenerative (Tube Side)
0.2 - 0.4	9.5×10^4 (b)	7.7×10^4 (b)
0.4 - 0.9	1.6×10^5	8.7×10^4
0.9 - 1.35	6.8×10^4	3.9×10^4
1.35 - 1.8	4.3×10^4	1.5×10^4
1.8 - 2.2	3.5×10^4	2.0×10^4
2.2 - 2.6	3.4×10^4	1.1×10^4
2.6 - 3.0	5.6×10^4	4.3×10^3
3.0 - 4.0	1.9×10^3	1.3×10^3
4.0 - 5.0	2.2×10^2	1.8×10^2
6.0 - 7.0	1.0×10^7	-
7.0 - 7.5	8.7×10^6	-

Seal Injection Heat Exchanger (Tube Side)

Energy Group (Mev/gamma)	Specific Source Strength (Mev/gm-sec)	
0.2 - 0.4	9.5×10^4 (b)	
0.4 - 0.9	1.6×10^5	
0.9 - 1.35	6.8×10^4	
1.35 - 1.8	3.8×10^4	
1.8 - 2.2	3.5×10^4	
2.2 - 2.6	3.4×10^4	
2.6 - 3.0	4.7×10^3	
3.0 - 4.0	1.9×10^3	
4.0 - 5.0	2.2×10^2	

TABLE 12.2-4 (Sheet 6 of 6)

RADIATION SOURCES - CHEMICAL
AND VOLUME CONTROL SYSTEM

Notes:

- (a) The letdown coolant volume is plant layout dependent.
- (b) Includes 80 kev xenon-133.
- (c) These sources correspond to a nominal operating level in the tank of 360 ft³ in the vapor phase and 240 ft³ in the liquid phase.
- (d) Homogeneous sources with the following dimensions and compositions:

<u>Filter</u>	<u>Source Dimensions</u>		<u>Source Composition</u> <u>(Volume Percent)</u>
	<u>Radius</u>	<u>Length</u>	
Reactor coolant	3.375	19	67% air, 33% water
Seal water return	3.375	19	67% air, 33% water
Seal water injection	1.375	21	11% air, 89% water
Boric acid	3.375	19	67% air, 33% water
Boric acid polishing	3.375	19	67% air, 33% water

TABLE 12.2-5 (Sheet 1 of 5)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Recycle Evaporator Feed Demineralizer
(75 ft³ of Resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	8.2×10^5
0.4 - 0.9	3.1×10^7
0.9 - 1.35	4.0×10^6
1.35 - 1.8	7.5×10^5
1.8 - 2.2	5.7×10^3
2.2 - 2.6	3.6×10^3

Recycle Evaporator Condensate Demineralizer
(30 ft³ of Resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.6×10^4
0.4 - 0.9	9.8×10^3
0.9 - 1.35	2.0×10^3
1.35 - 1.8	9.9×10^2
1.8 - 2.2	8.5×10^1
2.2 - 2.6	5.8×10^1

TABLE 12.2-5 (Sheet 2 of 5)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Recycle Holdup TanksVapor Phase
(600 ft³)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.4 x 10 ⁵ (a)
0.4 - 0.9	2.0 x 10 ⁴
0.9 - 1.35	5.7 x 10 ³
1.35 - 1.8	1.8 x 10 ⁴
1.8 - 2.2	2.8 x 10 ⁴
2.2 - 2.6	5.4 x 10 ⁴
2.6 - 3.0	6.2 x 10 ²
3.0 - 4.0	8.8 x 10 ²
4.0 - 5.0	5.5 x 10 ¹

Liquid Phase
(100,000 gal)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	8.2 x 10 ⁴ (a)
0.4 - 0.9	2.7 x 10 ⁴
0.9 - 1.35	9.7 x 10 ³
1.35 - 1.8	1.2 x 10 ⁴
1.8 - 2.2	1.8 x 10 ⁴
2.2 - 2.6	3.0 x 10 ⁴
2.6 - 3.0	7.8 x 10 ²
3.0 - 4.0	6.3 x 10 ²
4.0 - 5.0	4.9 x 10 ¹

TABLE 12.2-5 (Sheet 3 of 5)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Recycle Evaporator Feed Filter^(b)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6

Recycle Evaporator Condensate Filter^(b)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	3.0×10^4
0.4 - 0.9	2.0×10^4
0.9 - 1.35	4.0×10^3
1.35 - 1.8	1.9×10^3
1.8 - 2.2	1.7×10^2
2.2 - 2.6	1.1×10^2

TABLE 12.2-5 (Sheet 4 of 5)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Recycle EvaporatorVent Condenser Vapor

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	2.6×10^6 (a)
0.4 - 0.9	3.7×10^5
0.9 - 1.35	1.0×10^5
1.35 - 1.8	3.2×10^5
1.8 - 2.2	5.2×10^5
2.2 - 2.6	9.8×10^5
2.6 - 3.0	1.1×10^4
3.0 - 4.0	1.6×10^4
4.0 - 5.0	1.0×10^3

Evaporator Concentrates

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	1.5×10^5
0.4 - 0.9	8.9×10^5
0.9 - 1.35	3.6×10^5
1.35 - 1.8	4.6×10^4
1.8 - 2.0	3.3×10^3
2.0 - 2.6	2.0×10^3

TABLE 12.2-5 (Sheet 5 of 5)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Notes:

- (a) Includes 80 kev Xenon-133.
- (b) Homogeneous sources with the following dimensions and compositions:

<u>Filter</u>	<u>Source Dimensions</u>		<u>Source Composition</u> <u>(Volume Percent)</u>
	<u>Radius</u>	<u>Length</u>	
Recycle evaporator feed	3.375	19	67% air, 33% water
Recycle evaporator condensate	3.375	19	67% air, 33% water

TABLE 12.2-6 (Sheet 1 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Waste Evaporator Condensate DemineralizerSource Strengths(for 30 cubic feet of resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	7.5×10^4
0.4 - 0.9	1.7×10^5
0.9 - 1.35	4.6×10^4
1.35 - 1.8	7.0×10^3
1.8 - 2.2	4.2×10^2
2.2 - 2.6	2.7×10^2
2.6 - 3.0	9.0×10^0
TOTAL	3.0×10^5

Waste Monitor Tank Demineralizer Source Strengths(for 30 cubic feet of resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	5.8×10^5
0.4 - 0.9	3.9×10^6
0.9 - 1.35	1.6×10^6
1.35 - 1.8	2.3×10^5
1.8 - 2.2	2.3×10^4
2.2 - 2.6	1.0×10^4
TOTAL	6.4×10^6

TABLE 12.2-6 (Sheet 2 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Reactor Coolant Drain Tank Liquid Phase Source Strengths
(for 175 gallons of liquid)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	9.5×10^4 (a)
0.4 - 0.9	1.6×10^5
0.9 - 1.35	6.8×10^4
1.35 - 1.8	3.8×10^4
1.8 - 2.2	3.5×10^4
2.2 - 2.6	3.4×10^4
2.6 - 3.0	4.7×10^3
3.0 - 4.0	1.9×10^3
4.0 - 5.0	2.2×10^2
TOTAL	4.4×10^5

Reactor Coolant Drain Tank Vapor Phase Source Strengths
(For 23.4 cubic feet of vapor)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	4.1×10^5 (a)
0.4 - 0.9	4.5×10^3
0.9 - 1.35	6.7×10^2
1.35 - 1.8	2.0×10^3
1.8 - 2.2	3.5×10^3
2.2 - 2.6	7.4×10^3
2.6 - 3.0	4.1×10^1
3.0 - 4.0	2.1×10^1
TOTAL	4.3×10^5

(a) Includes 80 Kev Xe-133

TABLE 12.2-6 (Sheet 3 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Floor Drain Tank, Waste Monitor Tanks, and
Waste Holdup Tank Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.2 - 0.4	1.6×10^4
0.4 - 0.9	1.4×10^5
0.9 - 1.35	5.8×10^4
1.35 - 1.8	1.7×10^4
1.8 - 2.2	2.6×10^3
2.2 - 2.6	7.8×10^2
2.6 - 3.0	3.0×10^1
TOTAL	2.3×10^5

Chemical Drain Tank Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/gm-sec)</u>
0.4 - 0.9	2.4×10^4
0.9 - 1.35	6.2×10^2
1.35 - 1.8	6.2×10^2
TOTAL	2.8×10^4

TABLE 12.2-6 (Sheet 4 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Spent Resin Storage Tank Source Strengths

Energy Group (Mev/gamma)	Source Strength (Mev/gram-sec)	
	<u>Resin</u>	<u>Liquid</u>
0.2 - 0.4	2.2×10^7	2.2×10^3
0.4 - 0.9	2.0×10^8	2.0×10^4
0.9 - 1.35	2.3×10^7	2.3×10^3
1.35 - 1.8	5.8×10^6	5.8×10^2
1.8 - 2.2	1.7×10^5	1.7×10^1
2.2 - 2.6	1.0×10^5	1.0×10^1
2.6 - 3.0	2.3×10^4	2.3×10^0
3.0 - 4.0	6.7×10^3	6.7×10^{-1}
4.0 - 5.0	7.8×10^2	7.8×10^{-2}
	TOTAL	2.5×10^8

Waste Evaporator Feed, Floor Drain Tank, and
Waste Monitor Tank Filter Source Strengths

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -sec)
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6
	TOTAL
	1.4×10^7

Spent Resin Sluice Filter Source Strengths

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -sec)
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6
	TOTAL
	1.4×10^7

TABLE 12.2-6 (Sheet 5 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Waste Evaporator Condensate Filter Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Specific Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	2.7×10^5
0.4 - 0.9	5.3×10^5
0.9 - 1.35	1.8×10^5
1.35 - 1.8	2.4×10^4
1.8 - 2.2	1.7×10^3
2.2 - 2.6	1.0×10^3
TOTAL	1.0×10^6

Dimensions and Composition of Liquid Waste
Processing System Filters

<u>Filter</u>	<u>Source Dimensions</u> <u>Inches</u>	<u>Source Composition</u> <u>(Volume Percent)</u>
Spent resin sluice	Radius = 3.375 Length = 19.	Air - 67% Water - 33%
Waste evaporator feed	Same	Same
Waste evaporator condensate	Same	Same
Floor drain tank	Same	Same
Waste monitor tank	Same	Same

TABLE 12.2-6 (Sheet 6 of 6)

RADIATION SOURCES - LIQUID WASTE PROCESSING SYSTEM

Waste Evaporator Gas Space Source Strengths

Energy Group (Mev/gamma)	Source Strength (Mev/cm ³ -sec)
0.2 - 0.4	negl.
0.4 - 0.9	negl.
0.9 - 1.35	negl.
1.35 - 1.8	negl.
1.8 - 2.2	negl.
2.2 - 2.6	negl.
2.6 - 3.0	negl.
3.0 - 4.0	negl.
TOTAL	negl.

Waste Evaporator Concentrates Source Strengths

Energy Group (Mev/gamma)	Source Strength (Mev/gm-sec)
0.2 - 0.4	1.5×10^5
0.4 - 0.9	8.9×10^5
0.9 - 1.35	3.6×10^5
1.35 - 1.8	4.6×10^4
1.8 - 2.2	3.3×10^3
2.2 - 2.6	2.0×10^3
TOTAL	1.5×10^6

TABLE 12.2-7 (Sheet 1 of 2)

RADIATION SOURCES - GASEOUS WASTE PROCESSING SYSTEM.

Gaseous Waste Processing System Source Strengths
for Chiller, Gas Surge Tank (300 ft³), and Compressor

<u>Energy Group (Mev/gamma)</u>	<u>Source Strength (Mev/cm³-sec)</u>	
	<u>Chiller</u>	<u>Gas Surge Tank and Compressor</u>
0.2 - 0.4	4.2×10^5 (a)	2.6×10^6 (a)
0.4 - 0.9	7.6×10^4	3.7×10^5
0.9 - 1.35	1.4×10^4	1.0×10^5
1.35 - 1.8	4.2×10^4	3.2×10^5
1.8 - 2.2	7.3×10^4	5.2×10^5
2.2 - 2.6	1.6×10^5	9.8×10^5
2.6 - 3.0	9.1×10^2	1.1×10^4
3.0 - 4.0	5.4×10^2	1.6×10^4
4.0 - 5.0		1.0×10^3
TOTAL	7.8×10^5	5.0×10^6

(a) Includes 80 kev Xe-133.

TABLE 12.2-7 (Sheet 2 of 2)

RADIATION SOURCES - GASEOUS WASTE PROCESSING SYSTEM

Gaseous Waste Processing System Source Strengths
for Charcoal Guard and Adsorber Beds
 (for 75 ft³ Guard Bed and 150 ft³ Adsorber Beds)

<u>Energy Group (Mev/gamma)</u>	<u>Source Strength (Mev/cm³-sec)</u>	
	<u>Guard Bed</u>	<u>Adsorber Beds</u>
0.2 - 0.4	5.0 x 10 ⁶ (a)	3.7 x 10 ⁶ (a)
0.4 - 0.9	6.3 x 10 ⁴	3.4 x 10 ⁴
0.9 - 1.35	9.3 x 10 ³	5.6 x 10 ³
1.35 - 1.8	2.8 x 10 ⁴	1.7 x 10 ⁴
1.8 - 2.2	4.9 x 10 ⁴	3.0 x 10 ⁴
2.2 - 2.6	1.0 x 10 ⁵	6.3 x 10 ⁴
2.6 - 3.0	5.2 x 10 ²	3.1 x 10 ²
3.0 - 4.0	1.9 x 10 ²	9.8 x 10 ¹
TOTAL	5.3 x 10 ⁶	3.8 x 10 ⁶

Gaseous Waste Processing System Source Strengths
For HEPA Filter

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.2 x 10 ⁶ (a)
0.4 - 0.9	8.1 x 10 ²
0.9 - 1.35	7.9 x 10 ⁰
1.35 - 1.8	1.7 x 10 ¹
1.8 - 2.2	2.1 x 10 ¹
2.2 - 2.6	1.2 x 10 ²
TOTAL	1.2 x 10 ⁶

TABLE 12.2-8

RADIATION SOURCES - SOLID WASTE PROCESSING SYSTEM

Solid Waste Source Strengths at Time of Processing

Energy Group (Mev/gamma)	Spent Resin Source Strength (Mev/cm ³ -sec)	Evaporator Concentrates Source Strength (Mev/gram-sec)
0.2 - 0.4	2.2×10^7	1.5×10^5
0.4 - 0.9	2.0×10^8	8.9×10^5
0.9 - 1.35	2.3×10^7	3.6×10^5
1.35 - 1.8	5.8×10^6	4.6×10^4
1.8 - 2.2	1.7×10^5	3.3×10^3
2.2 - 2.6	1.0×10^5	2.0×10^3
2.6 - 3.0	2.3×10^4	-
3.0 - 4.0	6.7×10^3	-
4.0 - 5.0	7.8×10^2	-
TOTAL	2.5×10^8	1.5×10^6

Solid Waste Source Strengths Following 30 Days Decay

Energy Group (Mev/gamma)	Spent Resin Source Strength (Mev/cm ³ -sec)	Evaporator Concentrates Source Strength (Mev/gram-sec)
0.2 - 0.4	2.0×10^6	1.8×10^4
0.4 - 0.9	1.8×10^8	5.1×10^5
0.9 - 1.35	1.6×10^7	6.9×10^4
1.35 - 1.8	4.3×10^6	1.2×10^4
TOTAL	2.0×10^8	6.1×10^5

TABLE 12.2-9

RADIATION SOURCES - SPENT FUEL PIT COOLING SYSTEM

Spent Fuel Pit Demineralizer Source Strengths
(For 75 cubic feet of resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	1.0×10^4
0.4 - 0.9	6.5×10^5
0.9 - 1.35	3.0×10^5
1.35 - 1.8	5.9×10^3

Spent Fuel Pit Filter Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6

Spent Fuel Pit Filter Dimensions and Composition

<u>Filter</u>	<u>Source Dimensions</u> <u>(inches)</u>	<u>Source Composition</u> <u>(volume percent)</u>
Spent fuel pit	Radius = 3.375 Length = 19	Air - 67% Water - 33%

TABLE 12.2-10 (Sheet 1 of 3)

RADIATION SOURCES - STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

Steam Generator Blowdown Demineralizer Source Strengths
(For 75 cubic feet of resin)

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	2.9×10^5
0.4 - 0.9	3.9×10^7
0.9 - 1.35	2.2×10^6
1.35 - 1.8	9.6×10^5
1.8 - 2.2	1.7×10^3
2.2 - 2.6	5.4×10^2

Steam Generator Blowdown Spent Resin Storage Tank Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength (Mev/gram-sec)</u>	
	<u>Resin</u>	<u>Liquid</u>
0.2 - 0.4	2.9×10^5	2.9×10^1
0.4 - 0.9	3.9×10^7	3.9×10^3
0.9 - 1.35	2.2×10^6	2.2×10^2
1.35 - 1.8	9.6×10^5	9.6×10^1
1.8 - 2.2	1.7×10^3	1.7×10^{-1}
2.2 - 2.6	5.4×10^2	5.4×10^{-2}

TABLE 12.2-10 (Sheet 2 of 3)

RADIATION SOURCES - STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

Steam Generator Blowdown Prefilter Source Strengths

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	9.9×10^1
0.4 - 0.9	1.6×10^3
0.9 - 1.35	6.5×10^2
1.35 - 1.8	9.6×10^1
1.8 - 2.2	1.3×10^1
2.2 - 2.6	5.8×10^0
2.6 - 3.0	2.8×10^0

Steam Generator Blowdown Outlet and Spent Resin
Sluice Filter Specific Activity

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Source Strength</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	2.9×10^5
0.4 - 0.9	3.9×10^7
0.9 - 1.35	2.2×10^6
1.35 - 1.8	9.6×10^5
1.8 - 2.2	1.7×10^3
2.2 - 2.6	5.4×10^2

TABLE 12.2-10 (Sheet 3 of 3)

RADIATION SOURCES - STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

Dimensions and Composition of SGBPS Filters

<u>Filter</u>	<u>Source Dimensions (inches)</u>	<u>Source Composition (volume percent)</u>
Blowdown prefilter	Radius = 5.0 Length = 36	Air - 66% Water - 32% Stainless Steel - 2%
Blowdown outlet Spent resin sluice	Radius = 3.375 Length = 19	Air - 67% Water - 33%

TABLE 12.2-11

RADIATION SOURCES - RESIDUAL HEAT REMOVAL SYSTEM

<u>Energy Group (Mev/gamma)</u>	<u>Source Strength (Mev/gm-sec)</u>	
	<u>4 Hours After Shutdown</u>	<u>8 Hours After Shutdown</u>
0.2 - 0.4	7.6×10^4 (a)	6.5×10^4 (a)
0.4 - 0.9	5.8×10^4	2.8×10^4
0.9 - 1.35	2.4×10^4	1.1×10^4
1.35 - 1.8	7.0×10^3	2.3×10^3
1.8 - 2.2	5.8×10^3	1.3×10^3
2.2 - 2.6	6.1×10^3	1.3×10^3
2.6 - 3.0	4.5×10^2	9.6×10^1
3.0 - 4.0	1.8×10^2	3.7×10^1
4.0 - 5.0	3.0×10^1	6.5×10^0

(a) Includes 80 kev xenon-133.

TABLE 12.2-12

CORE AVERAGE GAMMA RAY SOURCE STRENGTHS AT
VARIOUS TIMES AFTER SHUTDOWN

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.2 - 0.4	1.8×10^9	1.6×10^9	8.4×10^8	6.0×10^8	1.5×10^8
0.4 - 0.9	1.1×10^{10}	9.6×10^9	6.5×10^9	5.6×10^9	3.5×10^9
0.9 - 1.35	2.0×10^9	1.3×10^9	6.7×10^8	5.0×10^8	1.5×10^8
1.35 - 1.8	3.7×10^9	3.3×10^9	2.7×10^9	2.3×10^9	6.5×10^8
1.8 - 2.2	3.0×10^8	2.3×10^8	1.5×10^8	1.3×10^8	5.0×10^7
2.2 - 2.6	2.5×10^8	1.8×10^8	1.6×10^8	1.4×10^8	3.9×10^7
2.6 - 3.0	6.2×10^6	3.2×10^6	2.7×10^6	2.3×10^6	6.8×10^5
3.0 - 4.0	4.3×10^6	1.3×10^6	1.1×10^6	9.2×10^5	2.7×10^5
4.0 - 5.0	2.0×10^5	1.0×10^4	-	-	-
Energy Group (Mev/gamma)	3 Months	6 Months	1 Year	5 Years	
0.2 - 0.4	5.1×10^7	3.0×10^7	1.7×10^7	2.0×10^6	
0.4 - 0.9	2.0×10^9	1.0×10^9	4.6×10^8	1.2×10^8	
0.9 - 1.35	4.2×10^7	2.8×10^7	2.1×10^7	7.8×10^6	
1.35 - 1.8	4.5×10^7	1.7×10^7	1.2×10^7	2.2×10^6	
1.8 - 2.2	1.8×10^7	1.3×10^7	8.4×10^6	2.4×10^5	
2.2 - 2.6	1.5×10^6	1.2×10^4	-	-	
2.6 - 3.0	2.6×10^5	2.0×10^2	-	-	
3.0 - 4.0	1.0×10^4	-	-	-	

TABLE 12.2-13

SPENT FUEL GAMMA RAY SOURCE STRENGTHS AT
VARIOUS TIMES AFTER SHUTDOWN

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.2 - 0.4	1.9×10^9	1.6×10^9	8.6×10^8	6.2×10^8	1.6×10^8
0.4 - 0.9	1.2×10^{10}	1.0×10^{10}	6.8×10^9	5.9×10^9	3.8×10^9
0.9 - 1.35	2.2×10^9	1.5×10^9	8.5×10^8	6.6×10^8	2.2×10^8
1.35 - 1.8	3.6×10^9	3.2×10^9	2.6×10^9	2.2×10^9	6.4×10^8
1.8 - 2.2	4.0×10^8	3.2×10^8	2.4×10^8	2.0×10^8	7.7×10^7
2.2 - 2.6	2.3×10^8	1.8×10^8	1.5×10^8	1.3×10^8	3.8×10^7
2.6 - 3.0	5.6×10^6	3.1×10^6	2.6×10^6	2.2×10^6	6.5×10^5
3.0 - 4.0	4.0×10^6	1.2×10^6	1.0×10^6	8.9×10^5	2.6×10^5
4.0 - 5.0	1.6×10^5	8.2×10^3	-	-	-
	3 Months	6 Months	1 Year	5 Years	
0.2 - 0.4	5.6×10^7	3.3×10^7	1.9×10^7	2.8×10^6	
0.4 - 0.9	2.3×10^9	1.3×10^9	7.1×10^8	2.1×10^8	
0.9 - 1.35	6.4×10^7	4.5×10^7	3.5×10^7	1.4×10^7	
1.35 - 1.8	5.6×10^7	2.7×10^7	2.0×10^7	3.8×10^6	
1.8 - 2.2	2.2×10^7	1.4×10^7	9.1×10^6	2.6×10^5	
2.2 - 2.6	1.5×10^6	1.1×10^4	-	-	
2.6 - 3.0	2.5×10^4	1.9×10^2	-	-	
3.0 - 4.0	1.0×10^4	-	-	-	

TABLE 12.2-14

CORE AVERAGE AND SPENT FUEL NEUTRON SOURCE
STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

<u>Time After Shutdown</u>	<u>Core Average (n/watt-sec)</u>	<u>Spent Fuel (n/watt-sec)</u>
12 hours	29.5	71.4
24 hours	29.5	71.4
100 hours	29.3	71.4
1 week	29.3	71.2
1 month	28.3	69.5
3 months	26.7	65.7
6 months	24.5	61.4
1 year	22.0	55.9
5 years	17.4	44.8

83 to 93 percent of the neutron source strength is due to the spontaneous fission of curium-242 and curium-244. The curium spontaneous fission neutron spectrum is quite similar to that of californium-252. The californium-252 spontaneous fission neutron spectrum may be expressed by a Watt formula as follows:

$$x(E) = 0.37 \exp(-0.88E) \sinh(\sqrt{2.0E})$$

where E is the neutron energy and $x(E)$ is normalized so that

$$\int_0^{\infty} x(E) dE = 1$$

TABLE 12.2-15 (Sheet 1 of 2)

IRRADIATED Sb-Be SECONDARY SOURCE ROD
SOURCE STRENGTHSGamma Ray

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm ³ -sec)					
	<u>1 Day</u>	<u>1 Week</u>	<u>1 Month</u>	<u>6 Months</u>	<u>1 Year</u>	<u>5 Years</u>
0.2 - 0.4	3.0×10^{10}	2.9×10^{10}	2.5×10^{10}	1.1×10^{10}	3.7×10^9	2.2×10^7
0.4 - 0.9	1.1×10^{13}	7.0×10^{12}	4.6×10^{12}	8.1×10^{11}	9.7×10^{10}	1.8×10^8
0.9 - 1.35	6.7×10^{11}	4.8×10^{11}	3.4×10^{11}	6.0×10^{10}	7.0×10^9	-
1.35 - 1.8	7.6×10^{12}	7.1×10^{12}	5.5×10^{12}	9.7×10^{11}	1.2×10^{11}	-
1.8 - 2.2	9.8×10^{11}	9.1×10^{11}	7.0×10^{11}	1.2×10^{11}	1.5×10^{10}	-

The secondary source cross-sectional area is 3.96 cm² per rod.

The Sb-Be material density is 3.38 gm/cm³.

TABLE 12.2-15 (Sheet 2 of 2)

IRRADIATED Sb-Be SECONDARY SOURCE ROD
SOURCE STRENGTHS

<u>Time After Shutdown</u>	<u>Neutron</u>
	<u>Neutron Source Strength</u> <u>(n/cm³-sec)</u>
1 day	4.5×10^8
1 week	4.2×10^8
1 month	3.2×10^8
6 months	5.8×10^7
1 year	6.8×10^6
5 years	-

The secondary source rod cross-sectional area is 3.96 cm^2 per rod.

The average neutron energy is 30 kev.

The Sb-Be material density is 3.38 gm/cm^3 .

TABLE 12.2-16

IRRADIATED INCORE DETECTOR AND DRIVE CABLE MAXIMUM
WITHDRAWAL SOURCE STRENGTHS

<u>Energy Group</u> <u>(Mev/gamma)</u>	<u>Incore Detector</u> <u>(Mev/cm³-sec)</u>	<u>Drive Cable</u> <u>(Mev/cm³-sec)</u>
0.2 - 0.4	8.1×10^{10}	1.9×10^9
0.4 - 0.9	3.3×10^{11}	1.4×10^{11}
0.9 - 1.35	2.4×10^{11}	2.9×10^{10}
1.35 - 1.8	2.4×10^{11}	1.1×10^9
1.8 - 2.2	6.2×10^{10}	1.3×10^{11}
2.2 - 2.6	6.6×10^{10}	4.2×10^{10}
2.6 - 3.0	3.4×10^{10}	4.5×10^9
3.0 - 4.0	2.1×10^{10}	1.1×10^9
4.0 - 5.0	1.1×10^{10}	-
5.0 - 6.0	3.0×10^9	-

The effective diameter and length of the incore detector are 0.305 and 3.56 cm, respectively.

The effective cross-sectional area of the drive cable is 0.0302 cm^2 .

TABLE 12.2-17

IRRADIATED IN-CORE DETECTOR DRIVE CABLE SOURCE STRENGTHS

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm ³ -sec)						
	8 Hours	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.2 - 0.4	1.9×10^9	1.8×10^9	1.6×10^9	9.1×10^8	2.4×10^7	4.2×10^5	-
0.4 - 0.9	2.9×10^{10}	1.5×10^{10}	1.4×10^{10}	1.3×10^{10}	7.9×10^9	4.9×10^9	1.9×10^8
0.9 - 1.35	2.8×10^{10}	2.8×10^{10}	2.7×10^{10}	2.4×10^{10}	1.7×10^{10}	1.5×10^{10}	9.1×10^9
1.35 - 1.8	7.7×10^7	4.2×10^7	3.9×10^7	3.1×10^7	7.3×10^6	1.2×10^6	-
1.8 - 2.2	1.5×10^{10}	2.1×10^8	-	-	-	-	-
2.2 - 2.6	5.0×10^8	6.8×10^6	-	-	-	-	-
2.6 - 3.0	5.3×10^8	7.2×10^6	-	-	-	-	-
3.0 - 4.0	1.2×10^8	1.7×10^6	-	-	-	-	-

The drive cable effective cross-sectional area is 0.0302 cm².

TABLE 12.2-18

IRRADIATED INCONEL 600 (0.10 WEIGHT PERCENT Co)
FLUX THIMBLE SOURCE STRENGTHS

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm ³ -sec)						
	12 Hours	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.2 - 0.4	7.2×10^9	7.1×10^9	6.1×10^9	3.4×10^9	8.1×10^7	8.0×10^5	-
0.4 - 0.9	1.5×10^{11}	1.5×10^{11}	1.4×10^{11}	1.1×10^{11}	2.1×10^{10}	4.6×10^9	1.7×10^7
0.9 - 1.35	4.6×10^{10}	4.6×10^{10}	4.5×10^{10}	4.5×10^{10}	4.2×10^{10}	3.9×10^{10}	2.3×10^{10}
1.35 - 1.8	2.0×10^{10}	1.6×10^9	1.4×10^9	1.2×10^9	2.7×10^8	4.5×10^7	-
1.8 - 2.2	1.7×10^8	6.8×10^5	-	-	-	-	-
2.2 - 2.6	5.6×10^7	2.2×10^4	-	-	-	-	-
2.6 - 3.0	6.0×10^7	2.4×10^4	-	-	-	-	-
3.0 - 4.0	1.4×10^6	5.5×10^3	-	-	-	-	-

The flux thimble cross-sectional area is 0.466 cm^2 .

TABLE 12.2-19

INTEGRATED GAMMA RAY AND BETA SOURCE STRENGTHS AT VARIOUS TIMES
FOLLOWING A MAXIMUM CREDIBLE ACCIDENT (TID-14844 Release Fractions)

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt)				
	0.5 Hour	1 Hour	2 Hours	8 Hours	1 Day
0.2 - 0.4	1.3×10^{12}	2.3×10^{12}	4.0×10^{12}	1.3×10^{13}	3.3×10^{13}
0.4 - 0.9	8.4×10^{12}	1.5×10^{13}	2.4×10^{13}	5.3×10^{13}	8.6×10^{13}
0.9 - 1.35	3.7×10^{12}	6.4×10^{12}	1.1×10^{13}	2.5×10^{13}	3.8×10^{13}
1.35 - 1.8	3.6×10^{12}	6.3×10^{12}	1.0×10^{13}	2.1×10^{13}	3.0×10^{13}
1.8 - 2.2	1.9×10^{12}	3.2×10^{12}	5.2×10^{12}	1.1×10^{13}	1.3×10^{13}
2.2 - 2.6	2.0×10^{12}	3.7×10^{12}	6.1×10^{12}	1.2×10^{13}	1.4×10^{13}
2.6 - 3.0	3.4×10^{11}	5.5×10^{11}	8.4×10^{11}	1.3×10^{12}	1.4×10^{12}
3.0 - 4.0	3.6×10^{11}	4.6×10^{11}	5.8×10^{11}	8.0×10^{11}	8.3×10^{11}
4.0 - 5.0	1.6×10^{11}	1.6×10^{11}	1.7×10^{11}	2.0×10^{11}	2.0×10^{11}
5.0 - 6.0	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}
Beta	1.3×10^{13}	2.2×10^{13}	3.6×10^{13}	8.9×10^{13}	1.5×10^{14}
	1 Week	1 Month	6 Months	1 Year	
0.2 - 0.4	1.3×10^{14}	2.3×10^{14}	2.6×10^{14}	2.6×10^6	
0.4 - 0.9	1.7×10^{14}	2.7×10^{14}	5.3×10^{14}	6.4×10^8	
0.9 - 1.35	4.8×10^{13}	5.3×10^{13}	6.0×10^{13}	6.4×10^7	
1.35 - 1.8	4.6×10^{13}	7.2×10^{13}	8.6×10^{13}	8.9×10^6	
1.8 - 2.2	1.4×10^{13}	1.5×10^{13}	1.8×10^{13}	2.0×10^5	
2.2 - 2.6	1.5×10^{13}	1.7×10^{13}	1.8×10^{13}	1.8×10^{13}	
2.6 - 3.0	1.4×10^{12}	1.5×10^{12}	1.5×10^{12}	1.5×10^{12}	
3.0 - 4.0	8.4×10^{11}	8.5×10^{11}	8.5×10^{11}	8.5×10^{11}	
4.0 - 5.0	2.0×10^{11}	2.0×10^{11}	2.0×10^{11}	2.0×10^{11}	
5.0 - 6.0	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	
Beta	3.9×10^{14}	6.5×10^{14}	1.1×10^{15}	1.4×10^{15}	

TABLE 12.2-20

SOURCE STRENGTH IN THE RESIDUAL HEAT REMOVAL LOOP AT VARIOUS TIMES
FOLLOWING AN EQUIVALENT FULL CORE MELTDOWN ACCIDENT

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/gm-sec-watt)				
	0 Hour	0.5 Hour	1 Hour	2 Hours	8 Hours
0.2 - 0.4	2.6×10^{-1}	1.8×10^{-1}	1.6×10^{-1}	1.5×10^{-1}	1.3×10^{-1}
0.4 - 0.9	3.1	2.4	1.9	1.4	5.3×10^{-1}
0.9 - 1.35	1.5	9.4×10^{-1}	8.2×10^{-1}	6.5×10^{-1}	2.9×10^{-1}
1.35 - 1.8	1.1	6.3×10^{-1}	5.4×10^{-1}	4.3×10^{-1}	1.9×10^{-1}
1.8 - 2.2	1.0×10^{-1}	6.5×10^{-2}	5.2×10^{-2}	3.8×10^{-2}	1.4×10^{-2}
2.2 - 2.6	2.2×10^{-1}	4.4×10^{-2}	3.6×10^{-2}	2.7×10^{-2}	1.2×10^{-2}
2.6 - 3.0	2.7×10^{-1}	4.5×10^{-3}	2.1×10^{-3}	5.7×10^{-4}	-
3.0 - 4.0	1.5×10^{-1}	3.1×10^{-2}	1.5×10^{-2}	4.1×10^{-3}	-
4.0 - 5.0	1.6×10^{-1}	4.0×10^{-4}	2.0×10^{-4}	-	-
5.0 - 6.0	1.2×10^{-3}	-	-	-	-
Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/gm-sec-watt)				
	1 Day	1 Week	1 Month	6 Months	1 Year
0.2 - 0.4	1.1×10^{-1}	6.5×10^{-2}	9.4×10^{-3}	2.0×10^{-4}	1.2×10^{-4}
0.4 - 0.9	2.8×10^{-1}	5.4×10^{-2}	2.6×10^{-2}	7.2×10^{-3}	3.1×10^{-3}
0.9 - 1.35	7.0×10^{-2}	3.6×10^{-3}	1.0×10^{-3}	1.9×10^{-4}	1.4×10^{-4}
1.35 - 1.8	5.1×10^{-2}	1.6×10^{-2}	4.4×10^{-3}	1.2×10^{-4}	8.4×10^{-5}
1.8 - 2.2	3.3×10^{-3}	8.6×10^{-4}	3.4×10^{-4}	9.0×10^{-5}	5.7×10^{-5}
2.2 - 2.6	3.2×10^{-3}	9.2×10^{-4}	2.7×10^{-4}	-	-

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

Specific design features for maintaining personnel exposure as low as reasonably achievable (ALARA) are discussed in this subsection. The design feature recommendations given in Regulatory Guide 8.8, Paragraph C.2, are utilized to minimize exposures to personnel.

12.3.1.1 Plant Design Description for ALARA

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of Subsection 12.1.2 and are outlined in this subsection.

12.3.1.1.1 Equipment and Component Designs for ALARA

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

12.3.1.1.1.1 Nuclear Power Block (NPB) Equipment

A. Reactor Vessel

The reactor vessel design includes an integrated head package which combines the head lifting rig, control and gray rod drive mechanism (CRDM's/GRDM's), water displacer rod drive mechanisms (WDR's) seismic supports, lift columns, reactor vessel missile shield, CRDM cooling system and power and instrumentation cabling into an effective, one-package reactor vessel head design. Mounted directly on the reactor vessel head, the system helps to minimize the time, manpower and radiation exposure associated with head removal and replacement

during refueling. Integral in the design is permanent shielding for reducing work area dose rates for the CRDM drive shafts.

The conventional top mounted instrumentation ports/conoseal thermocouple arrangement is replaced with a combination thermocouple/incore detector system on the WAPWR. This improvement eliminates the need to disassemble and reassemble the instrument port conoseals at each refueling, which has historically been a relatively high radiation exposure task.

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to ensure a smooth transition, and pipe branch locations are selected to ensure no interference from one branch to the next. All weld-to-pipe interfaces require a smooth, high quality finish.

Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with quick disconnect clasps to facilitate removal of the insulation for inspection of the welds.

B. Reactor Coolant Pumps

The reactor coolant pump design includes assembled cartridge seals which reduce the time required for replacement. The cartridge seal packages are expected to be capable of operating for 2 years without inspection or maintenance. Also, the reactor coolant pump casing will be manufactured by a cast method without major welds that require inservice inspection (ISI); thereby reducing personnel exposure associated with performing ISI.

C. Steam Generators

The Westinghouse APWR steam generator incorporates many design features to facilitate maintenance and inspection in reduced radiation

fields. [

(a,c)

] The steam generator manways (entrance to channel head) are sized for easier entrance and exit of workers with protective clothing, and to facilitate the installation and removal of tooling.

The specification of low cobalt tubing material for the WAPWR steam generator is an extremely important feature of the design - not only in terms of reduced exposure relative to the steam generator - but to the total plant radiation source term. The previous limit on the amount of cobalt in steam generator tubing was 0.1 weight percent and has been considered to be a major source of Co-60 activity in the plant. This limit has been substantially reduced to [] weight percent for the WAPWR design.

(a,c)

Other significant improvements to the steam generator design include a [] which should eliminate or substantially reduce the need for sludge lancing, and reduces tube and tube support degradation. Additional changes to increase steam generator reliability will also reduce occupational radiation exposure. [

(a,c)

(a,c)

D. Reactor Coolant Pipe Connections

To minimize crud buildup in branch lines, piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe wherever possible.

The resistance temperature detector (RTD) bypass manifold and its associated piping and valves has been replaced with an N-16 Transit Time Flow/Power Meter. This eliminates maintenance on the RTD system components and results in lower radiation fields about the RCS components.

12.3.1.1.1.2 Auxiliary Equipment

A. Filters

Filters that accumulate radioactivity are supplied with the means to perform cartridge replacement with semi-remote tools. Adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

Radioactive filters are located in a centralized location in the reactor external buildings, with a remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment from the site for burial. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. Each filter is contained in a shielded compartment and provided with vent and drain valving, and compartment drainage capabilities. A design criteria for the filter handling system is that it be simple with a minimum of components susceptible to malfunction.

B. Demineralizers

Demineralizers for highly radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks prior to solidification and so that fresh resin can be loaded into the demineralizer remotely. Underdrains are designed for full system pressure drop. The demineralizers and piping are designed with provisions for flushing.

C. Evaporators

Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations. Space is provided to allow removal of heating tube bundles. The highly radioactive

evaporator components are separated from those that are less radioactive. Instruments and controls are located in accessible low background radiation areas.

D. Pumps

Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease of removal.

E. Tanks

In general, horizontal and flat-bottom tanks are sloped downward to the tank drain. Overflow lines are directed to the waste collection system to control any contamination within plant structures. For tanks outside structures, which can potentially contain radioactive fluids, dikes are used to contain overflows.

F. Heat Exchangers

Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials to minimize leakage. Impact baffles are provided, and tube side and shell side velocities are limited to minimize erosive effects. Wherever possible, the radioactive fluid passes through the tube side of the heat exchanger.

G. Instruments

Instrument devices are located in low radiation zones and away from radiation sources whenever practical. Primary instrument devices, which for functional reasons are located in high radiation zones are designed for easy removal to a lower radiation zone for calibration.

Transmitters and readout devices are located in low radiation zones, such as corridors and the control room, for servicing.

Integral radiation check sources for response verification for airborne radiation monitors and safety-related area radiation monitors are provided.

H. Valves

To minimize personnel exposures from valve operations, motor-operated, air-operated, or other remotely actuated valves are used where justified by the activity levels and frequency of use. Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation. Valves for clean, non-radioactive systems are separated from radioactive sources and are located in readily accessible areas.

Recognizing that valve maintenance can be a major source of personnel radiation exposure, the WAPWR design reflects a reduced number of valves and piping in the auxiliary systems. This is evidenced in the integrated safeguards system (ISS) design, whereby the independent trains require less interconnecting piping and valves.

I. Piping

The piping in pipe chases is designed for the lifetime of the unit. Wherever radioactive piping is routed through areas where routine maintenance is required, pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection or maintenance requirements. Butt welds are used to the

fullest extent possible in radwaste piping utilized for transport of spent resins or slurries. Piping containing radioactive material is routed to minimize radiation exposure to the unit personnel.

J. Floor Drains

Floor drains and properly sloped floors are provided for each room or cubicle containing serviceable components containing radioactive liquids. If a radioactive drain line must pass through a plant area requiring personnel access, shielding is provided as necessary to ensure radiation levels consistent with the required access.

K. Sample Station

The sample station for routine sampling of process fluids is located as shown in Figure 12.3-3. Proper shielding and ventilation are provided at the local sample stations to maintain low radiation fields in proximate areas and minimize personnel exposure during sampling. The sample room is located directly over the chemical drain tank to minimize runs of chemical drain lines. Also, the sample heat exchangers inside the sample room are locally shielded to reduce background radiation levels inside the room.

L. Clean Services

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

12.3.1.1.2 Facility and Layout ALARA Considerations

This subsection describes the design features utilized for standard type plant process and layout situations. These features are employed in conjunction with the general equipment described in paragraph 12.3.1.1.1 and include the features discussed in the following paragraphs.

A. Valve Galleries

Valve galleries are provided with shielded entrances for personnel protection. Floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete surfaces are covered with a smooth surface coating which will allow easy decontamination. The valve gallery shield walls are designed to minimize personnel exposure during maintenance of components within or adjacent to the gallery and to protect personnel who remotely operate the valves.

B. Piping Chases

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

When possible and practical, radioactive and nonradioactive piping are separated to minimize personnel exposure. Should maintenance be required, provision is made to isolate and drain radioactive piping and associated equipment. Potentially radioactive piping is located in appropriately zoned and restricted areas.

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. In radioactive systems, the use of nonremovable backing rings in the piping points is prohibited. Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Piping which carries resin slurries or evaporator bottoms is run vertically as much as possible. Horizontal runs carrying spent resin are sloped toward the spent resin tanks. Large radius bends are utilized instead of elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize crud traps.

C. Penetrations

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

D. Contamination Control

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

Decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of decontaminable paints and suitable smooth-surface coatings to the concrete floors and

walls. Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

The role of the ventilation systems in minimizing the spread of airborne contamination is discussed in Subsection 12.3.3 of this module.

E. Equipment Layout

In those systems where process equipment is a major radiation source pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. In general, control panels are located in low radiation zones.

Major components such as tanks, demineralizers, and filters in radioactive systems are isolated in individual shielded compartments insofar as practical. Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as ion exchangers or heat exchangers and tanks in the primary coolant system) completely enclosed shielded compartments with hatch openings or removable concrete block walls are used. Provision is made on some major plant components for removal of these components to lower radiation zones for maintenance.

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

systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is provided. When this is not practicable, written procedures are used which reduce the total time personnel are exposed to the radiation field. Also, access to high radiation areas is under the direct supervision of the unit health physics personnel.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard area is regulated and controlled by radiation zoning and access control under the direction of the plant health physics staff. Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. During plant operation, personnel gain access to radiation controlled areas through an access control point.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal, shutdown, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. The radiation zone categories employed and their descriptions are given in Table 12.3-1. The zoning for each plant area under normal operational and refueling outage conditions are shown in Figures 12.3-1 through 12.3-6. Elevation views of the reactor building which further illustrate the component locations and layout are shown in Figures 12.3-7 and 12.3-8. The radiation zones shown in the figures represent conservative estimates of the maximum general area dose rates at the various plant areas. Dose rates at certain locations within the area may exceed the zone value due

to component crud traps or radiation streaming, but design features are incorporated to minimize such effects and the higher dose rates are expected to be highly localized and/or intermittent. Actual in-plant zones and control of personnel access will be based upon surveys conducted by the plant health physics staff.

Areas which may require occupancy to permit an operator to aid in the long term recovery from an accident are considered in the design. Such areas include the control room, safety-related motor control centers and switchgear, post accident sampling system room, radiochemistry laboratory, and remote shutdown panels. Such radiation protection design features are described in Section 12.3.2 of this module. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates in the area, and appropriate time limits for presence in the area are imposed.

Ingress or egress of plant operating personnel to controlled access areas is controlled by the plant health physics staff to ensure that radiation levels and exposures are within the limits prescribed in 10 CFR 20. Any area having a radiation level that could cause a whole body exposure in any 1 hour in excess of 5 mrem, or in any 5 consecutive days in excess of 100 mrem, will be posted "Caution, Radiation Area." Radiation areas are provided with access alert barriers, e.g., chain, rope, door, etc. Any area having a radiation level that could cause whole body exposure in any 1 hour in excess of 100 mrem will be posted "Caution, High Radiation Area." High radiation areas (> 100 mrem/hr) are provided with locked or alarmed barriers. During periods when access to a high radiation area is required, positive control is exercised over each individual entry. To the extent practicable, the measured radiation level and the location of the source is posted at the entry to any radiation area or high radiation area.

The posting of radiation signs, control of personnel access, and use of alarms and locks are in compliance with requirements of 10 CFR 20.203.

12.3.2 SHIELDING

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this subsection.

12.3.2.1 Design Objectives

The design objective of the plant radiation shielding, in conjunction with a program of controlled personnel access to and occupancy of radiation areas, is to reduce personnel and population exposures to levels that are within the dose standards of 10 CFR 20 and are as low as reasonably achievable (ALARA). Shielding and equipment layout and design are considered in ensuring that exposures are maintained ALARA during anticipated personnel activities in areas of the plant containing radioactive materials, utilizing the design recommendations given in Regulatory Guide 8.8, paragraph C.2, where practicable.

Three plant conditions are considered in the nuclear radiation shielding design:

- A. Normal, full-power operation.
- B. Shutdown operation.
- C. Emergency operations (for required access to safety-related equipment).

The shielding design objectives for the plant during normal operation (including anticipated operational occurrences), for shutdown operations, and for emergency operations are listed below:

- A. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and site boundary occupants are ALARA and within the limits of 10 CFR 20.

- B. To ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.
- C. To reduce potential equipment neutron activation and to mitigate the possibility of radiation damage to materials.
- D. To provide sufficient shielding for the control room so that for design basis accidents (DBAs) the direct dose plus the inhalation dose will not exceed the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

12.3.2.2 General Shielding Design

Shielding is provided to reduce the radiation levels resulting from direct and scattered radiation to less than the upper limit of the radiation zone assigned to each area. General locations of the plant areas and equipment discussed in this subsection are shown in the radiation zone diagrams of Section 12.3.1 of this module.

The material used for most of the plant shielding is ordinary concrete with a bulk density of approximately 145 lb/ft³. Whenever poured-in-place concrete has been replaced by concrete blocks, the design ensures protection on an equivalent shielding basis as determined by the density of the concrete block selected. Water is used as the primary shield material for areas above the spent fuel storage area and in the refueling cavity during refueling operations.

12.3.2.2.1 Reactor Containment Building Shielding Design

During reactor operation, the reactor containment building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete containment wall and the reactor vessel and steam generator compartment shield walls

reduce radiation levels outside the containment to less than 0.25 mrem/hr from sources inside containment. The containment shield completely surrounds the nuclear steam supply system with a wall thickness which ranges from 1.5 meters to a minimum of 0.6 meters at the top of the dome.

For design basis accidents (DBAs), the containment shield and the control room shielding reduce the plant radiation intensities from fission products inside the containment to acceptable emergency levels, as defined by 10 CFR 50, Appendix A, General Design Criterion 19, for the control room. (See Subsection 12.3.2.2.7.)

Where personnel and equipment hatches or penetrations pass through the containment wall, additional shielding is provided to attenuate radiation to the required level defined by the outside radiation zone during normal operation and shutdown and to acceptable emergency levels as defined by 10 CFR 50, Appendix A, General Design Criterion 19, during DBAs.

12.3.2.2.2 Reactor Containment Building Interior Shielding Design

During reactor operation, many areas inside the containment are Zone V and normally inaccessible. However, shielding is provided to reduce dose rates to approximately 100 mrem/h or less in areas of the containment that potentially require access at power. These are the Zone IV or lower areas shown in Figures 12.3-1 to 12.3-6.

The main sources of radiation are the reactor vessel and the primary loop components, consisting of the steam generators, pressurizer, reactor coolant pumps, and associated piping. The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. Air cooling is provided to prevent overheating, dehydration, and degradation of the structural and shielding properties of the primary shield.

The primary shield is a large mass of reinforced concrete surrounding the reactor vessel and extending upward from the containment floor to the walls of the fuel transfer canal. The minimum concrete thickness of the primary shield is 2.5 meters. The primary shield meets the following objectives:

- A. In conjunction with the secondary shield, to reduce the radiation level from sources within the reactor vessel and reactor coolant system (RCS) to allow limited access to the containment during normal, full-power operation.
- B. After shutdown, to limit the radiation level from sources within the reactor vessel, to permit limited access to the reactor vessel, and to permit limited access to the RCS equipment.
- C. To limit neutron activation of component and structural materials.

The secondary shield is a reinforced concrete structure surrounding the RCS equipment, including piping, pumps, and steam generators. This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products circulating in the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation. The minimum thickness of secondary shield walls is 1.2 meters.

Components of the letdown portion of the chemical and volume control system (CVCS) in the containment are located in shielded compartments that are normally Zone V, restricted access areas. Shielding is provided for each piece of equipment in the letdown system consistent with its postulated activity (Section 12.2 of this module) and with the access and zoning requirements of adjacent areas. This equipment includes the regenerative heat exchanger, the letdown and excess letdown heat exchangers, and the letdown lines.

After shutdown, the containment is accessible for limited periods of time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/hr, depending on the location inside the containment. These dose rates result from residual fission products and neutron activation products (components and corrosion products) in the RCS.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling pool and the fuel transfer canal to ensure that radiation levels remain below zone levels specified for adjacent areas. Water shielding is provided over the spent fuel assemblies during fuel handling.

12.3.2.2.3 Reactor External Building Shielding

During normal operations, the major components in the reactor external building with potentially high radioactivity are those in the CVCS, steam generator blowdown, boron recycle, liquid radwaste, gaseous radwaste, and spent resin handling systems. Shielding is provided for each piece of equipment consistent with its postulated activity (Sections 11.1, 11.2, 11.3 of RESAR-SP/90 PDA Module 12, "Waste Management", and 12.2 of this module) and with the access and zoning requirements of adjacent areas.

Depending on the equipment in the area, the radiation zones vary from Zone I through V. Corridors are generally shielded to allow Zone II access, and operator areas for valve compartments are generally Zone III.

Removable sections of block shield walls and concrete plugs are utilized as necessary for equipment maintenance and spent filter cartridge replacement. Permanent or temporary shielding is used between equipment in compartments with more than one piece of equipment to permit access for maintenance. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of adjacent areas.

All emergency core cooling system (ECCS), residual heat removal (RHR) and containment spray system components, which are located outside the containment and used to recirculate radioactive coolant during normal or post-accident operation, are housed in dedicated, separated component areas (SCA). One of the unique features/benefits of the integrated safeguards system (ISS) of the WAPWR is the four separate independent subsystems and these features//benefits are maintained in the layout and shield design. Each SCA includes the piping penetration area, valve area and pump compartment area associated with one ISS module, as shown in Figures 12.3-1 to 12.3-3. Layout considerations, which have been addressed, include pump pull space, motor laydown area, work area, monorails, removable shield walls, distance to stairs and equipment hatches. The equipment and shielding layout is expected to result in a minimum of whole body exposure during normal maintenance and inspection and a low potential for significant surface contamination and airborne levels in areas where access to ISS equipment is required, particularly in a post-accident recirculation mode. This arrangement provides separation of the low and highly radioactive equipment, one personnel access point for each of the four independent and separate SCA's, and makes it possible to passively prevent a major loss of emergency water storage tank (EWST) water outside containment. Further, the separation of trains reduces the number of components which could potentially be damaged as a result of flooding.

12.3.2.2.4 Fuel Handling Building Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and storage area, and the shield walls surrounding the fuel transfer and storage areas, are sufficiently thick to limit radiation levels outside the shield walls in all accessible areas to Zone II. The building external walls are sufficient to shield external plant areas to Zone I.

All spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Water depths of greater than 3 meters above a fuel assembly during fuel handling are maintained in the reactor cavity, the fuel transfer canal, and the spent fuel

pool. This limits the dose at the water surface to less than 2.5 mrem/hr for an assembly in a vertical position at the maximum elevation. Normal water depth above the stored assemblies in the spent fuel pit is greater than 8 meters and the dose rate at the pool surface is significantly less than 2.5 mrem/hr. The minimum 1.5 meter thick concrete walls of the fuel transfer canal and spent fuel pool walls supplement the water shielding and limit the maximum radiation dose levels in working areas to less than 2.5 mrem/h.

The spent fuel pit cooling system (SFPCS) shielding is based on the activity discussed in Section 12.2 of this module, and the access and zoning requirements of adjacent areas. Equipment in the SFPCS to be shielded includes the SFPCS heat exchangers, pumps, and piping.

12.3.2.2.5 Radwaste Buildings Shielding Design

The radwaste building is not within the scope of the W APWR NPB and therefore the shielding design is the responsibility of the plant specific applicant. However, the radwaste building shielding design should be consistent with the radwaste source strengths presented in Section 12.2 of this module.

12.3.2.2.6 Turbine Building Shielding Design

Radiation shielding is not required for process equipment located in the turbine building.

12.3.2.2.7 Control Room Shielding Design

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the control room. Consideration is given to shielding provided by the containment structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated LOCA, so that radiation doses are limited to 5 rem whole body from contributing modes of exposure for the duration of the accident, in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

The design basis LOCA is described in Subsection 15.6.4 of RESAR SP/90 PDA Module 1, "Primary Side Safeguards System". The contribution from direct radiation from airborne fission products inside the containment to personnel doses inside the control room following a postulated LOCA is discussed in Subsection 15.6.4.4.6.3 of RESAR SP/90 PDA Module 1, "Primary Side Safeguards System". The shielding of the control room ensures compliance with 10 CFR 50, Appendix A, General Design Criterion 19. The control room location and shielding provisions are illustrated in Figure 12.3-5.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at the outside surfaces of the buildings are maintained below Zone 1 levels. Plant yard areas that are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on equipment activities under the plant operating conditions described in Chapter 11 of RESAR-SP/90 PDA Module 12, "Waste Management", and Section 12.2 of this module. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as possible the actual geometry and physical condition of the source or sources.

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, ion exchangers, evaporators, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

Industry-accepted computer codes - ANISN-W⁽¹⁾, DOT-IIIW⁽²⁾, SCAP⁽³⁾ and MORSE⁽⁴⁾ - are used for shielding analysis. ANISN is a multi-group one-dimensional discrete-ordinates transport code that solves the one-dimensional Boltzmann transport equation for neutrons and gamma rays in slab, sphere, or cylinder geometry. Using a finite-difference technique, ANISN allows general anisotropic scattering; i.e., an Lth order Legendre expansion of the scattering cross-sections. DOT-IIIW is a two-dimensional discrete-ordinates transport code which effects a solution of the Boltzmann transport equation. DOT-IIIW permits anisotropic scattering to be included and is suitable for both neutron and gamma ray deep penetration calculations in a wide variety of shielding problems. Monte Carlo techniques as described below may be used for more complicated geometries such as penetrations. SCAP is a single-scatter point-kernel general purpose code for estimating the penetration and scattering of gamma rays that originate in a volume-distributed source. ANISN and DOT are used primarily for primary shield design. SCAP and DOT are used primarily for configurations not conveniently modeled in one-dimensional geometries.

For final design, a three-dimensional model is used to simulate radiation streaming from the reactor vessel surface to the containment using the MORSE Monte Carlo program. The source terms used for the MORSE code are generated by the computer code DOT. The source terms are divided into 47 neutron energy groups and 20 gamma energy groups.

Neutron and gamma ray cross-sections are prepared using the AMPX-11⁽⁵⁾ processing system. The reference cross-section library is Vitamin-C.⁽⁶⁾

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements in each plant area are evaluated at the point of maximum radiation dose through any wall. Since the actual failed fuel fraction during normal plant operation is expected to be less than 0.25%, the actual anticipated radiation level in most plant areas is less than this maximum dose and consequently less than the radiation zone upper limit.

Where shielded entryways to compartments containing high-radiation sources are necessary, labyrinths are designed using methods summarized in DRNL-RSIC-21⁽⁷⁾, and the SCAP computer code. The labyrinths are constructed so that the scattered dose rate, plus the transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for each plant area.

12.3.3 Ventilation

12.3.3.1 Design Objectives

The plant ventilation systems, in addition to their primary function of preventing extreme thermal environmental conditions for operating personnel and equipment, will provide effective protection for operating personnel against possible airborne radioactive contamination in areas where this may occur.

The systems will operate to ensure that the maximum airborne radioactivity level for normal operation, including anticipated operational occurrences, are within the limits of 10CFR20, Appendix B, Table I, for areas within plant structures and on the plant site where construction workers and visitors are permitted. The maximum levels correspond to design-basis reactor coolant inventory. The average airborne radioactivity levels meet the requirements of 10CFR20 and 50 and in fact will be considerably smaller since average coolant inventories and actual equipment leakage will be small.

The systems will operate to ensure compliance with normal operation offsite release limits as discussed in Section 11.3 of RESAR SP/90 PDA Module 12, "Waste Management".

The control room ventilation system will also operate to provide a suitable environment for equipment and continuous personnel occupancy in the control room under post-accident conditions in accordance with 10CFR50, Appendix A, Criterion 19.

The expected airborne radioactivity levels for normal operations and anticipated operational occurrences, in areas within plant structures, including each building in the reactor facility, and on the plant site where personnel, construction workers, or site visitors are permitted, along with the assumptions and methods used to calculate these airborne radioactivity levels will be presented in the Applicant's Safety Analysis Report. A discussion of the resulting estimated doses will also be presented.

12.3.3.2 Design Guidelines

In order to accomplish the design objectives, certain general design guidelines are followed when possible and applicable.

1. Air movement patterns are provided from areas of lesser contamination potential to areas of progressively greater contamination potential prior to final exhaust.
2. Slightly negative pressures are maintained, where applicable, to prevent uncontrolled exfiltration of contamination. Slightly positive pressure is maintained in the control room to prevent infiltration of potential contaminants.
3. Valves and equipment are maintained as leaktight as possible in order to prevent leakage of radioactive water and subsequent air-borne contamination.
4. Individual air supplies are provided for each building in order to keep potentially contaminated air flows separate from noncontaminated air.
5. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered to prevent contamination of the control room.

The plant ventilation system is described in further detail in RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The radiation monitoring system consists of the following:

- A. Area radiation monitoring system (ARMS).
- B. Process and effluent radiation monitoring system (PERMS).
- C. Sampling system.
- D. Post-accident monitoring systems (PAMS) radiation monitors.

The PERMS, sampling systems, and PAMS (airborne) radiation monitors are described in Section 11.5 of RESAR-SP/90 PDA Module 12, "Waste Management".

12.3.4.1 Area Radiation Monitoring

The ARMS is provided to supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, 8.8, and 8.12.

The design of the fuel pool racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 70.24 and Regulatory Guide 8.12, are not needed.

12.3.4.1.1 Design Objectives

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

- A. To furnish records of radiation levels in specific areas of the plant.
- B. To warn of uncontrolled or inadvertent movement of radioactive material in the plant.

- C. To provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where substantial change in radiation levels might be of immediate importance to personnel frequenting the area.
- D. To annunciate and warn of possible equipment malfunctions and leaks in specific areas of the plant.
- E. To furnish information for making radiation surveys.

By meeting the above objectives, the area radiation monitoring system aids health physics personnel in keeping radiation exposures as low as reasonably achievable (ALARA).

The design objectives of the ARMS during postulated accidents are:

- A. To provide the capability to alarm and initiate a containment isolation phase A or containment ventilation isolation signal in the event of a loss-of-coolant accident (LOCA), fuel handling accident inside containment, or abnormally high radiation inside the containment (monitors A-2A&B and A-11A&B).
- B. To provide long-term post-accident monitoring of conditions at strategic locations. (See Subsection 11.5.5 of RESAR-SP/90 PDA Module 12, "Waste Management").

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 of a specific area.
- B. Potential for personnel to unknowingly receive high radiation doses.
- C. Potential for equipment malfunction.

- D. Access areas where, during normal plant operation (including refueling), radiation exposures could exceed the radiation limits due to system failure or personnel error.
- E. Access 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 a reactor coolant pressure boundary (RCPB) leak, or fuel handling accident.
- G. Normally or potentially radioactive areas.

12.3.4.1.3 General System Description

The area radiation monitors are located at selected locations throughout the plant to detect, indicate, and store information through their associated data processing module on the radiation levels and, if necessary, annunciate abnormal radiation conditions. The ARMS monitors are an integral part of the PERMS, which is described in detail in Section 11.5 of RESAR-SP/90 PDA Module 12, "Waste Management".

The ARMS consists of individual, locally mounted area monitors. Each monitor is composed of the requisite number of channels, with a channel consisting of a radiation detector and check source. The detectors for all area monitors are either gamma-sensitive Geiger-Mueller counter tubes or ionization chambers. If exposed to radiation in excess of full-scale indication, the area monitors indicate that the full-scale reading has been exceeded and remain at the full-scale value. If the radiation field causing the overload condition is removed, the system returns to its normal operating condition unless the detector has failed. An administrative procedure (positioning the check source) is initiated to ensure that radiation monitoring equipment has not been damaged. All channels associated with a monitor are served by a local dedicated data processing module. All channels are indicated and annunciated in the control room and indicated and alarmed near the detector

location (normally at the data processing module). Monitors A-2A&B and A-11A&B, which are safety-related, Class 1E, are also indicated and annunciated at the safety-related display console.

12.3.4.1.4 Data Processing Module and Display Console

A description of these components is given in Section 11.5 of RESAR-SP/90 PDA Module 12, "Waste Management".

12.3.4.1.5 Local Annunciation

All area monitors have local annunciation consisting of an audible alarm rated at 80 dB at 10 ft and a warning light at the local readout.

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 plant 120-V safety-related buses. Power to the channels that monitor only normal operations is supplied from the regulated 120-V instrumentation bus that is backed by the diesel generator.

12.3.4.1.7 Redundancy, Diversity, and Independence

Monitors designated safety-related are part of the safety-related portion of the PERMS and are designed for redundancy, diversity, and independence in accordance with Institute of Electrical and Electronic Engineers (IEEE) 344-1975, IEEE 336-1971, IEEE 279-1971, IEEE 308-1974, IEEE 323-1974, and IEEE 384-1974. All monitors which are Seismic Category I are also manufactured and rated to the above standards.

12.3.4.1.8 Area Monitor Description

Table 12.3-2 gives the conditions of service for the area monitors. A brief description of each area monitor's function is given below.

A. Control Room Area Monitor A-1

To continuously indicate the radiation levels in the control room. A high alarm signal warns control room personnel of a deteriorated radiological condition inside the control room.

B. Containment Low-Range Area Monitors A-2A and A-2B

To continuously indicate the radiation levels inside the containment building at the operating deck. During refueling operations a high radiation alarm indicates a fuel drop accident and isolates the containment ventilation system. During power operations, a high radiation alarm indicates a possible LOCA and isolates the containment ventilation system.

C. Radiochemistry Laboratory Area Monitor A-3

To continuously indicate the radiation levels in the radiochemistry laboratory. A high radiation alarm signal warns the occupants of the radiochemistry laboratory of a deteriorated radiological condition.

D. Fuel Handling Building Area Monitor A-5

To continuously indicate the radiation levels inside the fuel handling building. A high radiation alarm signal warns the occupants of the fuel handling building of a deteriorated radiological condition.

E. Sampling Room Area Monitor A-6

To continuously indicate the radiation levels in the sampling room. A high radiation alarm signal warns the occupants of the sampling room of a deteriorated radiological condition.

F. Seal Table Instrumentation Room Area Monitor A-7

To continuously indicate the radiation levels in the seal table room and establish radiological habitability prior to entry. A high radiation alarm signal warns occupants of the seal table room of a deteriorated radiological condition.

G. Containment Access Hatch Area Monitor A-9

To continuously indicate the radiation levels in the containment access hatch and establish radiological habitability prior to entry.

H. Containment High Range Area Monitors A-11A and A-11B

To indicate, along with A-2A and A-2B, the radiation levels inside the containment building at the operating deck following a design basis accident. A high alarm signal initiates containment isolation phase A to mitigate the consequences of a design basis accident (primarily a LOCA).

12.3.4.1.9 Range and Alarm Setpoints

The range, setpoints, and control function of the PERMS area monitors are given in Table 12.3-3. The setpoints are initial and are subject to modification as plant operating experience is developed.

Radiation zones are described in Table 12.3-1.

The control room monitor A-1 has a greater sensitivity than the other area monitors, since it is located in a Zone I radiation area; monitors A-2A&B and A-11A&B cover a wide range of radiation levels. During plant shutdown including refueling operations, the radiation level on and above the operating deck should be less than 5 mR/hr. The high end of the range is dictated by the design basis accident, a LOCA.

Each area monitor has two alarm setpoints, intermediate and high. (See Table 12.3-3.) If a monitor has a control function, i.e., A-2A&B and A-11A&B, the control function is triggered coincidentally with the high alarm setpoint. An intermediate alarm gives a visual indication in the control room and near the detector that the radiation level has reached the intermediate setpoint. A high alarm gives both a visual and audible indication near the detector (along with a visual indication and annunciation in the control room) that the high alarm setpoint has been reached.

For testing, each area monitor has a check source assembly which is operated from the control console and uses a sealed Sr-90 source. Inservice inspection, calibration, and maintenance of the ARMS monitors is discussed in Subsection 11.5.2.5 of RESAR-SP/90 PDA Module 12, "Waste Management".

12.3.5 References

1. Soltesz, R. G., et al, Final Progress Report, Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Vol. 4, "One-Dimensional Discrete Ordinates Transport Technique," WANL-PR(LL)034, August 1970.
2. Soltesz, R. G., et al, Final Progress Report, Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Vol. 5, "Two-Dimensional Discrete Ordinates Transport Technique," WANL-PR(LL)034, August 1970.
3. Soltesz, R. G., et al, Final Progress Report, Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Vol. 6, "Point Kernel Techniques," WANL-PR(LL)034, August 1970.
4. PISC Computer Code Collection CCC-203, MORSF-CG, General Purpose Monte Carlo Multi-Group Neutron and Gamma Ray Transport Code with Combinatorial Geometry.
5. ORNL RSIC PSR-63, "AMPX-II, Modular GDE System for Generating Coupled Multigroup Neutron Gamma Ray Cross-Section Libraries from Data in End of Format".
6. ORNL RSIC DLC-41, "Vitamin-C, 171 Neutron, 36 Gamma Ray Group Cross-Sections in AMPX Interface Formats for Fusion and LMFBR Neutronics".
7. Selph, W. E., "Neutron and Gamma Ray Albedos," ORNL-RSIC-21, Oak Ridge National Laboratory, February 1968.

TABLE 12.3-1

RADIATION ZONE DEFINITIONS

<u>Zone</u>	<u>Maximum Dose Rate</u> <u>(mrem/hr)</u>	<u>Description</u> ^(a)
I	0.25	Uncontrolled access, no limitations on occupancy
II	2.5	Controlled access, unlimited occupational access
III	5	Controlled access, frequent routine access
IV	100	Controlled access, infrequent non-routine access
V	> 100	Controlled access, access highly limited or not accessible

-
- a. Uncontrolled access: Where entry and exit by plant employees and visitors are not under the direct supervision of the plant health physics staff. These areas can be occupied by plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.

Controlled access: Where higher radiation levels and/or radioactive contamination, which have a greater probability of radiation health hazard to individuals, can be expected. Only individuals directly involved in the operation of the plant will, in general, be allowed to enter these areas. Entry and exit are authorized and supervised by the plant health physics staff.

TABLE 12.3-2

CONDITIONS OF SERVICE FOR AREA RADIATION MONITORS

<u>Area Monitor</u>	<u>Number of Detectors</u>	<u>Operating Temperature (°F)</u>	<u>Pressure</u>	<u>Relative Humidity (%)</u>	<u>Radiation Zone</u>	<u>Safety Classification</u>	<u>Location/ Elevation (Meters)</u>
A-1 control room	1	65-85	-1/8 in. to +1/2 in. WG	50 (max)	I	NNS	Control room at 100.0
A-2A,B containment low range	2	60-120	-1.5 in. WG to +3 psig	17.7 to 50	V	SC-3/1E	Containment at 100.0
A-3 radiochemistry laboratory	1	65-100	0 psig	40 to 60	III	NNS	Radiochemistry lab at 84.8
A-5 fuel handling building	1	40-104	-1/4 in. to 0 in. WG	20 to 95	II	NNS	Fuel handling build- ing at 100.0
A-6 sampling room	1	65-100	0 psig	40 to 60	IV	NNS	Sample room at 84.8
A-7 seal table instrumentation room	1	60-120	-1.5 in. WG to +3 psig	17.7 to 50	IV	NNS	Seal table room at 93.2
A-9 containment access hatch	1	60-120	-1.5 in. WG to +3 psig	17.7 to 50	IV	NNS	Inside containment personnel airlock at 100.0
A-11A,B containment high range ^(a)	2	60-120	-1.5 in. WG to +3 psig	17.7 to 50	"	SC-3/1E	Outside surface of shield wall at 100.0

^a These monitors are qualified for post-LOCA environment.

TABLE 12.3-3
RANGE AND SETPOINTS FOR AREA RADIATION MONITORS

Mon. Jr	Range (mR/hr)	Sensitivity (mR/hr)	Initial Setpoint		Control Function	Accuracy
			Intermediate	High		
A-1 control room	10^{-2} to 10^3	10^{-2}	0.10 mR/hr	0.25 mR/hr	No	± 20 percent of actual radiation field
A-2A,B containment low range	10^{-1} to 10^4 (both)	10^{-1}	5.0 mR/h(a) 0.40 R/h(b)	15.0 mR/h(a) 1.0 R/h(b)	Yes, isolates containment ventilation system	± 20 percent of actual radiation field
A-3 radiochemistry laboratory	10^{-1} to 10^4	10^{-1}	2.0 mR/hr	2.5 mR/hr	No	± 20 percent of actual radiation field
A-5 fuel handling building	10^{-1} to 10^4	10^{-1}	1.0 mR/hr	2.5 mR/hr	No	± 20 percent of actual radiation field
A-6 sampling room	10^{-1} to 10^4	10^{-1}	2.0 mR/hr	2.5 mR/hr	No	± 20 percent of actual radiation field
A-7 seal table instrumentation room	10^{-1} to 10^4	10^{-1}	50 mR/hr	100 mR/hr	No	± 20 percent of actual radiation field
A-9 containment access hatch	10^{-1} to 10^4	10^{-1}	10 mR/hr	15 mR/hr	No	± 20 percent of actual radiation field
A-11A,B containment high range	10^3 to 10^{11} (both)	10^3	2.0 R/hr	3.0 R/hr	Yes, initiates containment Phase A isolation	± 20 percent of actual radiation field

^aDuring refueling operations

^bDuring power operation

FIGURES 12.3-1 THROUGH 12.3-8
(FOLDOUTS)
PROPRIETARY

12.4 DOSE ASSESSMENT

Radiation exposures to operating personnel will be within 10CFR20 limits. Radiation protection design features described in Section 12.3 of this module, and the health physics program supplied in the plant specific applicant's safety analysis report will ensure that the occupational radiation exposures (ORE) to operating personnel during operation and anticipated operational concurrences will be as low as reasonably achievable (ALARA).

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive fluids. In addition, in some plant radiation areas there can be radiation exposure to personnel due to the presence of airborne radionuclides. Inplant radiation exposures during normal operation and anticipated operational occurrences are discussed in Subsection 12.4.1 and radiation outside the plant structures are addressed in Subsection 12.4.2.

12.4.1 Exposures Within the Plant Structures

12.4.1.1 Direct Radiation Dose Estimates

NUREG-0713 presents a compilation of occupational radiation exposures (ORE) received annually at domestic commercial nuclear power plants. From the most recent report⁽¹⁾, average collective doses were calculated for domestic plants with Westinghouse-supplied WSSSs. Figure 12.4-1 illustrates the trend in ORE for these plants over the period 1969 through 1982. It is noted that:

- o Specific spikes or peaks in the curve have been generally due to major repair operations. For example, the 1972 peak was due to steam generator tube sheet weld repair operations inside the channel heads of two plants. The manual repair operations resulted in over 500 man-rem alone.

- o In general, dips or valleys in the curve were due to new plant start-ups. For example, the 1973-1975 time period had twelve new plants come on-line. The low doses at these plants during their first year of operation lowered the average plant dose.
- o In 1982, the highest and lowest collective dose was about 1600 and 100, respectively. In addition, the average collective dose dropped to 650 man-rem.

Factors which may have contributed to this increase in plant collective doses include the following⁽²⁾:

- o Increasing plant radiation fields
- o Required or mandated modifications/back-fits
- o Premature failures of major components
- o Use of inexperienced workers
- o Management attitude

In the analysis of ORE data cumulative man-rem per cumulative MW_e -Yr of electricity generated accounts for plant size, and provides a relative measure of costs (man-rem) versus benefits (power production). Table 12.4-1 provides a cumulative summary (up through 1982) of collective doses for domestic plants with Westinghouse-supplied NSSSs. Also included on Table 12.4-1 is a measure of the effective operating time (MW_e -Yr/ MW_e).

As it can be seen from Table 12.4-1, cumulative man-rem per MW_e -Yr ranges from 0.30 to 2.68. The best performing plants operate in the range of 0.3 to 0.4 man-rem per MW_e -Yr. Major factors which have contributed to these excellent performance levels include low plant radiation fields, good layout and access provisions, and excellent operational practices and procedures. If

it is assumed that the WAPWR achieves the same performance levels as these "best plants," approximately 350 man-rem per year would be achieved $[(0.3 \text{ man-rem/MW}_e\text{-Yr}) * (1300 \text{ MW}_e) * (0.9 \text{ capacity factor})]$.

However, various design improvements have been incorporated into the WAPWR plant, and operational improvements can be expected to be incorporated into the plant. In addition to these improvements, additional ALARA features will be considered for the final WAPWR design. With these additional plant features, approximately 215 man-rem per year is believed to be a realistic design goal for the annual WAPWR cumulative plant exposure.

Radiation exposure estimates have been made for each of the following work categories:

- o Reactor Operations and Surveillance
- o Routine Maintenance
- o In-Service Inspection
- o Special Maintenance
- o Waste Processing
- o Refueling

Table 12.4-2 is a breakdown of the expected annual plant collective doses for each of these work categories. The values are based on a detailed breakdown of the doses incurred within each category based on feedback obtained from operating plants. Detailed dose predictive models have been developed which identify the various steps which comprise the operation, anticipated radiation levels in the work area, the required number of workers, and the time to perform each step. A discussion and further breakdown of the collective doses in the above work categories follows.

Reactor Operations and Surveillance

To support plant operations, the performance of various systems and components must be monitored. Examples of these activities include the following:

- o Routine inspections of plant components and systems
- o Unidentified leak checks
- o Operation of manual valves
- o Instrument readings
- o Routine health physics patrols and surveys
- o Decontamination of equipment or plant work areas
- o Calibration of electrical or mechanical equipment
- o Chemistry sampling and analysis

A special concern in this category are those operations performed in containment during power operation. At-power containment radiation fields are significantly higher than during plant shutdown. The frequency and duration of at-power containment entries vary greatly from plant to plant and are known to be influenced by the following:

- o Plant technical specifications
- o Containment design
- o Frequency and degree of unscheduled repairs
- o Specific utility practices

Table 12.4-3 provides a list of operations which may require containment access during power operation. Based on limited feedback from U.S. utilities and taking credit for design changes and reliability improvements, no more than 100 worker-hours per year should be required to maintain the plant during at-power operations.

Table 12.4-4 provides a breakdown of the collective doses for reactor operations and surveillance. From Table 12.4-4, the largest fraction of the total dose occurs during routine patrols and inspections of plant systems and components.

Routine Maintenance

In order to keep the plant operational, routine maintenance must be performed on all mechanical and electrical components. Table 12.4-5 provides a

breakdown of the collective doses for major routine maintenance items. From Table 12.4-5, the following are noted:

- o Maintenance on the reactor coolant and other plant pumps (RHR, RCDT, SFP cooling, CVCS charging, etc.) result in the largest fraction of the annual doses. This is due to the fact that the maintenance requires a significant amount of "hands-on" work on the radioactive pump components.
- o Valve maintenance contributes a significant fraction to the annual doses. This can be explained by: 1) the large number of valves in a plant; 2) the buildup of activity on the inside surfaces of valves; and 3) the frequent operation of particular valves.
- o Steam generator sludge lance and secondary side inspection results in relatively low collective doses.

To maintain the RCPs, six different maintenance/inspection operations are performed. Table 12.4-6 provides a description and schedule of these operations. An eighteen month fuel cycle and a cartridge style seal have been assumed. Table 12.4-7 presents a dose estimate for each of the maintenance inspection operations. Table 12.4-8 summarizes the doses incurred for all pump maintenance and inspection operations. From Table 12.4-8, approximately 2.3 man-rem per year per pump is required. Note that in-service inspection operations have been considered an integral part of the maintenance/inspection operations (sequences B through F).

In-Service Inspection

Section XI of the ASME Boiler and Pressure Vessel Code requires both a pre-service and periodic in-service inspection (ISI) be performed on all plant safety class components⁽³⁾. The Code also addresses component accessibility and inspectability and the extent of and methods employed for such examinations.

The WAPWR design has features which permit complete compliance with the Code requirements. Examples of these design features include the following:

- o All reactor internals are completely removable from the reactor vessel, allowing access to the entire inside surface of the vessel for inspection. The tools and storage space required to permit removal of the internals are also provided.
- o The reactor vessel closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inspection.
- o All reactor vessel studs, nuts, and washers are removed to dry storage during refueling, allowing inspection in parallel with refueling operations.
- o Removable plugs are provided in the primary shield just above the reactor vessel primary coolant nozzles for inspection of the welds joining the nozzles to the safe-ends and the welds joining the safe-ends to the primary coolant piping. Readily-removable insulation is provided over these weld areas.
- o Manways are provided in the steam generator channel head and in the moisture separator section for access for internal inspection.
- o A manway is provided in the pressurizer top head for access for internal inspection.
- o The insulation covering all component and piping welds and adjacent base metal is designed for ease of removal and replacement in areas where external inspection is planned.
- o Openings are provided in the operating deck concrete shielding above the main coolant pumps to permit removal of the pump motor for internal inspection access to the pumps.

- o The primary loop compartments are designed to allow personnel entry during refueling operations, with shielding provided between major components, to permit direct inspection access to the external portion of piping and components.

In addition to these features, sufficient space is provided around the various components to permit access by the examiners and their equipment. Allowances for component disassembly and insulation removal/installation are provided. Where feasible, permanent platforms or scaffolding provisions, service lines, ventilation systems, and handling features are provided to improve access.

Table 12.4-9 provides a detail breakdown of the collective doses for in-service inspection. Although some inspections are performed over varying time intervals, as shown in Table 12.4-10, the dose estimates have been annualized over a ten-year inspection period.

Special Maintenance

To support plant operation, repairs will be required on various systems and components. Since much of this work is performed on a non-routine basis, it has been included as part of the special maintenance category. Examples of such activities include the following:

- o Installation of new plant systems and components (R.V. level indication system, R.V. head vent, snubbers, etc.)
- o Modification of existing equipment
- o Repairs of components which have failed prematurely

Due to the nature of this work, frequency of repairs are very difficult to estimate. However, Table 12.4-11 provides a best estimate of the annual dose breakdown for special maintenance. From Table 12.4-11, the major contributors to the total doses are plant modifications, steam generator inspection/

repairs, and pump and valve repairs. In general, these repairs are all labor intensive and/or performed in a high radiation environment.

At most operating plants, special maintenance on the steam generators have resulted in significant personnel doses. Even though steam generator doses at new plants are expected to benefit from both design changes and improved secondary side chemistry, periodic primary to secondary tube leaks are assumed to occur. As a result, the following steam generator primary side special maintenance is assumed to be performed:

- o Mechanical tube plugging of three tubes per steam generator every outage.
- o A three percent eddy current tube inspection of each steam generator every outage (first outage excluded).

Waste Processing

During various plant operations, liquid, gaseous, and solid waste products are generated. Examples of such waste include the following:

- o Filter cartridges
- o Demineralizer resins
- o Tank sludges
- o Evaporator bottoms
- o Dry active wastes

Since these wastes are generally radioactive, appropriate disposal practices and procedures must be followed. A breakdown of the annual waste processing collective doses is presented in Table 12.4-12.

Refueling

For the WAPWR, an eighteen month interval is assumed. The refueling process is labor intensive and detailed planning and coordination is essential in

order to maintain personnel doses ALARA. In addition, the incorporation of advance technology into the refueling process will lead to reduced personnel doses.

Table 12.4-13 provides a dose estimate for refueling. From Table 12.4-13, the majority of the annual doses occur during reactor disassembly and reassembly. These estimates are based on the incorporation of improved refueling equipment and procedures as noted in Section 12.3.1. Since the WAPWR is designed for eighteen month fuel cycles the annual average refueling dose is approximately fourteen man-rem.

12.4.1.2 Airborne Radioactivity Dose Estimates

Due to leakages of radioactive fluids into the auxiliary, containment, radwaste, fueling, and turbine buildings, plant personnel are exposed to radionuclides released into the atmosphere of these buildings by the leaked fluids. These atmospheric contaminants contribute to the total body, thyroid, and lung doses.

The peak airborne concentrations for most areas in the plant are within the limits specified in 10 CFR 20. By use of appropriate respiratory equipment and/or limitation of occupancy time, personnel are allowed to enter areas where the airborne activity levels exceed 10 CFR 20 limits.

Refer to the plant specific applicant's safety analysis report for a discussion of the annual airborne radiation exposures.

12.4.2 Radiation Exposure Outside the Plant Structures

Refer to the plant specific applicant's safety analysis report for a discussion of radiation exposures outside the plant structures.

12.4.3 References

1. B.G. Brooks. "Occupational Radiation Exposure at Commerical Nuclear Power Reactors 1982." Washington, DC: U.S. Nuclear Regulatory Commission, NUREG-0713, Annual Report, December 1983.
2. "Actions Being Taken to Help Reduce Occupational Radiation Exposure at Commercial Nuclear Power Plants." GAO/EMD-82-91, August 24, 1982.
3. "Rules for In-Service Inspection of Nuclear Power Components." New York, NY: American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section XI, 1980.

TABLE 12.4-1

COLLECTIVE DOSE SUMMARY FOR DOMESTIC WESTINGHOUSE PLANTS

<u>Stations</u>	<u>MW_e</u>	Station Cumulative Summary ^(a,b)	
		<u>MW_e-Yr/MW_e</u>	<u>Man-Rem/MW_e-Yr</u>
Four-Loop Plants			
Maximum	1180	12.17	1.75
Minimum	175	.44	.32
Average	943	4.66	.84
Three-Loop Plants			
Maximum	890	9.13	2.68
Minimum	436	1.83	.73
Average	731	5.31	1.60
Two-Loop Plants			
Maximum	512	6.82	1.97
Minimum	470	6.84	0.30
Average	496	7.89	0.81
Overall Average	777	5.6	1.1

^aSource: NUREG-0713, Volume 4

^bStartup through 1982

TABLE 12.4-2

COLLECTIVE DOSE BREAKDOWN

<u>Category</u>	<u>Percent (%)</u>	<u>Annual Dose (man-rem)</u>
Reactor Operations and Surveillance	19	40
Routine Maintenance	26	56
In-Service Inspection	13	28
Special Maintenance	31	66
Waste Processing	5	11
Refueling	6	14
	<hr/>	<hr/>
	100	215

TABLE 12.4-3

OPERATIONS WHICH MAY REQUIRE CONTAINMENT AT-POWER ACCESS

SURVEILLANCE

- Routine Patrols
- Health Physics Surveys
- Patrols to Identify System Leaks

MECHANICAL REPAIRS

- Isolate System/Component Leaks
- Valve Adjustments/Repairs
- Leak Test Personnel Hatch
- Fan Cooler Repairs
- RCP Oil Additions
- Repair In-Core Detector Drive System
- Repair/Replace Radiation Monitoring Detectors

ELECTRICAL REPAIRS

- Repair/Replace Transmitters
- In-Core Thermocouple System Repairs at Junction Box
- Valve Operator Repairs

TABLE 12.4-4

DOSE ESTIMATE FOR REACTOR OPERATIONS AND SURVEILLANCE

<u>Work Description</u>	<u>Annual Dose (man-rem)</u>
OPERATION SUPERVISION	
- Routine Patrols and Inspections	15
- Valve Line-Ups (Manual)	2
- System Flushing and Testing	1
HEALTH PHYSICS	
- Job Coverage	5
- Routine Surveys	5
DECONTAMINATION	
- Equipment and Work Areas	7
CALIBRATION	
- Transmitters, Survey Instruments, Radiation Monitors, etc.	3
CHEMISTRY	
- Sampling	2
	—
TOTAL COLLECTIVE DOSE:	40

TABLE 12.4-5

DOSE ESTIMATE FOR ROUTINE MAINTENANCE

<u>Work Description</u>	<u>Annual Dose (man-rem)</u>
Valve Adjustment/Repacking	12.0
Pump Overhaul	12.0
RCP Seal Maintenance/Inspection (4)	9.2
SG Sludge Lance (4)	2.1
Demineralizer Resin Change-out	6.0
Filter Replacement	5.0
Calibrate/Repair Electrical Components	4.0
Miscellaneous Work	2.5
SG Secondary Side Inspection (4)	3.7
	<hr/>
TOTAL COLLECTIVE DOSE: 56.5	

TABLE 12.4-6

REACTOR COOLANT PUMP MAINTENANCE/INSPECTION SCHEDULE

Outage	Months Since Start-up	Years Since Start-up	Required Sequence ^a			
			Pump 1	Pump 2	Pump 3	Pump 4
1	18	1.5	A	B	A	B
2	36	3.0	C	A	B	A
3	54	4.5	E	C	E	B
4	72	6.0	B	E	C	E
5	90	7.5	A	B	A	C
6	108	9.0	F	A	D	A
7	126	10.5	A	D	A	D
8	144	12.0	B	A	B	A
9	162	13.5	E	C	E	B
10	180	15.0	B	E	C	E
11	198	16.5	A	B	A	C
12	216	18.0	D	A	D	A
13	234	19.5	A	F	A	D
14	252	21.0	C	A	B	A
15	270	22.5	E	B	E	B
16	288	24.0	B	E	C	E
17	306	25.5	A	B	A	C
18	324	27.0	D	A	F	A
19	342	28.5	A	D	A	D
20	360	30.0	C	A	B	A
21	378	31.5	E	C	E	B
22	396	33.0	B	E	B	E
23	414	34.5	A	B	A	C
24	432	36.0	D	A	D	A
25	450	37.5	A	D	A	B
26	468	39.0	C	A	B	A

^aThe following is a description of the work operations:

- A - Pump and motor inspections;
- B - Motor inspection with minor ISI;
- C - Pump and motor inspections with minor ISI;
- D - Motor inspection with major ISI (flywheel);
- E - Pump and motor inspection with major ISI (rotor/stator);
- F - Pump and motor inspection with minor ISI.

TABLE 12.4-7

EXPOSURE ESTIMATE FOR RCP MAINTENANCE/INSPECTION

<u>Work Description</u>	<u>Work Time*</u> <u>(Man-Hrs)</u>	<u>Dose Rate**</u> <u>(Rem/Hr)</u>	<u>Dose</u> <u>(Man-Rem)</u>
A - Pump and Motor Inspections	156	.022	3.4
B - Motor Inspection With Minor ISI	129	.016	2.1
C - Pump and Motor Inspections With Minor ISI	125	.030	3.7
D - Motor Inspection With Major ISI (Flywheel)	174	.016	2.8
E - Pump and Motor Inspection With Major ISI (Rotor/Stator)	387	.0155	6.0
F - Pump and Motor Inspection With Minor ISI	181	.020	3.7

*Represents work time in significant radiation fields.

**Effective dose rate over the range of work area dose rates.

TABLE 12.4-8

REACTOR COOLANT PUMP DOSE SUMMARY

<u>Sequence</u> ^a	<u>Required Inspections</u> ^a	<u>Dose/Inspection</u> ^b <u>(man-rem)</u>	<u>Total Dose</u> ^c <u>(man-rem)</u>
A	36	3.4	122.4
B	23	2.1	48.3
C	14	3.7	51.8
D	12	2.8	33.6
E	16	6.0	96.0
F	3	3.7	11.1
			<hr/>
TOTAL COLLECTIVE DOSE ^c :			363.2
ANNUAL COLLECTIVE DOSE:			2.3/RCP

^a Refer to Table 12.4-6 for inspection description and frequency.

^b Refer to Tables 12.4-7.

^c Value is for four pumps.

TABLE 12.4-9

DOSE ESTIMATE FOR IN-SERVICE INSPECTION

<u>Component</u>	<u>Annual Dose (man-rem)</u>
Valves Bodies and Boltings	10.0
SG Primary Side Inspections	4.1
Reactor Vessel and Head	3.8
Other Piping	3.5
Reactor Coolant Loop Piping and Supports	1.8
SG Shell	1.7
Heat Exchanger Shells	1.3
Pressurizer Shell	1.2
Pump Housings and Supports	0.4
Tank Shells and Supports	0.3
Filter Housings and Supports	0.1

TOTAL COLLECTIVE DOSE: 28

TABLE 12.4-10

EXPOSURE ESTIMATES FOR VARIOUS INSPECTION ACTIVITIES

<u>Activity</u>	<u>Work Time</u> <u>(Man-Hrs)</u>	<u>Average</u> <u>Dose Rate</u> <u>(Rem/Hr)</u>	<u>Frequency</u>	<u>Dose</u> <u>(Man-Rem)</u>
Steam Generator Secondary Side Inspection	13	.04	Each refueling outage	0.5/SG
Steam Generator Eddy Current Inspection	78	.07	Each refueling outage	5.5/SG
Reactor Vessel In-Service Inspection				
10-Year Inspections	1127	.016	Once Per 10 Years	18.5
40-Month Inspection	619	.016	Twice Per 10 Years	10.0

TABLE 12.4-11

DOSE ESTIMATE FOR SPECIAL MAINTENANCE

<u>Work Description</u>	<u>Annual Dose (man-rem)</u>
Plant Upgrades/Modifications	15.0
Valve Repairs	11.0
SG Primary Side Inspection (3)	10.7
Pump Repairs	9.5
Electrical Repairs	4.0
RCP Repairs and Inspection (4)	4.0
SG Tube Plugging (4)	5.6
Repairs to Tanks, Heat Exchangers, Piping, etc.	2.5
SG Secondary Side Repairs (4)	1.4
Pressurizer Repairs	1.2
CRDM Repairs	0.8
<hr/>	
TOTAL COLLECTIVE DOSE:	66

TABLE 12.4-12

DOSE ESTIMATE FOR WASTE PROCESSING

<u>Work Description</u>	<u>Annual Dose (man-rem)</u>
Radioactive Waste Handling	5
System Adjustments/Repairs	4
System Operation (Sampling, Valve Adjustments, Monitoring, etc.)	1
Laundry Operations	1
	—
TOTAL COLLECTIVE DOSE:	11

TABLE 12.4-13

DOSE ESTIMATE FOR REFUELING

<u>Work Description</u>	<u>Worktime (man-hrs)</u>	<u>Dose Rate (rem/hr)</u>	<u>Dose (man-rem)</u>
Refueling Operations (16 MAN-REM)			
o Preparation	158	.005	0.8
o Reactor Disassembly	208	.019	3.9
o Fuel Shuffle	362	.006	2.2
o Reactor Reassembly	272	.025	6.8
o Clean-Up	258	.009	2.3
Miscellaneous SFP Building Work	1000	.002	2.0
Equipment Checks and Repairs	1000	.002	2.0
TOTAL COLLECTIVE DOSE:			20
ANNUAL DOSE:			14

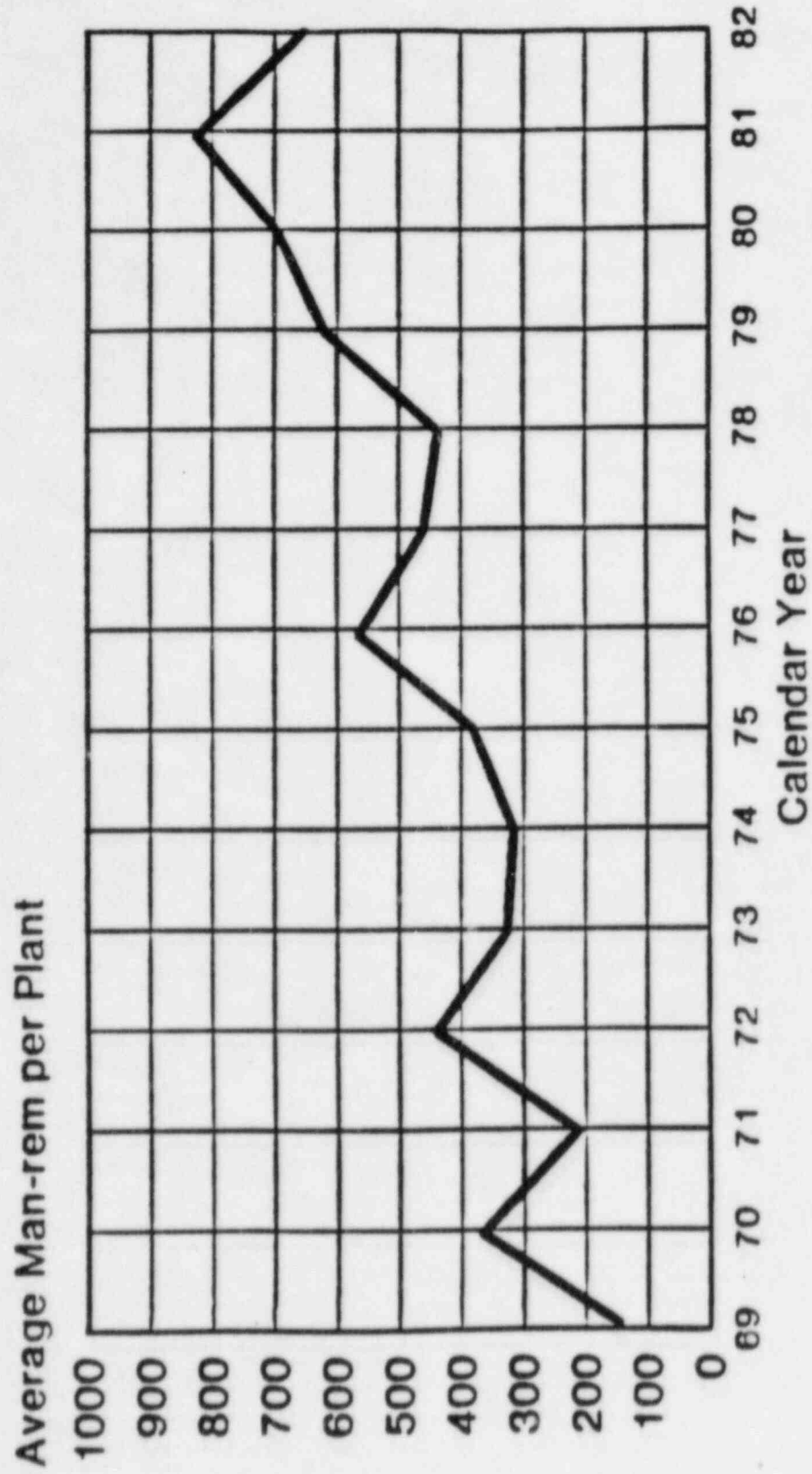


Figure 12.4-1. ORE Trend for Domestic Westinghouse Plants

12.5 HEALTH PHYSICS PROGRAM

Refer to the plant specific applicant's safety analysis report for a discussion of the applicant's health physics program.