11.1 SOURCE TERMS

Liquid and Gaseous Waste Activity

The normal sources of radioactive wastes are fission and activation products generated within the Primary Coolant System during Plant operation. The radioactive waste systems are designed to safely process radioactive wastes from the Plant with a primary coolant activity based on 1% failed fuel rods and continuous purification during Plant operation. The calculated primary coolant activity by isotope for the 1% failed fuel rod condition is presented in Table 11-1. This table also presents activity in the primary coolant as observed during normal power operation and indicates that the total activity level normally remains at less than 1% of design.

For the performance of alternative source term (AST) dose analyses, a Primary Coolant System source term that includes an expanded set of nuclides is required. Table 11-10 presents the recalculated and expanded source term for 1% failed fuel for AST analyses. Table 11-10 also includes corrosion products that were determined based on the guidance provided by ANSI/ANS-18.1-1999.

Sources and Accumulation of Wastes

For the purpose of design evaluation, the activity of the sources to the clean waste system was assumed to be that of primary coolant for the 1% defective fuel rod condition as given in Table 11-1. Table 11-2 lists the maximum expected quantity of clean waste of significant activity from the major waste sources and the conservative assumptions used to derive the waste quantities. The dirty waste system was historically designed to handle radioactive wastes generated by a steam generator blowdown rate of 20 gpm concurrent with 1 gpm primary to secondary leakage. It was assumed that dirty waste has an activity equal to 1% of the primary coolant activity. Further, as shown by Table 11-1, primary coolant activity normally is less than 1/100 that of the design basis activity. These considerations combine to provide for an extremely conservative system design relative to normal operating conditions.

11.2 LIQUID RADIOACTIVE WASTE SYSTEM

11.2.1 DESIGN BASES

11.2.1.1 Design Objective

The Liquid Radioactive Waste System is designed to collect, store, process, monitor, and dispose of all liquid radioactive wastes from the Palisades Plant. The principal design criterion is to ensure that the general public is protected from exposure to radioactive waste products in accordance with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix I.

11.2.1.2 Design Criteria

The Liquid Radioactive Waste System was originally designed and constructed as a CP Co Design Class 1 system and was located within the CP Co Design Class 1 auxiliary building, defined in Section 5.2. However, mechanical portions of the system not associated with "a loss of function which could cause an uncontrolled release of radioactivity" were designed as CP Co Design Class 2. Subsequent modifications to this system were undertaken during 1971-73 as described in Special Report 5 (see Reference 1) and Amendment 21 to Docket 50-255, dated February 26, 1971. Two buildings, the service building and the auxiliary building addition, were added during this time enclosing the new equipment. The auxiliary building addition is CP Co Design Class 1, the service building is CP Co Design Class 3, the liquid radwaste components are CP Co Design Class 2 and processing piping is CP Co Design Class 3, all per Section 5.2.

11.2.1.3 Codes

The original portions of the Liquid Radioactive Waste System were designed according to codes and criteria in effect at the time of initial design and construction of Palisades which was ASA B31.1-1961 for pressure piping and ASME, Section III, Class C, for vessels. These codes generally meet the suggested codes depicted for Quality Group D components and piping in NRC Regulatory Guide 1.26, Revision 1, issued September 1974, and Regulatory Guide 1.143, Revision 1, issued October 1979. A detailed listing of codes used on original components and piping is given in Table 11-3.

Liquid Radioactive Waste System modifications installed during the 1971-1973 service building addition were designed in accordance with the following applicable codes:

All nonatmospheric pressure bearing liquid radioactive waste process piping is designed to ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Nuclear Class 3, 1971 Edition, while all atmospheric radwaste drainage piping is designed to USAS B31.1.0-1967 Power Piping.

All tanks are designed to either ASME B&PV Code, Section III, Class 3, 1971, or API-620, 1970, or API-650, 1970, as follows:

ASME Section III: Pressure Vessels Over 15 Psig Design

API-620: 0 to 15 Psig Tanks (Tested at 5 Psig)

API-650: Atmospheric Tanks (Without Pressure Tests)

A more detailed listing of codes used for these components and piping will be found in Table 11-4.

In addition, it should be noted that the Systematic Evaluation Program (SEP) Topic III-1, "Quality Group Classification of Components and Systems (Palisades)," which was issued in December 1981 does not address liquid waste management system components and piping, all of which fall within Quality Group D, as defined in NRC Regulatory Guide 1.26. The Liquid Radioactive Waste System is now considered to be a Quality Group D system.

11.2.2 SYSTEM DESCRIPTION

The Liquid Radioactive Waste System is divided into three sections: (a) the clean waste section which processes high-activity, high-purity (low solids) liquid waste, (b) the dirty waste section which processes low-activity, low-purity (high solids) liquid waste and (c) the laundry waste section. The basic system is shown in Figures 11-1 and 11-2. Component ratings and descriptions are included in Tables 11-3 and 11-4.

11.2.2.1 Clean Waste Section

In the clean waste section of the Liquid Radioactive Waste System, wastes are collected, monitored, and processed by a combination of holdup (thereby permitting natural decay), filtration and ion exchange treatment (removal of insoluble particulates and soluble ions), and are stored for eventual discharge to Lake Michigan. Quantities of clean waste are obtained from the Chemical and Volume Control System's bleed letdown (primary coolant - see Figure 9-19), and are generated due to Plant start-ups and shutdowns which require that primary coolant boron concentrations be varied to maintain the necessary shutdown margins.

Liquid waste from the Chemical and Volume Control System first passes through an ion exchanger and purification filter before entering the clean waste section. In the clean waste section, this stream passes through the vacuum degasifier and is then discharged into one of the clean waste receiver tanks. The vacuum degasifier removes hydrogen and fission product gases and discharges them to the waste gas surge tank located in the Gaseous Radioactive Waste System.

The primary system drain tank collects liquid waste from sources within the containment building as listed in Table 11-5. The primary loop drains provide the major source of liquid to the drain tank. Liquid waste from the primary system drain tank is also passed through the vacuum degasifier before entry into one of the clean waste receiver tanks.

The equipment drain tank collects liquid waste from outside the containment building as listed in Table 11-6. Liquid waste from the equipment drain tank is discharged to the clean waste receiver tanks through a filter.

The radiochemistry lab drain tank collects the liquid waste generated while sampling the primary coolant for chemical and radiochemical analysis. Liquid waste from the radiochemistry lab drain tank is discharged to the Solid Waste Management System for solidification or can be sent to the dirty radwaste evaporator for volume reduction.

Four 50,000-gallon clean waste receiver tanks located inside the containment building provide temporary storage for collected liquid waste to allow for natural decay and to permit sampling of liquid waste activity. Liquid waste from the receiver tanks is discharged through the clean waste filter to remove insoluble particulates and through the radwaste demineralizers to remove soluble ions.

The effluent from the clean waste demineralizers passes to the Alternate Radwaste Processing System. The Alternate Radwaste Processing System has the capability to process radioactive waste either continuously or in batches.

The Advanced Liquid Processing System (ALPS) portion of the Alternate Radwaste Processing System provides for the removal of ionic impurities and suspended particulate from liquid radioactive waste through the utilization of ion exchange, chemical pretreatment and deep bed filtration. The system uses granular activated carbon deep bed filter(s) to remove total organic carbon content from the waste stream and a combination of cation / anion demineralizer vessels to remove dissolved cationic/anionic contaminants.

The Advanced Injection Method (AIM) portion of the Alternate Radwaste Processing System provides for the removal of small particulate and colloids, such as silica, organic compounds (such as color and odor producing substances), and metal oxides which are too fine to be easily removed by the ALPS system alone. The AIM system injects a polymer into the feed stream, which agglomerates the colloidal particles allowing for efficient removal by deep bed filtration.

The ALPS/AIM system will process clean liquid radwaste from the discharge of the Clean Waste Receiver Tanks through the radwaste demineralizers. The radwaste demineralizer effluent is conveyed directly to the ALPS/AIM system for processing. The discharge of the ALPS/AIM system is through the miscellaneous (dirty) liquid radwaste discharge line to the Miscellaneous Waste Distillate Tank. Once sampled the Miscellaneous Waste Distillate Tank is normally pumped to the Utility Water Storage Tank for waste dilution/release to Lake Michigan, within the limitations of the Offsite Dose Calculation Manual.

11.2.2.2 Dirty Waste Section

In the dirty waste section of the Liquid Radioactive Waste System, wastes are collected, monitored and processed by a combination of filtration and demineralization. The dirty waste drain tank collects liquid waste from the sources listed in Table 11-7.

Liquid from the dirty waste drain tank is processed through the miscellaneous waste filters while en route to the miscellaneous waste holdup tanks for temporary storage. Effluent from these tanks is fed to the Alternate Radwaste Processing System.

As with the clean waste system, the Alternate Radwaste Processing System effluent is discharged to the Miscellaneous Waste Tank, which is then normally pumped to the Utility Water Storage Tank for waste dilution/release to Lake Michigan, within limitations of the Offsite Dose Calculation Manual.

11.2.2.3 Laundry Waste Section

The laundry waste section is no longer used. Laundry waste services are now contracted offsite.

Both dry cleaning and wet wash machines were removed from service in 1989. All laundry is now processed offsite by a vendor. The drain and filter units are still in place. With use of the Alternate Radwaste Processing System, detergent wastes are no longer dumped into the drain system as they are drained into the normal dirty waste system.

11.2.3 RADIOACTIVE RELEASES

The Liquid Radioactive Waste System was designed to reduce radioactive materials in liquid discharges from the Palisades Plant, except laundry waste discharges, to be within the applicable limits set forth in Title 10 CFR, Part 50, Appendix I (10 CFR 50, Appendix I), as stipulated in Subsection 11.2.1.

A discussion on how the Liquid Radioactive Waste System's sections meet these requirements follows:

11.2.3.1 Clean Waste Section

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made and is considered historical.

The maximum anticipated annual quantity of clean liquid waste of significant activity to be processed through the liquid waste system before being recycled to the Plant is 724,300 gallons. As shown in Table 11-2, 586,600 gallons, or 81% of the total quantity of liquid waste, is obtained from the Chemical and Volume Control System and passes through one of three purification ion exchangers. The activity of this portion of the clean liquid waste is assumed to be reduced by a decontamination factor (DF) of 10 for each nuclide except noble gases and tritium for which a DF of 1 has been assumed.

Four 50,000-gallon clean waste receiver tanks provide temporary storage for liquid waste inside the containment building. It is expected that a 30-day hold period will normally be possible at any time during the fuel cycle. Sufficient capacity is always available for at least a 7-day hold period which allows for significant natural decay of radioactive isotopes.

The radwaste demineralizers contain a mixed bed H-OH resin. Activity of liquid waste passing through a demineralizer is assumed to be reduced by a DF of 10 for each nuclide except noble gases and tritium. Three demineralizers are installed in the clean waste stream, with one in service for normal plant operations. Fission product gases are assumed to be 100% removed by the vacuum degassing, by diffusion in the clean waste receiver tanks and by the evaporators. The filters upstream of the demineralizers are sized to substantially remove insoluble corrosion products. However, no credit in system evaluation has been assumed for use of these filters nor for the filtering effect of the radwaste demineralizers.

Historically the system contained an evaporator in which the decontamination factor of the evaporator has been conservatively assumed to be 10^3 for iodine and 10^4 for all other isotopes except tritium for which 1 has been assumed. Suppliers of radwaste evaporators have observed decontamination factors greater than 10^6 in testing and actual operating experience.

The distillate from the evaporator is expected to have a maximum concentration of 10^{-4} μCi/cm^3 (except for tritium) and 10 ppm of boron. This effluent can be processed through a polishing demineralizer to further reduce the boron or can be sent directly to the primary coolant storage tank prior to reuse in the Plant, or nonnormal discharge to the discharge mixing basin.

This ends the historical information contained in this subsection.

During normal operation, tritium builds up in the Primary Coolant System and the primary coolant makeup water storage tank due to recycling of primary coolant leakage. During refueling, a portion of the primary coolant is mixed with SIRW tank water and spent fuel pool water causing the tritium concentration to increase in these volumes also. Assuming a 40-year Plant lifetime and zero primary coolant leakage, the maximum tritium concentration was found to be 2.8 μCi/cm^3 in the primary coolant; 2.2 μCi/cm^3 in the refueling cavity water; 2.4 μCi/cm^3 in the SIRW tank; and 1.0 μCi/cm^3 in the spent fuel pool water.

Additionally, the concentration of tritium in the building air due to evaporation from the refueling cavity and spent fuel pool is monitored on a periodic basis. Chemistry and Radiation Protection personnel review the monitoring results and ensure the appropriate radiation safety controls are implemented in accordance with 10CFR20.

The amount of tritium released each year (over the Plant lifetime) due to evaporation in the containment building and spent fuel pool building is shown in Table 11-9.

Using the annual value of X/Q of 4.64 x 10^{-6} s/m^3, the maximum concentration of tritium at the site boundary from these sources is 3.3 x 10^{-11} μCi/cm^3. This is 0.033% of the 10 CFR 20 limit of 1 x 10^{-7} μCi/cm^3.

In order to estimate the maximum tritium dose to the general public, a hypothetical rupture of the SIRW tank was considered. The SIRW tank was chosen because its volume (33,420 ft^3) is larger than that of the primary system makeup tank or the spent fuel pool volume. The tritium activity in the SIRW tank was taken to be 2.8 μCi/cm^3, the maximum predicted reactor coolant concentration over a 40-year operating lifetime. Thus, the total quantity of tritium that would be released to Lake Michigan in the event of a hypothetical rupture is 2.6 x 10^3 curies.

The average dose to an individual in nearby South Haven consuming this water was calculated to be 0.056 mrem. The conservative assumptions which were used in calculating this number are as follows:

Dilution by a factor of 1,000 in Lake Michigan (see Reference 2)

Surface plume of 10-foot depth

Daily intake for South Haven water supply - 3×10^6 gallons (see Reference 3)

Current past water supply intake = 0.21 mi/h (see Reference 2)

South Haven water supply service area - 6.15×10^3 people (see Reference 3)

Average daily water consumption - 2,200 cm^3 (see Reference 4)

Subsequent to the above analysis, a revised value of the atmospheric dispersion parameter, X/Q, has been calculated based on new meteorological data and utilizing the Computer Code "XOQDOQ." The updated X/Q of 3.0×10^{-6} s/m^3 is approximately 65% of the value used in the above analysis, further attesting to the conservative nature of the results.

The concentration of radioactive materials in the SIRW tank or any temporary outside tank is limited to 1,000 times allowable effluent concentration. This limit may be exceeded for operational flexibility if the conditions of the Offsite Dose Calculation Manual (ODCM) are met. This limit is based on the dilution factor of 1,000 (before reaching the South Haven city water intake) assumed for tank ruptures. The bases for the limit originate in NUREG-0472, Revision 2, Radiological Effluent Technical Specifications for PWRs, July 1979. The specification of the limit has been changed to a concentration rather than the curie limit in NUREG-0472.

The bases for Liquid Holdup Tanks in NUREG-0472 reads; "Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less that the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area."

11.2.3.2 Dirty Waste Section

The information in this subsection pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or was added as changes to the systems were made and is considered historical.

The miscellaneous waste demineralizers are assumed to provide a DF of 10 except for noble gases and tritium. Similarly, the decontamination factor of the dirty waste evaporator is assumed to be 10^3 for iodine and 10^4 for all other isotopes except tritium.

Table 11-8 summarizes the performance of the dirty waste system (by isotope) under design operating conditions of 1% defective fuel. No credit has been assumed for any decay in the miscellaneous waste holdup tanks. This table shows that the effluent from the demineralizers which will be recycled to the utility water storage tank is of negligible activity.

11.2.3.3 Laundry Waste Section

Detergent waste is now drained and treated as normal dirty waste. The laundry release limit of 2.5 x 10-8 uCi/cm3 was removed from the Technical Specifications in Amendment 85, effective 1/1/85.

11.2.4 BALANCE OF PLANT (BOP) INTERFACE

The Liquid Radioactive Waste System interfaces with the remainder of the Plant as follows:

11.2.4.1 Clean Waste Section

The clean waste section obtains wastes as bleed from the primary loop letdown line, or inflow from the primary system drain tank, and equipment drain tank, which deposit into the clean waste receiver tanks.

Processed liquid effluent is normally discharged to the Utility Water Storage Tank for waste dilution/release to Lake Michigan.

11.2.4.2 Dirty Waste Section

The dirty waste section obtains wastes from open sumps and drains which empty into the dirty waste drain tank.

Processed liquid effluent is normally discharged to the Utility Water Storage Tank for waste dilution/release to Lake Michigan.

11.2.4.3 Laundry Waste Section

The laundry waste section obtains waste from the laundry drain tank. System is available but not used. There is no longer a limit for direct release of laundry to the environment (see 11.2.3.3).

11.2.4.4 Alternate Radwaste Processing System

Alternate Radwaste Processing System sample discharges and equipment drains are routed to the existing floor drain in the Alternate Radwaste Processing System process area.

Makeup and system flush water for the Alternate Radwaste Processing System is obtained from the primary system makeup tank.

11.2.5 SYSTEM EVALUATION

The release to the environment from the clean and dirty liquid waste sections is within the limits established in ODCM. The treated wastes are normally stored in the Utility Water Storage Tank for waste dilution/release to Lake Michigan, within limitations of the Offsite Dose Calculation Manual. The solid waste generated during the clean and dirty waste filtration and ion exchange is prepared for shipment in accordance with applicable rules, regulations and orders of governmental authorities having jurisdiction and turned over to a carrier or carriers licensed by governmental authorities having jurisdiction for shipment to an authorized disposal area or areas.

As shown in Figure 10-6, 11-1, and 11-5, discharged radwastes are released to the discharge mixing basin after proper monitoring. These wastes are mixed and diluted and then overflowed from the mixing basin into Lake Michigan.

Under normal operating conditions, such as (but not limited to) steam generator tube leakages, fire, or pipe breakage, clean and dirty wastes will be discharged to the environment after processing to reduce the discharged radioactivity to levels which are as low as practicable, and in any event, in accordance with the objectives of 10 CFR 50, Appendix I.

To assess the potential impact of these releases, the maximum individual doses from liquid effluents were calculated assuming 100% release of the liquid effluents from the clean and dirty waste systems.

Maximum individual doses from liquid effluents were calculated by the NRC LADTAP computer code, using models given in Regulatory Guide 1.109 (March 1976) (see Reference 5). Dose factors, bioaccumulation factors and the shore-width factor as given in Regulatory Guide 1.109 and in the LADTAP code were used, as were use factors for water and fish ingestion and for water-related activities.

Radioactive liquid wastes from Palisades are discharged to Lake Michigan after dilution with circulating water system discharge. This flow is via low velocity surface discharge at the shoreline.

Maximum individual doses were calculated for water and fish ingestion and for external exposure for shoreline use, swimming and boating. The water ingestion pathway was analyzed for an individual drinking water from the nearest municipal water intake, located at South Haven. This location is 5 miles from the Plant, where a dilution factor of 1,000 was assumed to apply (see Reference 2).

For the fish pathway, an effective dilution factor of 15 was used based on the following factors:

1. Sport fish likely to be taken in the area, rainbow trout, brown trout, lake trout and salmon, migrate on the order of 0.9 to 7 miles per day (see Reference 6). Thus, it is unlikely that a fish would be exposed to an undiluted concentration on an average basis.

2. According to Reference 6, it is unlikely that fish residing in a plume show increased concentrations as a result of their presence, since the majority of their uptake occurs through a food chain. Because of the migratory nature of these fish, the effective dilution appropriate for the food chain must be much greater than that existing at the Plant discharge.

For the purpose of this analysis, a factor of 15 is believed to be suitably conservative. Source terms and additional assumptions used in this analysis are presented in the Appendix I Analysis of May 1976 (Reference 17).

The maximum calculated doses to individuals from liquid effluents are summarized in Table 11-10, which also presents the pertinent LADTAP input data used in this analysis.

11.3 GASEOUS RADIOACTIVE WASTE SYSTEM

11.3.1 DESIGN BASIS

The design basis for the Gaseous Radioactive Waste System is to efficiently store gaseous isotopes for a time period which will permit sufficient radioactive decay prior to their discharge to the environment within limitations of the Offsite Dose Calculation Manual.

Design and construction codes for components and piping, and applicable nondestructive testing requirements are listed in Tables 11-3 and 11-4. The design criteria for the Gaseous Radioactive Waste System was identical to the liquid portion (Subsection 11.2.1.2) with the exception that the gas decay tanks added during the 1971-73 modification were CP Co Design Class 1, per Section 5.2.

11.3.2 SYSTEM DESCRIPTION

The Gaseous Radioactive Waste System is divided into two sections: (a) the gas collection header which collects low-activity gases from liquids which have been previously degassed and/or vented in other waste handling steps, and (b) the gas processing section which collects gases from potentially high-activity sources. The Gaseous Radioactive Waste System is shown in Figure 11-3. Component ratings and descriptions are shown in Tables 11-3 and 11-4.

11.3.2.1 Gas Collection Header

Gases which may contain potentially radioactive gases are passed through the gas collection header where they are discharged through a high-efficiency filter to the suction side of the main vent exhaust fans. The gases are diluted by ventilation exhaust air and are discharged through the ventilation stack to the atmosphere. The primary sources of low activity gaseous waste include the atmospheric vents of the liquid radwaste drain, collection and monitoring tanks, and containment building via "D" clean waste receiver tank (RUD-1018 removed). This vent path from containment to the collection header is an alternate path for venting containment. The primary containment vent path is through the "D" clean waste receiver tank to the radwaste area exhaust fans (V-14A&B).

11.3.2.2 Waste Gas Processing System

The waste gas processing system collects all potentially high-activity gaseous waste. The gas surge tank collects and absorbs surges from the demineralizer vents, quench tank vent, primary system drain tank vent, volume control tank vent, vacuum degasifier vent (which takes input from either of two degasifier pumps), equipment drain tank, and evaporator vents. The waste gas surge tank also collects vent gas from relief valves on the dirty waste tank, equipment drain tank, and waste gas decay tanks. The waste gas surge tank discharges to one of three compressors which compress the gas for storage and decay in one or more of six waste decay tanks. If activities greater than 1×10^{-5} μCi/cm^3 (Xe-133) have not been detected, the waste gas surge tank can be discharged through a high-efficiency filter directly to the ventilation stack.

If the surge tank is discharging directly to the ventilation stack, a high-radiation condition (as identified by a continuously operating monitoring system taking samples from the discharge line) will automatically close the discharge valve which is upstream of the stack. On occurrence of high surge tank pressure, a waste gas compressor starts automatically and, taking suction from the surge tank, discharges to the decay tanks.

Three of the six waste gas decay tanks have a volume of 100 cubic feet each and are designed for 120 psig. The remaining three decay tanks (added during the 1971-1973 auxiliary building addition) have a volume of 225 cubic feet each and are also designed for 120 psig. The total Primary Coolant System gas inventory and the gas inventory of the volume control tank can be stored in two of the smaller tanks if required for a cold, degassed Plant shutdown.

11.3.3 RADIOACTIVE RELEASES

As described in Subsection 11.3.2.2, gaseous effluents entering the Plant's ventilation stack are radiologically monitored and flow controlled so that 10 CFR 20, Appendix B, Table 2 values used for the design basis are not exceeded. This discharge is then immediately diluted by mixing airflow from one of the two continuously operating ventilation fans which conservatively transport 75,000 ft^3/min of air up the stack.

The waste gases are controlled to limit dose rates at the site boundary to the limits imposed by 10 CFR 50, Appendix I. The requirements of Appendix I to 10 CFR 50, including limiting conditions of operation and process monitoring operability and surveillance requirements were implemented by the Radiological Effluent Technical Specifications, Amendment No 85, of the Plant Technical Specifications (see Reference 10). The effluent specifications were then relocated to the Offsite Dose Calculation Manual per NRC Generic Letter 89-01.

A listing of gaseous isotopic activity at the Plant boundary expressed as a fraction of 10 CFR 20 values used for the design basis for no-holdup and 60-day holdup periods is given in Table 11-11. Further analysis pertaining to calculated maximum offsite activity levels and doses will be found in Subsection 11.3.5.

11.3.4 BOP INTERFACE

As shown in Figure 11-3, the Gaseous Radioactive Waste System interfaces with nuclear systems as it processes waste gases which empty into the gas collection header and waste gas surge tank.

The gaseous wastes leave the Plant in diluted form from the Plant's ventilation stack. These waste gases are either sent directly to the stack without holdup or are temporarily stored in any one of six waste gas decay tanks under pressure depending on their radioactivity and isotopic content.

True BOP interface occurs with the Component Cooling Water System because this system provides the heat sink for the sensible heat formed in the waste gas compressor heads and their respective aftercoolers while compressing the waste gases.

11.3.5 SYSTEM EVALUATION

Maximum individual doses from gaseous effluents were calculated by the NRC GASPAR (see Reference 7) computer code, using models given in Regulatory Guide 1.109 (March 1976). The basic source term and meteorological data entering into the calculations are described in the Appendix I Analysis, Palisades Plant (Reference 17).

Calculations of maximum individual doses from gaseous effluents have been made for the following exposure pathways:

1. External doses due to cloud immersion

2. External exposure to materials deposited on the ground

3. Internal exposure via food chain pathways, including vegetation, meat, cow milk and goat milk

4. Internal exposure via inhalation

All standard or default GASPAR parameter values were utilized, including dose conversion factors, food intake rates, stable element transfer coefficients and time delays.

Computed doses include the open site terrain correction factor for the appropriate distance, as given in Regulatory Guide 1.111 (March 1976). The occupancy and shielding factor of 0.7, as given in Regulatory Guide 1.109, was applied.

Maximum offsite air doses were determined, among overland locations, to be 0.95 mrem/yr for beta radiation and 0.36 mrem/yr gamma radiation at 0.48 mile in both the SSE and SSW directions.

Meteorological dispersion and deposition data were reviewed in conjunction with data pertaining to nearest residences, vegetable gardens, milk and meat animals within five miles to determined locations where specific exposure pathways would result in maximum doses. The GASPAR input data used to analyze doses at the locations so identified are presented in Table 11-12. Dose results for each location are presented in Table 11-13.

Including both the plume and ground contamination doses, the highest computed external dose rates are about 0.143 mrem/yr to the total body and 0.34 mrem/yr to the skin (both at 0.63 mile S). The highest computed dose due to nonnoble gas isotopes is 2.3 mrem/yr to the thyroid of a child (0.88 mile ENE at the nearest garden). Actual annual gaseous release data, analyzed by GASPAR using 5 year average, meteorological data along with yearly land use data, has demonstrated that gaseous effluents are a small fraction of 10 CFR 50, Appendix I limits.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASIS

The design basis for the Solid Waste Management System incorporates the applicable regulatory requirements, including the following:

Regulatory Guide 1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

Title 10, Part 20 (10 CFR 20), "Standards for Protection Against Radiation"

Title 10, Part 50 (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities"

Title 10, Part 50, Appendix I (10 CFR 50, Appendix I), "Numerical Guides for Design Objectives and Limiting Conditions for Operation Guides To Meet the Criterion as Low as is Reasonably Achievable for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents"

Title 10, Part 70 (10 CFR 70), "Domestic Licensing of Special Nuclear Material"

Title 10, Part 71 (10 CFR 71), "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions"

Title 49, Parts 170 and 171 (49 CFR 170 and 171), "Department of Transportation (DOT) Hazardous Materials Regulations"

NRC Information Notice No. 90-09, "Extended Interim Storage of Low Level Radioactive Waste by Fuel Cycle and Materials Licensees"(2/90).

NRC Generic Letter 81-38, "Storage of Low Level Radioactive Wastes at Power Reactor Sites"(11/81).

NRC Generic Letter 85-14, "Commercial Storage at Power Reactor Sites of Low Level Radioactive Waste Not Generated by the Utility"(8/85).

IE Circular No. 80-18, "10 CFR 50.59 Safety Evaluations for Changes to Radioactive Waste Treatment Systems"(8/80).

Design and construction codes used in this system are generally identical with those depicted in Subsection 11.2.1.3 for Quality Group D system components and piping. Onsite buildings, other than the Auxiliary and Service buildings, used to process and store radioactive waste are engineered structures, but are not seismically qualified.

In addition, the volume reduction processing equipment is located in a structure that meets the design criteria provided in Regulatory Guide 1.143. The isolation point of the Radwaste System (CV-2752) is located in a CPCo Design Class I structure.

Spent nuclear fuel is not covered by this section. Discussion on this subject will be found in Section 9.11 (Fuel Handling and Storage).

11.4.2 SYSTEM DESCRIPTION

The Solid Waste Management System is designed to collect, process, package and store for future offsite disposal low-level liquid wastes, spent ion-exchange resins and assorted solid wastes according to type of waste and levels of radiation activity present. The system is shown in Figure 11-4.

11.4.2.1 System Modifications

The quantities of wastes processed by the system were increased when modifications to the liquid and solid waste systems occurred during the 1972-1973 service building addition. These changes were brought about by legal commitments to change the Plant to a "near-zero" release Plant. These were the addition of ion-exchange resins obtained from the radwaste polishing demineralizer and the liquid concentrates (or "Bottoms") obtained as a by-product from the clean and dirty radwaste evaporators.

Radwaste Volume Reduction System

Liquid radwaste concentrates were previously processed using a VECTRA RVR-200 System supplied by Molten Metals Technology. The RVR-200 has been removed.

Alternate Radwaste Processing System

Alternate Radwaste Processing System liquid radwaste product solids are transferred and dewatered with the Standard Dewatering System (SDS). The SDS is an NRC approved (Reference 18), temporarily connected, portable system which is moved to the Alternate Radwaste Processing System equipment room, as needed, to provide a means of transferring carbon, ion exchange resins and precoats from ALPS/AIM filtration vessels into a High Integrity Container (HIC) for processing and ultimate shipping for burial.

The SDS system also provides processing of the HIC contents, which includes the removal of slurry water (dewatering), which acts as a transportation medium, from both carbon and resins. Dewatering the HIC contents reduces the volume of waste.

Containers of waste sent to a burial site will comply with the specific burial site criteria prior to shipment. Burial containers are highly resistant to radiation and all forms of chemicals that would normally be found in the waste

stream or ground soil. They are approved by the individual burial ground agreement states as meeting 10CFR61 waste form stability requirements.

11.4.2.2 Radioactive Waste Storage Facilities

Michigan was denied access to the three existing burial sites in November 1990. Michigan has also been expelled from the Midwest Compact. In mid 1995, the Barnwell, S. C. waste disposal site lifted the ban on Michigan radioactive waste and shipping activities resumed until July 1, 1998 when Barnwell, S. C. was closed to out-of-compact states. The majority of the radioactive waste stored on site was disposed of at the waste facility. The South Storage Building was emptied of waste and released for other activities.

Estimated Volumes and Activities

Approximately 2600 Curies of activity, dispersed in 2500 cubic feet of solid wastes are generated from the plant site in a normal year (ie, a year in which there are no extended outages or major nuclear repair work being performed).

These wastes can be separated into the following categories:

	Ci/yr	Ft^3/yr
Expended Filter Cartridges	10	50
Dewatered Ion-exchange Resin-shielded	125	227
Dry Active Waste	2	11000
Irradiated Hardware	2500	10
Dewatered Ion-exchange Resin-unshielded	0.1	450

Packaging

Solid wastes not being solidified by addition of immobilizing additives are processed as follows prior to offsite shipment:

1. Secondary side spent ion-exchange resin is packaged in liners, and gross dewatered to the turbine building sump or Auxiliary Building prior to shipment/storage.

2. Primary side spent ion-exchange resin is packaged in high integrity containers (HIC), dewatered in Auxiliary Building and transported/transferred in specially designed shipping casks.

3. Dry active wastes such as contaminated clothing, rags, paper, towels, gloves, shoe coverings, plastics, wood and metal are shipped in sealand containers to a vendor for volume reductio (incineration/supercompaction).

4. Expended filter cartridges are drained then transported to the shielded storage area in the East Radwaste building in specially designed casks. Filters are then transferred to HICs located in shielded vaults.

5. Irradiated hardware is drained in Auxiliary Building and stored or shipped in special containers. Irradiated hardware is also stored in the spent fuel pool.

6. Small amounts of liquid mixed waste and contaminated oil are stored in overpacks containing approved absorbents. This material may be shipped to licensed processors for incineration or other licensed treatment and processing.

7. Waste media (carbon/resins, precoats) from the ALPS/AIM system is packaged in liners and dewatered to the Auxiliary Building prior to shipment/storage.

Facility Description

1. The East Radwaste Facility consists of two adjacent buildings connected by a shared roll-up door. Radioactive waste (bags, filters, wood, metal, etc.) is transported to the East Facility to be processed. Resin is dewatered prior to leaving the Auxiliary Building. The Dry Active Waste (DAW) is contained to prevent the spread of radioactive material during transportation to the East Radwaste Facility.

   The West building is currently used primarily for storage of cement rad vaults for storage of packaged resin and filter HICs. This building is also used for temporary storage of large contaminated or retired plant equipment awaiting processing or packaging.

   The East building is primarily used for processing Dry Active Waste. The building is serviced by a HEPA ventilation unit which is operated during processing operation

   The East building also contains a built-in cement shielded vault system for storage of high level filters, resin and DAW. All items placed in the vaults for storage are packaged in High Integrity Containers or sealed containers to maintain a contamination free area. Semi-portable concrete vaults are available to be used to store higher level DAW, resin, and filters.

2. The South Storage Building is a 40' x 80' engineered steel building. In January 1992 the main floor of the building was elevated 24 inches (18 inches compacted sand with 6 inches cement cap) to prevent water intrusion from flooding, cooling tower overflows or excessive rainfall.

This building is being released for other activities, but could again designated as a storage building if waste disposal facilities are not available in the future. Container surveillance and radiation area monitoring to show compliance with Generic Letter 81-38 will be instituted if used as a radioactive waste storage building. If used in this manner, this building will be used only to store DAW in metal boxes, steel drums, incinerator ash in HICs, packaged in accordance with NRC and DOT requirements. These metal boxes are to be stored around the outer walls of the main floor. Every box is equipped with risers (feet) to allow containers to be raised off the floor to prevent inadvertent water accumulation to cause external corrosion and possible degradation of container integrity. Incinerator ash will be stored in the center section of the building.

3. Some solid radioactive waste materials are stored in the spent fuel pool until they can be shipped off site.

Radiological Consequences

1. Gaseous--Accident releases from the radioactive waste storage buildings are not considered credible because of material, packaging and steel and/or cement shielding. However, to show compliance with Generic Letter 81-38 criteria, three accident cases were run as well as direct dose calculations to the site boundary. The accident cases were a small fraction of 10CFR100 limits and direct dose to the site boundary was less than 1 mrem/yr as required by Generic Letter 81-38.

2. Liquid--There are no liquid effluent consequences in radwaste building because all waste in the building meets dry radioactive material status except for a small amount of liquid mixed waste and some contaminated oil. This waste will be over-packed with absorbent material and in no event could the criterion per IE Circular 80-18 of MPC at the nearest water supply be approached.

Containers

These containers are selected based on structural strength, the ability to maintain container integrity during processing, packaging, storage and transportation. They will also demonstrate minimal corrosion effects from exposure to internal environment over a long period of time. All containers awaiting shipping to be buried, are stored inside the building to protect against corrosion from external environment. These containers comply with the requirements of 10CFR71 and 49CFR as well as burial ground criteria to prevent the need for repackaging prior to shipment. HIC lids are equipped with passive vents to allow depressurization of hydrogen, but do not permit migration of radioactive material. Additional semi-portable cement rad vaults to shield filled HICs will be placed in East Radwaste Building on an as-needed basis.

Monitoring Equipment

1. Area Monitor--The east building will contain an area monitor calibrated to read out in mR/hr (equivalent to mrem/hr). This area monitor provides local alarm and will initiate a phone alarm to the control room when dose rates in the area reach or exceed alarm set points.
2. Air Monitoring--The storage area at the east radwaste building is equipped with a continuous air monitor. This monitor has an adjustable visual and audible alarm. The air monitor alarm will also initiate a phone alarm to the control room.

Effluent Monitoring

The processing/sorting area at the east radwaste facility is equipped with a portable ventilation unit with HEPA filter. The ventilation exhaust is equipped with a sample collection system. This system consists of a flow meter, vacuum pump and particulate filter sampler.

ALARA

Purification filter (F-54), activated incores and other higher level material will be stored in 36 inch concrete vaults in east radwaste building when not practicable to store in areas with less shielding. Activated incores can also be stored in casks that have been analyzed in regards to the requirements of Generic Letter 81-38. Higher level DAW, resin (not T-104) and filters will be stored in 18 inch thick semi-portable concrete vaults. The design resin liner would have a 12 inch reading of 20.1 mrem/hr in the 18 inch vault (Reference 1). One inch steel donut shield is available if concrete vault is not adequate. High integrity containers will be stored in a five inch concrete, or equivalent, vault. DAW boxes will be stacked with lower reading boxes on the outside to minimize dose outside the radwaste areas. These shielding materials and methods are provided to address ALARA principles and maintain low radworker exposure.

11.4.2.3 Alternate Radwaste Processing Systems Standard Dewatering System

The Alternate Radwaste Processing System, processes clean and miscellaneous (dirty) liquid radwaste.

The radioactive the influent is filtered through the ALPS subsystem granular activated carbon deep bed filter(s) to remove total organic carbon content from the waste stream and a combination of cation/anion demineralizer vessels to remove dissolved cationic/anionic contaminants.

The system via the AIM subsystem provides for the removal of small particulate and colloids, such as silica, organic components (such as color and odor producing substances), and metal oxides which are too fine to be easily removed by the ALPS system alone. The AIM system injects a polymer into the feed stream which agglomerates the colloidal particles, allowing for efficient removal by deep bed filtration.

The ALPS solids are consisting of carbon, resins and removed solids are collected in the Alternate Radwaste Processing System filtration vessels. As required, the contents of the filtration vessels are transferred and dewatered with the Standard Dewatering System (SDS) to a disposable High Integrity Container (HIC) for processing and ultimate shipping for burial.

The SDS system also provides processing for the HIC contents, which includes removal of slurry water, which acts as a transportation medium, from both carbon and resin. Dewatering the HIC contents reduces the volume waste for burial. Dewatering to meet disposal site criteria can be completed if the waste is sent directly to a burial site in lieu of using a vendor for processing. The SDS utilizes an approved vessel for storage and burial purposes.

11.4.3 RADIOACTIVE RELEASES

All liquid discharged to the environment from the Liquid Radioactive Waste System and solid wastes leaving the Solid Waste Management System are accounted for by quantity, activity and isotope inventory and are reported to the NRC on an annual basis. Packaged and immobilized low-level wastes meeting all applicable state and federal regulations could be shipped to licensed shallow-land burial grounds for internment. Waste shipments are made primarily by licensed commercial motor vehicles. All such shipments are reported to the NRC on an annual basis.

11.4.4 BOP INTERFACE

As part of the Alternate Radwaste Processing System, the SDS dewatering liquid effluent is routed to the existing floor drain in the process area for collection in the Liquid Radioactive Waste System.

Makeup water is supplied to flush the concentrates supply piping to the SDS obtained from the primary system makeup tank.

Station air is supplied to the SDS system to operate the system.

Building services required are HVAC, fire protection and electrical power.

11.4.5 SYSTEM EVALUATION

The portion of the Solid Waste Management System which processes liquid concentrates and spent ion-exchange resin is comprised of pressure retaining components to assure containment of these waste streams.

The Solid Waste Management System is capable of meeting all criteria listed in Subsection 11.4.1, "Design Basis."

11.4.6 REQUEST TO RETAIN SOIL IN ACCORDANCE WITH 10CFR20.302

(10CFR20.302 was in affect when this commitment was made.)

Consumers Power Company correspondence dated November 12, 1987 and January 25, 1988 requested authorization to dispose of contaminated soil in place as specified by 10CFR 20.302. The area known as the South Radwaste Area has been contaminated by numerous cooling tower overflows and contamination was redistributed by heavy rain showers. Although the majority of the radioactive material has been packed for waste shipment, a large volume of very low activity radioactive material remains. This volume of material would be very expensive to ship as waste. The NRC concurred with this request (Reference 16).

The specific area contaminated is noted as Area B on the survey grid map in Reference 11. The entire area is fenced and is about 12,000 sq ft of soil exposed with the remainder buildings and asphalt. The inhalation pathway is for breathing suspended soil from this area. The radworker could receive 8.03E-04 mrem/50year maximum organ (liver) dose and the infant could receive 3.16E-05 mrem/50year maximum organ (liver) dose, both of which are insignificant. Direct dose to a radworker is less than 2E-03 mrem/hr. Occupancy in this area should not average more than 2 hours/week or 100 hours/year, which would result in a dose of <1 mrem/year.

The radwaste activities which caused the contamination of the soil were completely relocated or have been replaced with other methodology. The South Storage Building has been deconned and is being used for other activities. Some fixed contamination is present in floor cracks and vaults. This has been documented for plant decommissioning.

The waste in this building has been shipped for disposal and the building released for other activities.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM

The process radiation and effluent radiological monitoring and sampling system was designed to assure that ionizing radiation levels are indicated and alarmed so that action, either automatic or manual, can be taken to prevent radioactive release from exceeding the limits of 10 CFR 20.

Detection devices are located in the various process systems and at selected positions throughout the containment and auxiliary buildings to monitor radiation levels and annunciate any abnormally high radiation activity. Instrument ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20.

11.5.1 DESIGN BASIS

Radiation monitoring devices using proven photomultiplier, scintillation and geiger-type ionization detection chambers have been utilized in the process and effluent radiological monitoring and sampling system.

Each process instrument has been selected according to the type of radiation to be measured, and sized to encompass the entire range of radiation activity which corresponds to the Plant's design power levels and failed fuel criteria. A listing of the process sample points, instrument sensitivity, and other information pertaining to these instruments is presented in Table 11-15 and shown in Figures 9-17, 11-5 and 11-6.

Additionally, all monitors in the stack-gas, containment air, off-gas, waste gas, engineered safeguards areas ventilating system discharge, radwaste ventilation and radwaste liquid discharge systems have been supplied with check sources. The check source is to simulate a radioactive sample and serve as a check for both the readout and detector.

11.5.2 SYSTEM DESCRIPTION

The process and effluent radiological monitoring and sampling system is a collection of radiological instrumentation and, therefore, cannot be described as a unique system by itself except for the stack monitoring subsystem (described in Subsection 11.5.3).

The process sampling systems and area radiation monitors, individual detectors, power supplies, and readout devices have been selected to operate in humidity and temperature ranges appropriate for their service areas.

Provisions are included to permit periodic testing while detection equipment is operational.

The aforementioned detection devices display their information in radiation monitoring equipment panels located inside the main control room. The panels provide mounting for indicators, power supplies and alarms for each of these radiation monitoring systems. Two of the panels are located beside the area radiation monitoring panel. The process liquids radiation monitoring panel and the gas radiation monitoring panel are fed by the instrument ac bus which, in the event of a loss of power, is fed by the diesel generators. The circulating water discharge monitor sample pump is powered from a normal lighting panel, with a low flow alarm powered from the instrument ac bus.

The type of detectors used and the information displayed are listed in Table 11-15. The sensitivity and alarm conditions for each instrument are also listed.

11.5.3 EFFLUENT MONITORING AND SAMPLING

Liquid effluents which are discharged from the Plant are monitored by a process sampling detector located in the circulating water discharge structure. The process sample is obtained from a continuously flowing (freeze protected) sample loop which is part of the monitoring system.

In 1983 a main steam relief monitoring system was installed to monitor accident releases in the event the atmospheric dump or safety valves lift. Two monitors, one viewing each main steam line, continuously monitor and record the activity present in the secondary steam.

Gaseous effluents leaving the Plant via the stack discharge system are monitored by the stack monitoring system. Abnormal gaseous releases detected by any of the process or area radiation monitors within the radiation controlled areas of the containment and auxiliary buildings are processed by engineered ventilation systems which ultimately discharge to the Plant's stack.

The Radioactive Gaseous Effluent Monitoring System (RGEMS) consists of normal range particulate/radioiodine filters, NaI gamma detector, scintillation chamber beta detector, and an accident range filter and ion chamber (refer to Figure 11-6). Flow through the system is provided by two 100% capacity diaphragm vacuum pumps. The flow is controlled by automatic flow control valves to provide isokinetic sample flow based on varying stack gas flows.

During normal operation, an isokinetic sample of the stack effluent is routed through a particulate/radioiodine filter then through the beta detector. The filter is continuously monitored by the NaI detector to detect any buildup on the filter. The filter is normally changed and counted on a weekly basis by Plant personnel.

On indication of abnormal stack effluent activity (alert level), a 15-second grab sample is automatically trapped in a sample bottle and an annunciator in the control room indicates the off-normal condition.

Following a high level indication, the normal sample loop is bypassed and the sample flow is split with approximately 2.0 standard liters per minute directed through the high-range filter and the balance through the ion chamber. To avert a too rapid buildup of activity on the filter, the capability exists to interrupt the sample flow through the filter periodically for periods ranging from 6 seconds to 54 seconds every minute. The continuous monitoring capability of the high-range noble gas monitor is not affected during filter flow interruption. A "high radiation" annunciator in the control room alerts the Plant operators to the condition.

Systems may be controlled either locally or remotely from the control room. Dual microprocessor controllers provide system control through the normal, alert and high operating modes. Normally the controller located in the control room provides full system control. In the event of failure the local controller takes control of the system functions.

Refer to Table 11-15 for details of the monitors.

11.5.4 SYSTEM EVALUATION

All process systems which contribute to Plant discharges are monitored prior to entering the various discharge systems. Each discharge system is also monitored, providing redundancy of radiation detection for Plant effluents. The radwaste area, containment air, waste gas, engineered safeguards pump room, and the off-gas radiation monitoring systems are backed up by the stack-gas monitoring system. The service water, radwaste liquids discharge, component cooling and the steam generator blowdown radiation monitoring systems are backed up by the circulating water discharge monitor.

Testing and maintenance for all systems, circuit testing of readout equipment and power supplies can be performed from the panels located in the control room. The circuit being tested or repaired is inoperative during that time and acts as if it were a tripped channel. The containment high-radiation monitors are continuously monitored while in service for loss of power, loss of detector high voltage and for loss of detector signal.

11.6 RADIATION PROTECTION

11.6.1 GENERAL

11.6.1.1 Radiation Exposure of Personnel

The basis for the shielding design for normal Plant operation is the Title 10, Code of Federal Regulations, Part 20 (10 CFR 20), entitled "Standards for Protection Against Radiation." The exposure of workers to licensed radioactive materials is limited to values in 10 CFR 20.

All areas of the Plant are subject to these regulations. The areas are zoned according to their expected occupancy by Plant personnel and their designed radiation exposure level under normal operating conditions.

Allowable design for all accessible areas of the Plant is not to exceed a total effective dose equivalent of 5 rems per calendar year. For all areas outside the Plant, the total effective dose equivalent should not exceed 0.1 rem per calendar year.

The control room shielding design provides adequate protection during a Maximum Hypothetical Accident (MHA) such that the accumulated dose will be within 10 CFR 50 Appendix A, GDC-19 limits as long as an individual remains in the control room. For exposures obtained during Plant ingress and egress as well as during excursions to selected Plant areas, limits of 10 CFR 100 will not be exceeded.

11.6.1.2 Radiation Exposure of Materials and Components

No regulations similar to those established for the protection of individuals existed at the time of FSAR preparation for materials and components. Materials were selected on the basis that radiation exposure would not cause significant changes in their physical properties which would adversely affect their operation during the design life of the Plant. Materials for equipment required to operate under accident conditions were selected on the basis of the additional exposure received through the time of their required operation in the event of an MHA.

In 1980, an analysis was performed to qualify safety-related electrical equipment for operation in a radiation environment following an accident. See Subsection 8.1.3.

11.6.2 RADIATION ZONING AND ACCESS CONTROL

The radiation zoning of the Plant areas established in the plant design is shown on Figures 11-7, 11-8 and 11-9.

The following list identifies the different zones used for the design of the Palisades Plant (the terms "controlled" and "uncontrolled" are used in a generic sense to describe access controls):

Zone Designation	Design Dose Rate (mrem/h on a 40 h/Week Basis)	Description
I	≤ 0.5	Uncontrolled, unlimited access
II	≤ 1.5	Controlled, unlimited access. 40 h/week
III	≤15.0	Controlled, limited access for routine tasks. 6 h/week
IV	≤ 100	Controlled, limited access for short periods. 1 h/week
V	> 100	Controlled occupancy for very short periods. Occupancy during emergencies. Normally inaccessible

UNCONTROLLED areas are those that can be occupied by Plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.

CONTROLLED areas are those 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.

ACCESSIBLE areas are those that will encounter radiation dose rates of less than 100 mrem/h and which can be entered either through open passages or unlocked doors. These areas can be entered by all individuals who have passed through the Plant access control station.

INACCESSIBLE areas are those where dose rates above 1,000 mrem/h can be expected. These areas are either blocked off completely or can be entered only through locked doors. Access is limited to an intermediate degree to areas where dose rates are between 100 mrem/h and 1,000 mrem/h. Access to areas greater than 1,000 mrem/hr is controlled by Chemistry and Radiation Protection personnel or, or in an emergency, by Operations supervision.

Access restrictions for controlled areas may be enforced by removable concrete shielding blocks, locked doors, chains, etc. Access is supervised commensurate with radiation health risk. Areas with dose rates in excess of 1000 mrem/hr may be barricaded, posted, and a flashing light used as a warning device consistent with Technical Specifications 5.7.

11.6.3 GENERAL DESIGN CONSIDERATIONS

The shielding design considers three conditions:

1. Normal full power operation. This also includes shielding requirements for certain off-normal conditions such as the release of fission products from leaking fuel elements.

2. Shutdown. This condition deals mainly with the radioactivity from the subcritical reactor core, with radiation from spent fuel bundles during onsite transfer, and with the residual activity in the reactor coolant and neutron-activated materials.

3. Maximum hypothetical accident. This includes design to reduce radiation streaming from the containment purge penetrations, personnel airlock, equipment lock and other accident sources. This permits access to areas required for emergency operations.

11.6.3.1 Specific Design Values

The material used for most of the Plant shielding is ordinary concrete with a bulk density of 144 lb/ft^3. Only in a very few instances has steel or water been utilized as primary shielding material.

11.6.3.2 Reactor Core Data

The reactor core power rating was assumed to be 2,650 MWt with 1% failed fuel and an average coolant velocity of 12.8 ft/s in the core. The core and fuel dimensions are discussed in Chapter 3.

11.6.4 SHIELDING DESIGN

11.6.4.1 Containment Building Shell

The containment shell serves two main shielding purposes:

1. During normal operation, it shields the surrounding Plant structures and yard areas from radiation originating at the reactor vessel and the primary loop components. Together with additional shielding inside the containment, the concrete shell will reduce radiation levels outside the shell to below 0.5 mrem/h in those areas which are occupied by personnel either on a permanent or routine basis.

2. In the event of a maximum hypothetical accident, the shell shielding will reduce Plant and offsite radiation intensities, emitted directly from released fission products, to acceptable emergency levels. The concrete roof of the containment will effectively reduce contributions due to sky-shine. The environmental consequences associated with an MHA are discussed in Section 14.22.

11.6.4.2 Containment Building Interior

During operation, most areas inside the containment will be inaccessible because of dose rates greater than 100 mrem/h. The containment air room is an exception. Personnel are able to enter this small portion of the containment which houses equipment and instruments that need inspection during operation. Shielding surrounding the air space (space adjacent to personnel hatch) keeps radiation levels down to less than 15 mrem/h.

Large sections over the steam generator chambers and the reactor transfer and storage pool are open and unshielded. These openings cause a high dose rate at the refueling floor. Neutrons streaming out of unfilled transfer and storage pool increase the containment internal dose rate. The missile shield over the reactor and the center part of the pool restricts these exposures in other areas. The reactor vessel which is the major radiation source is surrounded by a concrete shield.

Additional concrete shielding is provided around equipment that carries primary coolant water. The shielding is governed by strong N^{16} decay gamma radiation. Extra shielding is added for protection at the entrance to the containment air space (space adjacent to personnel hatch).

After shutdown, most of the containment becomes accessible but for limited lengths of time due to residual fission and activation product activities. At the refueling floor inside the containment, dose rates will range from 1.5 to 40 mrem/h, depending on the location at the refueling floor and the interval of time after shutdown. These dose rates are due to neutron-activated materials and fission and activation products inside the Primary Coolant System, and radioactivity absorbed on surfaces which had prior contact with primary coolant.

For the transport of spent fuel elements, concrete and gravel shielding provides protection for the areas close to the transfer route of the fuel. Shielding is provided around the reactor internals storage pool, the steam generator compartments and the clean waste receiver tanks. This shielding is designed for personnel protection during storage of activated reactor internals and for protection during the refueling operation.

11.6.4.3 Auxiliary Building (Including Radwaste Building Addition)

All radioactive areas can be reached through service corridors which are entered from the access control station. Radiation exposures are minimized during normal equipment operation by use of reach rods which penetrate through the shield walls or have other remote operators. Gauges and other instruments which need visual checking from time to time can be inspected from the corridor or on the local or main control boards. Plant operating personnel are thus able to perform duties necessary for normal operation of the Plant with a minimum of radiation exposure. The different systems are isolated from each other by individually shielded chambers. Systems and equipment can be isolated for maintenance or repair with no significant radiation interference from other systems or equipment.

During operation, the major radiation sources to be expected are the tanks, pumps and piping containing contaminated drainage. The concentrations of radioactive fission and activation products in these systems are expected to be generally of the low to medium type (10^{-3} μCi/cm^3 to 10 μCi/cm^3). Concrete shielding walls provide protection for the adjacent basement operation areas. Concrete shielding is provided around the waste gas decay tanks.

The spent resin holdup tank chamber is not accessible; however, a portion of the shielding constructed of solid concrete blocks is removable for access.

The dirty waste filter, spent fuel pool filter and demineralizer are shielded by concrete for protection of the adjacent uncontrolled areas, such as the diesel generator room. The use of dirty waste and fuel pool filters has been discontinued because of ALARA concerns.

Other equipment chambers contain volume control equipment, treated and filtered waste, laundry and radiation chemistry drain tanks and pumps, radwaste fan and filters, etc. A special area is designated and designed for the decontamination of equipment. This decontamination room can be sealed off by doors in order to prevent the spreading of contamination during the cleaning operations.

The area above the radwaste filters, demineralizers and spent resin tank, which is fenced off and controlled, serves for removal of the spent resin, filter and demineralizer components. Each filter and demineralizer can be reached by a removable concrete hatch. The adjacent service building may have to be shielded during removal of contaminated materials by construction of removable shielding shadow walls.

The Alternate Radwaste Processing System components which include the Advanced Liquid Processing System (ALPS), Advanced Injection Method (AIM) sub systems. These subsystems consist of one booster pump/control module, five demineralization/deep bed carbon filtration vessels, one AIM system control panel and one sample sink. These components are shielded by concrete for protection of the adjacent uncontrolled areas and the filtration vessels and radioactive waste supply line to the Alternate Radwaste Processing System area containing the equipment.

The Alternate Radwaste Processing System piping outside of the radiologically controlled Alternate Radwaste Processing System equipment room is shielded to maintain the uncontrolled general area dose rate less then 0.5 mrem/h to meet the design of a Zone I classification.

The counting room has been shielded in order to reduce background radiation. For the room containing the ventilation system, concrete is used to shield against radiation from the ventilation filters and also against radiation streaming out of the containment through the piping penetrations for the main steam lines.

After shutdown of the Plant, the shutdown heat exchangers will become significant radiation sources. Fission product activity and neutron-activated corrosion products present in the primary coolant are the radiation emitters. The heat exchangers are accessible with a 5 to 100 mrem/h dose rate.

The environment outside the containment will remain much the same during operation and shutdown of the Plant. The transfer of spent fuel is the primary source of radiation during periods of shutdown. The relative closeness of fully accessible and uncontrolled areas requires especially heavy shielding for the areas surrounding the spent fuel pool.

In the event of an MHA, the engineered safeguards pump rooms will become inaccessible, particularly when recirculating water from the containment sump. Shielding has been placed between these high sources and the Plant control room to prevent excessive direct radiation shine into the control room.

The areas close to the containment and close to the spray and injection systems at elevation 590 feet 0 inch will experience radiation levels as high as several tens of rem/hr and these areas will not be entered for some time after an MHA.

A concrete and steel radiation shield wall has been placed outside the equipment hatch to provide additional radiation protection for the spent fuel pool area and the control room.

The control room has concrete shielding for those sides which are in direct line of sight with the containment building. Together with the containment shell, the integrated whole body shine dose inside the control room would be less than 2.5 rem over a period of 30 days following an MHA.

11.6.4.4 Turbine Building

The turbine building is fully accessible and uncontrolled with dose rates much less than 0.5 mrem/h during normal Plant operation as well as during shutdown.

In the event of an MHA, some portions of the turbine building will not be entered by personnel since dose rates from the reactor building initially will be of the order of several rem/h for the unprotected portions of the building close to the reactor building. Other areas which require personnel entry for emergency procedures have been shielded to allow safe access.

11.6.4.5 General Plant Yard Areas

The radiation shielding design of the containment and auxiliary buildings protects all Plant yard areas from excessive radiation exposure. All yard areas which are frequently occupied by Plant personnel during normal operation and shutdown receive a radiation field of less than 0.5 mrem/h by design.

11.6.4.6 Other Buildings

Other buildings such as the feedwater purity building and the east and south radwaste buildings are designed so that shielding is provided around potential radiation sources.

Equipment storage buildings outside the protected area, such as the south and east storage buildings are controlled less than 0.5 mrem/hr at the building exterior or fence to control access.

In addition, adequate radiation shielding is provided at the interim old steam generator storage facility located outside the protected area boundary in a Zone I access area. The dose rate at the facility's exterior surface is less than 0.5 mrem/h to meet the design of a Zone I classification. The shielding design also meets the requirements of 40 CFR 190 for offsite dose.

11.6.5 AREA RADIATION MONITORING SYSTEMS

11.6.5.1 Design Basis

This system consists of monitors, instrumentation and alarms that serve to warn Plant personnel of increasing radiation levels in various Plant areas. Reliance is placed on the process radiation monitoring system for early warning of a Plant malfunction resulting in increasing radiation levels that might result in a health hazard.

All of the electronic circuitry except for the detectors (Geiger-Mueller tubes or ion chambers) is solid state. The circuits and their components have been selected to operate in humidity and temperature ranges appropriate for their service areas.

Detector ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20 and the access control zoning. The area radiation monitoring system instruments and detectors have been chosen for their proven reliability in other Plants, and spare items and portable units are provided to permit operation during prolonged maintenance.

The radwaste buildings are outside the plant protected area and are not connected to the area monitor system. Portable monitors with local readouts will be used with auto dialer phone alarm notification equipment because of remote locations. The monitors have been selected to operate in humidity and temperature ranges appropriate for their service areas.

11.6.5.2 System Description

Thirty four area monitors monitor locations within the auxiliary and containment buildings. They provide both indication and warning of radiation levels in both normally radioactive and non-radioactive areas. The 34 area radiation monitoring system detectors are wall mounted, coaxial ion chambers. The associated meters are designed with range and sensitivity suitable for their location. High radiation levels and individual circuit failures are alarmed both visually and audibly on the area radiation monitoring panel. Readouts and display equipment for the area radiation monitoring system are located on panels in the control room. The area radiation monitoring panels receive power from the preferred ac buses or the instrument bus.

Three additional monitors have been placed in the radwaste storage and processing buildings. Only two monitors are operational since the South Storage Building is not being used for Radwaste Storage at this time. These units are stand alone units with local readouts and alarms. If high radiation levels are reached in a storage building, the local monitor alarm will sound and also initiate an auto dialer phone notification to the plant main control room.

Pursuant to NUREG-0737, two high range gamma monitors have been installed in the containment building. The monitors are ion chambers with the readout range extended to 10^7 R/hour. These monitors are designed to provide a continuous readout of containment radiation levels for all conditions ranging from normal operation to hypothetical accident conditions. Calibration is performed by electronic equipment due to extreme exposure rate range of the instruments.

The monitors have been selected with ranges and sensitivities appropriate for their service areas. Alarms set points are adjustable to enable monitoring to within the requirements of 10CFR20 and the access control zoning. The location, range, and sensitivity of the monitors are listed in Table 11-16.

11.6.5.3 Testing and Maintenance

Circuit testing of readout equipment and the power source can be performed from the control room. The circuit being tested or repaired is inoperative during this time and acts as a tripped channel. Radiation sources are used to calibrate the detectors and circuits.

11.6.6 HEALTH PHYSICS

11.6.6.1 Facilities

Because the radiation protection and Plant water chemistry analyses are closely intermingled, laboratory complexes are used for most analytical work. This area, which is primarily located in the auxiliary building, is made up of the following:

1. A sample room is located adjacent to the containment building personnel air lock. Here, all samples of primary coolant are taken. They are then carried to the laboratory for radiochemical analyses.

2. A modified radwaste sampling area.

3. A controlled laboratory where highly radioactive samples are analyzed or diluted.

4. A controlled laboratory where intermediate-level radioactive samples are analyzed.

5. A counting room where station samples are counted for activity level.

6. A counting room for low-level environmental samples.

7. The Health Physics office.

8. A personnel decontamination facility associated with the access control area where personnel can be monitored for contamination and appropriate measures taken.

9. A change room for donning anti-contamination clothing.

The chemistry and controlled laboratories are provided with constant air-flow fume hoods which vent to the Plant stack. The auxiliary building also has a portable instrument storage area on the 611' elevation to provide instrumentation for use in radiation controlled areas of the Plant. This permits convenience in returning instruments as personnel return through this elevation to access control. The personnel decontamination facility near access control contains a shower and instruments for monitoring of decontamination activities.

All laundry is processed offsite. Storage for laundered clothes is provided in the anti-contamination clothing change room.

11.6.6.2 Tool and Equipment Decontamination Facility

A hot instrument shop provides an area for repair and maintenance of contaminated instrumentation.

A cask washdown pit for decontaminating large components and fuel transfer casks is located on the upper level of the spent fuel pool. This pit is also used for miscellaneous decontaminating of tools and equipment associated with fuel pool work.

An enclosure specifically designed and fabricated to handle radioactive material is located in the North Radioactive Material Storage Building inside the Protected Area. It is equipped with High Efficiency Particulate Air (HEPA) filtration and airborne radioactivity monitoring during use.

11.6.6.3 Calibration Facility

A calibration facility is provided in a separate room adjoining the turbine building at the 590-foot elevation. The room has a deep well and storage cells constructed of shielding blocks.

The room housing the calibration facility is kept locked and entrance is allowed only with the approval of the Chemistry and Radiation Protection Department.

The J.L. Shepherd Model 89 is an instrument used for calibrating portable radiation instruments and contains a 400 Ci Cs-137 and a 130 mCi Cs-137 source. The J.L. Shepherd is located in the instrument auxiliary building equipment cage. The cage is normally locked and the J.L. Shepherd is padlocked when not in use. Keys to the cage and the J.L. Shepherd are controlled by the Chemistry and Radiation Protection Department.

11.6.6.4 Radiation Control

Personal radiation exposure is kept to a practical minimum by a combination of shielding and access control.

11.6.6.5 Shielding

As previously discussed, shielding is designed to keep exposures to personnel to a practical minimum. As Plant operations progress, if it is found that the shielding in given locations is insufficient, either more shielding will be added to reduce exposure to the design rate or access will be limited to maintain exposures to a practical minimum.

11.6.6.6 Access Control

Access to the restricted area (as defined in 10CFR20, Section 20.1003) is limited on the basis of radiation levels or the presence of radioactive materials. Palisades has designated restricted areas as the Plant Protected Area (inside the security fence). The area within the site boundary is designated the owner controlled area. There are several secondary restricted areas outside the primary restricted area (e.g. East Radwaste, South Radioactive Material Buildings).

Any area in which radioactive material is stored, handled or processed and in which radiation levels are in excess of those defined as a "radiation area" in 10 CFR 20 has access controlled for purposes of radiation protection. These areas are designated “Radiologically Controlled Areas.” In general, all areas accessed through Door 105A are designated as radiation controlled areas. All entrances to these areas will be controlled.

Within a Radiologically Controlled Area, access to areas of higher radiation exposure rate are further controlled and defined as:

1. Radiation Area

   Any area, accessible to personnel, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 millirems in 1 hour at 30 centimeters (12 inches) from the radiation source, or from any surface that the radiation penetrates, is designated as a radiation area. Personal exposure in a radiation area are kept to a minimum by both administrative procedures based on accumulated dose records and by limiting the time spent in the area. Chemistry and Radiation Protection personnel perform routine surveys of, or utilize continuous remote monitoring of all accessible areas of the Plant to ascertain the current status of the radiation levels in these areas. Status sheets or their equivalent are readily available to display radiation levels and significant radiation sources in the area. Radiation areas may be isolated by a rope or chain, but they are all posted with signs, including the following words: CAUTION - RADIATION AREA

2. High-Radiation Area

   Any area accessible to personnel, in which radiation levels could result in an individual receiving a dose equivalent in excess of 100 millirem in one hour at 30 centimeters (12 inches) from the radiation source, or any surface the radiation penetrates is designated as a high radiation area and each entryway must be barricaded and conspicuously posted. The posting shall have the following notation: CAUTION-HIGH RADIATION AREA or DANGER-HIGH RADIATION AREA.

3. Locked High Radiation Area

   A high radiation area, in which radiation levels could result in an individual receiving a dose equivalent greater than 1 rem in one hour at 30 centimeters from a radioactive source, or any surface the radiation penetrates requires each entryway be conspicuously posted and locked or continuously guarded. Technical Specifications allow the locked and continuously guarded requirement to be met with a barricade and flashing lights under specific conditions.

4. Very High Radiation Area

Any area, accessible to personnel, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at 1 meter from the radiation source, or from any surface that the radiation penetrates, is designated as a very high radiation area. In addition to the above requirements for high radiation areas, the Chemistry and Radiation Protection Department shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to a very high radiation area. Access is only by permission of the Plant Manager, with controls, in addition to those for high radiation areas, which shall be specified on the Radiation Work Permit.

Any very high radiation areas identified at the plant shall be posted with signs, including the following words: GRAVE DANGER - VERY HIGH RADIATION AREA.

11.6.6.7 Facility Contamination Control

Contamination of general Plant areas is controlled by using the step-off pad technique. A double step-off pad may be employed for jobs involving high levels of contamination. Plastic and absorbent paper is used to carry contaminated tools and equipment and to facilitate rapid cleanup. Automated frisker stations and one or more portal monitors are located at access control so personnel can check themselves to assure that they are not contaminated, and thereby carry contamination to other parts of the plant. Where automated frisker stations are not installed, or out of service, monitoring at exits from Radiologically Controlled Areas with a G-M count rate meter and pancake probe or equivalent will be used. As a final check a portal monitor is normally used at the exit from the protected area in the security building (except during Site Emergency Plan activations and drills involving accountability actions).

Chemistry and Radiation Protection personnel make routine contamination surveys of all accessible areas of the Plant. Any area found contaminated to an undesirable level will be taped or roped off, signs posted and cleanup will commence as soon as practical if consistent with good radiological practices.

11.6.6.8 Personnel Contamination Control

Contamination of personnel is controlled and prevented by the use of several types of protective clothing when personnel enter contaminated areas.

1. Lab Coats - These are worn by laboratory personnel when analyzing radioactive samples, and by others when performing light work activity in, slightly contaminated areas as allowed by Chemistry and Radiation Protection personnel.

2. Cloth Coveralls - These are worn in most instances when entering contaminated areas.

3. Paper Coveralls - These are disposable and are sometimes used as an outer protector to the cloth coveralls.

4. Shoe Covers - Cloth covers are worn in most contaminated areas where dry contamination is encountered. In the case of wet contamination, either plastic shoe covers or rubbers are worn over the cloth shoe covers.

5. Gloves - Cloth and rubber gloves are worn in most contaminated areas.

6. Plastic Suits - These are worn in areas where the potential exists for liquid contamination of personnel and are generally worn over cloth coveralls.

7. Head Protection - Cloth caps can be worn in contamination areas during light work activity and under respirators, cloth hoods for high-level contamination and plastic hoods for wet contamination.

In cases where a double step-off pad is used, two sets of protective clothing are normally worn.

Normally, most of the Plant is accessible to personnel in street clothes. The bomb shelter change room just inside from access control is the main change area for the radiation controlled part of the station (areas beyond access control).

11.6.6.9 Airborne Contamination Control

Airborne contamination is minimized by keeping floor contamination low and by reducing leaks as much as possible. In the event of levels of airborne contamination approaching or exceeding levels as specified in 10 CFR 20, Appendix B, Table 1, Column 3, an evaluation is performed on the use of process or other engineering controls (e.g., close-capture and controlled-flow ventilation, containment, isolation, dilution).

In the event process or other engineering controls cannot be practically exercised, then other actions are taken to limit internal exposure, consistent with maintaining the total effective dose equivalent (TEDE) ALARA, such as limitation of working times, control of access, use of respiratory protection equipment or other controls. A protection factor is determined for each type of respiratory protection equipment in use. See Table 6-1 of Reference 8.

Continuous air monitors are used in areas of potential airborne radioactivity or air samples are taken with a portable sampler and analyzed to evaluate actual work conditions in such areas of potential airborne radioactivity. Continuous air monitors are set to alarm when the airborne radioactivity reaches the applicable derived air concentrations (DAC) established for Co-60 in 10 CFR, Part 20, Appendix B, Table 1, Column 3. In areas where continuous air monitors are not provided, samples taken with portable air samplers will provide the basis for determining concentrations to ensure appropriate control and tracking of internal exposures.

11.6.6.9.1 Respiratory Protection Program

A Respiratory Protection Program has been established by means of Plant procedures based on the requirements of 10 CFR 20. This program provides guidelines for personnel training in the proper selection and use of respiratory equipment. This program also sets forth the requirements for respirator control, inspection and maintenance.

Training

All personnel whose duties may require the use of a respirator are trained in the Respiratory Protection Program, in each of the following:

1. Recognition of the need for respiratory equipment, consistent with maintaining TEDE ALARA, including interpretation of airborne sampling results to identify airborne concentrations.

2. Restrictions on personnel entry into areas requiring respiratory protection and including frequency of entry and duration of stay for different levels of airborne concentrations.
3. Selection of the respiratory equipment which provides the most effective protection against the type and level of radioactive

airborne contaminant that may be present. A determination of the respiratory protection factors for each device.

4. Selection of respiratory equipment most suitable for the required work.

5. Preparation of the selected respiratory equipment for use.

   a. Inspection for cleanliness, damage and contamination from previous use.

   b. Instruction in the proper fitting procedure.

   c. Testing for proper fit.

The medical examination/evaluation shall be determined by a physician, or other licensed health care professional, prior to initial fit-testing for tight-fitting face pieces and prior to the first field use for loose-fitting devices, since no fit-test is required for these types. The individual must be re-evaluated every 12 months thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection devices.

Personnel having received training are required to remain current regarding changes to the Respiratory Protection Program. Formal retraining and requalification is performed for each individual on an annual basis.

Control, Inspection and Maintenance

A member of the Plant Chemistry and Radiation Protection staff has designated responsibility for the Respiratory Protection Program and maintains control of respiratory equipment when not in use:

1. To ensure the proper cleaning, handling and storage,

2. To prevent the use of contaminated, defective or faulty devices,

3. To determine if personnel requesting the equipment have been properly trained and fitted,

4. For proper equipment utilized for task and

5. For maintaining breathing air quality for self-contained and air line respiratory devices.

All respirators are inspected before and after each use. Respiratory equipment subjected to extended storage is inspected periodically.

All repairs and the steps required to maintain the respiratory devices ready for use, such as, cleaning, decontamination and filter cartridge replacement, are performed under the direction of the supervisor assigned responsibility for the Respiratory Protection Program.

11.6.6.10 External Radiation Dose Determination

All regularly assigned employees, contractors, and visitors frequenting radiation controlled areas of the plant are assigned individual dosimeters. Permanent dose records are kept on the above employees as required by 10CFR20. Primary thermoluminescent dosimeters are sent to a dosimetry analytical group for processing and interpretation. Personnel who have assigned dosimeters also may be assigned additional Plant dosimeters (normally a direct reading dosimeter or electronic dosimeter). These dosimeters are used to keep a daily total estimate of an individual's dose. Their use as a permanent record of an individual's dose will be restricted to times when thermoluminescent dosimeters are lost or damaged and when a large discrepancy exists between direct reading dosimeter/electronic dosimeter accumulations and thermoluminescent dosimeter interpretations. Assignment of permanent dose by other than primary dosimeter will be documented with an explanation.

Special high-range direct reading dosimeters, finger rings, wrist badges, neutron dosimeters, or thermoluminescent dosimeters are issued by Radiation Safety on the basis of need.

11.6.6.11 Internal Radiation Dose Determination

Internal radiation dose from the inhalation of airborne radioactive material is determined through tracking of DAC-hours and/or conduct of bioassays (including passive monitoring, whole body counting, or analysis of excreta). The effectiveness of process/engineering controls and the Respiratory Protection Program is evaluated by bioassays, or by urine analysis if tritium burdens require evaluation. At least one of these methods of uptake evaluation (DAC-hr tracking or bioassay) will be performed not less than once per year for all radiation workers who have been exposed to significant levels of airborne radioactivity or work routinely in contaminated areas of the plant (a documented passive monitoring program satisfies this requirement). Dose records of Plant radiation workers are available to their supervisors so that work can be planned accordingly and the effectiveness of the Health Physics Program can be evaluated. Bioassay results are maintained as a portion of these records in accordance with 10 CFR 20.2106.

11.6.7 RADIATION PROTECTION INSTRUMENTATION

11.6.7.1 Counting Room Instrumentation

The counting room instrumentation includes:

1. A multichanneled analyzer using a germanium crystal in a lead cave.
2. An alpha detector and scaler.
3. A liquid scintillation tritium counter.
4. A beta detector and scaler.

11.6.7.2 Portable Radiation Detecting Instrumentation

The portable radiation detecting instrumentation normally stored in the portable instrument storage area in access control includes:

1. A fast and thermal neutron count instrument.
2. High- and low-range G-M survey instruments.
3. Low-range ionization chamber instruments.
4. High-range ionization chamber instruments.
5. Extended probe high-range G-M tube.

11.6.7.3 Air Sampling Instrumentation

The portable and mobile air sampling instrumentation includes:

1. High-volume air samplers equipped with a particulate filter.
2. Low-volume air samplers that can be equipped with both particulate and halogen filters.
3. Lapel air samplers for breathing zone samples.
4. Continuous Air Monitors including alarm functions.

11.6.7.4 Personal Monitoring Instrumentation

Personal monitoring instrumentation may include one or all of the following:

1. Direct reading pocket ion chamber dosimeters with a range of 0-200mrem, 0-1.5rem, and 0-5rem.

2. Thermoluminescent dosimeters with a range of at least 10mrem to 1,000rem.

3. Electronic direct reading dosimeters with a range of at least 0-1000rem.

11.6.7.5 Emergency Instrumentation

Instruments are kept in special areas of the Plant which are accessible in the event of an emergency. These instruments are checked and calibrated at regular intervals to assure their proper functioning. These instruments include:

1. Radiation detection equipment capable of measuring field from 0.1 mrem/h-1,000 rem/h.

2. A portable G-M type instrument to be used as a check for low-level contamination.

3. A high-volume air sampler.

4. Direct reading dosimeters with a range of 0-50 rem.

11.6.8 TESTS AND INSPECTIONS

11.6.8.1 Shielding

Visual inspections of Plant shielding were made during the construction phase. Because of the shielding's massive structure, these inspections were limited to detecting major defects. With the reactor in operation, radiation surveys are made to assure that:

1. There are no defects or inadequacies in the shielding that might affect personal exposures during normal operation and maintenance of the Plant.

2. Areas of the Plant are correctly posted and barricaded as radiation and high-radiation areas.

These surveys consist both of gamma and neutron monitoring where appropriate. Continued routine radiation surveys of all areas of the Plant will assure integrity of the shielding.

A study was made of the adequacy of Plant shielding following a TMI-type accident. As a result of this study, shielding was added and emergency procedures were modified.

11.6.8.2 Area and Process Radiation Monitors

Each area and process monitor is periodically tested to determine that:

1. The calibration of the monitor ensures that control room readout instrumentation indicates true radiation levels.

2. The alarm scale trip points function properly and that the alarms function properly.

3. Equipment actions that occur upon a high radiation signal are verified.

11.6.8.3 Continuous Air Monitors

Each continuous air monitor is periodically tested to determine that:

1. The calibration of the monitor is correct and that readout in counts per minute can be converted to air contamination in $\mu Ci/cm^3$.

2. Airflow is constant.

3. Trip alarm points are set and function properly.

11.6.8.4 Radiation Protection Instrumentation

The following instrumentations are tested and calibrated periodically:

1. Counting room instrumentation.

2. Portable instrumentation.

3. Air samplers.

4. Thermoluminescent dosimeters (TLDs) are supplied by a contractor for dose recording and measurement. The contractor is certified by the National Voluntary Laboratory Accreditation Program (NVLAP) as required by 10 CFR 20. Direct reading dosimeters are supplied for daily dose tracking and are calibrated using standardized sources.

5. Emergency instruments.

11.6.9 CONTROL OF BYPRODUCT, SOURCE OR SPECIAL NUCLEAR MATERIAL (SNM) SOURCES

The control of byproduct, source or special nuclear material sources exceeding 100 millicuries is by approved Chemistry and Radiation Protection Department procedures which contain information described in Regulatory Guide 1.70.

The ability to handle sources has been demonstrated at Palisades since the Provisional Operating License was issued. Personnel qualifications, facilities, and equipment and procedures for handling have also been established. Surveillance leak testing to determine source leakage was incorporated into the Technical Specifications. Subsequently, the Technical Specifications were revised to relocate this leak testing requirement to the Offsite Dose Calculation Manual.

In the NRC safety evaluation for Amendment 98, the Staff noted they had reviewed the Palisades' personnel qualifications, facilities, equipment and procedures for handling byproduct, source and special nuclear material and found them consistent with Regulatory Guide 1.70.3 and meeting the requirements of 10 CFR Parts 30, 40 and 70. The Staff further found on the basis of the Palisades' radiation safety program, previous reviews, and information provided by NRC, Region III, that Palisades has an adequate Health Physics organization and radiation protection program, and that personnel are adequately trained to handle the sealed sources licensed for Palisades. The Staff concluded that incorporation of flexible, yet controlled licensed provisions for the receipt possession, and use of byproduct, source and special nuclear material into the Palisades Operating License is acceptable.

Some examples of Palisades' sources are:

Isotope	Quantity	Form	Use
PuBe	5 Ci	Sealed Source	Instrument Calibration
PuBe	1 Ci	Sealed Source	Instrument Calibration
Cs-137	10 Ci	Sealed Source	Instrument Calibration
Cs-137	400 Ci	Sealed Source	Instrument Calibration
Cs-137	120 mCi	Sealed Source	Instrument Calibration
Cs-137	250 mCi	Sealed Source	Instrument Calibration

The primary storage location for sources is the Calibration Facility but other controlled locations can be used as necessary for the operation of the facility (as described in Subsection 11.6.6.3).

11.6.10 RADIOACTIVE MATERIAL STORAGE FACILITIES

Storage of reusable radioactive materials is provided by buildings within the owner controlled area. These buildings (normally referred to as permanent radioactive material storage areas) are posted in accordance with 10 CFR 20. Access to these areas is positively controlled (e.g. locks, fences, stationed personnel) and movement of material to and from these buildings is overseen by Radiation Protection personnel. Designated permanent radioactive material storage areas include:

a. Auxiliary Building
b. Containment Building
c. North Storage Building
d. South Pole Barn
e. South Storage Building
f. East Radwaste Building
g. Feedwater Purity Building
h. Interim Steam Generator Storage Facility (ISGSF)
i. Independent Spent Fuel Storage Installations (ISFSIs)
j. Calibration Facility
k. East Storage Building
l. Dry Fuel Storage Building

New permanent storage areas may be added if they meet the same control requirements listed above.

Transitory radioactive material storage areas (normally referred to as temporary radioactive material storage areas) may be established as necessary based on operational needs. These areas are posted in accordance with 10 CFR 20 but do not require the control measures of permanent areas since the radiological risk is much lower. These areas are normally established inside the restricted area (protected area) using containers to prevent release of materials and exposure to the weather. If established outside the restricted area, continuous surveillance is provided for the area or the containers are designed to prevent unauthorized removal. Time periods for the use of temporary storage areas are to be as short as possible.