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11 WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 WASTE DISPOSAL SYSTEM

The system is designed to process wastes from both Units 3 and 4 and the term "plant" refers to these two nuclear units.

11.1.1 DESIGN BASES

Control of Releases of Radioactivity to the Environment

Criterion: The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment (10CFR Part50, Appendix A, Criterion 60)*. C26

The limits placed on plant radioactive effluent release by 10 CFR 20 and 10 CFR 50.67 have been considered in the design and operating plans for the plant, with the objective to maintain release concentration at the site boundary below natural background activity and thus only a minute fraction of 10 CFR 20 limits. To achieve these objectives the facility has been designed and will be operated as follows: C26

1. Liquid wastes will be collected in tanks and processed by the waste disposal demineralizers. The waste process provided can reduce activity well below established limits and represents a design for reducing activity to the lowest practicable value. Analyses of liquid prepared for release will be made to determine that activity levels have been minimized before release is permitted. The resulting activity after mixing with the

* Letter L-83-499, Amendment 103 and 97 - Radiological Effluent Technical Specifications and Radiological Environmental Monitoring", dated September 26, 1983. C26

circulating water will be near to or equal to natural background. The tritium is expected to be about 1% of MPC.

2. Gaseous wastes will be stored in decay tanks for natural decay. Gases will be released through the monitored plant vent, and at the site boundary the annual dose will be a small fraction of 10 CFR 20 limits. Cover gases in the nitrogen blanketing system will be reused to minimize the number of tanks released.

   The quantity of radioactivity contained in each gas decay tank is restricted to provide (a) assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem, and (b) assurance that the concentration of potentially explosive gas mixtures contained in the Gas Decay Tank System is maintained below the flammability limits of hydrogen and oxygen. Refer to Section 11.1.2 for the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program requirements. C30

3. Solid radioactive wastes will be packaged to minimize the number of containers shipped. Low level waste packaged for shipment may be stored on site in the Low Level Waste Storage Facility while awaiting shipment.

11.1.2 SYSTEM DESIGN AND OPERATION

The Waste Disposal System Process Flow Diagrams are shown in Figures 11.1-1 through 11.1-18 and Performance Data are given in Table 11.1-1. The Waste Disposal System is common to Units 3 and 4 with the exception of the reactor coolant drain tanks and reactor coolant drain tank pumps.

The Waste Disposal System collects and processes potentially radioactive reactor plant wastes prior to release or removal from the plant site within limitations established by applicable governmental regulations. The fluid wastes are sampled prior to release using an isotopic identification as necessary. Radiation monitors are provided to maintain surveillance over the release operation. Permanent record of Waste Disposal System releases is provided by radiochemical analysis of known quantities of waste. The system is capable of processing all wastes generated during continuous operation of the Reactor Coolant System assuming that fission products escape from one per cent of the fuel elements into the reactor coolant.

At least two valves must be opened to permit discharge of liquid or gaseous waste from the Waste Disposal System. One of these valves is normally locked closed. During release, the effluent is monitored, and the release terminated if the radioactivity level exceeds a predetermined value. Activity release limits are given in the Offsite Dose Calculation Manual in accordance with the Technical Specifications.

As secondary functions, system components supply hydrogen and nitrogen to RCS components as required during normal operation and provide facilities to transfer fluids from inside the containment to other systems outside the containment.

The waste disposal system is controlled primarily from a local control board in the auxiliary building and four local control boards in the radwaste facility with appropriate indicators and alarms. Off normal conditions are annunciated in the control room. All system equipment is located in or near the auxiliary building and in the radwaste facility except for the reactor coolant drain tank and pumps, which are located in the containment.

System Description

Liquid Processing

During normal plant operation the Waste Disposal System can process liquids from the following sources:

a) Equipment drains, floor drains, tank overflows, containment sumps, and leak-offs
b) Laboratory drains
c) Radioactive laundry and shower drains
d) Decontamination area drains
e) Resin transfer flush water
f) Refueling water from fuel transfer canal and/or reactor cavity
g) Holdup Tanks
h) Miscellaneous sources via molybdate holding tank

Additionally, each Unit's blowdown tank can be connected by a hose to the Waste Disposal System via the Waste Holdup Tank.

The system also collects and transfers liquids directly from the following sources to the Chemical and Volume Control System for processing:

a) Reactor coolant loop drains.
b) Reactor coolant pump seal leakage. C28
c) Excess letdown during startup.
d) Accumulators.
e) Valve and reactor vessel flange leakoffs.

These liquids flow to the reactor coolant drain tank and are discharged directly to the CVCS holdup tanks by the reactor coolant drain tank pumps which are operated automatically by a level controller in the tank. These pumps also return water from the refueling canal and cavity to the refueling water storage tank. There are one reactor coolant drain tank and two reactor coolant drain tank pumps inside each containment.

Waste liquids are collected by various drains and sumps. The liquid drains flow by gravity, or are pumped, to the waste hold up tank (See Figure 11.1-9). The activity level of waste liquid from the laundry area will usually be low enough to permit discharge from the site without processing. The liquid is pumped to one of the waste monitor tanks or monitor tanks where its activity can be determined for record before it is discharged through a radiation monitor. The liquid waste in the molybdate holding tank (item h, page 11.1-3) is typically pumped directly to the waste monitor tanks.

The liquids requiring cleanup before release are processed by the waste disposal demineralizer. The liquid from the waste disposal demineralizer is routed directly to one of three radwaste facility waste monitor tanks or one of two monitor tanks.

When one of the waste monitor tanks is filled, it is isolated, recirculated and sampled for analysis while one of the other two tanks is in service. If analysis confirms the activity level is suitable for discharge, the liquid is pumped through a flowmeter and a radiation monitor and then released to the circulating water system.

Otherwise, it can be returned to a waste holdup tank for reprocessing. Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by automatically closing the discharge control valve if the liquid activity level exceeds a preset value.

Gas Processing

During plant operation, gaseous wastes originate from:

a) degassing reactor coolant discharge to the Chemical and Volume Control System,
b) displacement of cover gases as liquids accumulate in various tanks,
c) miscellaneous equipment vents and relief valves, and
d) sampling operations and gas analysis for hydrogen and oxygen in cover gases.

During normal operation the Waste Disposal System supplies nitrogen from a Dewar vessel and hydrogen from a tube trailer to waste disposal components. Dual manifolds are provided, one for operation and one for backup. The system is sufficiently instrumented and alarmed to ensure continuous supply of gas.

Most of the gas received by the Waste Disposal System during normal operation is cover gas displaced from the Chemical and Volume Control System holdup tanks as they fill with liquid (see Figures 11.1-16, 11.1-17 and 11.1-18). Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. To prevent hydrogen concentration from exceeding the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or aerated liquids and the vent header itself is designed to operate at a slight positive pressure (1.0 psig minimum to 4.0 psig maximum) to prevent in-leakage. On the other hand, out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first compressor. From the compressors, gas flows to one of the gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select one tank for backup if the tank in operation becomes fully pressurized. When the tank in service becomes pressurized to approximately 100 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator of this event so that he may select a new backup tank. Pressure indicators are supplied to aid the operator in selecting the backup tank.

Gas held in the decay tanks can either be returned to the Chemical and Volume Control System holdup tanks, or discharged to the atmosphere if it has decayed sufficiently for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks in order to permit the maximum decay time before releasing to the environment.

However, the header arrangement at the tank inlet gives the operator freedom to fill, re-use or discharge gas to the environment simultaneously without restricting operation of the other tanks. During degassing of the reactor coolant prior to a refueling shutdown, it may be desirable to pump the gas purged from the volume control tank into a particular tank and isolate that tank for decay rather than re-use the gas in it. This is done by aligning the control to open the inlet valve to the desired tank and closing the outlet valve to the re-use header.

However, one of the other tanks can be opened to the re-use header at this time if desired, while still another might be discharged to atmosphere.

Before a tank can be emptied to the environment, it must be sampled and analyzed to determine the activity to be released. Once the activity has been recorded the gas can be discharged to the plant vent at a controlled rate through a radiation monitor. Samples are taken manually by opening an isolation valve from the gas decay tank discharge to the gas analyzer and collecting the gas in one of the sampling system gas sample vessels. If sampling has shown that sufficient decay has occurred, the isolation valve in the line from the tank to the gas analyzer is closed, the isolation valve in the plant vent discharge line is opened and the tank contents are released through the plant vent. During release a trip valve in the discharge line is closed automatically by loss of air flow from auxiliary building exhaust fans. In the event of a high activity level in the discharge line, the plant vent isolation valve RCV-014 will either be closed automatically (PVGM R-14 in service) or manually (RAD 6304 in service).

During operation, a gas sample is drawn from the particular gas decay tank being filled at the time, and analyzed to determine its hydrogen and oxygen content. The hydrogen analysis is for surveillance since the concentration range can vary considerably from tank to tank. Also, the capability exists for manual grab sample analysis of cover gases from tanks discharging to the waste gas vent header.

Radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments are calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The functionality and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of the ODCM is such that concentrations as low as $1X10^{-6}$ μCi/ml are measurable. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the Gas Decay Tank System. C30

Restricting the quantity of radioactivity contained in each Gas Decay Tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Amendments 282 and 276 to the Turkey Point Unit 3 and 4 operating licenses, respectively, relocated the requirements for explosive gas monitoring instrumentation, explosive gas mixtures, and the quantity of radioactivity in the Gas Decay Tanks from the Technical Specifications UFSAR Sections 12.12.4, 12.12.5, and 12.12.6, respectively, and to licensee controlled documents and established in the Technical Specifications the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program*. The Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program sets forth the requirements, methodologies and surveillances which ensure that the Gas Decay Tanks will be maintained within applicable flammability and radioactive release limits. The limits on the explosive gas concentration and radioactivity content in the Gas Decay Tanks imposed by the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program are unchanged from the limitations previously specified in the Technical Specifications. The program controls the quantity of radioactivity contained in Gas Decay Tanks and requires the gaseous radioactivity quantities to be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Postulated Radioactive Release Due to Waste Gas System Leak or Failure. The program limits the concentration of oxygen in the gas decay tank system (as measured in the inservice gas decay tank) to less than or equal to 2 percent by volume whenever the hydrogen concentration exceeds 4 percent by volume. The program requires continuous monitoring of the waste gases to verify these concentrations are within limit. Plant procedures implement the applicable requirements of the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program Plant that were previously specified in the Technical Specifications. Related procedures specify appropriate compensatory and corrective actions in the event the Gas Decay Tanks, the associated monitoring instrumentation, or portions thereof, become non-functional. Changes to these requirements are subject to the regulatory controls of 10 CFR 50.59.

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* NRC Letter dated September 11, 2018, Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Issuance of Amendments Regarding Technical Specifications Pertaining to Explosive Gas Monitoring, Gas Decay Tanks, and Standby Feedwater System (CAC Nos. MG0143 and MG0144; EPID L-2017-LLA-0272), (ML18214A125)

Solids Processing

The Waste Disposal System is designed to package all solid wastes in High Integrity Containers (HICs) for removal to disposal facilities. The HICs are designed to be placed into transfer casks for shipment off-site for disposal. The HICs are also designed to be stored in the Low Level Waste Storage Facility while awaiting shipment off-site for disposal. Refer to Figures 11.1-17, 11.1-18 and 11.1-19 for the spent resin processing flow diagrams.

The spent resins from the CVCS demineralizers are normally deposited in the spent resin storage tank. After resin in the spent resin storage tank has been agitated by bubbling nitrogen through the tank to the vent header, water is pumped through the tank at a controlled rate to sluice the slurry to the container area. There it is received in shielded containers and dewatered for disposal.

Provisions for dry bulk packaging of liquid waste system spent resins also exist. Spent resin is pumped as a water-resin slurry into a disposable container, which has connections for a dewatering line. The sluice water is removed by using a dewatering pump, which is piped to the waste hold-up tank through the floor drains.

All system components and piping can be internally decontaminated with flushing water from the primary water system. The permanently installed flushing water pipes can be isolated with manually operated valves.

Control valves and pumps handling radioactive fluids are functionally grouped together and located behind shield walls. The equipment is installed to permit easy access for maintenance work, tests, inspections, and replacement with minimum exposure to personnel.

Shielding is provided for each container as necessary to reduce the work area dose rates. The basis for all dose rate calculations is for one cycle of core operation with one percent defective fuel in each unit.

Components

Codes applying to components of the Waste Disposal System are shown in Table 11.1-2. Components summary data are shown in Table 11.1-3.

Laundry and Hot Shower Tanks

Three stainless steel tanks collect liquid wastes originating from the laundry. When a tank has been filled, its contents are pumped to one of the monitor tanks or waste monitor tanks after passing through a strainer and filter. If the radioactivity level is within permissible limits, the liquid is released to the circulating water system.

Reactor Coolant Drain Tanks

The reactor coolant drain tanks are all-welded austenitic stainless steel. There is one tank inside the containment of each of the two units. This tank serves as a drain collecting point for the Reactor Coolant System and other equipment located inside the containment.

Waste Holdup Tanks

The two waste holdup tanks can receive radioactive liquids from the Chemical and Volume Control System, floor drains, chemical drains, reactor coolant drain tanks, and laundry and hot shower tanks. The tanks are of stainless steel welded construction. The 24,300 gallons and 10,000 gallons waste hold-up tanks are located in the auxiliary building and radwaste facility, respectively. Contents of the auxiliary building tank can be transferred to the radwaste facility tank, but not vice-versa.

Spent Resin Storage Tank

The spent resin storage tank retains spent resin discharged from some of the demineralizers. Normally, the tank is filled over a long period of time, the contents are allowed to decay. A layer of water is maintained over the resin surface to prevent resin degradation due to heat generation from decaying fission products. The tank is all welded austenitic stainless steel.

Gas Decay Tanks

Six welded carbon steel tanks are provided to contain compressed waste gases (hydrogen, nitrogen, and fission gases). After a period for radioactive decay, these gases may be released at a controlled rate to the atmosphere through the plant vent. All discharges to the atmosphere will be monitored.

Compressors

Two compressors are provided for removal of gases to the gas decay tanks from all equipment that contains or can contain radioactive gases. These compressors are of the water-sealed centrifugal displacement type. The operation of the compressors is automatically controlled by the gas manifold pressure. Construction is primarily carbon steel. A mechanical seal is provided to minimize leakage of seal water. While one unit is in operation, the other serves as a standby for unusually high flows or failure of the first unit.

Waste Monitor Tanks

The contents of one of the three waste monitor tanks are analyzed for levels of radioactivity. If the activity is sufficiently low, the contents of the tanks are released to the circulating water system by one of two waste monitor pumps. Otherwise, the contents are returned to the waste holdup tanks for reprocessing. These tanks, are fabricated from stainless steel and meet the requirements of ASME Section VIII. Each tank provides the capability of storing 5,000 gallons of water.

Monitor Tanks

See description in Section 9.2.2.

Waste Disposal Demineralizer

Waste water in the waste holdup tank is processed primarily by the waste disposal demineralizer to reduce the level of activity. The liquid passes through a portable demineralization system which provides filtration and ion exchange before it is conveyed to the waste monitor tanks or monitor tanks.

Nitrogen Manifold

A dual manifold supplies nitrogen to purge the vapor space of various components to reduce the hydrogen. concentration or to replace fluid that has been removed. A large volume Dewar vessel which is maintained above a preset level, assures a continuous supply of gas. Additionally, bottled gas is provided for short-term maintenance and backup requirements.

Hydrogen Manifold

A dual manifold supplies hydrogen to the volume control tank to maintain the hydrogen partial pressure as hydrogen dissolves in the reactor coolant. A pressure controller, which is manually switched from one manifold to the other, assures a continuous supply of gas.

Gas Analyzer

Manual sampling and laboratory analysis is conducted to monitor the concentrations of oxygen and hydrogen in the cover gas of various Waste Disposal System tanks, Chemical and Volume Control System tanks and the pressurizer relief tank. Upon indication of a high oxygen level, provisions are made to purge the equipment to the gaseous waste system with an inert gas.

Continuous sampling of the gas decay tank being filled is performed by on-line equipment. A local alarm warns of a potentially explosive condition.

Pumps

Pumps used throughout the system for draining tanks and transferring liquids shown in Figures 11.1-1a and 11.1-1b are either canned motor or mechanically sealed types to minimize leakage. The wetted surfaces of all pumps are stainless steel or other materials of equivalent corrosion resistance.

Piping

In general, the permanent piping which carries liquid wastes is stainless steel. All gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

Valves

All valves exposed to gases are carbon steel. All other valves are stainless steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Stop valves are provided to isolate equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive wastes if the tanks might be overpressurized by improper operation or component malfunction. Tanks containing wastes which are normally of low radioactivity level are vented locally.

11.1.3 DESIGN EVALUATION

The following section was prepared as part of the licensing process for the plant. This section is historical and has not been updated in consideration of revisions to 10 CFR Part 20. Reference to 10CFR Part 20 refer to the pre-1990 version of 10 CFR.

<u>Liquid Releases</u>

Based on the estimated total liquid discharge to the Waste Disposal System in Table 11.1-4 and the capacity of the waste monitor and monitor tanks, the estimated number of yearly releases is 1000. This evaluation was performed for original plant licensing and is conservative with respect to actual operations.

The estimated annual liquid release is indicated in Table 11.1-5. The maximum activity discharge rate will be controlled to assure that the circulating water concentration during releases is as low as practicable below the requirements of 10CFR20.

The liquid waste processing facilities have been evaluated and demonstrated to be in compliance with 10CFR50, Appendix I requirements. This is addressed in supplementary licensing documents*.

<u>Liquid Wastes (Without Primary - Secondary Leakage)</u>

Liquid wastes are generated primarily by plant maintenance and service operations, and consequently, the quantities and activity concentrations of influents to the system, Tables 11.1-4 and 11.1-5, are estimated values. Therefore, considerable operational margin has been assigned between the estimated system load and the design capability as indicated by Table 11.1-4. A conservative estimate of activity released in the liquid phase is summarized in Table 11.1-5. This tabulation is generated as follows:

1. All liquid waste is initially at peak reactor coolant activity concentrations based on continuous full power operation with 1% defective fuel clad in each unit.
2. Allow 500 minutes for decay, the time required to process a 1000 gallon batch at 2 gallons per minute.**
3. Concentrate the waste to a bottoms activity concentration of 40 uc/cc, the packaging facility design limit.**

*Letter L-76-212, "Appendix I Evaluation" dated June 4, 1976 from R.E. Uhrig of Florida Power and Light to D. R. Muller of the USNRC.

**These values are based on original system design and operating characteristics. While changes have been made to the original system, actual releases continue to meet the guidelines of 10CFR20.

4. Divide demineralizer combined DF of at least 10^6 which yields 4×10^{-5} uc/cc in the waste condensate.

5. Multiply by the quantity released from both units, listed in Table 11.1-4, to obtain the total estimated annual release in Table 11.1-5.

6. Add to this the activity released through waste disposal by the CVCS monitor tanks. This is estimated to be less than 2 mc/yr.

7. The tritium estimate in Table 11.1-5 assumes that one percent of the tritium that is formed in the fuel (the predominant source) diffuses through the zircaloy clad and enters the reactor coolant. Tritium discharges will be evaluated and accounted for by analyzing a composite sample. All of the sources of tritium accumulating in the reactor coolant, shown in Table 9.2-6, are included in the annual release.

8. When the liquid in the waste monitor or monitor tanks has been properly determined to have an activity level low enough for discharge according to the release requirements of 10CFR20 for unidentified isotopes, the monitor tank pumps or waste monitor pumps are started and the liquid can be discharged to the seal wells of either Unit 3 or Unit 4 or both. The valves at the seal wells are electrically interlocked with the circulating water pumps to prevent liquid from being discharged into an inactive well, thus ensuring complete mixing at all times. Discharge piping is shown schematically in Figure 11.1-3.

9. A radiation monitor (described in Section 11.2.3) automatically closes the discharge from the waste monitor tank pumps or monitor tank pumps if the activity level exceeds the monitor set point. This ensures that the activity in the circulating water discharge canal will be below the release requirements of 10CFR20.

Liquid Waste With Primary - Secondary Leakage

The isotopic equilibrium activity concentration in the secondary coolant for any given radioisotope is related to the reactor coolant activity, the steam generator blowdown (cleanup) flow, the isotopic natural decay and the primary to secondary leakage flow by the following equation:

$$C_{si} = \frac{L_{ps} C_{pi} \mu c/cc}{\lambda_i V_s + F_s}$$

where:
C_{si} = Secondary coolant activity, μc/cc
C_{pi} = Primary coolant activity, μc/cc
L_{ps} = Primary to secondary leakage, gpm
F_s = Secondary blowdown flow, gpm
λ_i = Isotopic natural decay constant, min. $^{-1}$, and
V_s = Liquid volume of the secondary coolant, gal.

The relationship assumes that the reactor coolant equilibrium is independent of leakage rate. Consideration is given to I-131 as the major contributor to environmental activity release, because noble gas concentrations in the secondary will be quite low, being continuously entrained with the normal steam flow and released to the atmosphere through the air ejector.

The above relation is plotted in Figure 11.1-5 for I-131 as the primary to secondary leak rate versus the ratio of secondary to primary activity as a function of various blowdown flows.

The steam generators blowdown system, shown in Figures 10.2-41 and 10.2-42, consists of three independent blowdown lines (one per steam generator) which tie into a common blowdown flash tank. High activity liquid contained in the flash tank can be directed to the radioactive liquid waste system through manual valve alignment and a portable hose connection. The flashing component is discharged to the atmosphere or the shell side of the associated number 4 feedwater heaters. The blowdown tank liquid overflow discharge goes to the circulating water discharge canal. Upon indication of a high-radiation level, a radiation monitor provided in the header of the steam generators' blowdown sampling lines on each unit will actuate solenoid valves on the associated unit to automatically isolate the blowdown including the sampling lines, close the valve in the blowdown flash tank discharge line to the circulating water discharge canal and sound an alarm in the control room. Because the iodine preferentially remains in the liquid phase, the plant vent monitor which monitors the air ejector would be less sensitive than the liquid effluent monitor to iodine activity. C31

The steam generators blowdown flow is normally maintained at 1% or less of feedwater flow. This flow is adjusted as required to control the chemistry in the steam generators secondary side within established requirements. During startup or abnormal conditions the steam generator blowdown flow may be increased, as required, to approximately 6% of the feedwater flow.

The setpoints of the steam generator blowdown radiation monitors are selected to isolate the blowdown, as previously indicated, at an activity concentration that will limit the combined secondary coolant and radwaste releases to below 10CFR20 requirements. The alarm setpoints for these monitors are determined by and set in accordance with the methodology and parameters of the Turkey Point Offsite Dose Calculation Manual (ODCM). ODCM implementation is required by Technical Specification 6.8.

Blowdown was analyzed for radiological considerations in the original FSAR to occur routinely, typically on a daily basis over a one to several hour period at which times a flowrate of approximately 50 gpm is maintained. Assuming a permissible limit for the 624,000 gpm condenser cooling water iodine concentration at ten times the concentration limit for a one-hour blowdown, then, for several values of percent failed fuel, the allowable maximum primary to secondary leak rates can be read from Figure 11.1-5. At these limiting values, the combined secondary coolant and radwaste releases would be below 10CFR20 requirements provided blowdown did not exceed 2.5 hours per day.

In addition to the ranges of normal operating conditions with tolerable amounts of failed fuel and primary to secondary leak rates, the site boundary I-131 equivalent dose is estimated under the following assumptions:

a. Steam line break outside the containment under no load conditions,

b. Releasing the contents of one steam generator,

c. Secondary I-131 activity = Primary I-131 activity = 1.5 uCi/cc from 1% failed fuel, and

d. One tenth the iodine content in the steam generator reaches the site boundary.

The site boundary thyroid dose equals approximately 1.78 rem.

Gaseous Wastes

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged for the CVCS volume control when degassing the reactor coolant, and nitrogen from the closed gas blanketing system. The gas decay tank capacity will permit 45 days decay of waste gas before discharge. Table 11.1-6 contains an estimate of annual noble gas activity release based on the following assumptions:

For Xe-133:

1. The quantity of Xe-133 removed from the plant over a core cycle is determined assuming all gaseous waste is initially at peak reactor coolant activity concentration based on 1% defective fuel clad, and each unit at 2300 Mwt power with daily load reduction to 15% power.

2. Using the same reactor coolant activity concentrations as in (1), the total Xe-133 removed to the Waste Disposal System by degassing the Reactor Coolant System for three cold shutdowns are combined. The Cold shutdowns are assumed to occur at the following times: (a) during the second week of operation, (b) at the peak xenon level and (c) during refueling.

3. Using the same reactor coolant activity concentrations as in (1) the total Xe-133 removed from the reactor coolant to the Waste Disposal System as a result of 4 hot shutdowns occurring at equal intervals in the core cycle.

4. Sum items 1, 2 and 3 for two units to obtain the total Xe-133 removed to the Waste Disposal System and allow for 45 days decay to obtain the total estimated annual release of Xe-133.

For Kr-85:

Since there is not significant decay of Kr-85 during the operating periods involved, the total Kr-85 that enters the reactor coolant during the core cycle is assumed to be eventually released through the Waste Disposal System. In comparison to Kr-85 and Xe-133, there will be no significant activity release after 45 days of decay from the remaining gaseous wastes since the isotopes half lives are short and/or the quantities present in the reactor coolant are small.

Gaseous Release Rate

In order to illustrate the conservatism that is available for gaseous releases from Turkey Point, an estimate has been made of the maximum release rate that would conform to 10 CFR 20. Considering Xe-133 and Kr-85 as the only nuclides, Table 2, Column 1, in Appendix B of 10 CFR 20 gives effluent concentration values of 5×10^{-7} µCi/ml and 7×10^{-7} µCi/ml Respectively, applicable at the site boundary.

The average annual dilution factors for all 10 degree sectors of the site boundary are given in Figure 2D-1 and Table 2D-1, both in Appendix 2D of Section 2. For the three years of wind data taken at the site the largest dilution factor (X/Q) occurs in the 360 degree sector. The average value for the three year period, 1968-1970, is 1.02×10^{-6} sec/m^3. For purposes of calculating the allowable gaseous routine release rate limit, this value is used.

Using the above given effluent concentration value and X/Q value, the allowable average annual routine gaseous release limit is 0.49 Ci/sec, for Units 3 and 4 combined. In Table 11.1-6 the estimated release of Xe-133 and Kr-85 is 14,758 Ci/yr for Units 3 and 4 combined, and is equivalent to an average annual release rate of 0.47×10^{-3}) Ci/sec (which is much less than the 10 CFR 20 limit) using the conservative assumptions above.

The estimated annual releases are as follows:

	No.	Ci/release	Release time, hrs.
Min.	6	2460	7
Max.	20	760	2.1

The maximum release rate would be 97m Ci/sec. The site boundary effluent concentration will not be exceeded. Hold up for further natural decay of xenon for an additional month as feasible, and proportionally fewer releases per year, would about halve the total activity released.

The iodine activity release to the atmosphere from the secondary system and from the waste processing system under the limiting operating conditions of 1% failed fuel and the steam generator tube leakage (0.135 gpm) described in this section, and an expected 0.8 plant availability factor for Turkey Point Units 3 and 4 is estimated to be 259 millicuries per year. The corresponding maximum thyroid dose at the site boundary would be 0.38 millirem. The computations are based upon the data in Table 9.2-4, (un-updated FSAR), use a stripping and plateout fraction for iodine of 4 x 10^{-3}, include a 45 day gas decay tank holdup and yield an annual release from the steam system of 252 millicuries and from the waste processing system of 7 millicuries. Under normal expected operating conditions these activity releases will be less than one-one hundredth of those indicated.

The exposure of minors within the restricted area; if continuously present at the Scout Camp would be considerably below the limits established by 10 CFR 20.104, and 10 CFR 20.202 (pre-1990 10 CFR 20).

The maximum probable exposure for this on site facility would be for an individual in the Scout Camp area during a release when the wind is blowing into this sector. That exposure would be 0.007 rem, assuming:

$X/Q = 1.9 \times 10^{-4}$ sec/m^3

$\bar{E}$ = 0.205 (based on 48% contribution from Xe-133 and 52% from Kr-85 after 45 day gas storage)
S = 0.105 Ci/sec.

The provisions for monitoring iodine release paths are as follows:

1. Both the plant vent and Unit 3 spent fuel building exhaust vent have fixed filter iodine monitors.

2. The iodine release via the blowdown tanks will be calculated from the integrated flow through the blowdown flow meters and the quantity of iodine measured in the secondary side of the steam generators.

3. The iodine release from the hogging jets, main steam safety valves, and waterbox priming jets will be calculated from steam flow and the iodine measured weekly in the main steam samples. Steam flow will be calculated from time in use times maximum flow capacity of the device.

The testing and/or measurements outlined in 2 and 3 above shall only be made if iodine is detected in the secondary coolant by sampling required by the ODCM. C31

In addition, there are process radiation monitors for the plant vent, condenser air ejectors, and steam generator blowdown as described in Section 11.2 and listed in Table 11.2-7. The alarm set points are set low to alert the operator before a significant release could occur.

The gaseous waste processing facilities have been evaluated and the as-built arrangement and potential radioactive releases to the environment are demonstrated to be in compliance with 10CFR50, Appendix I requirements. This is addressed in supplementary licensing documents.*

Solid Wastes

Solid wastes can consist of spent resins, spent filters and miscellaneous materials. All solid wastes are packaged in containers for removal to a disposal facility. Low level waste packaged for shipment may be stored on-site in the Low Level Waste Storage Facility while awaiting shipment off-site to a disposal facility.

*Letter L-76-212, "Appendix I Evaluation", dated June 4, 1976 from R.E. Urhig of Florida Power and Light to D.R. Muller of the USNRC.

TABLE 11.1-1

WASTE DISPOSAL SYSTEM
PERFORMANCE DATA
(Two Units)

Plant Design Life	80 years
Normal process capacity, liquids	Table 11.1-3
Evaporator load factor	Table 11.1-4
Annual liquid discharge	
Volume	Table 11.1-4
Activity	
Tritium	Table 11.1-5
Other	Table 11.1-5
Annual gaseous discharge	
Activity	Table 11.1-6

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TABLE 11.1-2

WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS

Component	Code
Reactor Coolant Drain Tanks	ASME III, [1] Class C
Spent Resin Storage Tank	ASME III, [1] Class C
Gas Decay Tanks	ASME III, [1] Class C
Waste Holdup Tank, Auxiliary Building	No Code
Waste Holdup Tank, Radwaste Building	ASME III, [1] Class 3
Laundry and Hot Shower Tanks	No Code
Piping and Valves	USAS-B31.1 [2] Section I
Waste Gas Compressor	No Code
Waste Monitor Tanks	ASME VIII
Monitor Tanks	See Table 9.2-3
Molybdate Holding Tank	No Code
Low Level Waste Storage Facility	EPRI Guidelines & 2010 Florida Building Codes

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NOTES:

1. ASME III-American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels

2. USAS-B31.1-Code for pressure piping and special nuclear cases where applicable

TABLE 11.1-3

COMPONENT SUMMARY DATA

TANKS	Quantity	Type	Volume Each Tank	Design Pressure	Design Temperature °F	Material
Reactor Coolant Drain	1 per Unit	Horiz	350 gal.	25 psig	267	ss
Laundry & Hot Shower	3(3)	Vert	600 gal.	Atm	180	ss
Waste Holdup, Aux.Bldg.	1(3)	Horiz	3242 ft^3	Atm	150	ss
Waste Holdup, Rad. Fac.	1(3)		10000 gal	Atm	200	ss
Spent Resin Storage	1(3)	Vert	300 ft^3	100 psig	150	ss
Gas Decay	6(3)	Vert	525 ft^3	150 psig	150	cs
Waste Monitor Tank	3(3)	Vert	5000 gal	Atm	200	ss
Molybdate Holding Tank	1(3)	Horiz	3000 gal	Atm	150	cs
Monitor Tank			See TABLE 9.2-3			

PUMPS	Quantity	Type	Flow each unit gpm	Head ft.	Design Pressure psig	Design Temperature °F	Material(1)
Reactor Coolant Drain Unit 3	2 per unit	Horiz. cent.	150	175	100	267	ss
Reactor Coolant Drain Unit 4	2 per unit	Horiz. cent.	75/125	150/120	100	267	ss

NOTES:

1. Material contacting fluid.
3. Shared by Units 3 and 4.

TABLE 11.1-3

COMPONENT SUMMARY DATA

Pumps	Quantity	Type	Flow Each Unit gpm	Head Ft.	Design Press. psig	Design Temp. °F	Material(1)
Laundry	2	Horiz cent(2)	100	250	150	180	ss
Waste Evaporator Feed (Aux. Building)	1*	Horiz cent(2)	20	100	100	150	ss
Auxiliary Building Sump	14	Vert. Duplex	75	70	45	220	cs
Containment Sump	2	Vert. Duplex	75	70	45	220	cs
Radwaste Facility Sump	2	Vert.	35	70			ss
Waste Evaporator Feed	2*	Horiz cent	35/100	250/200	150	200	ss
Waste Monitor Tank	2*	Horiz cent	35/100	250/200	150	200	ss

Miscellaneous							
Waste Gas Compressors	2*	Horiz(2) cent	22(CFM)	-	-	-	-

(2) Mechanical Seal Provided
* Shared by Unit 3 and Unit 4

TABLE 11.1-4

ESTIMATED LIQUID DISCHARGE
TO WASTE DISPOSAL *

Source	Weekly Discharges Peak, During power, gal. Two units at power	During Refuel- ing, gal. One unit at power One unit refueling	Total Annual Discharge, gal. Two units
Laundry, shower, handwashes	12,200	112,850	1,593,590
Laboratories	600	600	31,200
Equipment Drains, leaks	3040	2490	154,780
Decontamination	1000	700	50,200
Totals	16,840	116,640	1,829,770
Evaporator load Factor, %	<6	<39	<12

* This table was developed as part of the original plant licensing process and is not updated.

TABLE 11.1-5

ESTIMATED LIQUID RELEASE BY ISOTOPE*
(TWO UNITS)

Isotope	Annual Release uc	Yearly Average uc/cc	Isotope	Annual Release uc	Yearly Average uc/cc
H 3**	2.90×10^{9}	1.28×10^{-6}	I 131	1.57×10^{4}	0.691×10^{-11}
Mn 542	$.16 \times 10^{0}$	0.95×10^{-15}	Te 132	1.66×10^{3}	0.731×10^{-12}
Mn 56	5.88×10^{1}	2.59×10^{-14}	I 132	4.86×10^{2}	2.14×10^{-13}
Co 58	6.58×10^{1}	2.9×10^{-14}	I 133	1.99×10^{4}	0.876×10^{-11}
Co 60	7.76×10^{0}	3.42×10^{-15}	I 134	7.52×10^{-2}	3.31×10^{-17}
Sr 89	2.67×10^{1}	1.18×10^{-14}	I 135	5.80×10^{-3}	2.55×10^{-12}
Sr 90	8.04×10^{-1}	3.54×10^{-16}	Cs 134	1.73×10^{3}	0.762×10^{-12}
Y 90	9.24×10^{-1}	4.07×10^{-16}	Cs 136	2.50×10^{2}	1.10×10^{-13}
Sr 91	6.86×10^{0}	3.02×10^{-15}	Cs 137	9.40×10^{3}	4.14×10^{-12}
Y 91	4.72×10^{1}	2.08×10^{-14}	Ba 140	6.34×10^{0}	2.79×10^{-15}
Y 92	1.08×10^{0}	0.476×10^{-15}	La 140	5.82×10^{0}	2.56×10^{-15}
Mo 99	1.96×10^{4}	0.863×10^{-11}	Ce 144	2.26×10^{1}	1.00×10^{-14}

Totals		
Tritium	2.90×10^{9} uc/yr	1.28×10^{-6} uc/cc
Other Waste Disposal	7.5×10^{4} uc/yr	3.30×10^{-11} uc/cc
Chemical and Volume Control System	2.0×10^{3} uc/yr	0.881×10^{-12} uc/cc

* These values are based on original system design and operating characteristics. While changes have been made to the original system, actual releases continue to meet the requirements of 10CFR20.

** Initial cycle.

TABLE 11.1-6

ESTIMATED ANNUAL GASEOUS RELEASE BY ISOTOPE

(TWO UNITS)

Isotope		Activity Environment Curies/yr
H 3		Negligible
Kr 85		7714
Kr 85m, 87, 88		Negligible
Xe 133		7044
Xe 133m, 135, 135m, 138		Negligible
	Total	14,758

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-1

REFER TO ENGINEERING DRAWING

5613-M-3061 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

LIQUID WASTE DISPOSAL SYSTEM
REACTOR COOLANT DRAIN TANK
AND PUMPS

FIGURE 11.1-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-2

REFER TO ENGINEERING DRAWING

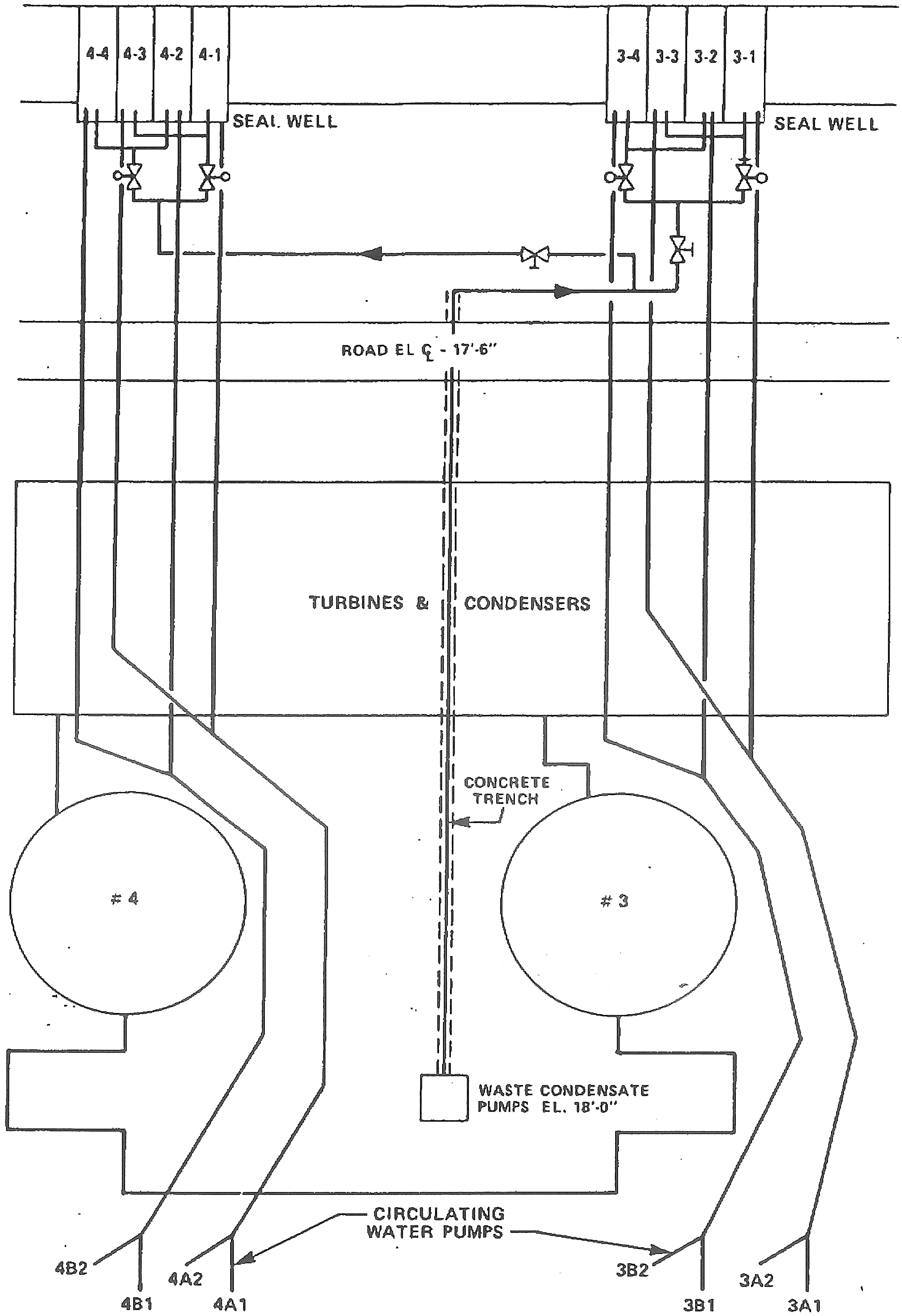
5613-M-3061 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

LIQUID WASTE DISPOSAL SYSTEM
CONTAINMENT DRAINS

FIGURE 11.1-2



REV. 9 (7/91)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3 & 4

DISPOSAL OF RADIOACTIVE LIQUIDS

FIGURE 11.1-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-4

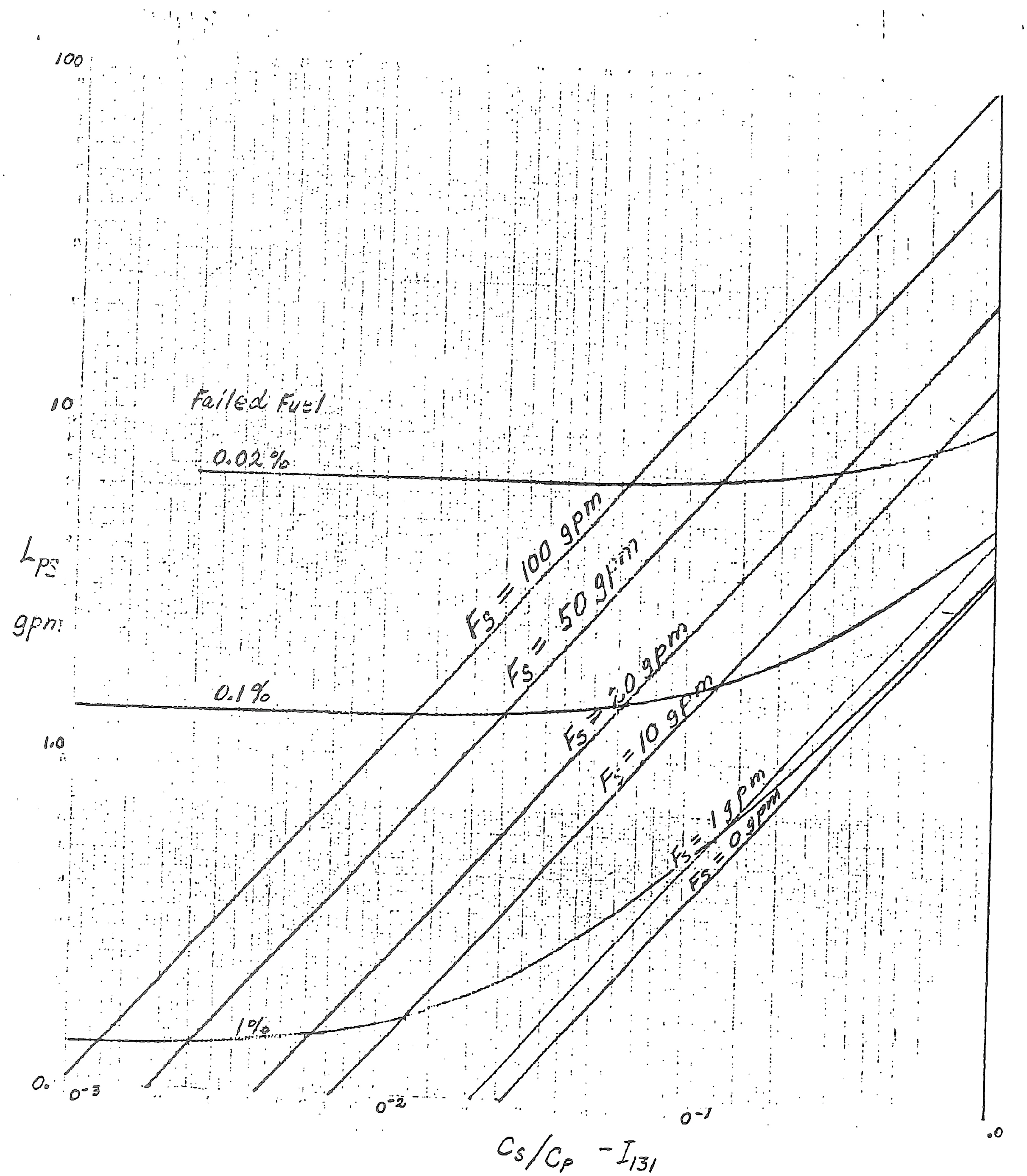
REFER TO ENGINEERING DRAWING

5614-M-3061 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

LIQUID WASTE DISPOSAL SYSTEM
REACTOR COOLANT DRAIN TANK
AND PUMPS
FIGURE 11.1-4



FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT

PRIMARY - SECONDARY ACTIVITY
RELATIONS STUDY

FIGURE 11.1-5

[Figure 11.1-6 - DELETED]

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-7

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEETS 9 & 11

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

RADWASTE SOLIDIFICATION SYSTEM
CEMENT HANDLING AND
CONTAINER FILLING
FIGURE 11.1-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-8

REFER TO ENGINEERING DRAWING

5614-M-3061 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

LIQUID WASTE DISPOSAL SYSTEM
CONTAINMENT DRAINS

FIGURE 11.1-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-9

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
WASTE HOLDUP & TRANSFER

FIGURE 11.1-9

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-10

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
LAUNDRY WASTE

FIGURE 11.1-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-11

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
DRAIN HEADERS AND SUMPS

FIGURE 11.1-11

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-12

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
POLISHING DEMINERALIZER

FIGURE 11.1-12

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-13

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 5

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
WASTE EVAPORATOR FEED

FIGURE 11.1-13

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-14

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 6

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
WASTE EVAPORATOR PACKAGE

FIGURE 11.1-14

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-15

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 7

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
LIQUID SAMPLING, MONITORING,
AND CHEMICAL ADDITION
FIGURE 11.1-15

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-16

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 8

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

LIQUID WASTE DISPOSAL SYSTEM
WASTE MONITOR TANKS

FIGURE 11.1-16

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-17

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 9

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SOLID WASTE DISPOSAL SYSTEM
SPENT RESIN STORAGE

FIGURE 11.1-17

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-18

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 10

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

SOLID WASTE DISPOSAL SYSTEM

HOLDUP AND MIXING

FIGURE 11.1-18

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-19

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 11

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SOLID WASTE DISPOSAL SYSTEM
CONTAINER FILL

FIGURE 11.1-19

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-20

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 12

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
GASEOUS WASTE DISPOSAL SYSTEM WASTE GAS COMPRESSORS **FIGURE 11.1-20**

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-21

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 13

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

GASEOUS WASTE DISPOSAL SYSTEM
WASTE GAS DECAY TANKS

FIGURE 11.1-21

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.1-22

REFER TO ENGINEERING DRAWING

5610-M-3061 , SHEET 14

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

GASEOUS WASTE DISPOSAL SYSTEM
GAS WASTE ANALYZERS

FIGURE 11.1-22

11.2 RADIATION PROTECTION

11.2.1 DESIGN BASES

Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity released to the environs of the plant have not been excessive. (1967 Proposed GDC 17)

The containment atmosphere, the plant vent, Unit 3 spent fuel pit exhaust, the condenser air ejector exhaust, the steam generator blowdown effluent, the main steam lines and the Waste Disposal System liquid effluent are monitored for radioactivity concentration during normal operations, anticipated transients, and postulated accident conditions. High radiation activity from any of these sources is indicated, recorded and alarmed in the control room.

Waste disposal system liquid effluent released to the circulating water system canal is monitored. For the case of leakage from the containment under MHA conditions, the area radiation monitoring system, supplemented by portable survey equipment provides adequate monitoring of releases. An outline of the procedures and equipment to be used in the event of a postulated accident are discussed in Section 11.2.2 and 12.3. The environmental monitoring program is described in Section 2.

Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (1967 Proposed GDC 18).

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release of gases and liquids.

The spent fuel pit cooling loop flow is monitored to assure proper operation, as shown in Section 9.3.

Ventilation systems exhaust air from the auxiliary building and radwaste facility and discharge to the atmosphere via plant vent through roughing and HEPA filters. Exhaust air from Units 3 and 4 Containments discharge to the atmosphere via the plant vent through roughing filters. Radiation monitors are in continuous service in this area and actuate a high-activity alarm on the control board annunciator as described in Section 11.2.3.

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (1967 Proposed GDC 68)

Auxiliary shielding for the Waste Disposal System and its storage components are designed to limit the dose rate to levels not exceeding 0.5 mr/hr in normally occupied areas, to levels not exceeding 2.5 mr/hr in periodically occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas. Actual doses in these areas varies with plant conditions and may exceed the design values. Dose to plant personnel is controlled administratively to maintain doses ALARA.

Gamma radiation is continuously monitored at various locations in the Auxiliary Building and fuel storage areas. A high level is alarmed locally and annunciated in the control room.

Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (1967 Proposed GDC 69)

All waste handling and storage facilities are contained and equipment is designed so that accidental releases directly to the atmosphere will not exceed the 10 CFR 50.67 guidelines; refer also to Sections 11.1.2, 14.2.2 and 14.2.3. The components of the Waste Disposal System are not subjected to any high pressures (see Table 11.1-3) or stresses. In addition, the tanks, which have a design pressure greater than atmospheric pressure, piping and valves of the system are designed to the codes given in Table 11.1.2. Hence, the probability of a rupture or failure of the system is exceedingly low. C26

11.2.2 PRIMARY AND SECONDARY SHIELDING

Design Basis

Radiation shielding is designed for operation at maximum calculated thermal power and to limit the normal operation levels at the site boundary to below those levels allowed for continuous non-occupational exposure. C26

Original design of the plant shielding was performed assuming a core power level of 2296 MWt and a 12-month fuel cycle length. The plant shielding was re-evaluated for the power uprate assuming a core thermal power of 2652 MWt and an 18-month fuel cycle. Taking into consideration the conservative analytical techniques used to establish the original shielding design and the plant Technical Specifications, which restrict the reactor coolant activity to levels significantly less than 1% fuel defects, it is concluded that the increase in the core power level and in the fuel cycle length will have no significant impact on plant shielding adequacy and safe plant operation. C26

In addition, the shielding and containment measures provided ensure that in the event of the maximum hypothetical accident, the evaluated off site and control room operator dose results will remain below the applicable limits in 10CFR 50.67 and Regulatory Guide 1.183. C26

Operating personnel are protected by adequate shielding, monitoring, and operating procedures. Each area is classified according to the dose rate allowable in the area. The allowable dose rate is based on the expected frequency and duration of occupancy. All areas capable of personnel occupancy are classified as one of five zones of radiation level as shown in Fig. 11.2-1 and 11.2-2. The classification of occupancy of the zones is listed in Table 11.2-1. Typical Zone I areas are the offices, control room, the turbine area and turbine service areas. Zone II areas include the passageways and local control spaces in the Auxiliary Building and the operating floor of the containment during reactor shutdown. Areas designated Zone III include the sample rooms, valve room, fuel handling areas, and intermittently occupied work areas.

Typical Zone IV areas are the shielded equipment compartments in the Auxiliary Building and the reactor coolant loop compartments after shutdown. Zone V areas are high radiation controlled areas. These areas are those containing high radiation components such as gas decay tank, mixed bed demineralizers, spent resin tank, waste container storage area, and volume control tank.

All radiation areas are appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

The unit shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel transfer shielding, and the auxiliary shielding.

Primary Shield

The primary shield is designed to:

1. Reduce the neutron flux incident on the reactor vessel to limit the radiation induced increase in transition temperature.

2. Attenuate the neutron flux sufficiently to prevent excessive activation of components.

3. Limit the gamma flux in the reactor vessel and the primary concrete shield to avoid excessive temperature gradients or dehydration of the primary concrete shield.

4. Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after shutdown.

5. Reduce the induced secondary radiation leakage to obtain optimum division of the shielding between the primary and secondary shields.

Secondary Shield

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen - 16 activity, which is produced by neutron activation of oxygen during passage of the coolant through the core.

Accident Shield

The accident shield ensures safe radiation levels for desired component access outside the containment following a maximum hypothetical accident.

Fuel Handling Shield

The fuel handling shield permits the safe removal and transfer of spent fuel assemblies and Rod Control Cluster Assemblies (RCCAs) from the reactor vessel to the spent fuel pit. It is designed to attenuate radiation from spent fuel, RCCAs and reactor vessel internals to less than 2.5 mr/hr at the refueling cavity water surface and to less than 15 mr/hr within the spent fuel area.

Auxiliary Shielding

The function of the shielding is to protect personnel working near various system components in the Chemical and Volume Control System, the Residual Heat Removal System, the Waste Disposal System and the Sampling System. The shielding provided for the auxiliary building is designed to limit the dose rate to less than 0.5 mr/hr in normally occupied areas, and below 2.5 mr/hr in periodically occupied areas.

Shielding Design

Primary Shield

The primary shield consists of the core baffle, water annuli, barrel-thermal shield, all of which are within the reactor vessel, the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced concrete structure extending from the base of the containment to an elevation of 58'-0". The lower portion of the shield has a minimum thickness of 7.0 feet of regular concrete (ρ=2.3 g/cc) and is an integral part of the main structural concrete support of the reactor vessel; it extends upward to the refueling floor, with vertical walls 4 and 5 feet thick, to form an integral portion of the refueling cavity.

The primary shield neutron flux are listed in Table 11.2-2. The flux listed are those occurring on the horizontal mid-plane of the core.

At locations other than the horizontal midplane of the core the intensity of both the neutron and the gamma flux begins to decrease.

Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of interior walls in the containment, the operating floor at elevation 58'-0" and the floor at elevation 30'-6".

Certain interior walls within the containment also serve as the accident shield.

The main portion of the secondary shield consists of 2' to 3'6" walls, which surround the coolant loops and steam generators at the 30'-6" and 58'-0" elevations. The secondary shield will attenuate the radiation levels in the reactor coolant loop compartment from a value of 25 rem/hr. to a level of less than 1 mrem/hr outside the containment.

The original secondary shield design parameters are listed in Table 11.2-3. With the 1995 thermal uprate and the 2012 Extended Power Uprate (EPU), core power has increased beyond the original design basis for the secondary shield. However, survey history over the plant's operational lifetime shows that the original design remains adequate to limit the dose rate outside the containment building to well within the 1 mrem/hr limit established above, when uprate scaling factors are applied. C26

Accident Shield

The accident shield consists mainly of the containment structure. The containment structure is a reinforced post-tensioned concrete cylinder 3 ft. 9 in. thick capped by a reinforced and post-tensioned concrete dome 3 ft. 3 in. thick.

Shielding has been provided within the containment in excess of that required for operational reasons to limit the post accident dose which otherwise might be present within the containment at penetration areas.

Additional shielding has also been provided within the Auxiliary Building to permit post accident access to the RHR system area.

The control room is shielded so that the post accident integrated dose from direct radiological shine to personnel in that room will be less than 2 Rem. The impacts of radiological shine on post-accident whole body gamma dose to control room personnel were accounted for in the analyses to address NUREG-0578, Item 2.1.6.b and NUREG-0737, Items II.B.2.2 and III.D.3.4. Based on the results of these analyses, a 1-1/2 inch steel shadow shield was installed between the Unit 3 54-inch purge valve and north control room wall via PC/M 80-63. The NRC commitments related to this modification are contained in References 4 and 5. The total control room operator shine dose consequence results for EPU conditions in Table 11.2-12 and Section 14.3.5.1 include the effects of this shadow shield. C26

Fuel Handling Shield

The refueling cavity is irregularly shaped, formed by the upper portions of the primary shield concrete, and other sidewalls of varying thicknesses. A portion of the cavity is used for storing the upper and lower internals packages. The walls vary in thickness, from 4 to 5 ft.

The refueling cavity, flooded with borated water to elevation 56'-10" during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 23 ft. above the reactor vessel flange. This height ensures that a minimum of 9 ft. of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate from only the active fuel is less than 12 mrem/hr at the water surface.

The spent fuel assemblies and RCC assemblies are remotely removed from the containment through the horizontal spent fuel transfer tube to be placed in the spent fuel pit. Concrete, 3' to 4'6" thick, shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during a time a spent fuel assembly is passing through the main concrete support of the containment and the transfer tube.

Radial shielding as the spent fuel is raised for transfer to the spent fuel storage pit is provided by the water and concrete walls of the fuel transfer canal. Actual dose rates in the area adjacent to the spent fuel storage pit may exceed design values during spent fuel transfer. Administrative procedures ensure that dose to personnel is maintained ALARA.

Fuel is stored in the spent fuel pit of the Auxiliary Building which is located adjacent to the containment. Shielding for the spent fuel storage pit is provided by 5' 6" thick concrete walls to elevation 32' 10"; above this elevation the walls are tapered in places to a thickness of 3 ft. The pit is flooded to a level such that the water height is 23 feet above the stored spent assemblies. During spent fuel handling a minimum of 7 feet 11 inches is maintained above the top of a fuel assembly.

Radiation from spent fuel has increased since the original shielding design due to the 1995 and 2012 power uprates and the transition to an 18 month fuel cycle. However, survey history over plant's lifetime demonstrates that the original shielding design for fuel handling continues to provide adequate protection for operators in these areas when uprate scaling factors are applied, within the limits established above. C26

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which contain reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Access to the Auxiliary Building is allowed during reactor operation. Equipment is shielded so that compartments may be entered without having to shut down,or to decontaminate equipment in an adjacent room.

The shield material provided throughout the Auxiliary Building is regular concrete (p=2.3 g/cc). The principal auxiliary shielding provided is tabulated in Table 11.2-6.

In addition, in some cases the installation of temporary or permanent lead shielding may be necessary to reduce area dose due to localized hot spots.

11.2.3 RADIATION MONITORING SYSTEM (RMS) C29

The Radiation Monitoring System is designed to perform two basic functions:

a. Warn of any radiation health violation which might develop.
b. Give early warnings of a malfunction which might lead to an unsafe health condition or unit damage.

Instruments are located at selected points in and around the unit to detect, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the control room. The Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff to provide adequate information and warning for the continued safe operation of the units and assurance that personnel exposure does not exceed 10 CFR 20 guidelines.

C29

The Radiation Monitoring System is divided into the following sub systems:

a. Process Radiation Monitoring System (PRMS) C29
Monitors various fluid streams in operating systems.

b. Area Radiation Monitoring System (ARMS) C29
Monitors radiation levels at various locations within the operating area of the two units.

c. Environmental Radiation Monitoring System
Monitors radiation exposure in the area surrounding the units.

d. Containment High Range Radiation Monitoring System (CHRRMS)
Monitors the radiation inside containment during post-accident conditions. C29

Process Radiation Monitoring System

This system consists of channels which monitor radiation levels in various operating systems. The output from most channel detectors is transmitted to the Radiation Monitoring System cabinets located in the control room area where the radiation level is indicated by a numerical display and recorded by a multipoint recorder. High radiation level alarms are annunciated in the control room and indicated on the Radiation Monitoring System cabinets. C26

Each channel (except the R-*-15 Steam Jet Air Ejection Monitor Channels) contains a completely integrated modular assembly, which may include the following: C26

a) Level Amplifier

Amplifies the energy of the radiation pulse to provide a discriminated output to the log level amplifier.

b) Log Level Amplifier

Accepts the shaped pulse of the level amplifier output, performs a log integration, (converts total pulse rate to a logarithmic analog signal) and amplifies the resulting output for suitable indication and recording,

c) Power Supplies

Power supplies are contained in each drawer and/or monitoring skid for furnishing the positive and negative voltages for the transistor circuits, relays and alarm lights and for providing the high voltage for the detector.

d) Test-Calibration Circuitry

These circuits provide a precalibrated signal to perform channel test, and a solenoid operated radiation check source to verify the channel's operations. An annunciator light on the control board indicates when the channel is in the test calibrate mode. In lieu of a check source, RD-3-20 and RM-3-20 utilize an internally generated test function to ensure the rate meter is functioning properly. C26

e) Radiation Level Numerical Display

This display, mounted on the drawer, indicates in counts per minute on a digital display (R-14,R-*-17A/B, R-18, and R-*-19) or logarithmically C26 in mR/hr on an analog display (R-4-20). The display is in µCi/cc from 10^{-11} to 10^{-5} for R-*-11 and 10^{-6} to 10^{-1} for R-*-12. The level signal is also recorded. The display for RD-3-20 and RM-3-20 indicate in mR/hr C26 on a digital display.

f) Indicating Lights

These lights indicate high-radiation alarm levels and circuit failure. An annunciator on the control board is actuated on high radiation. R-3/4-20 also annunciate on the control board when a channel failure occurs. C26

g) Bistable Circuits

Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level over the range of the instruments), and one to alarm on loss of signal (circuit failure).

h) A remotely operated long half-life radiation check source is furnished in each channel except R-3-20. The energy emission ranges are similar C26 to the radiation energy spectra being monitored. The source strength is sufficient to cause a definite display increase above background.

R-3-20 utilizes an internally generated test function which is used to compare to a baseline generated for the monitor. This comparison ensures that no degradation of the signal occurs without notification C26 to the operator. The operator utilizes the test function in a manner similar to the check source and it performs a similar function.

C26

The R-3-15 and R-4-15 channels are modular assemblies with the following elements:

a) Radiation Detector

A plastic Beta scintillation detector generates current pulses when exposed to radiation in the Steam Jet Air Ejector effluent.

b) Local Processing & Display Unit (LPDU)

The LPDU is located in the Load Center Room, Turbine Building Elev. 31'. It provides high voltage supply to the detector and processes the current pulses generated by the detector. The following elements are integral to the LPDU:

- Preamplifier and amplifier circuits necessary to process the detector pulses
- 1024 multi-channel analyzer (MCA) necessary to determine counting in a specific range of energies
- Software algorithms for using the detector counting to compute other measurements (e.g., volume activity, leakrate, etc.)
- AC/DC power supply for the LPDU electronics and the detector
- Display of process measurements as well as indication of alarms and faults via lights and buzzer.

C26

c) Remote Display Unit (RDU)

The RDU is located in the Control Room. It provides a numerical and graphical display of the primary channel measurements computed by the LPDU, including count rate in CPM, volume activity in μCi/cc and/or leak rate in GPD. The following elements are integral to the RDU:

- Status LEDs and associated relays to indicate the following operating conditions:
  Operate/fault
  Test
  Alert alarm
  High alarm
  High/High alarm

- Programmable logic circuit which drives the automated check source sequence (see below)
- Internal power supply for the electronics

d) Data Logger

The data logger records the RDU analog output during a check source test. The analog output is associated with detector counting in CPM. The data logger is configured to convert the analog signal to count a rate value and store the measurements to non-volatile memory which can be downloaded and reviewed to track detector performance.

e) Check Source Circuit

Provides an automated test of channel operability using a solenoid activated radioactive source. The source sequence is activated using the RDU keypad. During the sequence, radiation alarms are disabled, and the "Test" LED on the RDU is illuminated to indicate the channel is under testing. The radiation measurements during the test are archived by a data logger adjacent to the RDU so that the detector performance can be tracked over time.

f) Booster Relay Plate

The booster relay plate contains five DPBT relays corresponding to the status relays of the RDU:

- Operate
- Test
- Alert alarm
- High alarm
- High/High alarm

C26

The relays are used to drive annunciators corresponding to alarm states. The "operate" relay can be used to drive an annunciator if the channel experiences a fault. The relay plate contains an integral DC power supply which drives the coils of the booster relays and the data logger.

g) Field Junction Box

A field junction box houses a booster relay which drives the solenoid operated check source along with a key-switch for local or remote actuation of the check source. A terminal strip in the junction box permits user interface with the input/output signals of the LPDU:

- Relay contacts
- Analog input
- Analog output
- Serial links

The Process Radiation Monitoring system consists of the following radiation monitoring channels:

Containment High Range Radiation Monitoring System (CHRRMS) Monitors (RaD-3-6311A & B, RaD-4-6311A & B) C29

Two high range radiation monitors and associated instrument channels are provided for in-containment post-accident monitoring in compliance with NUREG-0737, Item II.F.1. These monitors are shown on Figures 11.2-1 and 11.2-2.

Each channel monitors the containment radiation levels from 10^0 to 10^8 R/hr inside the containment. Each detector is a gamma ionization chamber installed inside the containment. The signal processor, which supplies indication and recording in the control room, the high range radiation module and recorder are located outside the containment. Two alarms are provided to alert the control room operator upon high radiation within the containment.

Also a failure trip is provided to activate upon loss of power, high voltage, or signal from the detector. A sustaining signal is generated within the detector corresponding to 1 R/hr. A failure alarm will occur if the signal from the detector falls below this value. This feature assures knowledge of the monitor's integrity at all times.

The safety-related redundant monitoring instrumentation channels are energized from independent Class 1E power sources, and are physically separated in accordance with Regulatory Guide 1.75.

Containment Air Particulate Monitors (R3-11 & R4-11)

R3-11 and R4-11 are provided to measure air particulate beta radioactivity in each containment and to ensure that the release rate through each containment vent during purging is maintained below specified limits. Each monitor has a measuring range of at least 10^{-9} to 10^{-6} μCi/cc. High radiation level for the channel initiates closure of the containment purge supply and exhaust duct valves and containment instrument air bleed valves, and initiates control room ventilation isolation. The alarm setpoints for these monitors are determined from Technical Specifications (Table 3.3-3) and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

The sample is drawn from the containment ductwork through a closed, sealed system monitored by a beta scintillation counter - filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size on its constantly moving surface, and is viewed by a photomultiplier-scintillation crystal combination. The samples are returned to the containment after it passes through a series connected (CH R3-12) or (CH R4-12) gas monitor.

Each detector assembly is in a completely enclosed housing. The detector is a hermetically-sealed photomultiplier tube - scintillation crystal combination. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity.

A backup containment air sampling system consisting of tubing, two isolation valves, a flow indicator, quick connects, conduit and a 120 volt receptacle provides easily accessible connections and a secure mounting location for a portable sampling pump to take "grab" samples. This system provides an alternate means to sample the containment atmosphere in the event that either RD-11 or RD-12 malfunctions.

Containment air sampling to support personnel entry at power can be performed with R-11 and R-12, or via the backup containment air sampling system described above.

The filter paper mechanism, an electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

To reduce moisture in the Containment Air Particulate supply line and Containment Radioactive Gas Monitor return line, R-11 and R-12 lines are heat traced. Heat tracing on the R-11 and R-12 lines is not required to be operable and does not adversely affect the function of the Containment Air Particulate Monitors (R-3-11 and R-4-11) or the Containment Radioactive Gas Monitors (R-3-12 and R-4-12). C28

Containment Radioactive Gas Monitors (R3-12) & (R4-12)

Each monitor is provided to measure gaseous beta radioactivity in the respective containment and, to ensure that the radiation release rate during purging is maintained below specified limits. High gas radiation level initiates closure of the containment purge supply and exhaust duct valves and containment instrument air bleed valves, and initiates control room ventilation isolation. C26

Each monitor has a measuring range of at least 10^{-6} to 10^{-3} µCi/cc. The alarm setpoints for these monitors are determined from Technical Specifications (Table 3.3-3) and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

The detector skid draws a continuous air sample from the containment atmosphere. After it passes through the air particulate monitor (R-*-11), it is drawn into the gaseous beta detector through a closed, sealed system. The sample is constantly mixed in the fixed, shielded volume, where it is viewed by the beta scintillation photomultiplier detector. The sample is then returned to the containment.

The detector assembly is in a completely enclosed housing containing a beta scintillating detector mounted in a constant gas volume container. Lead shielding is provided to reduce the background level to a point where it does not interfere with the detector's sensitivity. A locally mounted electronic assembly transmits the signals to the remote indication, alarm, and control circuits.

The containment air particulate and radioactive gas monitors have assemblies that are common to both channels. They are described as follows:

a) The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detector.

b) The pump consists of:
   1. A pump to obtain the air sample.
   2. A flowmeter to indicate the flow rate.
   3. A flowmeter to indicate the flow adjustment.
   4. A flow alarm assembly to provide low and high flow alarm signals.

c) Selector valves are used to direct the sample to the detector for monitoring and to block normal flow when the channel is in maintenance or "purging" condition.

d) A temperature sensor and pressure sensor are used to protect the system from high sample stream temperatures and pressures. This unit automatically closes the sample inlet and outlet valves upon a high temperature and/or pressure condition.

e) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample. This facilitates detector calibration by establishing the background level and aids in verifying sample activity level.

f) The control and indicating assembly in the control room provides remote access to radiation monitoring functions. This assembly provides monitoring functions, readout display of monitored data, and status alarm indication for each channel.

g) The electronic mass flow measurement system is calibrated from 0 to 5 standard cubic feet per minute. A local sight flow gauge is provided for reference only, and must be compensated for actual flow rate.

Alarm lights are actuated by the following:

a. Flow transmitter (low and high flow).
b. The pressure/temperature sensors (high pressure/high temperature).
c. The filter paper sensor (paper drive malfunction).
d. Failure of any microprocessor controlled self test

On both units, one common alarm light is turned off, an annunciator is actuated, and supplemental information is available to the control room operator.

<u>Plant Vent Gas Monitors (R-14 and RaD 6304)</u>

The plant vent gas monitors detect radiation passing through the plant vent to the atmosphere. Each detector consists of a thin-walled, self-quenching type Geiger-Mueller tube (high sensitivity beta-gamma detector) operated in parallel with an impedance matching network.

Monitor R-14 has a maximum sensitivity of 5 x 10^{-7} µCi/cc. The alarm setpoint for this monitor is determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

Remote indication and annunciation of R-14 is provided on the Waste Disposal System control board in the Control Room. On high radiation level alarm the gas release valve in the Waste Disposal System is automatically closed.

Monitor RaD 6304 covers a range from 10^{-7} to 10^{5} µCi/cc for Xe-133. It transmits a pulse signal to the control console in the computer room. High radiation, intermediate radiation and rate of rise alarms are provided. RaD-6304 also functions to collect halogens and particulates on filter elements for later analysis in compliance with NUREG-0737, Item II.F.1.2, "Sampling and Analysis of Plant Effluents", and Regulatory Guide 1.97.

Condenser Air Ejector Monitors (R3-15, R4-15)

Each channel monitors the discharge from the air ejector exhaust header of the condenser for gaseous radiation which is indicative of a primary to secondary system leak.

R-*-15 use a single inline beta scintillator. All detectors monitor a fixed volume sufficiently shielded to prevent background radiation from reducing maximum sensitivity. R-*-15 has a range of 1.0E-07 to 1.0E-01 µCi/cc for Kr-85. The alarm setpoints for these monitors are determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8

Gaseous radioactive effluent releases via the steam jet air ejectors on the main condensers are monitored for iodine, particulate, and noble gas activity by RaD-6304 plant vent monitor. The ODCM requires the gaseous effluent from the steam jet air ejector to be continuously sampled and analyzed weekly for radioactive iodine and particulates during plant operating Modes 1-4, when primary-to-secondary leakage is detected. The ODCM requires the steam jet air ejector to be continuously monitored for noble gas activity releases during Modes 1-4.

Component Cooling Liquid Monitors (R3-17A, R3-17B, R4-17A & R4-17B)

Each channel continuously monitors the component cooling loop of the Auxiliary Coolant System for radiation indicative of a leak of reactor coolant from the Reactor Coolant System and/or the residual heat removal loop in the Auxiliary Coolant System. A scintillation counter is located in an inline well. A high-radiation level alarm signal initiates closure of the valve located in the component cooling head tank vent line to prevent radioactive gas release.

The measuring range of each monitor is 10^{-5} to 10^{-2} μCi/cc. The alarm setpoints for these monitors are determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

Waste Disposal System Liquid Effluent Monitor (R-18)

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. A scintillation counted and holdup tank assembly monitors these effluent discharges. Remote indication and annunciation are provided on the Waste Disposal System control board.

The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cc. The alarm setpoint for this monitor is determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

Steam Generator Liquid Sample Monitors (R3-19 & R4-19)

Each channel monitors the liquid phase of the secondary side of the steam generators for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air removal gas monitor. Samples from the bottom of each of the steam generators are mixed in a common header and the common sample is monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, blowdown is automatically isolated. Each steam generator is sampled in order to determine the source of the activity. This sampling sequence is achieved by manually obtaining steam generator liquid samples at the primary sample sink for laboratory analysis after allotting sufficient time for sample equilibrium to be established.

A high-radiation level signal will close the isolation valves in the sample lines, the discharge from the blowdown tank to the circulating water discharge (environment), and the blowdown recovery flow control valves.

The measuring range of each monitor is 10^{-5} to 10^{-2} μCi/cc. The set point is selected to transfer the blowdown as noted above, at an activity concentration equivalent to no more than 6.1 x 10^{-8} μCi/cc in the circulating water. The alarm setpoints for these monitors are determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8.

In channels R-18, and R-19, a photomultiplier tube-scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used for liquid effluent radiation actuation. Lead shielding is provided to reduce the background level so it does not interfere with detector's sensitivity. The in-line, fixed volume container is an integral part of the detector unit.

Main Steam Line Monitor (RAD-6426)

The Main Steam Line High-Range Noble Gas Effluent Monitor (RAD-6426) was installed at Turkey point as a result of actions required following the accident at TMI. RAD-6426 is used in post-accident monitoring as required to meet the requirements of Regulatory Guide 1.97, Revision 3. Monitor RAD-6426 is identified a Type E (Effluent Release Monitoring), Category 2 Variable (instrumentation designated for indicating system operating status).

The function of RAD-6426 is to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident, and to provide the plant operator and emergency planning agencies with information on plant releases of noble gases. RAD-6426 is not included in the current Probabilistic Risk Assessment (PRA) and is not a Maintenance Rule risk-significant component.

The Main Steam Line Monitor design uses two Geiger-Muller detectors within one assembly with overlapping ranges placed adjacent to each steam line, upstream of the Atmospheric Dump Valves and Main Steam Safety Valves, to detect high-energy gammas that penetrate the pipe wall. Each detector assembly is shielded in order to protect the detectors from background radiation. The detector assembly response to the high-energy gammas is then analytically correlated to the total noble gas volumetric activity in the steam line. Each detector assembly has a range from 10^{-1} to 10^{3} μCi/cc to meet R.G 1.97 requirements. The output from the Main Steam Line Monitors does not go to the Radiation Monitoring Cabinets in the Control Room, but is an input to the Distributed Control System (DCS), which provides the monitor information to displays (ERDADS) in the Control Room, Technical Support Center, and Emergency Offsite Facility. C27

As a Category 2, Type E instrument, RAD-6426 does not meet any of the 10CFR 50.36(c)(2)(ii) screening criteria for inclusion in the Technical Specifications Post Accident Monitoring Table.

As a result, a License Amendment (Reference 6) was approved to relocate the Main Steam Line Monitor Limiting Conditions for Operation and Surveillance Requirements from the Technical Specifications to the UFSAR and related procedures.

The functionality of the monitor is determined by performance of procedures for channel checks, functional testing and channel calibration on a frequency equivalent to the previous Technical Specification Surveillance Requirements. Specifically, a channel check is required on a monthly basis, and a channel calibration is required on a refueling basis. Performance of these surveillances is governed by plant procedures, in conjunction with the preventative maintenance program. The related procedures contain instructions for notifications and compensatory actions during the times that the monitor is not functional. The monitor is required to be functional in Modes 1, 2 and 3. C26

Reactor Coolant Letdown Line Activity Monitors (R3-20 & R4-20)

One channel for each unit is provided for detection of fuel clad failure which consists of a fixed position gamma sensitive GM detector for RD-4-20, local indication and signal transmission to a radiation monitoring rack in the control room, where it is indicated and alarmed on high activity level. RD-3-20 utilizes a gamma sensitive ion chamber detector. A remotely operated check source is included for R-4-20. The detector is located on the CVCS reactor coolant letdown outside the Containment Building where background radiation is relatively low and the flow transit time from the core is greater than 40 second to permit 7.2 second N-16 activity to decay to an acceptable level. A channel alarm induced by a rapid rise in coolant activity signals the requirement to take and count a coolant sample. The alarm setpoints for these monitors are determined by and set in accordance with the methodology and parameters of the Turkey Point ODCM. ODCM implementation is required by Technical Specification 6.8. For R-3-20 an internally generated test function is utilized as described previously. C26 C26 C26

Spent Fuel Pool Vent Monitor - Unit 3 (RaD-3-6418)

The Spent Fuel Pool Vent Monitor detects radiation passing through the Unit 3 spent fuel pool vent to atmosphere. A beta-gamma sensitive Geiger-Mueller tube is used to monitor the gaseous radiation level. Monitor RaD-3-6418 covers a range from 10^{-7} to 10^{5} microcuries per cc for Xe-133. Indication and alarms are provided on the console in the cable spreading room. RaD-3-6418 also functions to collect halogens and particulates on filter elements for later analysis in compliance with NUREG-0737, Item II.F.1.2, "Sampling and Analysis of Plant Effluents", and Regulatory Guide 1.97.

Area Radiation Monitoring System (ARMS)

C29

This system consists of channels which monitor radiation levels in various areas. These areas are as follows:

Detector Tag No.	Channel No.	Area Monitor
RD-3-1401	1	Unit 3 Cntmt Personnel Access Hatch
RD-3-1402	2	Unit 3 Cntmt Refueling Floor El. 58'
RD-3-1403	3	Unit 3 Cntmt Incore Instr. Equip.
RD-4-1404	4	Unit 4 Cntmt Personnel Access Hatch
RD-4-1405	5	Unit 4 Cntmt Refueling Floor El. 58'
RD-4-1406	6	Unit 4 Cntmt Incore Instr. Equip.
RD-3-1407	7	Unit 3 Spent Fuel Pit Transfer Canal
RD-4-1408	8	Unit 4 Spent Fuel Pit Transfer Canal
RD-1409	9	Aux. Bldg. Laundry Tank and Pump Room
RD-1410	10	Aux. Bldg. Chemical Storage Area
RD-4-1411	11	Unit 4 Cask Handling Facility
RD-3-1412	12	Unit 3 Cask Handling Facility
RD-3-1413	13	Aux. Bldg. Outside Unit 3 Sample Room
RD-4-1414	14	Aux. Bldg. Outside Unit 4 Sample Room
RD-3-1415	15	Aux. Bldg. North End of N/S Corridor
RD-4-1416	16	Aux. Bldg. South End of N/S Corridor
RD-1417	17	Aux. Bldg. East End of E/W Corridor
RD-1418	18	Aux. Bldg. West End of E/W Corridor
RD-3-1419	19	Unit 3 Spent Fuel Pit Exhaust
RD-1420	20	Control Room
RD-3-1421	21	Unit 3 Spent Fuel Building North Wall
RD-4-1422	22	Unit 4 Spent Fuel Building South Wall
RD-3-1423	23	Unit 3 New Fuel Building
RD-4-1424	24	Unit 4 New Fuel Building

System Description

Each of the channels is identical, and each channel is comprised of a detector, preamplifier, local indicator and a remote cabinet mounted indicator in the Control Room.

Channels 21 - 24 provide accidental criticality monitoring in accordance with 10 CFR 50.68(b) (Reference 1). Upon implementation of the license amendment for the installation of the spent fuel pool cask racks (References 2 and 3), the spent fuel pool licensing basis was changed to 10 CFR 50.68(b) from compliance with 10 CFR 70.24. With respect to radiation monitoring, 10 CFR 50.68(b) states that "radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions."

<u>The Detector</u>

This is composed of a matched ion chamber and preamplifier pair. Calibration constants determined by the manufacturer are used by the preamplifier assembly to optimize the combined detector and preamplifier response curve. The calibration constants for a channel are entered at the remote indicator in the Control Room. The ion chamber and the preamplifier are mounted separately. The preamplifier converts current from the ion chamber into an analog logarithmic DC output to drive the local meter. The preamplifier also converts the ion chamber current into a digital signal which is transmitted to the remote indicator. Any failure of the preamplifier will activate an alarm at the channel indicator in the Control Room. High voltage for the ion chamber is developed and controlled in the preamplifier assembly.

The ARMS detectors inside containment are only rated for an external air pressure of 15 psig. Therefore, the ARMS is not rated for post-accident conditions. Post-Accident monitoring is done by the CHRRMS instead. C29

<u>The Local Indicator</u>

The DC output from the detector passes through this indicator which is located in a separate box from the preamplifier. Radiation levels are indicated on a logarithmic scale which is calibrated from 10^{-1} to 10^{7} mR/hr. High radiation levels actuate a horn and a red flashing light locally. The lowest decade of the meter scale is corrected for "live zero".

The Remote Indicator

This is a cabinet mounted module which accepts the signal from the preamplifier via a digital highway. The signal is processed to provide two visual displays, one current output, and an alarm relay output. Other outputs are available for future modifications. Visual displays are a five digit display of the radiation value and a multi-color bargraph indicator which covers the range of 10^{-1} to 10^{7} mR/hr. The bargraph has three LED segments per decade. The bargraph will change color in the event of an alarm condition. Front panel alarm indicators and rear panel output relays for alarm annunciation are also included. Front panel pushbuttons are provided to turn power on/off, display alarm limit set points, to acknowledge alarms, and to activate a check source function. Analog outputs of 4-20 ma is connected for recorder and computer monitoring. Analog output of 0 to 10 VDC is available but not connected. A communication loop transfers data between the remote indicator and the preamplifier.

Five LEDs are used to provide visual indication of status on the front face of the remote indicator. They are as follows:

1. HIGH Alarm Red LED
2. WARN Alarm Amber LED (Not Used)
3. Fail Alarm Red LED
4. Range Alarm Red LED
5. Check Source Green LED

Only the high alarm is connected to an annunciator window on the "Common X" panel. The high alarm will flash until acknowledged. The WARN set point is not used. The fail and range alarms are not connected to annunciation. The check source LED is lit while the check source function is activated.

Radiation Monitoring System Cabinet

All of the remote indicators are centralized in one cabinet which is located conveniently in the Control Room. The cabinet houses 24 remote indicators and a 30 point recorder. Each remote indicator is provided 120 VAC power within the cabinet. Each remote indicator provides all power to its channel including the local preamplifier, local alarm light and local horn. The recorder sequentially records the outputs from the remote indicators at a chart speed of 2 inches per hour and a print rate of 10 seconds per point.

Health Physics Program

Facilities and Access Provisions

The facility has been divided into four basic areas:

1. Controlled Area – As defined in 10CFR20, means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason. This area, which includes the cooling canals, comprises the site within the boundary shown in Figure 2.2-4.

2. Restricted Area – As defined in 10CFR20, means an area to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area. Restricted areas are located within the security fence shown in Figure 11.2-3.

3. Generating Station Area - This area, also referred to as the protected area (PA), is that within the security fence, shown in Figure 11.2-3, and is occupied by the two nuclear units and their associated structures. Access to the Generating Station Area is through a guarded gate.

4. Radiation Controlled Area - The Radiation Controlled Area (RCA) is shown in Figure 11.2-3. This area includes that in which radioactive materials and radiation above 0.5 mrem/hr may be present. The Radiation Controlled Area includes the auxiliary building, Units No. 3 & 4 containment, fuel handling buildings, waste handling facility building, dry storage warehouse, the steam generator storage building, and the Low Level Waste Storage Facility. It does not include the rod control switchgear rooms. Access to the Radiation Controlled Area is limited to those individuals authorized for entry. Entry into the Radiation Controlled Area is through a clearly marked Radiation Control Point. Restricted Areas may be established outside the RCA (but within the Controlled Area) in accordance with 10CFR Parts 19 and 20. C27

Any area inside the Radiation Controlled Area in which radioactive materials and radiation may be present shall be surveyed, classified and conspicuously posted with the appropriate radiation caution sign. The Radiation Controlled Area dress facility is employed as a protective clothing change area and storage area.

Personnel decontamination showers are located in the Decontamination Shower Facility, located in the north end of the dress facility.

All personnel monitor themselves on leaving the radiation controlled areas.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any designated high radiation area or contaminated area. These measures include the following:

1. Areas accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.100 rem in one (1) hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, are barricaded and conspicuously posted as "high radiation areas." Administrative controls require the issuance of a Radiation Work Permit (RWP) prior to entry to any high radiation areas or contaminated area.

2. Locked doors are provided to prevent unauthorized entry into those areas in which the radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem in one (1) hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. Doors shall remain locked, except during periods of access by personnel under an approved RWP. For individual high radiation areas that are located within large areas, such as, the pressurized water reactor (PWR) containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

3. Any individual or group of individuals permitted to enter a high radiation area is provided with or accompanied by one or more of the following:

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

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

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

4. All personnel are required to wear protective clothing for entry into designated contamination areas. The areas involved are decontaminated as necessary to prevent the spread of contamination. Decontamination is performed under the direction of health physics personnel.

Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the interpretation of thermoluminescent dosimeters (TLD). Direct reading dosimeters (which include both self-reading pocket ionization chambers and digital alarming dosimeters) provide day-by-day indication of external radiation exposure.

All plant assigned personnel subject to occupational radiation exposure are issued beta-gamma thermoluminescent dosimeters (TLDs) and are required to wear them at all times practical while within the Radiation Controlled Area. Neutron sensitive TLDs are issued to personnel whenever a significant neutron exposure is possible.

Plant assigned personnel are issued TLDs at the entrance to the Radiation Controlled Area and return them prior to leaving at the end of the day. The TLDs are processed on a routine basis. Personnel TLDs may also be processed for administrative exposure control purposes or when it appears that an overexposure may have occurred.

Direct reading dosimeters are issued, in addition to the TLD badge, to personnel working in the Radiation Controlled Area. Direct reading dosimeters are read, recorded and re-zeroed regularly. Dosimeter records furnish the exposure data for the administrative control of radiation exposure.

Special or additional personnel monitoring devices are issued as may be required under unusual conditions. For example, finger rings may be prescribed for monitoring exposure to the hands.

Non-qualified personnel entering the Radiation Controlled Area are escorted by qualified personnel and are issued personnel monitoring devices as appropriate prior to entering the Radiation Controlled Area. An escort may not be required for those who have received the necessary radiation protection training when this arrangement is approved by the Health Physics Supervisor and authorized by the Plant Manager - Nuclear.

Personnel Protective Equipment

The nature of the work to be done is the governing factor in the selection of protective clothing to be worn in the Radiation Controlled Area. The protective apparel available include shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel such as plastic or rubber suits, face shields, and respirators are also available. health physics-trained personnel shall evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Process or other engineering controls (e.g., containment or ventilation) are used, to the extent practical, to control the concentrations of radioactive material in the air. When it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, the following are used, consistent with maintaining the total effective dose equivalent (TEDE) as low as reasonably achievable:

a. Control of access;
b. Limitation of exposure times;
c. Use of respiratory protection equipment; or
d. Other controls.

Respiratory protection equipment selected provides a protection factor greater than the multiple by which peak concentrations of airborne radioactive materials in the working area are expected to exceed the values specified in 10 CFR 20. If the selection of a respiratory protection device with a protection factor greater than the multiple defined in the preceding sentence is inconsistent with the goal of keeping the TEDE as low as reasonably achievable, respiratory protection equipment with a lower protection factor may be selected only if such a selection would result in keeping TEDE as low as reasonably achievable.

Respirator devices available for use include:

1. Full-face respirator (filter, filter/charcoal canister, or supplied air)
2. Air-fed hoods (supplied air)
3. Self-contained breathing apparatus

Self-contained or supplied air breathing apparatus are available for use in a situation involving exposure to gaseous activity or oxygen deficient atmospheres.

Respirators are maintained by checking for mechanical defects, contamination, and cleanliness by health physics trained personnel.

Monitoring Instrumentation

A Health Physics Room and Radiochemistry laboratory are provided for the health physics and chemistry personnel. These facilities include both laboratory and counting rooms. These are equipped to analyze routine air samples and contamination swipe surveys. Areas are available for the storage of portable radiation survey instruments, respiratory protection equipment and contamination control supplies.

A portal monitor is located at the personnel exits from the Protected Area and provides a final radiation survey of all personnel leaving the Protected Area.

The types of portable radiation survey instruments available for routine monitoring functions are listed in Table 11.2-9.

Survey instruments are calibrated periodically, and maintenance records are provided for each instrument according to plant operational procedures.

11.2.4 EVALUATION

Evaluation of LOCA Control Room Dose

This section describes the shielded dose determined during the reanalysis of events performed under the Regulatory Guide 1.183 (Reference 7) methodology with Alternative Source Term (AST).

The total control room dose requires the calculation of direct shine dose contributions from:

- the radioactive material on the control room filters,
- the radioactive plume in the environment, and
- the activity in the primary containment atmosphere C26
  - through the containment walls, and
  - through the purge line penetration (that has line of sight to the control room).

The limiting contribution to the total dose to the operators from direct radiation sources such as the control room filters, the containment atmosphere, and the released radioactive plume were calculated for a LOCA/MHA. The 30-day direct shine dose to a person in the control room, considering occupancy, is provided in Table 11.2-12.

Direct shine dose is determined from three different sources to the control room operator after a LOCA/MHA. These sources are the containment walls, the purge duct penetration area (different shielding than containment walls, but same source term), the control room make-up and recirculating air filter and the external cloud that envelops the control room. All other sources of direct shine dose are considered negligible. The MicroShield 5 code is used to determine direct shine exposure to a dose point located in the control room. The exposure results from the series of cases for each source location were then corrected for occupancy using the occupancy factors specified in Regulatory Guide 1.183. The cumulative exposure and dose are subsequently calculated to yield the total 30-day direct shine dose from each source.

Operator dose during a design basis LOCA for actions outside the control room are evaluated in detail in UFSAR Section 14.3.5.

<u>Evaluation of Vital Area Access Outside the Control Room</u>

The Turkey Point shielding design ensures that radiation to personnel performing vital accident mitigating steps outside the Control Room is within the 5 rem dose limit of 10 CFR 50, Appendix A, GDC 19, in compliance with Item II.B.2 of NUREG-0737.

Operating procedures have been evaluated to identify the subset of actions required for accident mitigation that occur outside the Control Room envelope. For each of these actions, a conservative post-accident dose rate has been developed based on the following: C26

- Mission location
- Time after accident
- Contributing sources of radioactivity
- Available shielding

Table 11.2-13 identifies each of the credited actions occurring outside the Control Room envelope and the maximum dose an operator could receive performing each mission. The doses presented in Table 11.2-13 include the transit path from the Control Room to each vital access area. In all cases, the resulting dose is within the 5 rem limit of GDC 19.

The principal contributors to mission dose outside containment are radionuclides from three distinct fluid volumes present during accident conditions: the containment atmosphere, pressurized RCS sample or letdown fluid, and depressurized sump fluid.

The total isotope inventory for the Turkey Point whole-core source term, presented in Table 14.3.5-7, is multiplied by appropriate core release fractions to obtain the radionuclide inventory specific to each of these three sources:

Source A - Containment atmosphere - 100% noble gases and 25% halogens

Source B - Sample or letdown streams - 100% noble gases, 50% halogens, and 1% remainders

Source C - Sump fluid - 50% halogens and 1% remainders

To varying degrees, these sources contribute to mission dose due to each mission area's proximity to the Containment Building wall, containment penetrations, or sample or recirculating sump fluid piping. In addition, dose from containment atmosphere leakage to the environs surrounding the mission areas is also considered. C26

The dose contribution from immersion in the post-accident cloud of containment atmosphere leakage (Source A) credits the operator with wearing a self contained breathing apparatus (SCBA) that eliminates at least 95% of inhaled radioactive contamination.

As shown in Figures 9.9-1, 9.9-4 and 9.4-5, the control room has its own independent ventilating system. In the event of a MHA the control room ventilating system is automatically placed in the recirculating mode as discussed in Section 9.9. The control room ventilation system is supplied by emergency power.

The radiation sources used with the original auxiliary shielding design criteria resulted from a loss of coolant accident caused by a double-ended rupture of a reactor coolant loop where the engineered safety features function to prevent melting of fuel cladding and to limit the metal-water reaction to a negligible amount. This would result in only the fission products which are in the fuel rod gaps being released to the containment. It was assumed that all gap activity, except that of the noble gases, would be absorbed in the sump water which flows in the residual heat removal loop and associated equipment. C26

Mission dose evaluations prepared for EPU conditions are based on recirculating sump water containing 1% failed fuel and 50% of the core halogen inventory (“Source C” described above). Gamma energy release rates for Source C are presented in Table 11.2-11.

The radioactivity in the containment could be an additional source of radiation to the Auxiliary Building following a loss-of-coolant accident. However, the radiological exposure rate in the Auxiliary Building from this source would be less than one percent of that from heat removal system piping. Operator dose from streaming radiation through containment penetrations is considered on a case-by-case basis.

An evaluation was made of direct radiation levels surrounding recirculation piping of varying size. The evaluation was based on the radiation sources and evaluation parameters tabulated on Table 11.2-11. The results of the evaluation are presented in Figure 11.2-4, showing dose rates and 31-day integrated dose as a function of distance from a 20-ft length of recirculation piping.

If maintenance of equipment near the recirculation loop is absolutely essential to the continued operation of the engineered safety features during the recirculation phase, local shielding would permit some operations in the vicinity of the loop.

If maintenance directly on the loop proper is required, such operations would be limited in duration as radiation levels adjacent to equipment containing the sump water and fission products might be as high as 200 to 300 rem per hour shortly after the initiation of recirculation. Any such emergency maintenance operations described above could be carried out behind portable shielding and using portable breathing equipment to limit the inhalation hazard from possibly leaking components.

11.2.5 TEST AND INSPECTION CAPABILITY

Complete radiation surveys are made throughout the containment and Auxiliary Building during initial phases of start-up for comparison with future periodic surveys. Survey data are compared to design levels up to rated full power. Survey data are evaluated and reviewed to ensure that operating personnel will not be exposed in excess of applicable limits.

Checks of the waste liquid effluent monitors response to a test source are made periodically.

The plant vent gas monitor is calibrated during shutdown, and normal response of each monitor can be tested at any time using a, remotely operated test source to verify the instruments response and alarm functions.

11.2.6 REFERENCES

1. 10 CFR 50.68, “Criticality Accident Requirements”.

2. NRC Letter to FPL, “Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Temporary Spent Fuel Pool Cask Racks” (TAC Nos.MB 6909 and MB 6910), License Amendments 226/222, November 24, 2004.

3. FPL Letter L-2003-213 to the NRC, “Turkey Point Units 3 and 4 – RAI Response for Addition of Spent Fuel Pool Cask Area Rack Amendment”, September 8, 2003.

4. FPL Letter L-80-16 to the NRC, “NUREG-578 Short Term Requirements,” January 11, 1980.

5. FPL Letter L-81-285 to the NRC, “Post TMI Requirements - Control Room Habitability”, July 9, 1981.

6. NRC Letter (Accession No. ML12024A104) Jason C. Paige (NRC) to Mano Nazar (FPL), “ Turkey Point, Units 3 and 4 - Issuance of Amendments Regarding High Range - Noble Gas Effluent Monitors, Main Steam Lines Accident Monitoring Instrumentation (TAC Nos. ME6891 and ME 6892)”, dated June 15, 2012. C26

7. USNRC, Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants”, July 2000. C26

TABLE 11.2-1

RADIATION ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Max. Dose Rate (1% Fuel Defects) mrem/hr
I	Normal Occupancy	$\leq$ 0.5
II	Periodic Occupancy	$\leq$ 2.5
III	Short Specific Occupancy	$\leq$ 15.
IV	Minimal Occupancy	$\leq$ 100
V	Controlled Access	> 100

TABLE 11.2-2

PRIMARY SHIELD NEUTRON FLUX AND DESIGN PARAMETERS

Original Calculated Neutron Flux / Shield Horizontal Mid-plane[1]

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Energy Group	Incident Flux n/cm ^{2}-sec	Leakage Flux n/cm ^{2}-sec
E >1 Mev.	1.2×10^9	1.15×10^2
5.3 Kev. $\leq$ E $\leq$1 Mev.	1.7×10^{10}	2.75×10^2
.625 $\leq$ E $\leq$ 5.3 Kev.	9.8×10^9	4.75×10^2
E < .625 ev.	4.1×10^9	2.4×10^5

Original Core Design Parameters[1]

A.	Thermal Power Rating (100% power)	≈ 2300 MWt [1]
B.	Effective Dimensions	
	1. Height	12.0 ft
	2. Diameter	9.98 ft
C.	Volume Fractions	
	1. UO_2	0.3022
	2. Zircaloy	0.0933
	3. Water	0.5980
	4. 304 Stainless Steel	0.0061
	5. 718 Inconel	0.0004
D.	Operating Times (Equivalent Full Power Hours)	
	1. Initial cycle	11,700
	2. Equilibrium cycle	8,200
E.	Mode of Operation	Base Load
F.	Fraction of Fuel Rods with Cladding Defects	0.01

NOTE :

1. Plant shielding was designed at the plant's initial power rating. Radiological impacts due to the 1995 thermal power uprate were analyzed and found not to be significant. The design was also shown to be adequate for the 2012 extended power uprate, due to the low-leakage fuel management methods that minimize flux near the core periphery. C26

TABLE 11.2-3

ORIGINAL SECONDARY SHIELD DESIGN PARAMETERS

Core power density	86.4 w/cc[1]
Reactor coolant liquid volume	9400 ft^3
Maximum purification letdown rates	120 gpm
Average water temperature in core	580 °F
System operating pressure	2250 psia
Reactor coolant transit times:	
Core	0.9 sec.
Core exit to steam generator inlet	2.0 sec.
Steam generator inlet channel	0.6 sec.
Steam generator tubes	3.2 sec.
Steam generator tubes to vessel inlet	2.7 sec.
Vessel inlet to core	2.1 sec.
Total Out of Core	10.6 sec.
Total power dose rate outside secondary shield	< 1 mr/hr [1]

NOTE :

1. Plant shielding was designed at the plant's initial power rating. Radiological impacts outside the secondary shield due to thermal power uprate and extended power uprate were analyzed and found not to be significant.

TABLE 11.2-4

ACCIDENT SHIELD DESIGN PARAMETERS

DELETED

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TABLE 11.2-5

REFUELING SHIELD DESIGN PARAMETERS

DELETED

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TABLE 11.2-6

PRINCIPAL AUXILIARY SHIELDING

Component	Concrete Shield Thickness, Ft. - In.
Demineralizers	3-6
Charging pumps	1-6
Holdup tank	2-6
Volume control tank	3-0
Reactor coolant filter	2-9
Gas stripper	2-6
Gas decay tanks	4-0
Waste gas compressor	2-8
Waste evaporators	2-0
Waste holdup tank, Aux. Bldg.	1-0 to 1-6
Waste holdup tank, Rad. Fac.	2-0
Distillate demineralizers	1-0
Waste monitor tanks	1-0
Waste holdup/mixing tanks	3-0
Cement mixers	3-0

Design parameters for the auxiliary shielding include:

Core thermal power	2652 MWt
RCS Activity	NOTE 1 C26
Dose rate outside auxiliary building and radwaste facility	<1 mr/hr
Dose rate in the building outside shield walls	<2.5 mr/hr

NOTE 1: The auxiliary shielding design was re-evaluated and found acceptable for EPU using scaling factors to compare original to EPU dose results. RCS activity at uprate conditions is assumed to be consistent with full power operation at the Technical Specification limit for RCS Dose Equivalent (DE) I-131. C26

TABLE 11.2-7

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES

Channel	Sensitivity Range	Detected Isotopes
Process		
R3-11 & R4-11	1.0×10^{-9} to 1.0×10^{-6}*	I^{131},I^{133},Cs^{134},Cs^{137}
R3-12 & R4-12	1.0×10^{-6} to 1.0×10^{-3}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133}
R-14	5.0×10^{-7} to 1.0×10^{-4}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133}
R3-15 & R4-15	1.0×10^{-7} to 1.0×10^{-1}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133} C26
R3-17A, R3-17B, R4-17A, R4-17B	1.0×10^{-6} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R-18	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R3-19, R4-19	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R3-20, R4-20	1.0×10^{o} to $1.0 \times 10^{+5}$**	Kr^{85},Ar^{41},Xe^{133},Xe^{135}
Area Rl thru R24	1.0×10^{-1} to $1.0 \times 10^{+7}$**	

Notes: * is given in µCi/cc
** is given in mr/hr

Prefixes R3 or R4 designate Unit #3 or Unit #4. Channels without prefix number monitor both units.

TABLE 11.2-7a

RADIATION MONITORING, SYSTEM CHANNEL ALARM SET POINTS

[TABLE INTENTIONALLY LEFT BLANK]

TABLE 11.2-8

DETECTING MEDIUM CONDITIONS

Channel	Medium	Temperature Range, C
Area:		
R-1 through R-24	Air	10-50
Process:		
R3-11	Air	10-50
R4-11	Air	10-50
R3-12	Air	10-50
R4-12	Air	10-50
R-14	Air	4-50
R3-15	Air	10-50
R4-15	Air	10-50
R3-17 A&B	Water	4-71
R4-17 A&B	Water	4-71
R-18	Water	15-71
R3-19	Water	15-71
R4-19	Water	15-71
R3-20**	Water	10-80
R4-20*	Water	10-80

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* Detector mounted on outside of pipe carrying medium.
** Detector mounted external to pipe carrying medium.

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TABLE 11.2-9

PORTABLE RADIATION SURVEY INSTRUMENTS

Type
Low Range beta-gamma Survey Meter
Intermediate Range beta-gamma Survey Meter
High Range beta-gamma Survey Meter
Personnel Monitoring beta-gamma Survey Instruments
Neutron Survey Meter
High Volume Air Particulate Sampler
Low Volume Air Particulate Samples
Beta-gamma and gamma Portal Monitors
Direct Reading Dosimeters (includes both self-reading pocket ion chambers and digital alarming dosimeters)
Low Level gamma Scintillation Survey Meters
Alpha Scintillation Survey Meters

TABLE 11.2-10

INSTANTANEOUS RADIATION SOURCES RELEASED TO THE
CONTAINMENT FOLLOWING TID-14844
ACCIDENT RELEASE - Mev/sec

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TABLE 11.2-11

LOCA ACTIVITY SOURCES IN CIRCULATING IN RESIDUAL HEAT
REMOVAL LOOP AND ASSOCIATED EQUIPMENT - Mev/sec

Time After Release

Energy MeV	0 hr	1 hr	2 hr	8 hr	24 hr	168 hr	31 day
0.01	1.26E+17	3.37E+16	2.80E+16	1.93E+16	1.28E+16	3.14E+15	8.74E+14
0.025	7.61E+16	1.69E+16	1.38E+16	8.94E+15	5.71E+15	1.95E+15	5.79E+14
0.038	9.69E+16	2.29E+16	1.91E+16	1.45E+16	1.18E+16	5.76E+15	1.15E+15
0.058	1.17E+17	2.51E+16	2.02E+16	1.27E+16	7.86E+15	2.21E+15	7.76E+14
0.085	1.81E+17	3.99E+16	3.08E+16	2.34E+16	2.09E+16	1.11E+16	1.47E+15
0.125	2.70E+17	9.65E+16	8.42E+16	6.66E+16	5.21E+16	1.27E+16	3.11E+15
0.225	8.75E+17	2.50E+17	2.52E+17	2.95E+17	2.10E+17	2.72E+16	3.93E+15
0.375	1.72E+18	6.32E+17	5.63E+17	4.70E+17	4.09E+17	2.31E+17	3.09E+16
0.575	6.90E+18	3.61E+18	2.96E+18	1.56E+18	8.07E+17	9.89E+16	3.50E+16
0.85	1.08E+19	4.17E+18	2.58E+18	6.26E+17	2.81E+17	1.09E+17	6.80E+16
1.25	9.46E+18	3.35E+18	2.69E+18	1.23E+18	2.94E+17	1.29E+16	3.94E+15
1.75	3.17E+18	1.47E+18	1.14E+18	5.33E+17	1.58E+17	5.63E+16	1.56E+16
2.25	1.37E+18	2.64E+17	1.93E+17	7.56E+16	1.77E+16	4.20E+15	1.71E+15
2.75	8.21E+17	4.08E+16	2.28E+16	5.42E+15	4.31E+15	3.27E+15	9.01E+14
3.5	7.21E+17	4.78E+16	1.57E+16	3.79E+14	4.69E+13	3.60E+13	1.13E+13
5	5.82E+17	8.99E+14	2.43E+14	1.24E+12	2.32E+10	3.35E+08	3.27E+08
7	1.16E+15	5.46E+07	5.43E+07	5.43E+07	5.43E+07	5.40E+07	5.26E+07
9.5	3.07E+12	8.48E+06	8.47E+06	8.47E+06	8.47E+06	8.42E+06	8.21E+06

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TABLE 11.2-12

CONTROL ROOM DIRECT SHINE SHIELDED DOSE RESULTS USING AST
(EPU CONDITIONS)

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SOURCE	DIRECT SHINE DOSE (rem)
Containment	
Walls	0.060
Purge Duct	0.337
External Cloud	0.277
CR Recirculation Filters	0.054
Total	0.728

TABLE 11.2-13

VITAL AREA ACCESS MISSION DOSES

MISSION	UNIT/ TRAIN/ EQPT	DOSE (rem)
Verify Cold Leg Recirculation Capability	Unit 3, Train A, MCC 3C	1.92
	Unit 3, Train B, MCC 3B	0.96
	Unit 4, Train A, MCC 4C	1.68
	Unit 4, Train B, MCC 4B	0.96
Close Radiation Shield Doors	Unit 3	1.04
	Unit 4	1.12
Reset Pressurizer Heater Lockout Relay (shift to “Emergency Mode”)	Unit 3	3.06
	Unit 4	3.17
SI Accumulator Isolation MOV Breaker Operation	Unit 3, MCC3A	3.06
	Unit 3, MCC3B	0.46
	Unit 3, MCC3C	1.59
	Unit 4, MCC4A	3.17
	Unit 4, MCC4B	1.37
	Unit 4, MCC4C	0.46
Recovery from Failed Open Cold Leg Recirculation Valve	Unit 3	1.27
	Unit 4	4.80
Recovery from Failed Open Cold Leg Direct RHR Injection Valve	Unit 3	3.76
	Unit 4	3.76
Recovery from Failed closed MOV-3/4-869	Unit 3	1.27
	Unit 4	4.80
AFE Control Valve Backup Nitrogen Bottle Change-out	Unit 3	2.10
	Unit 4	2.10
EDG Fuel Oil Replacement	Unit 3	2.19
	Unit 4	2.19
EDG Lube Oil Replacement	Unit 3	2.02
	Unit 4	2.02

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FINAL SAFETY ANALYSIS REPORT

FIGURE 11.2-1

REFER TO ENGINEERING DRAWING

5610-M-50

C26

Revised 04/17/2013

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

RADIATION ZONE DIAGRAM PLAN
FULL POWER OPERATION WITH 1%
FAILED FUEL

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FIGURE 11.2-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.2-2

REFER TO ENGINEERING DRAWING

5610-M-51

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

AREA RADIATION ZONE PLAN

FULL POWER OPERATION

WITH 1% FAILED FUEL

FIGURE 11.2-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.2-3

REFER TO ENGINEERING DRAWING

5610-C-2

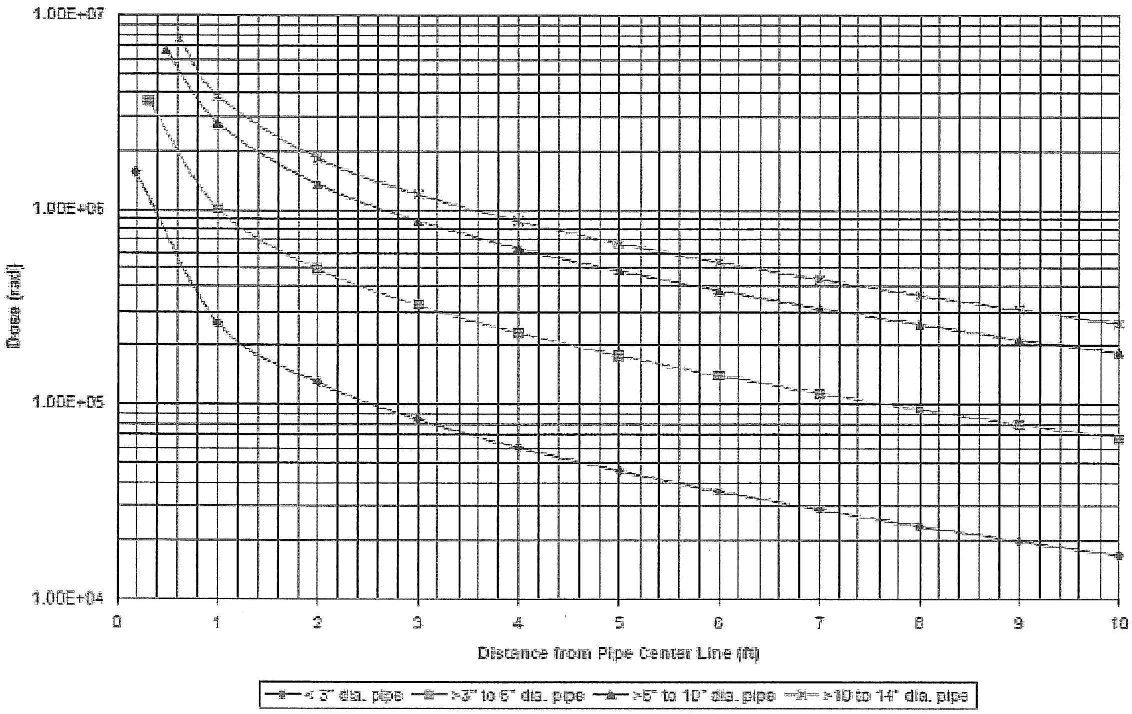
REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

GENERAL STATION AREA

FIGURE 11.2-3

31- day Integrated EPU Dose from Post - Accident Recirculating 20’ Lines



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T=0 hr Post LOCA Dose Rates (R/hr) from 20' Pipe				
Distance (ft)	Nominal Pipe Diameter (inches)			
	3	6	10	14
Contact	1.11E+05	2.57E+05	4.70E+05	5.39E+05
1	1.87E+04	7.18E+04	1.98E+05	2.73E+05
2	9.22E+03	3.54E+04	9.64E+04	1.31E+05
3	5.95E+03	2.30E+04	6.26E+04	8.55E+04
4	4.26E+03	1.66E+04	4.52E+04	6.23E+04
5	3.23E+03	1.27E+04	3.46E+04	4.80E+04
6	2.54E+03	1.00E+04	2.74E+04	3.82E+04
7	2.05E+03	8.11E+03	2.22E+04	3.12E+04
8	1.68E+03	6.69E+03	1.84E+04	2.59E+04
9	1.41E+03	5.61E+03	1.54E+04	2.18E+04
10	1.19E+03	4.76E+03	1.31E+04	1.86E+04

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LOCA RECIRCULATION PIPING T=0 HR DOSE RATE and 31 - DAY INTEGRATED DOSE

FIGURE 11.2-4

FINAL SAFETY ANALYSIS REPORT

FIGURE 11.2-5

DELETED

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SENSITIVITY OF DOSE TO ACTIVITY IN THE RESIDUAL HEAT REMOVAL WATER

FIGURE 11.2-5

11.3 RADIOACTIVE MATERIALS SAFETY

11.3.1 MATERIALS SAFETY PROGRAM

Procedures, facilities and equipment for handling and processing of radioactive liquid, gaseous and solid wastes are described in Section 11. Procedures, facilities and equipment for the safe handling and storage of new fuel assemblies and spent fuel assemblies are described in Section 9.5. Various radioactive sources are employed to calibrate and/or check the process and effluent radiation monitors and the area radiation monitors described in Section 11.2.3 and the portable radiation survey instruments listed in Table 11.2-9. Check sources that are integral to the area, process and effluent monitors are handled and stored by employing the normal Health Physics Operating procedures. The same consideration applies to radionuclide sources of exempt quantities which are used to periodically check the radiation monitoring equipment.

Radioactive sources purchased or prepared by the Chemistry Department or under the direction of the Radiochemist for the calibration, testing or standardization of laboratory counting equipment will be stored under administrative control in the radiochemistry laboratory or in a designated storage area.

Radioactive sources purchased by the Health Physics Department for the calibration, testing or standardization of laboratory counting equipment shall be stored under administrative control in the health physics counting room, the health physics calibration facility, the radiochemistry laboratory or in a designated storage area.

If a sealed source containing greater than 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material is found to be leaking greater than or equal to 0.005 microcuries, it shall be immediately removed from service. The source will be disposed of in accordance with plant waste disposal procedures or repaired. Records shall be maintained current which will include, but are not necessarily limited to date received, supplier, isotope, quantity, and date of ultimate C30 disposal or consumption of the source at which time it will be removed from the inventory list. All documentation accompanying the purchase of any sealed source shall become a permanent part of the Health Physics Department records.

Radioactive sources and materials are subject to controls for the purpose of radiation protection. These controls include:

a) Monitoring for external dose rate and removable contamination upon receipt at the plant and prior to shipment away from the plant. Both the packaging surface and the transport vehicle are monitored prior to shipment away from the plant.

b) Each sealed source obtained by license is labeled as to the quantity of activity, isotope and source identification number. The radiation symbol is affixed to all of the above sources except those which are contained in area and process monitoring components. Radioactive sources are stored under administrative control when not in use.

c) Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

    i. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

        1) With a half-life greater than 30 days (excluding Hydrogen 3), and
        2) In any form other than gas.

    ii. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

iii. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

d) Records on the results of inventories, leak tests and the receipt and final disposition dates shall be maintained for accountable sealed sources. The Health Physics Supervisor is responsible for the accountability and documentation of accountable sources.

e) Radiation work permits which provide detailed instructions for all work in radiation, high radiation, and airborne radioactivity areas. Radiation work permits are described in the Turkey Point Plant Radiation Protection Manual.

In the event of an inventory discrepancy of sealed sources, the Health Physics Supervisor will investigate and determine if the loss may result in a substantial hazard to persons in unrestricted areas. If required the loss will be reported in accordance with the requirements of 10 CFR 20.

The sealed sources will be handled and used in accordance with the Turkey Point Plant Radiation Protection Manual. Recognized methods for the safe handling of radioactive materials are implemented to maintain potential external and internal doses at levels that are as low as reasonably achievable (ALARA). The radioactive materials safety program is described in the Turkey Point Plant Radiation Protection Manual.

11.3.2 FACILITIES AND EQUIPMENT

The radiochemistry laboratory consists of a 31'- 6" x 18' room containing a fume hood, cabinets, counter-tops, and a counting room, along with necessary chemistry hardware. The radiochemistry counting room is located in the Health Physics Control Building.

The fume hood is a five foot wide radioactive model of the Fisher Conserv-Air. Air is drawn by a blower-motor which provides a 125 FPM face velocity and a 2300 FPM duct velocity. Filtering is provided by a 24" x 24" x 2" prefilter followed by a 24" x 11-1/2" CWS type filter with a 99.95% efficiency rating for 0.3 to 0.5 micron-sized particles. This exhaust is fed into the plant vent exhaust plenum. Two additional filters (a 1" prefilter and a 10" DOP tested absolute filter) filter this exhaust before it is released to the plant vent. This exhaust is continuously monitored by particulate, iodine, and gaseous detectors.

Equipment and facilities for the sampling of radioactive liquids and gases are described in Section 9.4. The area radiation monitoring and the process and effluent monitoring systems are detailed in Subsection 11.2.3. Health physics instrumentation is listed in Table 11.2-9.

The health physics facilities include: (a) office area for health physics supervisors and support personnel; (b) records area; (c) computer room; (d) areas for controlling Rca access; (e) material release building; (f) instrument calibration and radioactive source storage area; and (g) counting room.

11.3.3 PERSONNEL AND PROCEDURES

The key person responsible for the supervision of the handling and monitoring of the materials is the Health Physics Supervisor whose experience and qualifications are listed in the Plant Technical Specifications.

The radiation safety instructions to working personnel appropriate to the handling and use of radioactive materials are listed in the Turkey Point Plant Radiation Protection Manual.

Radioactive sources that are subject to the material controls described in the Radiation Protection Manual will only be used or handled by or under the direction of chemistry and radiation protection personnel. Each individual using these sources are familiar with the radiological restrictions and limitations placed on their use. These limitations protect both the user and the source.

A comprehensive basic Health Physics Training Program is given to all personnel assigned to Turkey Point Units 3 and 4 with unescorted access to the RCA. Supervisors are responsible for ensuring that their employees receive adequate on the job radiation protection training. The amount and type of training depends on the kind of work they perform and where they work. Orientation lectures on radiation and radiation protection are given to all new employees. In the course of their work, employees will receive additional training in radiation protection practices from supervisors, senior co-workers and chemistry and radiation protection personnel.

All personnel must pass a Health Physics examination before they are allowed access to the radiation control area unescorted. Those persons who have not successfully completed the Health Physics Training program and examination are escorted.

11.3.4 REQUIRED MATERIALS

A listing of isotopes, maximum quantities, forms and uses for all purchased byproduct, source and special nuclear materials is given in Table 11.3-1. Instrumentation check and calibration sources between atomic numbers 3 and 83 having less than 100 mCi beta/gamma activity or 100 milligrams of source or special nuclear material have been excluded from this listing.

TABLE 11.3-1

BYPRODUCT, SOURCE AND SPECIAL NUCLEAR MATERIALS: RADIOACTIVE SOURCES LISTING

Isotope	Quantity	Form	Use
U-235	See Section 3.2	See Section 3.2	Reactor fuel
U-238	See Section 3.2	See Section 3.2	Reactor fuel
Pu-Be	4 @ = 100 Ci ea	Each source contains two capsules inserted between Sb-Be pellets, all sealed in a stainless steel tube	These neutron sources are no longer used in the core. They are stored in the spent fuel pool.
Any byproduct material with atomic numbers between 3 and 83	Not to exceed 150 millicuries of each radio-nuclide	Gas or Liquid	Calibration of Analytical Instrumentation used for radio chemical analysis
Tritium	Up to 1 Curie	Liquid	Calibration of Analytical instrumentation used for radio-chemical
analysis.			
Americium	Up to 1 mci	Liquid	Calibration of Analytical instru-mentation used for radio-chemical
analysis.			
Pu-Be	Up to 10 ci	Solid Capsule	Instrument check for ex-core reactor instrumentation Calibrate Health Physics neutron survey instrument-ation
Americium 241 Beryllium	Up to 10 ci	Solid Capsule	Calibration of Health Physics Equipment

11.4 Radiological Administrative Controls

The following programs shall be established, implemented, and maintained:

11.4.1 In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

11.4.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

11.4.3 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.