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12.1 SHIELDING

12.1.1 DESIGN OBJECTIVES

Original radiation shielding was designed for continuous safe operation at a core power level of 2700 MWt and a 12-month fuel cycle with system activity levels stemming from fuel cladding defects in the equivalent of one percent of the fuel rods.

The plant shielding was re-evaluated for the extended power uprate assuming a core thermal power of 3030 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% failed fuel (due to fuel defects), it is concluded that the increase in the core power level and current operation with an 18-month fuel cycle will have no significant impact on plant shielding adequacy and safe plant operation.

In addition, the shielding ensures that, in the event of a maximum hypothetical accident (MHA), the integrated radiation exposures off-site and in the control room, due to the contained activity, do not result in radiation doses to plant personnel or the general public in excess of 10 CFR 100 limits.

The applicable portions of 10 CFR 20 and 10 CFR 100 are used as the bases for defining the following limits on acceptable exposures in areas within and beyond the site boundary for normal operation and for accident conditions, respectively.

Location	Max Whole Body Dose Rate (mrem/hr)
a) Site Boundary	
Normal Operation	0.001
Following MHA	25 Rem in 2 hrs
b) Service Building	
Normal Operation	0.05
c) Turbine Building	
Normal Operation	0.05
d) Reactor Auxiliary & Fuel Handling Buildings	
Continuous Occupancy	
Outside Controlled Access Areas	0.5
Inside Controlled Access Areas	2.5

Location	Max Whole Body Dose Rate (mrem/hr)
Controlled Occupancy	
Occupancy for 6 hr/wk	15.0
Occupancy determined by Health Physics Staff	100.0
e) Limited Access Areas in Containment Structure during Operation at Full Power	100.0
f) Control Room	
Normal Operation	0.5
Following MHA	3 rem integrated whole body dose over 90 days after accident

Areas with dose rates greater than 100 mrem/hr are isolated and access to these areas is controlled in accordance with 10 CFR 20 by means described below.

The calculated maximum dose rate levels (based on 1 percent failed fuel) in the plant are shown in Figures 12.1-1 through 12.1-5. For purposes of determining length of occupancy, the plant has been divided into the following zones:

Zone	Access	Dose Rate (mr/hr)
I	No restriction	<0.25
II	Occupational Access	0.25 $\rightarrow$ 2.5
III	Periodic Access	2.5 $\rightarrow$ 15.0
IV	Limited Access	15.0 $\rightarrow$100.0
V	Restricted Access	$\geq$100.0

Zone II is a restricted radiation area to which plant personnel can have continuous access during the regular 40 hr/wk work schedule, without exceeding the allowable whole body dose of 5 rems per calendar year.

Zones III and IV are posted with "Caution-Radiation Area" signs. The radiological conditions will be conspicuously posted at the entrance to the area.

Zone V is posted with "Caution-High Radiation Area" signs, and access to it is strictly controlled. Zone V represents inaccessible restricted areas, entry to which is permissible only in conformance with the requirements specified in the Radiation Protection Manual. Each high radiation area in which radiation levels are such that an individual could receive a dose equivalent in excess of 0.100 rem in 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" prior to entry. Locked doors are provided to prevent unauthorized entry into these areas in which the intensity of radiation is greater than 1000 mr/hr.

Neutron and gamma radiation surveys are performed in all accessible areas of the plant as required to determine shielding adequacy. Areas such as the containment operating floor, reactor vessel head, reactor coolant loop compartments and spent fuel handling areas are surveyed prior to access for refueling or after shutdown and a time-limited work schedule established.

The effects of neutron streaming within containment during power operation was performed at the pre-EPU power level as discussed below to confirm that the permanent reactor cavity seal ring shield is equivalent to or better than the water bag shield used previously. Radiation surveys performed on the containment operating floor during power operation will serve to confirm the adequacy of the permanent seal ring shield.

The two-dimensional calculation of an energy and angular dependent planar source for subsequent use in the three-dimensional Monte Carlo analysis of the seal ring shield configuration was carried out in R,Z geometry using the DORT discrete ordinates code [7] and the SAILOR cross-section library [2]. The SAILOR library is a 67 energy group (47 neutron, 20 gamma ray) ENDF/B-IV data set produced specifically for light water reactor applications. In this analysis, anisotropic scattering was treated with a P_3 expansion of the scattering cross-sections and the angular discretization was modeled with a 166 angle asymmetric quadrature with fine resolution in the upward direction.

The R,Z discrete ordinates calculation extended radially from the interior of the reactor core to a depth of 65 centimeters into the concrete biological shield and axially from the bottom of the active fuel to an elevation just below the primary loop nozzles. From the nozzle elevation upward, the reactor/shield geometry is not amenable to accurate analysis using two-dimensional techniques, thus, dictating the coupling to the three-dimensional Monte Carlo. The core source used in the discrete ordinates calculation was representative of past St. Lucie fuel management and was treated conservatively in the R,Z calculation to produce radiation levels in the reactor cavity representative of the azimuthal maximum. This approach should introduce some conservatism in the analysis.

The planar source for input to the Monte Carlo analysis was taken at an elevation of 203.04 cm relative to the midplane of the reactor core. The source consists of 67 particle energy groups at each of 126 radial intervals across the geometry. The upward directed angular distribution of neutrons and gamma rays was described in 131 angular intervals.

This azimuthally uniform, position and energy dependent, source is then process into cumulative probability distributions using the DOMINO code [3]. The cumulative probability distributions are in turn randomly sampled using the MCNPBQ code [4] to produce a source particle file. The source particle file contains the particle type (neutron or photon), initial position, direction, and energy for a large number of particles which are started and tracked in the geometry by the MCNP4A Monte Carlo N-Particle Transport Code [5].

The MCNP4A problem geometry included the reactor vessel, mirror insulation, reactor vessel nozzles and loop piping, a portion of the reactor vessel head, studs, and nuts, the PCI head insulation, the Westinghouse permanent reactor cavity seal ring shield, the concrete surrounding the reactor and refueling cavity, the steam generator doghouses, the missile shield, and the operating floor. Note that the reactor structure above the reactor head (CRDMs, drive shafts and housings, ventilation ductwork, etc) was not included. This introduces a further degree of conservatism in the model. The Monte Carlo calculations use continuous energy cross-sections derived from ENDF-B/V [6].

The original method of refueling operation required the use of borated water bags to provide neutron shielding. This was reflected in previous versions of the FSAR. These water bags were temporary shields used only during power operations. Bags were installed surrounding the reactor vessel head and were supported by a seismic Class I removable steel structure. The addition of a permanent neutron shield in the reactor cavity has eliminated the requirement to install and remove the shielding during the outage and thus reduces personnel exposure.

The original neutron shield design made use of neutron shield water bags supported on a Seismic Class I steel framework. The water bags and support steel were installed in the refueling pool around the reactor vessel head. The water bags and support steel were disassembled and removed prior to refueling operations and reassembled prior to plant start-up. This neutron shield has been replaced by a Permanent Cavity Seal/Shield Ring (PCSR). The Permanent Seal Ring spans the annulus between the reactor vessel seal ledge and the refueling pool floor at the 36" elevation. The neutron shield is located in the reactor cavity directly below the seal ring. The shield consists of a stainless steel ring filled with 13½" thick borated concrete. The seal ring is a Seismic Class I structure and the neutron shield is a Seismic Class II/I structure. The neutron shield is provided with removable plugs to permit access into the reactor cavity for maintenance activities. The removable plugs incorporate a labyrinth design to permit air flow through the shield from the Reactor Cavity Cooling System to maintain cavity temperatures. Credit is also taken for air flow around the neutron shield in this design.

Table 12.1-8 lists the projected occupation radiation exposure savings (man-rem) at an 80 percent plant availability factor due to the neutron shielding with the original water bags. The doses listed in Table 12.1-8 are conservative because the permanent neutron shield is more efficient and personnel exposure during removal and replacement of the shield is eliminated.

12.1.1.1 <u>Design Considerations For Limiting Exposures</u>

The following design considerations were used in equipment and facility layout for limiting radiation exposure during plant operation, inspections, and maintenance. These considerations included shielding for refueling operations for waste management system operations, and for maintenance and in-service inspection. The criteria listed below together with appropriate procedures as outlined in Sections 12.1.5 and 12.3.2 are employed to maintain occupational radiation exposures as low as practicable:

a) Basis for Shield Design

Shielding design and the assignment of the radiation zones shown in Figures 12.1-1 through 12.1-5 are based on the operating or shutdown condition of each system, whichever is more conservative.

b) Segregation of Radiation Zones

Clean areas are separated from potentially radioactive areas by two control points, one of which serves as the access control and monitoring station for general entry, and the other of which services as a monitoring station for personnel egressing from the controlled area to the clean locker room. Personnel access to and from these areas can only take place through these control points, ensuring that radioactivity is not carried to clean areas.

Special attention is devoted to the segregation of potentially radioactive systems from systems and areas which do not see any radioactivity. Even though non-radioactive areas (personnel facilities) and systems (electrical systems, control room systems) are located in the auxiliary building, they are kept separated from radioactive systems by suitable shielding walls so that the maximum dose rate in these areas will not exceed 0.5 mr/hr. The only means of ordinarily gaining access from non-radioactive to radioactive areas is through the access control points. However, temporary checkpoints and controlled areas may be established to facilitate maintenance operations. The shielding provided and the access checkpoints are, shown in the auxiliary building general arrangements.

c) Shielding Cubicles / Component Shielding

Major sources of radioactivity are located in individually shielded cubicles to ensure safe inspection and maintenance. Labyrinth layout eliminates streaming through access opening in the cubicles. Maintenance and repairs may thus be accomplished in one cubicle without the necessity of the shutdown and decontamination of equipment in adjacent cubicles. Cubicles are for unencumbered work around the equipment and piping housed therein.

Efforts are expended to separate major components in a given system from other major components by use of shielding walls. Tanks, filters, heat exchangers and demineralizers are the major sources of activity in any given system; special effort was expended in isolating these components in shielded cubicles. Valves and pumps should not have more activity than the piping associated with them; thus the extent to which they are housed in individual cubicles is dictated by the projected piping activity. The extent to which each tank, filter and demineralizer is housed in individual cubicles is dictated by the projected activity count and the ability for taking components housed in the same cubicle out of service at the same time. Tanks which are known to continuously contain potentially high radioactivity are housed in individual cubicles. Examples are holdup tanks, boric acid holdup tanks and spent resin tanks. Tanks which are fairly low activity (e.g., floor drain tanks, laundry tanks and condensate tanks) or tanks which have the potential for occasional high activity (e.g., boric acid makeup tanks) are, housed in common cubicles with space provided for portable shields.

No shielding cubicle totally encloses any given component or components. Labyrinths are designed to prevent direct radiation shine through the opening to accessible areas, and to minimize scattered contributions. Labyrinth walls as well as the shielded cubicle walls are generally designed extending between floors so that the floors above and below also act as shielding barriers. Partial height shielding walls are used in cubicles housing horizontal tanks and pumps, the activity of which is either expected to be low or concentrated at the tank bottom. See Appendix 12B for an example of calculations of scattered radiation contribution to dose rates through partial height walls. The height of the shielding barriers in such cases is determined by geometric considerations, i.e., a person standing at the farthest point from the cubicle in question, (in an accessible area) cannot see any of the radioactive components within the cubicle including piping. As an example, the equipment and chemical drain tanks are separated from their respective pumps by a wall which is 10 feet tall. Anyone in the pump cubicle is not directly exposed to any of the piping in the tank cubicle. Similarly the wall separating the pumps from the corridor is sufficiently high so that a person standing next to the equipment drain pumps wall on the north side of the accessible corridor cannot see any radioactive piping housed in either the tanks or pumps cubicle.

Cubicles housing filters are partial height cubicles provided with a roof. The thickness of the roof is sufficient to allow a person to stand on it during filter cartridge removal operations.

Partial height shielding walls are also employed in valve pits to allow personnel entrance for maintenance and repair; and in certain labyrinth walls (such as the charging pumps walls) allow for pump removal by monorail. Partial height walls at the gas compressors entrances are utilized as supports for a 1’0" thick concrete slab designed to serve as a pipe chase for radwaste piping which runs overhead outside the gas compressors cubicles.

No special provisions are made to shield individual valves or groups of valves in valve cubicles with the exception of valves in systems handling high activity such as the letdown lines, the demineralizer resin lines and spent resin lines. Valves associated with the letdown line are housed in shielded cubicles. Other valves are housed in the equipment cubicles and are operated by extension stems through the shielding walls. Drain valves for equipment handling low level activity, such as condensate tanks, are not provided with extension stems since the exposure received during actuation of such valves would be insignificant. Care is taken, however, to locate valves away from tank bottoms where radioactive crud may accumulate so that the operator will not be forced to crouch under the tank to operate the valve.

d) Piping and Ductwork (See item e for "field run" piping)

Radioactive piping and ducting is run to minimize radiation exposure to personnel. Generally this minimization involves:

1) minimizing the routing pipes through areas which must be kept accessible at all times, e.g., corridors

2) avoiding the routing of high activity pipes through low-radiation zones

3) using shielded pipe chases when 1) and 2) cannot be avoided

4) separating radioactive and nonradioactive piping for maintenance and inspection purposes

5) slanting pipelines wherever possible to assist in removing crud deposits from the line both prior to and during maintenance

6) Avoiding rapid changes in direction in piping where possible and avoiding elbows or bends in ducting. Thus providing the least chance of buildup of contamination.

7) Utilizing consumable insert or open root piping welds for pipes 2-1/2 inches in diameter and greater to minimize the internal weld profile, thereby reducing the potential for contaminant entrapment. Piping 2 inches in diameter or smaller is socket welded to reduce inner surface cavities at the pipe connection.

Some specifics illustrating how these general guidelines are implemented are provided below:

1) Radiation zones are shown in Figures 12.1-1 through 12.1-5. No radioactive process line is normally routed through areas where the design radiation levels are less than 2.5 mr/hr. unless calculations show that the contact dose rate in such lines is below 0.5 mr/hr. In some isolated cases, such as the floor and equipment drain system, it might be necessary to briefly run the process lines through less than 2.5 mr/hr. areas; In such cases these lines will be run high and supported so that they can be wrapped in suitable shielding material. The criterion used to establish the height above the floor at which these lines will be run will be that the dose rate received by a person standing on the floor at the closest point to the line will be a small fraction of 2.5 mr/hr. The fraction is in turn determined by other equipment in the area in the sense that the addition of all contributions to the dose rate at the particular point will not exceed 2.5 mr/hr.

2) Within each cubicle, efforts are made to minimize piping which can not be isolated. However, there are headers serving identical components running through cubicles (e.g., gas decay tanks, gas header, waste gas compressor headers, etc.), cross-tying piping serving redundant components (holdup tank piping), and collection headers for drainage (floor and equipment). These pipes are routed through normally inaccessible cubicles to minimize exposures which would otherwise result by routing them through areas requiring more frequent access. In this manner, the need for pipe chases is minimized. The cubicles used for routing radioactive piping are those housing tanks since they are considered low maintenance items, with infrequent inspection requirements.

3) Pipe chases are utilized for piping which is known to be highly radioactive (e.g., spent resin lines, concentrator bottoms) and in places where radioactive piping must cross accessible areas. Where pipe chases are not practical, additional supports are provided for any given radioactive line so that the necessary shielding can be wrapped directly onto the line. The pipe chases are designed so that the maximum radiation dose rate outside the chase is less than 2.5 mr/hr.

4) All piping which may potentially carry activity is routed on drawings which are reviewed by the shielding engineer. In addition, shielding walk-through surveys are periodically made during the construction phase of the plant. Any unacceptable pipe routing, as well as shielding deficiencies, which are detected during such inspections are corrected.

e) Field Run Piping

Piping 2 inches in diameter or smaller which is "field run" is classified in two categories: special and miscellaneous.

Special piping includes seismic Class I piping, radioactive waste process piping, and other piping which by nature of its fluid content can be expected to carry radioactivity in sufficient quantity to require special consideration.

Miscellaneous piping includes drains, vents, sampling and instrumentation lines, some of which are potentially radioactive.

Routing of special class piping is accomplished in compliance with the same radiation protection criteria used for routing of larger pipes (refer to part (c) above). Drawings indicating the routing are made either by the Ebasco New York office personnel or by designers in the field (hence the "field run" connotation). In either case the drawings are reviewed both by a shielding engineer (either New York based or assigned to the field) and Ebasco main office supervisory personnel, who sign such drawings.

Miscellaneous piping is not routed on drawings. Any such piping, however, which can potentially carry fluid of sufficient activity to pose an exposure hazard, are identified (e.g., sampling lines) and the field forces alerted so they can run it in shielded areas whenever possible, to minimize exposures during operation and maintenance.

For portions of such piping which cannot be run in shielded areas, shielding sketches are prepared by the field forces and submitted to the shielding engineer for confirmation of the shielding adequacy.

f) Curbs and Drains

All cubicles with the exception of the spent resin tank cubicle and the boric acid holdup tank cubicle are provided with drains to their respective equipment and floor drain collection header. Floor drains in the spent resin and boric acid holdup tank cubicles are plugged and the equipment drains raised so that rupture of the tanks will not cause transport of resins or concentrates through the floor or equipment drainage system. Curbing is provided in those cubicles to contain any accidental spills. Backgassing through the drain system from the collection tanks is prevented by providing a loop seal between the tank and the header. Additional loop seals are provided in the drain system, some of which are upstream of the header.

g) Shield Discontinuities

Shielding discontinuities caused by shield plugs and concrete hatch covers are provided with off-sets to reduce radiation streaming.

h) Protective Coatings

Protective coatings for floors and walls are used to minimize the buildup of surface damage and contamination from leaking process fluids. Chemical and radiation resistant paints, coating or sealers minimize the retention of contamination on these surfaces and make them amenable to various decontamination procedures. The surfaces of all cubicles which contain radioactive materials that have a potential for contamination including containment, are painted with paints suitable for easy decontamination. The types of coating systems used are described in detailed Engineering specifications.

i) Sampling

A shielded sampling station is provided for the major radioactive samples (reactor coolant). Sampling will take place under a hood with sufficient shielding provided between the sample piping and the actual sample location. Local samples are not taken from their own cubicles. Care is taken, however, to locate the sampling points in the lowest practical radiation area.

j) Waste Management System Process Control

Waste management system operations are conducted primarily from panels which are located in accessible areas and shielded from radioactive components. The dose rate is not expected to exceed 2.5 mrem/hr. However not all radioactive processes and process lines are controlled and operated from panels; in many instances operation of a given process line requires operation of several valves and instruments which are not located in a panel. In all cases, however, the operation of process lines can be accomplished from a shielded position. The boric acid and waste concentrators are examples of systems which are operable from panels.

These panels are located in an accessible area where the average radiation level does not exceed 0.5 mr/hr at the hot spot outside the shield wall. Sluicing of resins, draining of radioactive tanks, etc. are operations which are initiated by valves, the operation of which is accomplished by extension stems through the shielding walls. The radiation level at the extension stem operator is less than 2.5 mr/hr. Instrumentation is generally accessible from locations outside shielding walls, and valving for isolating these instruments is generally made accessible by one of two means, namely, (i) locating isolation valves outside of shield walls, or (ii) utilizing extension stems that penetrate the shield wall.

k) Refueling Operations

All refueling operations are performed under water. The refueling cavity and spent fuel pool are designed so that the combination of concrete and water shielding results in personnel exposures below 2.0 mrem/hr.

l) Maintenance and Inspection Considerations

The plant is designed so that personnel are not normally expected to work in areas where they may be exposed to radioactive fluids in process equipment or piping resulting in dose rates exceeding 2.5 mr/hr. Personnel, however, may be required to enter areas of higher radiation levels to perform required maintenance and repair of system components. Under these conditions, however, the components requiring work will have been drained to remove as much of the activity as possible. Residual activity may remain in the form of crud. For lines and equipment which might be expected to build up considerable crud activity (resin sluice, concentrator bottoms), demineralized water connections are provided for flushing.

Access to a given component of a system does not require exposure to major components from other systems, but might require exposure to major components of the same system. Access to all major components, except the boric acid condensate tanks, is directly from low radiation areas (2.5 mr/hr or less). Access to the boric acid condensate tanks is through the condensate pumps, and access is more frequent in the pump area than in the tank area.

Clean service lines do not share pipe chases with radioactive lines. Valving in clean service lines entering radioactive cubicles is placed outside the cubicle. Both precautions ensure maintenance of clean service lines with a minimum exposure.

Within any given cubicle there might be pipes which are operating while the particular component is being maintained. Such piping will contribute to the overall dose received by the maintenance and repair personnel in the cubicle. However, during design, efforts were expended to minimize piping in any given cubicle other than that associated with the component. For instance, in the charging pump area, in any given cubicle the only pipe processing radioactive fluid while the pump is down for maintenance is the common header.

Filter cubicles are designed so that filter cartridges can be removed and inserted in a shielded cask with a minimum of exposure. Unbolting of the filter vessel head is performed from the roof of the filter cubicle. Once the head is unbolted, a shielded cask is transported by monorail over the opening on the cubicle roof, and the cartridge is pulled through the bottom of the cask.

Each major component, excluding large tanks, has the capability of being removed from the cubicle for maintenance or replacement. Knockout panels are provided wherever existing openings in cubicles are not sufficiently large to allow removal.

Cubicles housing radioactive components having high maintenance requirements are designed with extra space provided to allow for efficient inspection, disassembly, reassembly, and removal. All equipment is designed and located to be accessible for maintenance.

Requirements for portable shielding are minimized by providing permanent shielding wherever feasible.

Pump lubrication is generally provided by the fluid being pumped. Pump maintenance during operation is not required. Nonetheless care is taken to house pumps in fairly low activity areas or in areas in which the radiation level is essentially a direct function of the pump operation, so that once the pump is taken out of service for maintenance, the radiation level will drop to low values (e.g., charging pumps, holdup pumps).

Specific decontamination areas in both reactor and auxiliary buildings are designed to aid in decontamination procedures. However, decontamination is done in place wherever possible. Provisions are made to reduce the extent of contamination encountered during maintenance and repair of that portion of the system. In the event that flushing or remote chemical cleaning of contaminated systems becomes necessary, provisions set forth in the unit procedures are utilized.

Most inservice inspection is accomplished by remote techniques so that personnel exposure is minimized.

Accessways are provided to normally inaccessible areas between the primary and secondary shield to allow for inservice inspection. The primary shield wall reduces residual core radiation to 2.5 mrem/hr during shutdown. Contributions from the steam generators and piping is of the order of 100 mrem/hr in the absence of large localized crud activities. The shielding thus provides limited entry into the secondary shield area for inspection and short maintenance. Longer stays may require decontamination of a reactor coolant loop.

Areas that are potentially radioactive are served with ventilation systems to reduce the level of airborne activity. Stationary and portable monitoring equipment is available to monitor the level of airborne activity.

As part of the Post-TMI Short Term Lessons Learned Studies, a design review of plant shielding and environmental qualification of equipment for spaces/systems which may be used in post-accident operations was performed for SL-1 (item 2.1.6.b). See Reference 1.

The objective of the review in part was to insure that required access to vital areas and equipment needed for post-accident operations will not be compromised by high radiation fields.

Radiation fields were considered resulting from gaseous nuclide releases from the core into the containment atmosphere as a result of the postulated LOCA and from all equipment and piping outside the containment that might contain primary coolant or radioactive gases during post-accident operations.

Source terms for liquid systems were assumed to be 100% of the core inventory of noble gases, 50% of the core inventory of halogens, and 1% of other nuclides contained in the primary coolant. Source term for containment atmosphere followed the guidance of Regulatory Guide 1.4. Appropriate dilution and decay factors were applied.

Vital areas requiring access during post accident operation were identified plus the time period of occupancy and the number of persons were specified. Dose rates were calculated for numerous areas in the RAB revealing that in general access would be possible although administrative controls must be utilized to limit such access to short periods of time. Certain areas were identified which would require modifications to reduce exposure during post-accident operations. See Section 12.1.6.

12.1.2 DESIGN DESCRIPTION

Plant site plan and general arrangement drawings showing shielding and location of equipment are shown on Figures 1.2-1 through 1.2-19.

Design criteria for penetrations through shielding walls and for acceptable radiation levels at valve stations for process equipment containing radioactive fluids are identical to the shielding criteria for the shield walls and process equipment involved.

The physical models used to determine the shielding required are approximations to the physical layout, i.e., cylindrical sources were used to approximate tanks, line sources for pipes and so forth. The methodology outlined in T. Rockwell III, "Reactor Shielding Design Manual," are employed to determine dose rates from the modelled equivalent sources. Conservative assumptions were made in the shielding analysis such as: waste tanks completely full, neglect wall thickness of some pipes and neglect any air attenuation. Source data utilized in the calculation of shielding thicknesses are given in Section 12.1.3. Refer to Appendix 12.A for an illustration of the Rockwell Methodology in determining dose rates outside a shield wall.

The shielding calculations also utilize point kernel integration codes such as ISOSHLD. Concrete scattering problems are solved by utilizing the Chelton-Huddleston dose albedo formula. The complex geometry sources, simulated by a multiplicity of point sources, and the overall scattered dose rate are evaluated by integrating the results for a single point source over all the point sources. The scattered dose from a single point source is evaluated from the differential dose albedo formula by integrating over the polar and azimuthal angles defined by the extent of the scattering surface.

12.1.2.1 <u>Reactor Building Primary Shield</u>

The primary shield function is to limit radiation emanating from the reactor vessel; this radiation during operation consists of neutrons (both fast and slow) emitted from the core, prompt fission gammas, fission product gammas, and gamma radiation resulting from neutron capture in the core internals and vessel. Following shutdown, only fission product gammas, and gamma radiation from neutron activation of the coolant and corrosion products are present.

The primary shield consists of 7'-3" of reinforced concrete surrounding the reactor vessel. The annular cavity between the primary shield and the reactor vessel is air cooled to prevent overheating and dehydration of the concrete shield (see Section 9.4.8).

The primary shield arrangement and thickness (shown in Figures 1.2-8, 1.2-9 and 1.2-10) are designed to:

a) attenuate the neutron flux to prevent excessive activation of unit components and structures

b) reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between the primary and secondary shield

c) reduce residual radiation from the core to a level which permits access to the region between the primary and secondary shields at a reasonable time after shutdown

d) permit access during shutdown for inspections

12.1.2.2 Reactor Building Secondary Shield

The secondary shield reduces the radiation activity from the reactor coolant system to a level which allows limited access to the containment during operation and to supplement primary shielding. The controlling radiation source in the design of the secondary shield is N-16 resulting from the (n,p) reaction with the oxygen in the coolant.

The secondary shield is reinforced concrete 4'-0" thick surrounding the reactor coolant piping, pumps, steam generators, and pressurizer.

In addition, a partial 2'-0" thick reinforced concrete wall is located midway between the primary and secondary shields in the main steam and feedwater lines penetration region, to prevent streaming of gamma radiation from the reactor coolant system through the penetration openings.

The entire secondary shield system is shown in Figures 1.2-8, -9 and -10.

12.1.2.3 Shield Building

The steel containment structure is enclosed by a reinforced concrete shield building with 3'-0" thick cylindrical walls and a 2'-6" thick dome. In conjunction with the primary and secondary shields, the shield building limits the radiation level outside the structure from all sources within the containment to less than 0.5 mrem/hr at 2700 Mwt with 1 percent failed fuel (due to fuel defects).

Taking into consideration the conservative analytical techniques used to establish the Reactor Building secondary shield and the Shield Building and pre-EPU survey dose rates outside the containment wall, the secondary shield and Shield Building were determined to be adequate for extended power uprate operation.

The combination of shield structures ensures that radiation doses at the site boundary due to direct radiation from the activity within the containment are below the recommended guideline values of 10 CFR 100 in the event of the MHA.

12.1.2.4 Fuel Handling Building Shielding

Shielding is provided for radiation protection of plant personnel during all phases of spent fuel removal, storage and preparation for off-site shipment. Operations requiring shielding of personnel are: spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, and spent fuel transfer cask loading prior to transfer to the Independent Spent Fuel Storage Installation (ISFSI).

All spent fuel removal, transfer and shipping cask loading operations are done under a minimum of 10'-4" of borated water.

The refueling cavity above the reactor vessel flange is flooded to elevation 60 ft to provide 24 ft of water shielding above the reactor vessel flange. This height assures 132 inches of water above the active portion of a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate from the spent fuel assembly is less than 2 mrem/hr at the water surface.

The fuel assembly removed from the reactor vessel is moved to the upender and horizontally transferred to the fuel pool by the fuel transfer mechanism inside the fuel transfer tube. A 6'-0" concrete shield around the refueling cavity protects the refueling personnel from radiation from the spent fuel assemblies and reactor internals.

The fuel pool is flooded to provide shielding as the fuel is being withdrawn from the fuel transfer tube and being raised by the spent fuel handling machine for insertion in the spent fuel rack. The concrete sides of the fuel pool are 6'-0" thick to ensure a dose rate of less than 0.5 mrem/hr at the outer surface of the structure.

Taking into consideration the conservative analytical techniques used to establish the fuel handling shielding and pre-EPU survey dose rates in the area, the fuel handling shielding was determined to be adequate for EPU operation.

12.1.2.5 Reactor Auxiliary Building Shielding

The reactor auxiliary building shield walls are designed to protect personnel working near various system components, such as those in the chemical and volume control, waste management, shutdown cooling, and sampling systems.

In addition, major pieces of potentially radioactive equipment are housed within shielded compartments so that access to any component for repair or inspection is permissible without shutdown and decontamination of the entire system.

The criterion of isolating major components in individual cubicles is adhered to within limitations imposed by equipment removal considerations and the availability of temporary shields.

Although not shown in Figure 12.1-3a the boric acid makeup pumps are shielded from the boric acid makeup tanks. These tanks are normally low activity tanks (less than 10 mr/hr contact). Should crud buildups occur, the maintenance of these tanks would require the use of part-height temporary shields for protection against radiation from the bottom of the tank. Similar considerations apply to the chemical and equipment drain tanks. The demineralizers, shown in Figure 12.1-4, are also low maintenance items. Access to them is infrequent and can be preceded by flushing of the units concerned.

The partial shield walls utilized in radiation Zone V (note that area is Zone V only when plant is operating or if the equipment is seriously crudded) areas to separate high frequency maintenance items such as pumps and valves from tanks and other potentially radioactive components are of sufficient height so that maintenance can be performed with a minimum exposure to direct radiation. Systems designed to mitigate the effects of airborne radiation are discussed in Section 12.2.

The flash tank and ion exchanger valve enclosures have been deleted. The deletion was dictated by considerations of valve operability and as well as consideration of personnel exposure during servicing versus operation of these valves.

The valve enclosures have been deleted so that more space and unencumbered access could be made available to maintenance personnel. Individual line shielding may be employed as required, to minimize exposures during operation.

Access to the flash tank, ion exchanger and letdown system valves is controlled. Access to the spent resin tank valve enclosure is not controlled since there is no unshielded piping carrying activity continuously in the vicinity of the enclosure.

Airborne contamination levels are determined prior to servicing or operation of the valves in controlled access areas by suitable sampling. The level of airborne contamination in valve enclosure areas where access is unrestricted is monitored periodically during health physics surveys.

The concrete thickness of each compartment shield wall is sufficient to reduce the dose rate outside the compartment at nearby normally accessible areas to less than 2.5 mrem/hr. The individual enclosures also serve to contain any major spill of radioactive liquid from any of the tanks or other components in the system.

A floor drainage system conveys any liquid spills to floor sumps from which they are pumped to the waste management system for processing.

The floor drains are embedded in concrete so as to contain any spills from leaking lines and/or to shield the line in access areas.

Taking into consideration the conservative analytical techniques used to establish compartment shield walls and plant technical specifications that restrict coolant activity to levels significantly less than 1% failed fuel (due to fuel defects), the compartment shield walls were determined to be adequate for EPU operation.

12.1.2.6 Control Room Shielding

Direct radiation contribution to the dose rates in the control room following a MHA stem from the following:

a) Airborne activity in the containment (TID 14844 releases are assumes to be uniformly dispersed throughout the containment free volume

b) shield building ventilation system filters

c) hydrogen purge system filters

The control room layout and location relative to the sources of direct radiation is shown is Figures 12.1-6 and 12.1-7 respectively. The nearest filter is located at least 70 ft away with a minimum of 4 ft of concrete intervening. The shield thickness between the control room and the containment located 31 ft away, is a minimum of 6 ft equivalent concrete (2 inch thick steel pressure vessel, 3 ft thick concrete shield building and 2 ft thick control room wall).

The cumulative dose to the control room following a MHA is shown in Figure 12.1-8. The dominant contribution comes from the containment.

12.1.2.7 Emergency Core Cooling System Area Shielding

Shielding of the emergency cote cooling system rooms has been designed for normal operation and shutdown conditions. Under normal plant operating conditions only the reactor drain pump and the low pressure safety injection pumps are sources of radiation. Sufficient shielding is achieved with 1 foot thick concrete walls. A 2 foot thick concrete wall separates the two rooms. One room houses one low pressure safety injection pump, one high pressure safety injection pump, one containment spray pump and two reactor drain tank pumps; the other room houses one low pressure safety injection pump, two high pressure safety injection pumps* and one containment spray pump. Refer to Figure 1.2-12.

This arrangement allows inspection and servicing of the safety feature systems during plant operation and shutdown, and also some accessibility during the long term cooling following the postulated accident.

12.1.2.8 Materials and Structural Requirements

The concrete for the primary and secondary shield, shield building, reactor auxiliary building and fuel handling building shield walls has a density of 138 lb/ft^3. Since the primary and secondary shield walls serve as support for the reactor coolant system components and provide missile protection and support for the refueling apparatus, reinforced concrete is used.

12.1.2.9 Decontamination Area

The equipment decontamination area is located near the hat machine shop since most of the need for decontamination will occur prior to utilization of the shop facilities. The decontamination area (Figure 1.2-13) is equipped to handle the decontamination of small and medium sized equipment and tools.

12.1.3 SOURCE TERMS

Plant shielding is designed to attenuate neutron and gamma, radiation emanating from the following sources:

a) reactor vessel

b) reactor coolant loops

c) auxiliary systems equipment

d) spent fuel assemblies

e) radioactive material released during postulated accidents

The gamma and neutron sources utilized in the design of the primary shield are shown in Tables 12.1-1 and 12.1-2. The gamma sources include capture gammas from the reactor core, internals and vessel.

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* The HPSI pump 1C has been abandoned in place.

The activity of N-16 in the reactor coolant at the vessel outlet nozzles determines the amount of shielding required around the reactor coolant loops during full power operation.

During shutdown the major sources of activity in the reactor coolant loops are the fission and corrosion products reported in Table 12.1-3, corrected to account for the change in coolant density from operating of shutdown conditions.

The fission product gamma radiation source strength in the containment at various times following the maximum hypothetical accident is shown in Table 12.1-4.

Fission product and corrosion activity also determine the shielding requirements for the auxiliary systems. The total quantity of the principal nuclides in process equipment that contains or transports radioactivity is identified for selected locations in Tables 12.1-5 and 12.1-6. The selected locations are indicated by the numbers within ellipses on Figures 11.2-1, 11.2-3, 11.2-4, 11.3-1, 9.3-4 and 9.3-5. Table 12.1-6A lists the individual components of the waste management system that are expected to contain significant amounts of radioactivity. Expected maximum values, i.e., those used as a design basis for shielding requirements, correspond to the equivalent activity values calculated to exist due to fuel cladding defects in 1.0 percent of the fuel rods and are shown in Table 12.1-5. Expected average values correspond to fuel cladding defects in 0.1 percent of the fuel rods and are shown in Table 12.1-6.

The activity in the chemical and volume control system prefilter, ion exchangers and afterfilter is determined by assuming a letdown flow of 40 gpm and retention of at least 90 percent of the nonvolatile fission products, except cesium, yttrium, molybdenum and tritium; for which no removal is assumed. No credit is taken for radioactive decay.

A zinc injection system was installed in the Fall of 2009. The zinc injection system will displace cobalt from resident RCS oxide films. This will result in a temporary increase in the RCS coolant activity from ^{58}CO and ^{60}CO. The elevation in the radiocobalt activity will be temporary, lasting only until the plant oxide layers are fully conditioned. The period of increase in activity will be approximately two operating cycles depending upon the rate of zinc injection. The temporary increased activity in the RCS coolant as a result of zinc injection will cause an increase in resin and filter activity for approximately two operating cycles. Increased resin and filter usage during the first two cycles will occur, but resin depletion during cycle operation is not expected.

In the boron recovery system, the activity in the preconcentrator ion exchangers is determined by assuming retention of at least 90 percent of all incoming radionuclides, except tritium, and a batch flow rate of 20 gpm. The corresponding average annual flow rate is 1.5 gpm based on processing 780,000 gallons, equivalent to 10 reactor volumes, from sources listed in Table 11.2-1. Reactor coolant pump seal flow is not included in this average. A decontamination factor (DF) of 104 (ratio of bottoms to distillate activity) is assumed for the boric acid concentrator** for all nuclides (except tritium). The influent activity is taken to be concentrated in the bottoms by a factor of 300. No operation of the boric acid condensate ion exchanger** is assumed.

In the waste management system, the activity present in the gas decay tanks is that of the noble gases present in the coolant, with no credit taken for radioactive decay.

Outside storage tanks which may contain some radioactivity during normal operation are the primary water storage tank, the refueling water tank and the condensate storage tank. * The tank locations are shown on the site plot plan, Figure 1.2-2. Maximum expected radionuclide inventories

* Three Steam Generator Blowdown Treatment Monitor Tanks, which are located outside, may also contain radioactivity during normal operation.

** The Boric Acid Concentrators and supporting components are no longer used.

for the three tanks are presented in Table 12.1-3A. Refueling water tank inventories were calculated based on the completion of the refueling cycle assuming clean-up by the fuel pool purification system filter and ion exchanger. No decay in the tank was assumed.

Table 12.1-3A also provides the surface dose rates and site boundary dose rates resultant from these tanks. The dose rates provided are overly conservative since the maximum activity was assumed to exist continuously throughout the year.

The dose rates from outside storage tanks reported in Table 12.1-3A for site boundary have been evaluated at the closest distance between the specific outdoor tanks and the site boundary. These distances are approximately 5000 ft north for the primary water tank, 5150 ft north for the condensate storage tank and 5300 ft north for the refueling water tank. At these distances the total dose rate (direct plus air scattered) as a function of distance is a monotonically decreasing function.(1) This insures that the total dose rates from the tanks at points beyond the side boundary do not exceed that at the site boundary.

If tank surface dose rates are ever in violation of the plant health physics program, measures involving either shielding or restrictive access will be taken to assure that beyond the restricted area the dose rate is less than 2.5 mrem/hr.

A remote radiation controlled area for the long term storage of certain byproduct material that is used primarily during plant outages has been developed and is located between the site intake and discharge canals. Reference 10 discusses the requirements for this remote facility.

No radioactive waste or shipping casks are stored outside of the radiation controlled area. See Section 11.5.7.

The routing of piping carrying potentially radioactive matter in quan-

1) Reactor Shielding for Nuclear Engineers, N. H. Schaeffer, Editor USAEC, TID-25951, 1973 Chapter 8 and 9.

tities significant enough to affect the amount of exposure to plant personnel is done during the design stage to ensure that the piping is adequately shielded. Any such piping designed in the field is checked for shielding adequacy.

12.1.4 AREA MONITORING

12.1.4.1 Design Bases

The area radiation monitoring system is designed to:

a) warn of abnormal gamma radiation levels in areas where radioactive material may be stored, handled or inadvertently introduced

b) warn plant personnel whenever abnormal concentrations of airborne radioactive materials exist

c) supplement other systems, including the process radiation monitoring system (Section 11.4), the leak detection system (Section 5.2.4), and the airborne radioactivity monitoring system (Section 12.2.4) in detecting abnormal migrations or accumulations of radioactive material

d) initiate a containment isolation signal in the event of abnormal radiation inside the containment

The area radiation monitoring system is a complex of radiation monitors and alarms which provide operating personnel with a continuous record of radiation levels at selected locations within the plant. The system assists in detecting unauthorized or inadvertent movement of radioactive material in the plant and in helping operating personnel decide on deployment of personnel in the event of an accident resulting in the release of radioactive material.

12.1.4.2 System Description

The area radiation monitoring system consists of 31 channels that are located at selected places in and around the plant to detect and record the radiation levels and, if necessary, annunciate abnormal conditions. The areas where the gamma monitors are located are shown in Table 12.1-7. The control room panel provides instrumentation for indication, annunciation and recording of all 31 channels. Where necessary, local indication and alarm annunciation is provided in the area of the sensor.

The gamma area radiation monitoring system is shown as a functional block diagram in Figure 12.1-9. A typical channel consists of a gamma sensitive Geiger Muller (GM) detector, an indicator, alarm unit, power supply, shared multipoint recorder, and where necessary, a local auxiliary unit providing indication and audio/visual alarm status in the vicinity of the sensor.

Channels which are provided with auxiliary units are indicated in Table 12.1-7. Each auxiliary unit consists of a five decade readout meter (10-1 mr/hr to 104 mr/hr, a red high-high radiation light, and a high radiation horn which activates with the alarm light.

The control room log readout module associated with each channel has a similar dynamic range of five decades. It is provided with a 0-10 mv recorder output, a 0-50 mv computer output and a meter output required to operate field mounted indicators.

Control room readout modules channels 3, 4, 5 and 6 (CIS) are provided with a 4-20 ma dc output for containment isolation. Each sensor associated with CIS initiation is protected to allow continuous operation under the temperature, pressure and chemical conditions associated with LOCA containment environment.

12.1.4.3 Design Evaluation

Each channel provides five full decades of dynamic range. The energy response has been empirically determined by the manufacturer to be + 15 percent from 100 Kev to 1.5 Mev. Each detector is provided with a live zero to preclude spurious fail alarms when the background radiation level in which the detector is operating is less than 0.1 mr/hr or 1 mr/hr for channels 3, 4, 5 and 6. Natural background radiation and check source contribution hold the system in the non-failure condition. If for any reason the system stops responding to radiation a failure condition will be indicated on the readout module. A built-in anti-saturation circuit prevents the system readings from falling off full scale during overrange conditions.

Channels 3, 4, 5 and 6 of the area radiation monitoring system, and part of the CIS instrumentation have an energy response spectra as described by Figure 12.1-10. The source used in the calibration of this instrumentation was Cs^{137}. The energy response spectra indicates the ratio of indicated/true value to energy using a baseline calibration of Cs^{137} (0.662 Mev). The sensitivity of the remaining area radiation monitors is as indicated on Figure 12.1-11.

To prevent a loss of monitoring of an entire area of the plant due to a loss of power, each of the thirty-one channels is connected to a separate electronic power supply in conformance with the NRC non-common mode failure criteria. The four containment area radiation monitors which initiate CIS are fed from four Class 1E Instrument Buses (MA, MB, MC, and MD) with annunciator power on the uninterruptible 125 V DC system. All other monitors have power supplies fed from an interruptible source and are inoperative during a loss of off-site power until the non-essential load section of the Reactor Area Motor Control Centers (1A6 and 1B6) can be re-connected to the essential section of these motor control centers.

These four containment area radiation monitors identified by channel numbers 3, 4, 5 and 6 in Table 12.1-7 are designed as seismic Class I and can withstand LOCA containment environment for a period of at least 15 minutes after an accident. See Section 3.11 for a discussion of the environmental qualification of these components.

In addition to qualification under the LOCA environment conditions, the four containment monitors are physically and electrically separated from each other in accordance with the criteria set forth in IEEE 279, August 1968, and IEEE 308, November 1970. The output signals from the monitors (labeled MA, MB, MC and MD) feed the engineered safety features logic to make up the CIS. The monitors are located within the containment at 90 degree intervals along the containment vessel wall. The radiation monitors level alarm is set at 10 R/hr. A setpoint of 10 R/hr was selected based on response time for the high containment pressure setpoint of 5 psig since both signals feed the CIS circuitry.

Each of the 31 readout modules in the control room indicates three alarm conditions which operate according to the following sequence:

a) High-High Alarm-Reset Pushbutton/Light

Red light ON indicates high-high alarm or alarm circuit malfunction. Depressing the red pushbutton/light resets the high-high radiation alarm when the Function Selector Switch is in the OPERATE position and displays the alarm setpoint on the meter in ALARM position.

b) High Alarm-Reset Pushbutton/Light

Amber light ON indicates high alarm or alarm circuit malfunction. Depressing the amber pushbutton/light resets the high radiation alarm when the Function Selector Switch is in the OPERATE position and displays the alarm setpoint on the meter when the Function Selector Switch is in the ALARM position.

c) Fail/Safe-Check Source Pushbutton/Light

Blue light OFF indicates a module or detector failure. Depressing the blue pushbutton/light activates the check source on some detectors.

The Failure/Reset Alarm described above provides continuous monitoring of area radiation monitoring system integrity.

12.1.4.4 Testing and Inspection

An internal trip test circuit adjustable over the full range of the trip circuit is a design feature. The test signal is fed into the indicator and trip unit so that a meter reading is provided in addition to a real trip. Some monitors utilize a check source that is remotely controlled from the control room.

The function of the containment area monitors is discussed in Section 7.3. Environmental qualifications of the containment area radiation monitors are discussed in Section 3.11.

12.1.4.5 <u>High Range Containment Radiation Monitors</u>

In compliance with NUREG-0737, Item II.F.1, Attachment 3, permanently installed high range containment radiation monitors for post-accident monitoring of containment radiation are provided.

The conformance of the In-Containment High Range Radiation Monitoring System with the requirements of NUREG-0578, its clarifications and NUREG-0737 is detailed in Table 12.1-9. This table includes the criteria, safety evaluation and remarks, and applicable guidance from NUREG-0737, Item II.F.I.

12.1.4.5.1 Design Basis

a) Measurement and indication capability is provided in the range, from personnel safety levels to
10^8 R/hr.

b) Monitors are located to "view" a large segment of the containment free volume.

c) Safety related redundant instrumentation channels are provided to meet the single failure criteria.

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

e) The monitoring instrumentation inputs are from sensors that directly measure containment radiation and provide input only to the containment radiation monitors.

f) Testing and calibration requirements are specified in the plant technical specifications.

g) Continuous indication and recording of containment radiation is provided in the control room.

12.1.4.5.2 Design Description

The In-Containment High Range Monitoring System utilizes redundant detection channels. Each channel is composed of an in-containment ambient radiation detector transmitting to a control room rack mounted signal processor and recorder.

The two in-containment radiation detectors are located at the 90 ft. elevation at azimuth 0^0 and 180^0.

12.1.4.6 <u>Fuel Storage Criticality Monitoring</u>

On August 14, 1997, St. Lucie Units 1 & 2 were granted an exemption from the requirements of 10 CFR 70.24 concerning criticality accident monitors (References 8 & 9). This exemption was superceded when St. Lucie Plant elected to comply with a new regulation, 10 CFR 50.68.

In December of 1998 the NRC issued 10 CFR 50.68, Criticality Accident Requirements, a new regulation which gave licensees the option of complying with the requirements of 10 CFR 70.24 or 10 CFR 50.68(b). St. Lucie plant elected to forgo an existing exemption to 10 CFR 70.24 and to comply with 10 CFR 50.68(b) (Reference 11). 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."

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12.1.5 OPERATING PROCEDURES

Permanent shielding is installed wherever radioactive sources are expected to occur and wherever operating and maintenance functions are normally required. Both regular and special temporary radiation shields are available for use in all normal operational situations where radioactivity containing equipment must be transported or repaired.

Typically, quantities of lead sheets, and 2 percent antimony-enriched lead brick will be available inside or in close proximity to the radiation controlled area to be used for special temporary shielding. Solid hydrogenous sheets of virgin and borated poly material will also be available for special neutron shielding. Depending on the circumstances, temporary shielding can either be fabricated in the work shop or erected in situ using lead bricks and sheets if the conditions permit.

Procedures have been written that provide guidelines for the use of special temporary shielding in areas where extensive maintenance is required and the radiation levels are such that the estimated time of occupancy would cause the individual's exposure to approach or exceed the administrative guidelines. These procedures detail the type and amount of shielding to be used based on types of radiation, geometry of the radiation source, and the dose rates that are desired outside of the shielding.

Procedures have been written to ensure that regulatory and administrative exposure limits and guidelines are not exceeded. Daily or weekly exposure summary reports are posted detailing an individual's official plus estimated dose for the current quarter and year and allowable dose remaining before exceeding administrative exposure guidelines. Exposures are governed by procedures such as Radiation Work Permits.

12.1.6 TMI SHIELDING STUDY

12.1.6.1 Introduction

Following the requirement of NUREG-0737 Item II.B.2 "Plant Shielding", a design review of the St. Lucie Unit 1 plant shielding was performed. This assures safe personnel access to the vital equipment or areas required for mitigation or monitoring of an accident.

In compliance with Item II.B.2 of NUREG-0737, radiation source terms are specified, systems assumed to contain high levels of radioactivity as a result of a postulated accident are listed, vital areas requiring access are identified, and dose rates and doses in vital areas are presented.

12.1.6.2 Source Terms

Source terms used are consistent with the requirements of Section II.B.2 of NUREG-0737, which are based on the guidelines of Regulatory Guides 1.4, 1.7 and Standard Review Plan 15.6.5:

a) It was assumed that 100 percent of the core inventory of noble gases, 50 percent of the inventory of halogens and 1 percent of the inventory of other nuclides would be released to the reactor coolant water.

b) It was assumed that 100 percent of the core inventory of noble gases and 25 percent of the inventory of halogens would be released to the containment atmosphere.

c) Time and source dilution factors were applied where appropriate in determining source terms in various systems at different times following an accident.

12.1.6.3 Radioactive Systems

The systems identified as potentially containing high levels of radioactivity in a post-accident situation and which were considered in the shielding design review undertaken assure access to vital areas, are listed in Table 12.1-10. All other systems, such as the Chemical and Volume Control System, and the Waste Management System are not necessary for post-LOCA operation. Degassing of the Reactor Coolant System will be done using the Reactor Head Vent System (see Section 5.7) rather than the letdown portion of the Chemical and Volume Control System. The Waste Management System will be isolated and, therefore, not employed since radioactive leakages and drains will be routed back into the containment via the ECCS Leakage Collection and Return System (LCRS) as described in Section 11.2.2.1.

12.1.6.4 Vital Areas Requiring Occupancy/ Access

An extensive review was undertaken to identify vital areas of the plant to which personnel access following an accident must be assured. A vital area is defined as an area which may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. A list of these areas (with accompanying occupancy, dose rate, and dose information) appears in Table 12.1-11. As indicated in Table 12.1-11, in cases where the review revealed that high dose rates or accumulated dose would preclude access, means for remote operation, additional shielding or plant modifications were provided.

The result of the review process was to assure that access to vital areas could be accomplished consistent with NUREG-0737 requirements of: (1) less than 15 mrem/hr (averaged over 30 days) for areas requiring continuous occupancy, and (2) GDC-19 requirements of less than 5 rem for the duration of the accident for areas requiring irregular occupancy. Exposure to other operating EC289683 personnel performing necessary functions in vital areas was limited to 3 rem whole body and 18-3/4 rem extremities. At EPU conditions, these exposures will be limited to 4.4 rem whole body and 27.4 rem extremities.

12.1.6.5 Dose Rate and Dose Calculations

Dose rate calculations were performed in areas identified as vital areas, and along potential access routes. Time-dependent sources were determined as stated in Section 12.1.6.2, and appropriate geometry factors were applied to pipe and equipment of the systems identified in Section 12.1.6.3. The shielding effect of the equipment, fluid, and shield wall arrangement were considered. The effect of rebar, embedded plates, or any structural steel was neglected, which, when combined with the very conservative, "worst case" source terms, resulted in very conservative calculated dose rates.

Acceptance criteria for doses in vital areas considered the appropriate dose limit, required occupancy time, round-trip access route, required frequency of access and a safety margin.

Additionally, integrated doses were estimated for safety equipment required for post-accident operation to insure that GDC 4 requirements were met.
Acceptance criteria for equipment dose were either equipment specification, actual equipment qualification, or generic material damage data.

REFERENCES FOR SECTION 12.1

1. R. E. Uhrig (FPL) to D. G. Eisenhut (NRC) Re: St. Lucie Unit 1, Docket No. 50-335, NUREG-0578 Short Term Requirements, L-80-17 dated 1/11/80.

2. ORNL RSIC DLC-76, "SAILOR Coupled, Self-Shielded, 47-Neutron, 20-Gamma-Ray, P3 Cross Section Library for Light Water Reactors," July 1987.

3. LRA-80-701, "DOMINO Program Description," T. C. Chan and R. K. Disney, December 11, 1980.

4. RA-83-530, "MORQ Computer Program User's Manual," R. K. Disney, January 13, 1984 [Code renamed MCNPBQ].

5. ORNL RSIC CCC-200, "MCNP4A Monte Carlo N-Particle Transport Code System," December 1993.

6. ORNL RSIC DLC-105, "MCNPDAT MCNP Version 4 Standard Neutron Cross Section Data Library Based on ENDF/B-V," December 1992.

7. ORNL RSIC CCC-543, "TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport Version 2.7.3," May 1993.

8. L. A. Weins (NRC) to T. F. Plunkett (FPL), "ISSUANCE OF EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 70.24 ST. LUCIE UNITS 1 AND 2," dated 8/14/97.

9. J. A. Stall (FPL) to Document Control Desk (NRC), "Application for Exemption from 10 CFR 70.24: Request for Additional Information," L-97-171, July 10, 1997.

10. Safety Evaluation PSL-ENG-SENS-98-011, Revision 1, “On-Site Storage of Radioactive Materials in a Remote RCA,” March 1998.

11. St. Lucie Plant Management Action Item (PMAI) PM98-11-092.

TABLE 12.1-1
(Historical)

MAXIMUM GAMMA SPECTRA (γ/cm^2-sec)
OUTSIDE REACTOR VESSEL

E(mev)	Side.	Top	Bottom
10.00	1.40(+7)*	1.57(+3)	7.34(+3)
9.00	1.22(+8)	1.16(+4)	1.08(+7)
8.00	2.29(+8)	1.23(+4)	1.93(+7)
7.00	2.28(+8)	2.07(+4)	3.42(+7)
6.00	2.41(+8)	2.53(+4)	4.91(+7)
5.00	2.99(+8)	3.21(+4)	6.71(+7)
4.00	3.80(+8)	4.19(+4)	8.88(+7)
3.00	5.74(+8)	5.47(+4)	1.25(+8)
2.00	9.27(+8)	8.02(+4)	1.94(+8)
1.38	5.46(+8)	4.82(+4)	1.14(+8)
1.00	5.01(+8)	4.33(+4)	1.07(+8)
0.75	6.88(+8)	5.97(+4)	1.51(+8)
0.50	7.45(+8)	9.66(+4)	2.51(+8)
0.25	9.65(+8)	1.18(+5)	3.21(+8)

*Denotes power of ten.

TABLE 12.1-1A

PROTECTIVE COATINGS

DATA INTENTIONALLY DELETED

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TABLE 12.1-2
(Historical)

MAXIMUM NEUTRON SPECTRA (n/cm^2-sec)
OUTSIDE REACTOR VESSEL

E(mev)	Side	Top	Bottom
18	9.04(+3)*	Insignificant	1.09(-2)
14	5.22(+5)		8.08(-1)
10	1.34(+7)		1.35(+1)
8	2.88(+7)		1.00(+1)
6	5.13(+7)		1.06(+1)
4	4.72(+7)		1.16(+1)
3	3.30(+7)		1.33(+1)
2	3.49(+7)		1.17(+1)
1	3.99(+7)		1.53(+1)
0.33	4.67(+7)		3.42(+1)

*Denotes power of ten.

TABLE 12.1-3
(Historical)

REACTOR COOLANT ACTIVITY
SHUTDOWN CONDITIONS (70 F)

Nuclide	Half Life	Specific Activity (µCi/cc) Anticipated Operational Occurrence (1.0% Failed Fuel)	Specific Activity (µCi/cc) Normal Operation (.1% Failed Fuel)	Nuclide	Half Life	Specific Activity (µCi/cc) Anticipated Operational Occurrence (1.0% Failed Fuel)	Specific Activity (µCi/cc) Normal Operation (0.1% Failed Fuel)
H-3	12.3y	0.182	0.148	I-I33	21h	7.81	0.781
Br-84	32m	6.43(-2)*	6.43(-3)	Xe-133	5.3d	250.0	25.0
Kr-85m	4.4h	2.05	0.205	Te-134	42m	3.62(-2)	3.62(-3)
Kr-85	10.8y	1.22	1.22(-1)	I-134	52m	0.855	8.55(-2)
Kr-87	76m	1.12	1.12(-1)	Cs-134	2.1y	0.138	1.38(-2)
Kr-88	2.8h	3.58	0.358	I-135	6.7h	3.72	0.372
Rb-88	18m	3.52	0.352	Xe-135	9.2h	10.4	1.04
Rb-89	15m	8.83(-2)	8.83(-3)	Cs-136	13d	3.52(-2)	3.52(-3)
Sr-89	51d	7.0(-3)	7.0(-4)	Cs-137	30y	0.441	4.41(-2)
Sr-90	28.8y	3.6(-4)	3.6(-5)	Xe-138	17m	0.497	4.97(-2)
Y-90	64h	1.41(-3)	1.41(-4)	Cs-138	32m	0.952	9.52(-2)
Sr-91	9.7h	4.9(-3)	4.9(-4)	Ba-140	12.8d	8.42(-3)	8.42(-4)
Y-91	59d	0.153	1.53(-2)	La-140	40.2h	8.07(-3)	8.07(-4)
Mo-99	67h	2.80	0.280	Pr-143	13.6d	8.05(-3)	8.05(-4)
Ru-103	39.6d	5.7(-3)	5.7(-4)	Ce-144	285d	5.7(-3)	5.7(-4)
Ru-106	367d	3.42(-4)	3.42(-5)	Co-60**	5.2y	7.16(-4)	7.16(-4)
Te-129	67m	3.46(-2)	3.46(-3)	Fe-59**	45d	2.94(-5)	2.94(-5)
I-129	1.7(7)y	9.95(-8)	9.95(-9)	Co-58**	71d	6.43(-3)	6.43(-3)
I-131	8.0d	5.48	0.548	Mn-54**	312d	3.8(-5)	3.8(-5)
Xe-131r	12d	2.04	0.204	Cr-51**	27d	5.24(-3)	5.24(-3)
Te-132	78h	0.455	4.55(-2)	Zr-95**	65d	1.29(-6)	1.29(-6)
I-132	2.3h	1.41	0.141				

* Denotes Power of ten
** Corrosion products

TABLE 12.1-3A
(Historical)

ESTIMATED MAXIMUM NUCLIDE ACTIVITIES IN OUTSIDE STORAGE TANKS

Isotope	Refueling Water Tank (Ci)	Tank (Ci)	Primary Water Storage Tank (Ci)		Condensate Storage	
	1% ff	0.1% ff	1% ff	0.1% ff	1% ff	0.1% ff
Kr-85	1.55(-7)*	1.55(-8)	6.86(1)	6.86	-	-
Sr-89	9.42(-3)	9.42(-4)	9.97(-6)	9.97(-7)	5.21(-4)	5.21(-5)
Sr-90	7.65(-4)	7.65(-5)	1.02(-6)	1.02(-7)	2.26(-5)	2.26(-6)
Sr-91	2.82(-11)	2.82(-12)	-	-	3.32(-5)	3.32(-6)
Y-90	9.65(-2)	9.65(-3)	2.50(-6)	2.50(-7)	1.13(-11)	1.13(-12)
Y-91	2.13(1)	2.13	3.55(-5)	3.55(-6)	8.10(-3)	8.10(-4)
Mo-99	1.00(1)	1.00	5.71(-3)	5.71(-4)	1.97(-2)	1.97(-3)
Ru-103	7.97(-2)	7.97(-3)	7.03(-5)	7.03(-6)	2.61(-4)	2.61(-5)
Ru-106	5.59(-3)	5.59(-4)	8.63(-6)	8.63(-7)	2.49(-5)	2.49(-6)
Te-129	3.36(-1)	3.36(-2)	-	-	1.48(-3)	1.48(-4)
I-129	1.68(-8)	1.68(-9)	-	-	7.80(-9)	7.80(-10)
I-131	2.89(-1)	2.29(-2)	1.24(-6)	1.24(-7)	9.48(-2)	9.48(-3)
Xe-131n	1.84(1)	1.84	1.68(1)	1.68	-	-
Te-132	2.07(-1)	2.77(-2)	1.52(-4)	1.52(-5)	3.69(-3)	3.69(-4)
I-132	2.95(-3)	2.95(-4)	-	-	1.13(-2)	1.13(-3)
I-133	3.07(-5)	3.07(-6)	3.9(-10)	3.90(-11)	1.82(-2)	1.82(-3)
Xe-133	7.18(1)	7.18	5.22(2)	5.22(1)		
Cs-134	2.30(1)	2.30	3.69(-3)	3.69(-4)	1.44(-6)	1.44(-6)
I-135	5.58(-12)	5.52(-12)	-	-	2.82(-3)	2.82(-4)
Xe-135	3.60(-6)	3.60(-7)	9.12(-10)	9.12(-11)	-	-
Cs-136	2.46	2.46(-1)	1.56(-4)	1.56(-5)	2.73(-5)	2.73(-6)
Cs-137	7.44(1)	7.44	1.25(-2)	1.25(-3)	1.05(-2)	1.05(-3)
Xe-138	-	-	-	-	-	-
Cs-138	-	-	-	-	8.79(-5)	8.79(-5)
Ba-140	5.88(-3)	5.88(-4)	3.66(-6)	3.66(-7)	3.47(-2)	3.47(-3)
La-140	6.72(-2)	6.72(-3)	8.30(-7)	8.30(-8)	2.11(-4)	2.11(-5)
Pr-143	1.21(-1)	1.21(-2)	3.77(-5)	3.77(-6)	5.29(-5)	5.24(-6)
Ce-144	9.23(-2)	9.23(-3)	1.39(-4)	1.39(-5)	3.53(-5)	3.53(-6)
Co-60	1.58(-2)	1.58(-2)	1.99(-4)	1.99(-4)	2.04(-4)	2.04(-5)
Fe-59	6.49(-4)	6.49(-4)	3.91(-6)	3.91(-6)	5.55(-5)	5.55(-5)
Co-58	1.42(-1)	1.42(-1)	1.08(-3)	1.08(-3)	4.07(-5)	4.07(-5)
Mn-154	8.37(-3)	8.37(-4)	9.40(-6)	9.40(-6)	3.65(-4)	3.65(-4)
Cr-51	1.16(-1)	1.16(-1)	4.96(-4)	4.96(-4)	2.03(-4)	2.03(-4)
Zr-95	2.85(-5)	2.85(-5)	2.10(-7)	2.10(-7)	1.55(-11)	1.55(-11)
Dose rate at tank surface (mrem/hr)	16.0	4.6	1.05	0.11	0.339	0.071
Dose rate at site boundary (mrem/yr)	3.78	0.79	1.63(-l)	3.4(-2)	1.19(-2)	2.5(-3)

* Numbers in denote power of 10

TABLE 12.1-3A (Cont'd)

Assumptions:

Core power level = 2700 Mwt
Primary-secondary leakage rate = 20 gpd
Steam generator blowdown rate (continuous)= 0.14 gpm
Total BMS waste generated = 843,000 gal
Fuel pool purification system flow rate = 40 gpm
Condensate tank activity = 15% directly from hotwell

Note: Three steam generator blowdown treatment monitor tanks, which are located outside, may also contain radioactivity during normal operation.

TABLE 12.1-4
(Historical)

FISSION PRODUCT GAMMA SOURCE IN CONTAINMENT BUILDING (MEV/SEC)
FOLLOWING MAXIMUM HYPOTHETICAL ACCIDENT

	Energy Interval (Mev)						
Time	0.1 - 0.4	0.4 - 0.9	0.9 - 1.35	1.35 - 1.8	1.8 - 2.2	2.2 - 2.6	>2.6
0	2.11 (18)*	1.26 (19)	5.00 (18)	9.39 (18)	6.10 (18)	4.39 (18)	1.99 (18)
0.5 hr	2.01 (18)	1.11 (19)	4.56 (18)	3.44 (18)	3.17 (18)	3.56 (18)	2.54 (17)
1 hr	1.94 (18)	9.81 (18)	3.98 (18)	3.05 (18)	2.14 (18)	2.95 (18)	9.32 (16)
2 hr	1.84 (18)	7.78 (18)	3.22 (18)	2.44 (18)	1.47 (18)	2.09 (18)	3.00 (16)
8 hr	1.43 (18)	4.39 (18)	1.48 (18)	1.08 (18)	4.76 (17)	3.95 (17)	1.07 (16)
24 hr	7.93 (17)	3.05 (18)	5.95 (17)	4.27 (17)	1.59 (17)	7.08 (16)	1.76 (14)
1 wk	2.10 (17)	7.32 (17)	1.12 (17)	1.20 (17)	4.03 (16)	2.13 (16)	1.19 (14)
1 mo	2.81 (16)	8.76 (16)	2.17 (15)	1.66 (16)	1.00 (15)	1.27 (15)	3.78 (13)
2 mo	4.10 (15)	5.39 (16)	3.68 (14)	2.95 (15)	5.29 (14)	1.95 (14)	1.22 (13)
4 mo	5.15 (14)	2.98 (16)	1.36 (14)	3.34 (14)	4.42 (14)	2.34 (13)	5.29 (12)

*Denotes power of ten

Note: Gamma source term values from this table up to 720 hours post-release remain applicable for operation at 2700 Mwt with extended burnup.

TABLE 12.1-5
(Historical)
WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING ANTICIPATED OPERATIONAL OCCURRENCES (1.0% failed fuel) (µCi/cc)

Nuclide	CVCS 6 WMS I	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	wms 15	WMS 18,19,20,21
H-3	1.2(-1)*	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)
Br-84	4.11(-3)	4.11(-3)	4.11(-6)	4.11(-6)	0	0	0
Kr-85m	1.57(0)	1.57(0)	1.57(0)	7.85(-1)	0	0	0
Kr-85	1.14(0)	1.14(0)	1.14(0)	5.7(-1)	4.72(-1)	4.72(-1)	2.36(-1)
Kr-87	8.0(-1)	8.0(-1)	8.0(-1)	4.0(-1)	0	0	0
Kr-88	2.68(0)	2.68(0)	2.68(0)	1.34(0)	0	0	0
Rb-88	5.96(-1)	5.96(-1)	5.96(-1)	5.96(-1)	0	0	0
Rb-89	9.02(-3)	9.02(-3)	9.02(-3)	9.02(-3)	0	0	0
Sr-89	3.8(-3)	3.8(-3)	3.8(-5)	3.8(-5)	3.36(-5)	3.36(-6)	1.18(-8)
Sr-90	1.84(-1)	1.84(-1)	1.84(-3)	1.84(-3)	1.84(-3)	1.84(-4)	9.2(-7)
Y-90	6.71(-4)	6.71(-4)	6.71(-4)	6.71(-4)	6.58(-5)	6.58(-6)	3.29(-8)
Sr-91	1.69(-3)	1.69(-3)	1.69(-5)	1.69(-5)	3.36(-12)	3.36(-13)	1.18(-15)
Y-91	2.48(-2)	2.48(-2)	2.48(-2)	2.48(-2)	2.23(-2)	2.23(-3)	1.12(-5)
Mo-99	1.21(0)	1.21(0)	1.21(0)	1.21(0)	1.25(-1)	1.25(-2)	6.25(-5)
Ru-103	3.24(-3)	3.24(-3)	3.24(-4)	3.24(-4)	2.77(-4)	2.77(-5)	1.39(-7)
Ru-106	1.67(-4)	1.67(-4)	1.67(-5)	1.67(-5)	1.64(-5)	1.64(-6)	8.2(-9)
Te-129	1.6(-2)	1.6(-2)	1.6(-3)	1.6(-3)	0	0	0
I-129	4.0(-8)	4.0(-8)	4.0(-11)	4.0(-11)	4.0(-11)	4.0(-14)	2.0(-16)
I-131	1.7(0)	1.7(0)	1.7(-3)	1.7(-3)	7.84(-4)	7.84(-7)	3.92(-9)
Xe-131m	1.18(0)	1.18(0)	1.18(0)	5.9(-1)	2.89(-1)	2.89(-1)	5.78(-2)
Te-132	2.31(-1)	2.31(-1)	2.31(-3)	2.31(-3)	3.39(-4)	3.39(-5)	1.2(-7)
I-132	3.46(-1)	3.46(-1)	3.46(-4)	3.46(-4)	0	0	0
I-133	1.88(0)	1.88(0)	1.88(-3)	1.88(-3)	1.51(-6)	1.51(-9)	7.55(-12)
Xe-133	1.97(+2)	1.97(+2)	1.97(+2)	9.85(+1)	2.49(+1)	2.49(+1)	4.98(0)
Te-134	4.61(-3)	4.61(-3)	4.61(-4)	4.61(-4)	0	0	0
1-134	6.95(-2)	6.95(-2)	6.95(-5)	6.95(-5)	0	0	0
Cs-134	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-3)	1.13(-5)

TABLE 12.1-5 (Cont'd)

Nuclide	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4*5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
N-135	6.39(-1)	6.39(-1)	6.39(-4)	6.39(-4)	0	0	0
Xe-135	6.49(0)	6.49(0)	6.49(0)	3.25(0)	0	0	0
Cs-136	3.47(-2)	3.47(-2)	3.47(-2)	3.47(-2)	2.15(-2)	2.15(-4)	1.08(-6)
Cs-137	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-3)	4.79(-5)
Xe-138	3.4(-1)	3.4(-1)	3.4(-1)	1.7(-1)	0	0	0
Cs-138	3.48(-1)	3.48(-1)	3.48(-1)	3.48(-1)	0	0	0
Ba-140	4.64(-3)	4.64(-3)	4.64(-5)	4.64(-5)	2.85(-5)	2.85(-6)	1.93(-8)
La-140	4.58(-3)	4.58(-3)	4.58(-4)	4.58(-4)	1.1(-4)	1.1(-5)	5.5(-8)
Pr-143	4.01(-3)	4.01(-3)	4.01(-4)	4.01(-4)	2.53(-4)	2.53(-5)	1.27(-7)
Ce-144	2.57(-3)	2.57(-3)	2.57(-4)	2.57(-4)	2.46(-4)	2.46(-5)	1.23(-7)
Co-60**	8.64(-4)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	4.32(-7)
Fe-59**	3.54(-5)	3.54(-6)	3.54(-6)	3.54(-6)	3.08(-6)	3.08(-6)	1.54(-8)
Co-58**	7.74(-3)	7.74(-4)	7.74(-4)	7.74(-4)	7.14(-4)	7.14(-4)	3.57(-6)
Mn-54**	4.57(-5)	4.57(-6)	4.57(-6)	4.57(-6)	4.47(-6)	4.47(-6)	2.24(-8)
Cr-51**	6.32(-3)	6.32(-4)	6.32(-4)	6.32(-4)	5.02(-4)	5.02(-4)	2.51(-6)
Zr-95	1.56(-7)	1.56(-8)	1.56(-8)	1.56(-8)	1.42(-8)	1.42(-8)	7.1(- 11)

*Denotes power of ten

**Corrosion products

TABLE 12.1-6
(Historical)

WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING NORMAL OPERATIONS (0.1% failed fuel) (μCi/cc)

Nuclide	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS
H-3	8.4(-2)*	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)
Br-84	4.11(-4)	4.11(-4)	4.11(-7)	4.11(-7)	0	0	0
Kr-85m	1.57(-1)	1.57(-1)	1.57(-1)	7.85(-2)	0	0	0
Kr-85	1.14(-1)	1.14(-1)	1.14(-1)	5.7(-2)	4.72(-2)	4.72(-2)	2.36(-2)
Kr-87	8.0(-2)	8.0(-2)	8.0(-2)	4.0(-2)	0	0	0
Kr-88	2.68(-1)	2.68(-1)	2.68(-1)	1.34(-1)	0	0	0
Rb-88	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	0	0	0
Rb-89	9.02(-4)	9.02(-4)	9.02(-4)	9.04(-4)	0	0	0
Sr-89	3.8(-4)	3.8(-4)	3.8(-6)	3.8(-6)	3.36(-6)	3.36(-7)	1.18(-9)
Sr-90	1.84(-2)	1.84(-2)	1.84(-4)	1.84(-4)	1.84(-4)	1.84(-5)	9.2(-8)
Y-90	6.71(-5)	6.71(-5)	6.71(-5)	6.71(-5)	6.58(-6)	6.58(-7)	3.29(-9)
Sr-91	1.69(-4)	1.69(-4)	1.69(-6)	1.69(-6)	3.36(-13)	3.36(-14)	1.18(-16)
Y-91	2.48(-3)	2.48(-3)	2.48(-3)	2.48(-3)	2.23(-3)	2.23(-4)	1.12(-6)
Mo-99	1.21(-1)	1.21(-1)	1.21(-1)	1.21(-1)	1.25(-2)	1.25(-3)	6.25(-6)
Ru-103	3.24(-4)	3.24(-4)	3.24(-5)	2.77(-5)	2.77(-5)	2.77(-6)	1.39(-8)
Ru-106	1.67(-5)	1.67(-5)	1.67(-6)	1.67(-6)	1.64(-6)	1.64(-7)	8.2(-10)
Te-129	1.6(-3)	1.6(-3)	1.6(-4)	1.6(-4)	0	0	0
I-129	4.0(-9)	4.0(-9)	4.0(-12)	4.0(-12)	4.0(-12)	4.0(-15)	2.0(-17)
I-131	1.7(-1)	1.7(-1)	1.7(-4)	4.7(-4)	7.84(-5)	7.84(-8)	3.92(-10)
Xe-131m	1.18(-1)	1.18(-1)	5.9(-2)	2.89(-2)	2.89(-2)	2.89(-2)	5.78(-3)
Te-132	2.31(-2)	2.31(-2)	2.31(-4)	2.31(-4)	3.39(-5)	3.39(-6)	1.2(-8)
I-132	3.46(-2)	3.46(-2)	3.46(-5)	3.46(-5)	0	0	0
I-133	i.88(-1)	1.88(-1)	i.88(-4)	1.88(-4)	1.51(-7)	1.51(-10)	7.55(-13)
Xe-133	1.97(+1)	1.97(+1)	1.97(+1)	9.85(0)	2.49(0)	2.49(0)	4.98(-1)
Te-134	4.61(-4)	4.61(-4)	4.61(-5)	4.61(-5)	0	0	0
I-134	6.95(-3)	6.95(-3)	6.95(-6)	6.95(-6)	0	0	0
Cs-134	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-4)	1.13(-6)

TABLE 12.1-6 (Cont'd)

Nuclides	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
I-135	6.39(-2)	6.39(-2)	6.39(-5)	6.39(-5)	0	0	0
Xe-135	6.49(-1)	6.49(-1)	6.49(-1)	3.25(-1)	0	0	0
Cs-135	3.47(-3)	3.47(-3)	3.47(-3)	3.47(-3)	2.15(-3)	2.15(-5)	1.08(-7)
Cs-137	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-4)	4.79(-6)
Xe-138	3.4(-2)	3.4(-2)	3.4(-2)	1.7(-2)	0	0	0
Cs-138	3.48(-2)	3.48(-2)	3.48(-2)	3.48(-2)	0	0	0
Ba-140	4.64(-4)	4.64(-4)	4.64(-6)	4.64(-6)	2.85(-6)	2.85(-7)	1.93(-9)
La-140	4.58(-4)	4.58(-4)	4.58(-5)	4.58(-5)	1.1(-5)	1.1(-6)	5.5(-9)
Pr-143	4.01(-4)	4.01(-4)	4.01(-5)	4.01(-5)	2.53(-5)	2.53(-6)	1.27(-8)
Ce-144	2.57(-4)	2.57(-4)	2.57(-5)	2.57(-5)	2.46(-5)	2.46(-6)	1.23(-8)
Co-60**	8.64(-4)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	4.32(-7)
Fe-59**	3.54(-5)	3.54(-6)	3.54(-6)	3.54(-6)	3.08(-6)	3.08(-6)	1.54(-8)
Co-58**	7.74(-3)	7.74(-4)	7.74(-4)	7.74(-4)	7.14(-4)	7.14(-4)	3.57(-6)
Mn-54**	4.57(-5)	4.57(-6)	4.57(-6)	4.57(-6)	4.47(-6)	4.47(-6)	2.24(-8)
Cr-51**	6.32(-3)	6.32(-4)	6.32(-4)	6.32(-4)	5.02(-4)	5.02(-4)	2.51(-6)
Zr-95**	1.56(-7)	1.56(-8)	1.56(-8)	1.56(-8)	1.42(-8)	1.42(-8)	7.1(-11)

*Denotes power of ten
**Corrosion products

TABLE 12.1-6A

WMS COMPONENT SHIELDING DATA
(Data Corresponds To Initial Plant Operations)

Component	Location	Radioactive Volume	Source Strength (μCi/cc)	Max (3) Estimated Contact Dose Rate (r/hr)	Max Surface Dose Rate Outside Shield (mr/hr)
Equipment Drain Tank	Fig. 1.2-12 South West Corner	1000 gal	Table 12.1-5 WMS 3	5	2.5
Aerated Waste Storage Tank	Fig. 1.2-12 North West Corner	40,000 gal	Table 12.1-5 WMS 3		
Chemical Drain Tank	Fig. 1.2-12 South West Corner	1000 gal	Table 12.1-5 WMS 6	5	2.5
Laundry Tank	Fig. 1.2-12 between RAK-RAJ & RA4	Experience shows negligible activity.			
Pre-concentrator Ion Exchanger (5)	Fig. 1.2-13 RAB between RAI & RA3	32 ft^3	Table 11.5-4 WMS Pre Concentrator	10	2.5
Preconcentrator Filters (4)	Fig. 1.2-12 North of RA5 between RAE & RAD	- (2)	Table 11.5-6 WMS Pre Concentrator	10	2.5
Laundry Filter (4)	Fig. 1.2-12 On RA4, between RAK & RAJ	- (2)	Table 12.1-5 WMS 1 0	(less than waste filter)	2.5
Waste Concentrator (7)	Fig. 1.2-13 RAD & Sect. B-11		Table 11.5-2 Waste Concentration	1	2.5
Spent Resin Tank	Fig. 1.2-12 between RA3-RA4 & RAB-RAA	3200 gal Purification	Table 11.5-4	100	2.5

TABLE 12.1-6A (Cont'd)

Component	Location	Radioactive Volume	Source Strength (µCi/cc)	Max (3) Estimated Contact Dose Rate (r/hr)	Max Surface Dose Rate Outside Shield (mr/hr)
Drumming Tanks	Fig. 1.2-13 South East Corner	55 gal each	Table 11.5-4 Purification	10	2.5
Flash Tank	Fig. 1.2-13 RAD-RAZ	400 gal	Table 12.1-5 WMS-5	2	2.5
Hold-up Tanks	Fig. 1.2-12 between RAH-RAF	40,000 gal each	Table 12.1-5 WMS-8	5	<2.5
Boric Acid Condensate Ion Exchanger (5), (7)	Fig. 1.2-3 North of RA3 on RAB	$32 ft.^3$	Table 12.1-5 10xWMS-18	20	2.5
Boric Acid Condensate Tanks (7)	Fig. 1.2-12	7300 gal South East Corner	Table 12.1-5 WMS-18	0.0025	2.5 (No shield)
Boric Acid Holding Tank (7)	Fig. 1.2-12	2400 gal RAC-RA3	Table 11.2-7 BRS-16	3	2.5
Waste Condensate Tanks	Fig. 1.2-12 between RA4 and RAC-RAH	1725 gal	See Note 6 WMS 14	0.0025	2.5 (No Shield)
Waste Filter (4)	Fig. 1.2-12 On RA4 between RAI-RAH	-(2)	See Note 6 WMS 3 & 12	10	2.5
Waste Ion Exchangers (5)	Fig. 1.2-13 North of RA3 East of RAC Drumming Station	 2 @ $32 ft.^3$ 2 @ $50 ft.^3$	See Note 6 WMS 13	10	2.5

TABLE 12.1-6A (Cont'd)

Component	Location	Radioactive Volume	Source Strength (μCi/cc)	Max (3) Estimated Contact Dose Rate (r/hr)	Max Surface Dose Rate Outside Shield (mr/hr)
Gas Decay Tanks	Fig. 1.2-12 North East Corner	144 ft^3 each	Table 11.3-1 Position No.	10 8	<2.5
Gas Surge Tank	Fig. 1.2-12 North of RA2-RAC	10 ft^3	Table 11.3-1 Position No.	6 6	2.5
Gas Lines & & Compressors	Fig. 1.2-12 North East Corner		Table 11.3-1	~5	2.5

NOTES:

(1) All ion exchanger, and filter DF's were conservatively assumed to be equal to CVCS ion exchanger and filter DF's respectively.

(2) No Volume assumed - a point source was employed for calculation.

(3) Contact doses were not calculated. There are estimates based on extrapolation to zero shield thickness of the calculations performed to design the adequate shield thickness.

(4) Activity calculated by assuming 100 percent retention of all corrosion products listed in the given table column for one core cycle.

(5) Activity calculated by assuming 100 percent retention of all fission products in the given table column, for one core cycle.

(6) Table 11.2-12 used to contain source strength for these items; however, this table was deleted from the UFSAR in Amendment 1. Reference to this table is kept here for historical purposes.

(7) No longer used.

TABLE 12.1-7

AREA RADIATION MONITORING SYSTEM

Channel No.	Location and Figure No.	Mr/hr Range[1]	Local Alarm Indicator
(1)	Control room Figure 1.2-17		None
(2)	Operating deck area Figure 12.1-2A		1
(3)	*CIS monitor Figure 12.1-2A		None
(4)	*CIS monitor Figure 12.1-2A		None
(5)	*CIS monitor Figure 12.1-2A		None
(6)	*CIS monitor Figure 12.1-2A		None
(7)	Fuel pool area Figure 12.1-4		1
(8)	Refueling canal area Figure 12.1-4		1
(9)	Fuel pool pump area Figure 12.1-4		1
(10)	Boric acid preconcentrator filter area Figure 12.1-3A		1
(11)	Waste filter area Figure 12.1-3A		1
(12)	Laundry filter area Figure 12.1-3A		1
(13)	Waste gas compressor area Figure 12.1-3A		1
(14)	Charging pump area Figure 12.1-3A		1
(15)	Holdup drain pump area Figure 12.1-3A		1
(16)	Sample room area Figure 12.1-3B		1
(17)	Ion exchanger valve area Figure 12.1-3B		1
(18)	Ion exchanger valve area Figure 12.1-3B		1

TABLE 12.1-7 (Cont'd)

Channel No.	Location and Figure No.	Mr/hr Range(1)	Local Alarm Indicator
(19)	Drumming station area Figure 12.1-3B		1
(20)	Purification filter area Figure 12.1-3B		1
(21)	Spent resin tank area Figure 12.1-3A		1
(22)	ECCS equipment area Figure 12.1-3A		1
(23)	Decontamination area Figure 12.1-3B		1
(24)	HVAC room Figure 12.1-3C		2
(25)	Chemical drain pump area Figure 12,1-3A		1
(26)	Flash tank pump Figure 12.1-3B		1
(27)	Boronometer enclosure Figure 12.1-3B		1
(28)**	New fuel storage area Figure 12.1-4		1
(29)**	Aerated waste storage area Figure 12.1-3A		1
(30)**	Boric acid concentrator area Figure 12.1-3B		2
(31)**	Fuel pool filter area Figure 12.1-4		1

* Asterisk indicates the area monitors that are classified as Class 1E electrical equipment and supply the containment isolation signal (CIS). For further discussion of the ESFAS refer to Section 7.3.

** Channel numbers are specifically for identification of sensors on the referenced figures. Sensor identification on P & ID's and CWD's is provided by the notation RE 1-31 except for channels 2, 28, 29, 30 and 31 which are labeled RE-36, 37, 38, 39 and 2 respectively.

1 Instrument ranges are selected in accordance with standard engineering practices.

TABLE 12.1-8
(Historical)

OCCUPATION RADIATION EXPOSURE BUDGET FOR
THE REACTOR VESSEL HEAD NEUTRON STREAMING
SHIELD (BORATED WATER BAGS)
(Assumed 80% Plant Availability Factor)

		Avg. Neutron Dose Rate (mRem/hr)	Avg. Gamma Dose Rate (mr/hr)	Estimated Exposure Time (mRem-hr/wk)	Exposure (man-rem/yr)
A.	OUTSIDE CONTAINMENT				
1.	No Shield	2.5	2.5	5	1.2
2.	Shield	0.5	0.5	5	0.21
B.	INSIDE CONTAINMENT				
1.	Operating Floor - No Shield	2500	500	0.25	39
	- Shield	100	100	0.25	2.6
2.	Other Areas - No Shield	100	50	0.5	3.9
	- Shield	25	50	0.5	2.0
3.	Refueling, Removal & Replacement of Shield	0	33	18	0.60
4.	Stored Activated Support Structure [(a)]	0	0.5	770	0.72(b)
C.	ESTIMATED MAN-REM SAVED DUE TO SHIELD	-	-	-	38

a) Assumes 4 people present for 2 weeks
b) 15 days only
c) The dose rates for item B.3 & B.4 of this table were determined when St. Lucie 1 used temporary shielding of the vessel area with borated water bags. The addition of a Permanent Cavity Seal/Shield ring (beginning of Fuel Cycle 13) eliminated the removal and replacement operation of the water bags. Therefore, the dose rates and doses for these items above are conservative.
d) The dose rates for item B.3 & B.4 of this table were determined when St. Lucie 1 used temporary shielding of the vessel area with borated water bags. The addition of a Permanent Cavity Seal/Shield ring (beginning of Fuel Cycle 13) eliminated the removal and replacement operation of the water bags. Therefore, the dose rates and doses for these items above are conservative.

TABLE 12.1-9

HIGH RANGE CONTAINMENT RADIATION MONITORS
CONFORMANCE WITH TMI RELATED CRITERIA

Design Criteria	Safety Impact/Remarks	NUREG-0737 Reference
Redundant monitors located in widely separated spaces in the containment	Location 90' elev 0^0, 180^0 above steam generators	Position, Changes #2 Clarification #1, #3 Table 11.F.1-3
Safety related instruments and displays	Safety related due to environmental qualification, redundancy and continuity of power during accident and reference to RG 1.97 Rev 2	Position, Changes #5 References, Table 11.F.1-3
Energy response of 60 key to 3MeV with linearity of $\pm$20% in range of 100 key to 3MeV	$\pm$20% linearity requirement new - no problem foreseen	Changes #4 Clarification #5 Table 11.F.1-3
Radiation range of 1R/hr to 10^7 R/hr	Gamma-only detector range at 1 R/hr to 10^8 R/hr	Position, Changes #1, Clarification #2, Table 11.F.1-3
Calibration		
-Pre-Installation (for replacement detectors)	Certify for at least one point per decade of range between 1 R/hr and 10^3 R/hr	Table 11.F.1-3- Special Environmental Qualifications
-Post-Installation (For in-situ use)	Electronic calibration Corresponding to 10^5R/hr for all range decades above 10 R/hr. Use a check-source giving a continuous reading of 1 R/hr (approximate value, each detector unique), to calibrate below 10 R/hr.	Change #6, Table 11.F.1-3-Special Calibration

TABLE 12.1-10

SYSTEMS POTENTIALLY CONTAINING HIGH LEVELS OF RADIOACTIVE MATERIALS

Containment Spray System

Safety Injection System

- Low Pressure Safety Injection
- High Pressure Safety Injection

Shutdown Cooling System

Post Accident Sampling Systems*
- Liquid Sampling
- Hydrogen Analyzer

Ventilation Systems
- Shield Building Ventilation System
- Control Room Emergency Ventilation System
- ECCS Area Ventilation System

Containment Building

* The PASS is no longer used

TABLE 12.1-11

AREAS IDENTIFIED IN SHIELDING REVIEW AS REQUIRING ACCESSIBILITY FOLLOWING AN ACCIDENT

	AREA	LOCATION	OCCUPANCY REQUIREMENTS	MAXIMUM DOSE RATE AND DOSE	REMARKS
(1)	Control Room	RAB EL. 62.00'	Continuous	<15 mrem/hr 1 hr. after accident <5 rem for duration of accident	
(2)	Technical Support Center	RAB EL. 62.00' Control Room Envelope	Continuous	<15 mrem/hr 1 hr. after accident <5 rem for duration of accident	Units 1 and 2 share the TSC
(3)	Valve Station: LPSI Pump Suction Isolation Valves V3432, V3444 LPSI Pump Disch. Isolation Valves V3206, V3207	RAB EL. -10.0' ECCS Room	Two men for 10 minutes each	>100 rem/hr- 1 hr. after accident ~17 rem per person (Historical)	Manual valves are fitted with motor operators operated from the Control Room. Therefore, access will no longer be required.
(4)	Valve Station: Containment Spray Isolation Valve I-MV-07-3A I-MV-07-3B	RAB EL. 19.50' Penet. Room	One man for 10 minutes	10-100 rem/hr. 1 hr. after accident ~10 rem per person (Historical)	Same as item (3) Formerly I-V07162 Formerly I-V07165
(5)	Valve Station: Shutdown Cooling Heat Exchanger Inlet Isolation- V3452, V3453 Outlet Isolation- V3456,V3457 I-MV-03-2	RAB EL. -0.50' SDC Heat Ex. Corridor	One man for 10 minutes	10-100 rem/hr- 1 hr. after accident ~10 rem per person (Historical)	Same as item (3) Formerly I-V-07008
(6)	Valve Station: Shutdown Cooling Warm-up Valve I-MV-03-1A I-MV-03-1B	RAB EL. 19.50' Electrical Penet. Room	One man for 5 minutes	10-100 rem/hr. 1 hr. after accident ~10 rem per person (Historical)	Installed new motor operated valves operated from the Control Room. No access req'd. V 3400 Locked Open V 3484 Locked Open

TABLE 12.1-11 (Cont'd)

	AREA	LOCATION	OCCUPANCY REQUIREMENTS	MAXIMUM DOSE RATE AND DOSE	REMARKS	
(7)	Valve Station: CVCS Charging Aux. HPSI HDR Crosstie V2340	RAB EL. -0.50' Charging Pump Cubicles	One man for 5 minutes to establish HPSI Hot Leg Injection path to Pressurizer	>100 mrem/hr. 1 hr. after accident ~17 rem per person	Install valve stem extension, to operate valve from outside of pump cubicles.	EC289683
(8)	Post-Accident* Sampling System (PASS)	RAB EL. 19.50' East Corridor	Infrequent access to collect grab samples	<100 mrem/hr 1 hr. after accident <100 mrem per visit	Refer to Subsection 9.3.7.1 for a description of the PASS.	

Studies Subsequent to TMI Shielding Study: EC289683

AREA	LOCATION	OCCUPANCY REQUIREMENTS	MAXIMUM DOSE RATE AND DOSE	REMARKS
Electrical Equipment Room	RAB EL. 43’	1 hour installation 1-3 teams as required	Varies along route 2.5 hr. after accident 3.7 rem per person with transit SCBA	Install jumpers to power cross-train Hot Leg Injection MOVS

* The PASS is no longer used.

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - OUTSIDE OF REACTOR, REACTOR AUXILIARY, AND FUEL HANDLING BUILDINGS
FIGURE 12.1-1

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 18', 23' AND 62'
FIGURE 12.1-2a

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 45'
FIGURE 12.1-2b

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION -0.5'
FIGURE 12.1-3a

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION 19.5'
FIGURE 12.1-3b

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION 43'
FIGURE 12.1-3c

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION 62'
FIGURE 12.1-3d

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - FUEL HANDLING BUILDING FLOOR ELEVATIONS 19.5', 48' & 62'
FIGURE 12.1-4

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM DOSE RATE LEVELS - FUEL HANDLING BUILDING VERTICAL SECTIONS
FIGURE 12.1-5

Withheld Under 10 CFR 2.390

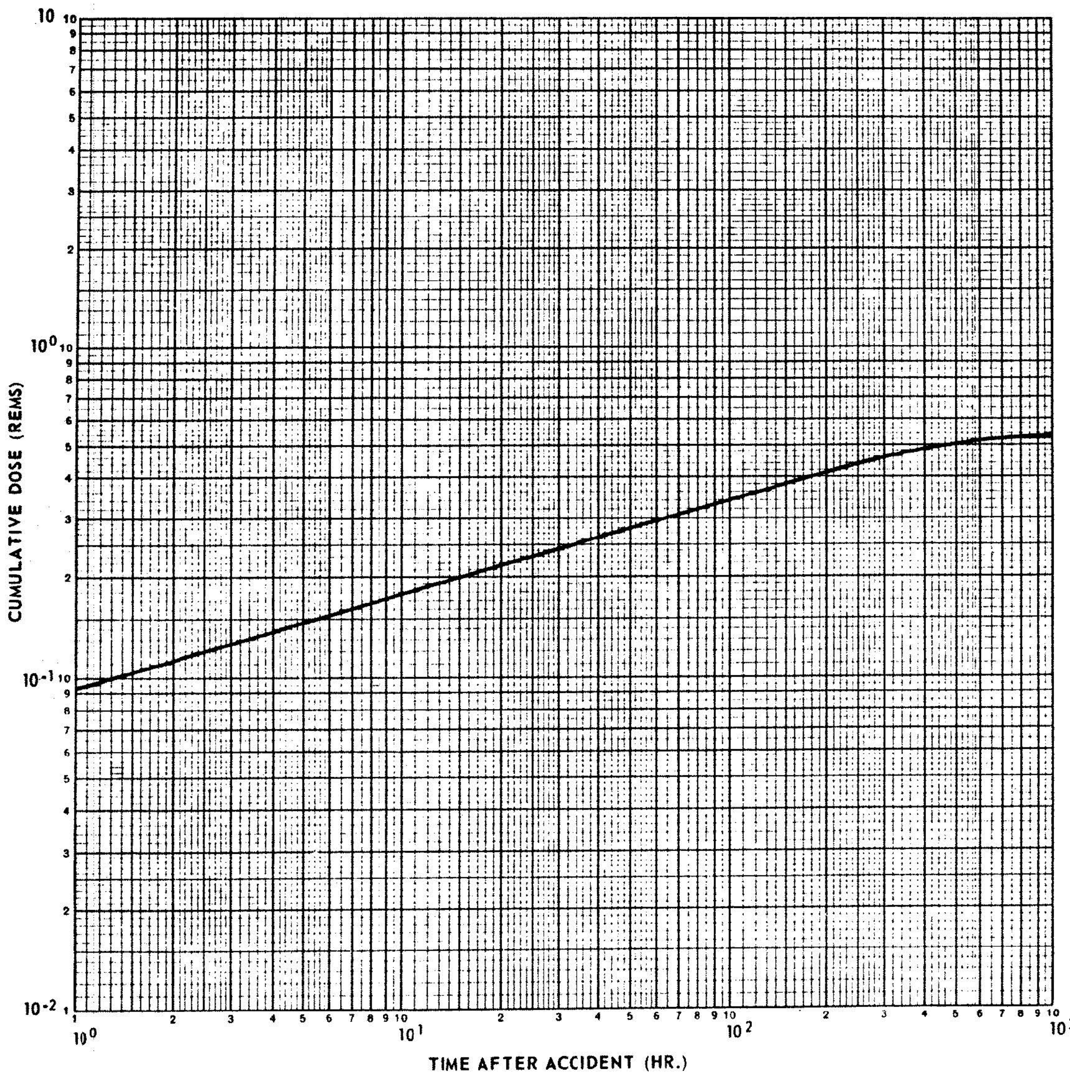
FLORIDA POWER & LIGHT COMPANY St. Lucie Plant
CONTROL ROOM LAYOUT ISOMETRIC FIGURE 12.1-6

Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

ISOMETRIC VIEW OF CONTROL ROOM
SHIELDING AND POST-LOCA
RADIATION SOURCES

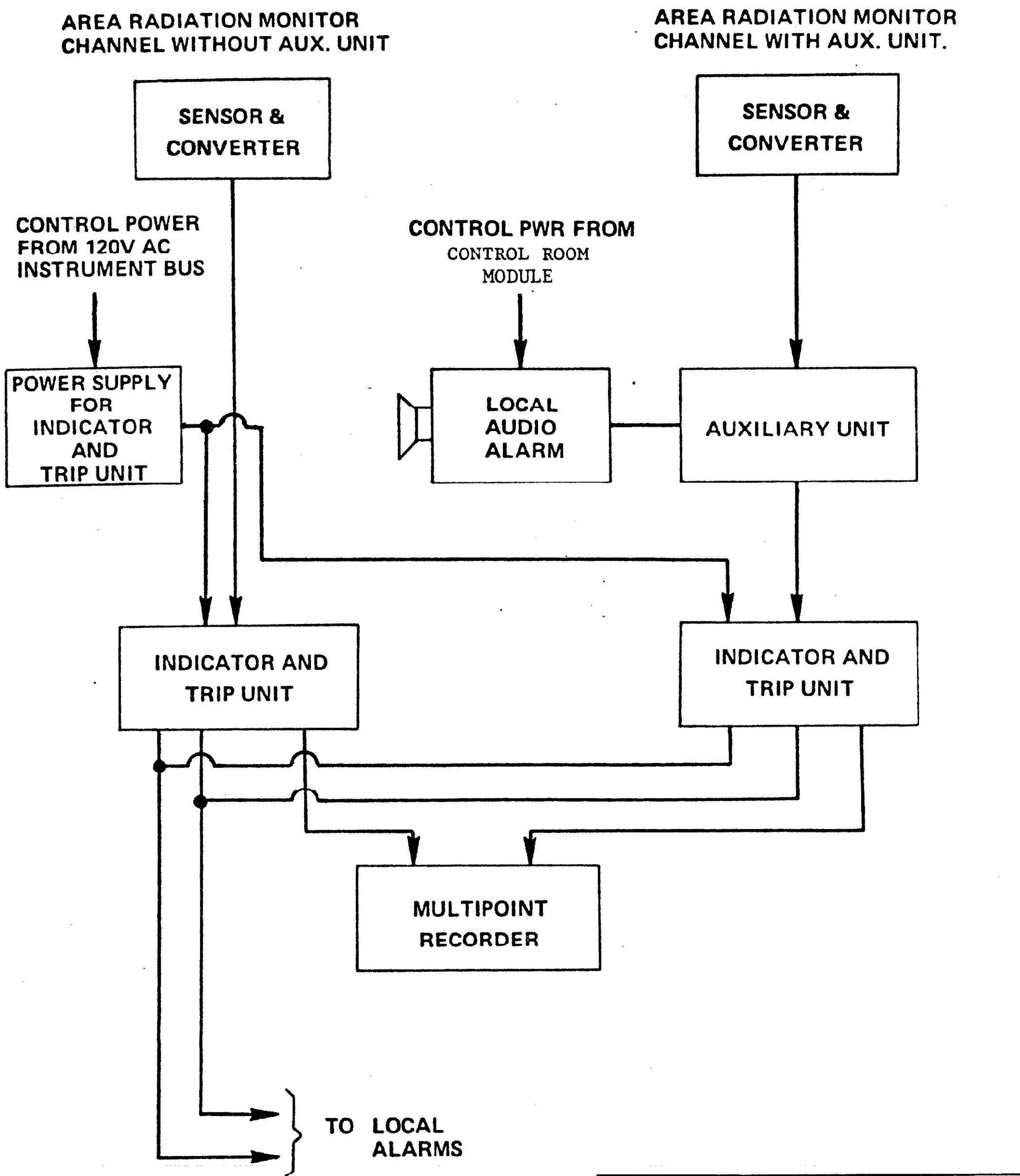
FIGURE 12.1-7



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

CUMULATIVE POST-MHA WHOLE BODY DOSE IN THE CONTROL ROOM

FIGURE 12.1-8

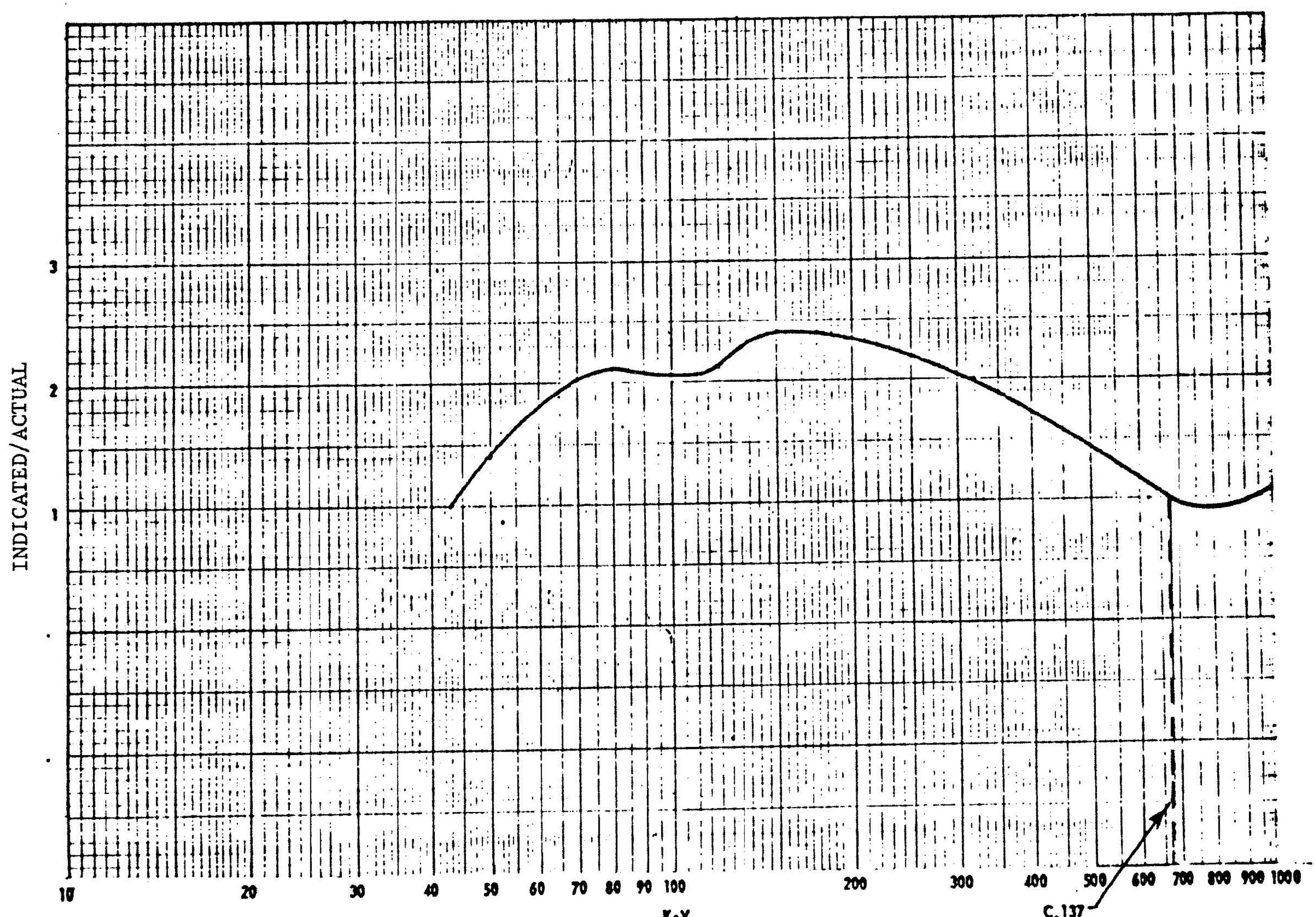


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

AREA RADIATION MONITORING SYSTEM
FUNCTIONAL BLOCK DIAGRAM

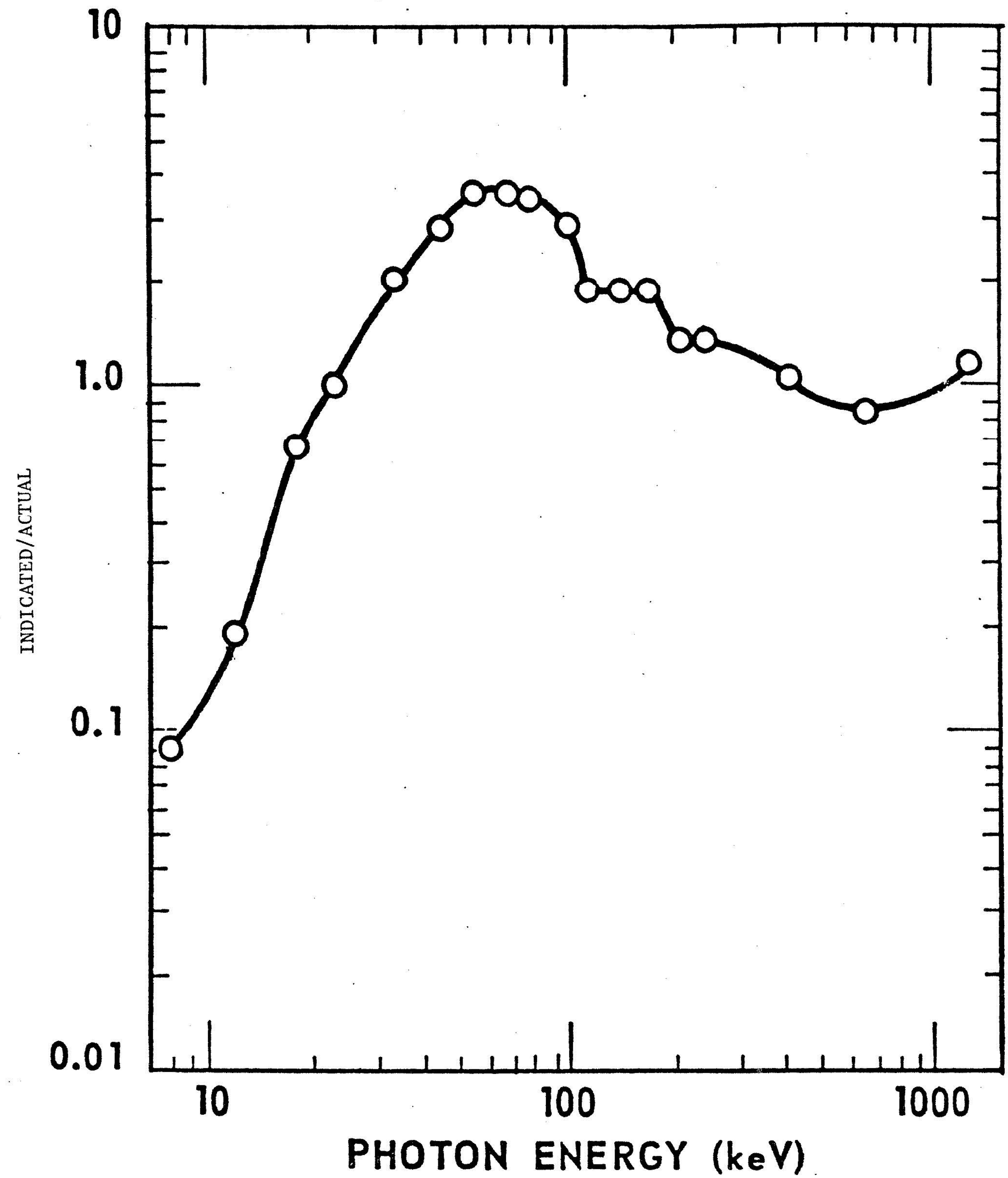
FIGURE 12.1-9



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

ENERGY RESPONSE CURVE

FIGURE 12.1-10



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

SENSITIVITY OF GEIGER MULLER DETECTORS

FIGURE 12.1-11

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The design objectives for the plant ventilation systems as they relate to radiation protection are to:

a) limit airborne radioactivity in the buildings to levels that will permit access required for operation, maintenance, inspection and testing without exceeding the requirements of 10 CFR 20.

b) assure migration of air toward areas having sources of airborne radioactivity.

The plant ventilation systems are designed in accordance with ORNL NSIC-65 which provides for features that facilitate inspection and maintenance. See Table 12.2-0 for a listing of other industry standards and an explanation how ORNL-NSIC-65 was applied to the design. Table 12.2-1 summarizes the areas subjected to contamination by airborne radioactivity and shows the effect of ventilation system operation on the summation of the ratios of nuclide activity, (C), to their respective maximum permissible concentrations, (MPC), as set forth in 10 CFR 20, Appendix B.

The information developed is based on Table 11.1-1 using data for 0.1 percent failed fuel.

12.2.2 DESIGN DESCRIPTION

12.2.2.1 General

Charcoal adsorbers, where referred to in the system descriptions, consist of banks composed of tray type units rated at 333 cfm capacity each. Trays are of stainless steel construction and consist of two carbon beds of two inch thickness, separated by an interspace.

The charcoal is iodine impregnated with the capability of trapping a minimum of 99.9 percent of the iodines at accident concentrations with 5 percent as methyl iodide when operating at 70 percent relative humidity and 150°F. Test clips are provided for periodic testing to monitor charcoal for aging effects. Freon tests are periodically applied in accordance with ANSI - N510 - 1975 to assure minimum in-place leak test efficiencies of 99.9 percent.

High efficiency particulate (HEPA) air filters, where referred to, consist of banks assembled in 24" wide x 24" high x 11-1/2" deep extended media dry type units. Each unit is rated at 1000 cfm. Filters are fabricated from an all glass media with maximum 5 percent binder, closely pleated over aluminum separators, and have a minimum efficiency of 99.97 percent when tested with thermally generated DOP droplets of uniform 0.3 micron size. Filters located within the containment are provided with composition-asbestos separators in lieu of aluminum. Periodic in-place leak tests are applied, using DOP aerosol to assure filter integrity.

Medium efficiency filters, where referred to, are of the disposable dry media type, having a minimum National Bureau of Standards atmospheric type test efficiency of 50-55 percent. Filter cartridges are 24"x24"x36" deep or 24"x24"x18" deep.

Although filter replacement is infrequent, design features enable maintenance personnel to enter the casing through an access door located downstream of the filter. Each cell can be removed by loosening the holddown clamps and extracting via the handle provided. Health Physics personnel will survey filters for contamination level prior to handling by maintenance personnel.

12.2.2.2 Containment Airborne Radioactivity Removal System

The containment airborne radioactivity removal system consists of two units as shown on Figures 9.4-1 through 9.4-3. The system is designed to remove airborne radioactivity in particulate and iodine form by recirculating containment atmosphere through high efficiency filters and charcoal absorbers. The system is used for radioactivity removal during normal operation only and serves no function for post-LOCA dose reduction. Each of the two units located at elevation 23' includes a suction connection from the containment cooling system ring duct header located at elevation 98'; a bank of HEPA air filters, 2 units wide by 5 units high; a bank of unit tray type charcoal absorbers, 2 units wide by 15 units high; and a belt driven, single width single inlet, freely discharging centrifugal fan. Fan and filters for each of the two units are rated at 10,000 cfm with a fan static pressure of 6 in. wg which permits rated operation with fully loaded filters.

The containment airborne radioactivity-removal system is manually energized from the control room. Charcoal adsorber temperature and HEPA filter pressure drop are annunciated and monitored in the control room. Low flow annunciation is also provided.

All active components of the system are suitable for operation in 120°F ambient temperature and 1 rad/hr of radiation. Ambient temperatures are limited to a maximum of 120°F by normal operation of three containment fan cooling units.

The system design and calculational evaluation presented below is based on normal plant operation only, assuming a 1200 lb/day reactor coolant leakage into the containment and design parameters based on original plant design. This information is retained for historical purposes. Specific equilibrium activities of the coolant are listed in Table 11.1-1. Other assumptions made in the calculation model are:

a) Failed fuel is 0.1 percent.

b) Iodine partition coefficient is 0.10.

c) 100 percent of all noble gases become airborne at the leakage or generation rate.

d) Tritium, which is in water molecular form, concentrates in the atmosphere up to the level determined by the specific humidity (lb of water vapor per pound of dry air at 120°F dry bulb and 50 percent relative humidity).

e) Mixing efficiency of all airborne constituents is 90 percent.

f) Because a quantity of leakage remains in liquid form, there is little opportunity, barring liquid surface disturbances, atomization or residue following complete evaporation, for particulates to become airborne. Therefore, the assumption of all particulates being airborne is conservative.

After 90 days with the assumed leakage rate of 1200 lb/day of reactor coolant into the containment with all airborne radiation cleanup systems inactive, the sum of C/HPC for all nuclides is 1860. At this time, the sum of all nuclide activities is 0.00193 μCi/cc. Iodines alone amount to 647 MPC, particulates amount to 1142 MPC, and noble gases including tritium amount to 71 MPC. Pulldown time for the airborne radioactivity removal system is a function of air change rate which, when both units are operating, is 0.48 containment net free volume air changes per hour. With this capacity and following 26 hours of system operation, the gross C/MPC level is reduced to a minimum of 80.5, with gross activity at 0.00185 μCi/cc. However, within 10 hours, 99 percent of the total pulldown from 1860 to 71 MPC has occurred. At minimum MPC level, iodine, particulate and noble gas concentrations in MPC's have been reduced to 7.5, 0.7, and 71.3 respectively.

12.2.2.3 Containment Purge System

The containment purge system, which exhausts the containment atmosphere to the environment, is rated at 42,000 cfm and is operated following reduction of iodine and particulate activity by the containment airborne radioactivity removal system. The system is shown on Figures 9.4-1 and 9.4-2. It serves to further reduce the residual iodine and particulate activity as well as to reduce the activity of the nonfilterable noble gases and trititm. Where only short term access to the containment is required, the system is not operated. For extended time, as on a weekend shutdown, the purge exhaust system is used. Although it is likely that the reactor will be subcritical at these times, the calculational model conservatively assumes full power operation.

The suction side of the purge system is connected through a 48" x 48" duct to the containment cooling system ring duct header to assure uniform purging of the containment. A 36" x 14" branch duct is connected through an automatic damper to forty air inlets located above the water line in the refueling cavity. This exhaust branch is activated during refueling. During a normal nonrefueling purge, the containment air from the ring header is drawn through butterfly isolating valves FCV-25-4, 5 and 6 into a filter casing that is common to two belt-driven, single width single inlet, top annular upblast fans that discharge to the plant vent. Each fan is rated at 42000 cfm and 10 1/2 in. wg static pressure.

The filter casing and high efficiency filter mounting frames, common to both fans, are of all welded construction and include, in the direction of flow, a set of medium or high efficiency prefilters and a bank of HEPA filters or V-bank carbon adsorber cells, 7 units wide by 6 units high. This enables the flexibility to operate the purge system on either a particulate filtration system (with the HEPAs) or elemental iodine removal system (with carbon adsorbers).

The air makeup side of the purge system includes, in the direction of flow, a 12' wide x 10' high air intake louver, a bank of medium efficiency filters, 5 cells wide by 4 cells high, and three 48 in. diameter butterfly type isolating valves, designated FCV-25-1, -2, -3.

The system is manually energized from the control room. When the switch is moved to the “start” position, exhaust butterfly valves FCV-25-4, -5 and -6 open and through valve limit switches start a fan. Fan motor starter interlocks and a differential pressure switch permit opening of makeup air butterfly valves FCV-25-1, -2, and -3 only when a slightly negative pressure differential has been established in the containment. This prevents unfiltered flow back through the makeup air valves. The purge fans are designed to trip off if a high containment vacuum condition occurs. All containment purge system isolation valves (FCV-25-1, -2, -3, -4, -5 and -6) close automatically on activation of the containment isolation signal.

Restrictions on the use of containment purge and vent system isolation valves have been imposed by the Commission, Item II.E.4.2 of Reference 1. In addition, for operation at Extended Power Uprate (EPU) conditions, valves I-FCV-25-1, 2, 3, 4, 5 and 6 are not operated and remain closed in Modes 1 through 4 (Ref. EPU LAR L-2010-259 Attachment 5, Section 2.7.7). Leakage testing of containment purge and vent valves is performed in accordance with the Containment Leakage Rate Testing Program.

Calculational assumptions noted for the containment airborne radioactivity removal system apply as well to the purge system, which may be energized following radioactivity cleanup. The final values for MPC and activity concentration resulting from the recirculation phase of cleanup (i.e., the airborne radioactivity removal system) are used as the initial input to the computer calculation for the purge phase cleanup.

The result is that after operation of the airborne radioactivity removal system and the purge system, the sum of C/MPC for all nuclides is equal to 3.73. This corresponds to a gross activity level of 0.108×10^{-4} μ Ci/cc. Iodines alone amount to 2.3 MPC, particulates amount to 0.5 MPC, and noble gases including tritium amount to 0.9 MPC. Approximately one containment volume per hour is processed by the purge system. This permits 99 percent of the total cleanup to the above levels to occur after 9.6 hours of operation. This information is retained for historical purposes.

12.2.2.4 Reactor Auxiliary Building Main Ventilation System

The reactor auxiliary building main ventilation system is of the non-circulating type and consists of a conventional central air supply system with a Class I seismic rating and a nonseismic Class I filtered exhaust. The system is shown on Figure 9.4-1.

The air supply system includes a screened outside air intake louver, a bank of medium efficiency filters, and two belt-driven fans that discharge into an air duct distribution system. The fans are double width double inlet, top angular upblast type. Supply fans HVS 4A and 4B are each rated for full system flow which is 66,615 cfm and 3 in. wg. static pressure. A system of ductwork exhaust areas having sources of airborne radioactivity at a rate that is sufficient to reduce activity to levels permitting required weekly access during power operation. Refer to Table 12.2-1 for a listing of these areas. In most cases, the air flow required to limit temperature rise exceeds radioactivity control requirements.

The main exhaust system includes exhaust ductwork to the main plenum; a bank of 12 wide by 6 high medium efficiency prefilters; a bank of HEPA filters 12 wide by 6 high; and two belt-driven, single width single inlet, upblast centrifugal fans (HVE-10A and 10B) which discharge to the plant vent through backdraft dampers. Each of the two exhaust fans have a design capacity of 74,600 cfm at 9.32 in. wg. static pressure. The exhaust system is not required to operate following an accident, this function being performed by emergency exhaust system fans HVE-9A and 9B.

Supply and exhaust fans, except for HVE-9A and 9B, are manually energized from the control room.

Air flow stoppage is annunciated by means of flow switches located in the discharge of each fan. HEPA filter pressure drop is indicated in the control room and annunciated when in excess of 3 in. wg.

Other areas within the reactor auxiliary building that are free of airborne radioactivity contaminating sources are exhausted directly to the environment. Such areas include laboratories, offices, electrical equipment areas, cold locker rooms, battery rooms, cold shower, cold area toilet, clean clothes issue room and cable enclosure.

The total volume of the reactor auxiliary building is 978,770 cu ft of which 638,370 cu ft are areas exhausted to the plant vent.

In most cases the minimum air change rate in any space within the reactor auxiliary building is 4 volumes per hour. The overall air change rate for portions of the building exhausted to the plant vent is 6.73 volumes per hour. Air change rates for specific areas having sources of airborne contamination are listed in Table 12.2-1.

12.2.2.5 Fuel Handling Building Ventilation System

The fuel handling building ventilation system reduces plant personnel doses due to potential airborne activity resulting from diffusion of fission products through the fuel pool water. The system consists of (a) a separate air supply and air exhaust network serving the fuel pool area and (b) a separate air supply and air exhaust network serving the fuel pool equipment areas and the new fuel storage area. The system is shown on Figures 9.4-1 through 9.4-3.

Each air supply system includes a wall intake, disposable roughing filters, and a cabinet type centrifugal fan. The first supply system is rated at 9800 cfm; the second supply system is rated at 8600 cfm. These supply systems provide hourly air changes of 6.8 volumes and 4.97 volumes for the fuel pool and equipment areas, respectively.

The fuel pool is exhausted by 36, 8" x 4" air inlet ducts spaced 4'-0" on centers around the pool periphery and approximately 1' above the pool surface. Four such inlet ducts are located 8'-0" on centers in the south wall of the refueling canal. The fuel pool exhaust system includes the concrete-embedded welded steel pool exhaust ducts; the duct connection to the filter casing; and all welded filter casing that includes a bank of medium efficiency filters, 3 wide x 3 high, and a bank of HEPA filters 3 wide x 3 high; a bank of charcoal adsorbers 3 wide x 10 high, and two belt-driven, single width single inlet, upblast fans. Each fan is rated to deliver full system airflow of 10,350 cfm at 7-1/4 in. wg. static pressure which corresponds to an average velocity of 1295 ft/min through the 36 fuel pool exhausts. Discharge is to a 42" dia x 30'-0" long vent stack, via a concrete penthouse on top of the FHB, exhausting at 109.50 ft. elevation.

Air is supplied to the fuel pool at a rate of 9800 cfm from 17'-0" above the 62'-0" floor elevation forcing migration of all air downward to the fuel pool surface. The amount of air supplied, being less than that exhausted, prevents out-leakage from the pool area to the equipment spaces within the building and to the environment.

The fuel pool equipment areas are exhausted through a filter system of the same arrangement used in the fuel pool exhaust system except for charcoal adsorbers. A single width single inlet, upblast centrifugal fan exhausts the filter casing and discharges out the vent stack - The fan is rated for 8450 cfm flow and 8 in. wg static pressure. Fuel handling building-ventilation system exhaust is monitored for effluent flow and radiation. Radiation level is monitored by a system similar to the vent pipe monitoring system described in Section 12.2.4.2.

12.2.2.6 Control Room Ventilation System

The design of the control room reflects the limits of acceptable exposures to plant personnel established for design basis accidents. General Design Criterion 19 stipulates that radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 REM total effective dose equivalent for the duration of the accident. As stated in Section 12.1.1, the control room shielding is designed to limit the maximum whole body dose rate to 0.5 mrem/hr during normal operation and the maximum integrated whole body dose to 3 REM for 90 days following MHA. The actual expected cumulative whole body dose rate following a MHA is shown on Figure 12.1-8.

The control room air conditioning system consists of three split-system direct expansion air conditioning units, a ducted air intake and air distribution system, and an emergency filtration section with HEPA filters and charcoal absorbers. Each unit is sized for one-half total load. The system is shown on Figures 9.4-1 and 9.4-3.

Each of the three indoor units includes a fail-open electrically operated inlet damper, medium efficiency filters, direct expansion refrigerant cooling coil, heating coil, cabinet type centrifugal fan rated at 9500 cfm and 3.75 inch wg, static pressure, and a gravity damper to prevent recirculation through deenergized units. Each of the indoor sections is served by a weatherproofed, roof mounted condensing unit which consists of an air cooled condenser section assembled to a refrigerant compressor section.

The emergency filtration system processes control room air through charcoal filters if the plant site is subject to airborne contamination. This system includes a manual shut-off damper, a 1 unit wide x 2 unit high bank of HEPA filters, and a 1 unit wide x 6 unit high bank of charcoal adsorbers feeding into two full capacity centrifugal booster fans arranged in parallel, each rated at 2000 cfm and 5 inches wg. static pressure. Motorized inlet and gravity outlet dampers are provided at each booster fan. Two electric motor operated 12 inch butterfly isolation valves are located on each duct leading from the outside air intake hoods which are mounted on the north and south walls of the reactor auxiliary building at elevation 78'-9". All ductwork is of welded construction.

During normal operation the control room ventilation system operates with the emergency filtration system deenergized. Outside makeup air bypasses the emergency filter train and mixes with the control room return air before it is conditioned by the cooling or heating coils.

Upon receipt of a CIS, all four outside air intake valves and toilet area exhaust valves close within 35 seconds preventing entry of potentially contaminated outside air. The outside air intake valves also close upon receipt of a high radiation signal from radiation monitors located in the air intakes. The CIS also energizes both filtration booster fans which, through interlocks, open the respective fan inlet dampers. One of the two filtration booster fans may be manually deenergized and placed on standby. Control room air is then recirculated at a rate of 2000 cfm through the HEPA and charcoal filters, resulting in the cleanup of residual airborne radioactivity that may be present.

Based on a control room volume of 62,700 cubic feet which conservatively includes space above the hung ceiling, 1.9 air changes per hour are recirculated through the filters. Assuming 90 percent efficiency, activities of filterable nuclides are reduced to 1 percent of their initial levels in approximately 2 hours, assuming no in-leakage.

The outside air intakes may remain closed for a period of time dictated by maximum allowable CO_2 concentrations and minimum levels of oxygen. Because of the high ratio of control room volume per occupant, considerable time would pass before levels of either constituent could endanger personnel. CO_2 level should be kept at 1 percent or below for continuous exposure. Levels of

2 or 3 percent can be tolerated for one or two hours with some discomfort but not permanent harm. Oxygen should not be less than 17 percent for prolonged occupancy. With the outside air intake closed, and assuming 10 occupants, the CO_2 level would reach 1 percent in 37 hours and the oxygen level would reduce to 17 percent in 111 hours. This is based on a breathing rate, oxygen consumption and carbon dioxide production rate of 30 cubic feet per hour, 1.20 cubic feet per hour and 1.00 cubic feet per hour, respectively, and on a net control room volume of 36,900 cubic feet. Net control room volume excludes space above the hung ceiling and upon adjacent rooms and for this calculation, is conservative. Upon analysis of favorable post-LOCA site radioactivity information, one of the outside air intake ducts will be opened. Approximately 750 cfm of outside air is then mixed with 1250 cfm of return air and drawn through the HEPA and charcoal filter section. Determination of whether to draw outside makeup air from either the north or south side of the reactor auxiliary building is based on the readings of the radiation monitors located at each intake.

When air flow through any of the control room supply system discharge ducts or through either of two emergency filter booster fan discharges falls below a preset limit, it is annunciated in the control room.

See Sections 6.4, 9.4, and 15.4 for related information concerning the Control Room HVAC.

12.2.3 SOURCE TERMS

The source terms that form the basis for plant ventilation system design and operation are derived from Tables 12.1-3, 12.1-5 and 12.1-6, which list the specific activities in process fluids and Table 12.2-1, which lists estimates of leakage points and rates in the process systems. When combined properly, the tables yield the source terms necessary for the inhalation dose calculations discussed in Section 12.2.6.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

Radiation detection devices are provided in the plant to monitor normal radiation levels and to detect and enunciate any abnormal radiation condition. The plant airborne radioactivity monitoring system consists of the containment atmosphere radiation monitoring system and the vent pipe radiation monitoring system which are designed to provide such detection and annunciation by sampling and/or monitoring plant particulate and gaseous activity levels. The plant airborne radiation monitoring system is supplemented in the monitoring of plant radiation levels by the plant area radiation monitoring system for which the design bases and description are provided in Section 12.1.4.

12.2.4.1 Containment Atmosphere Radiation Monitoring System

The containment atmosphere radiation monitoring system is designed to provide a continuous indication in the control room of the particulate and gaseous radioactivity levels inside the containment. Radioactivity in the containment atmosphere indicates the presence of fission products due to a reactor coolant leak or leakage of a contaminated secondary fluid system. Refer to Figure 12.2-1 for the following discussion.

The sample flow is withdrawn from a ring duct downcomer through one of two redundant sample nozzles. The nozzles were originally designed as isokinetic; however, the Reference 2 calculation concluded that isokinetic sampling was not necessary in order to achieve representative particulate sampling.

The tubing from the redundant nozzles is merged just upstream of the monitors to enable the monitors to withdraw a sample from either of the independent nozzles.

Immediately downstream of the merge, the sample is split into two branches: one branch feeds the particulate monitor skid and the other branch feeds the noble gas monitor skid.

The particulate monitor skid again splits the sample into two branches: one branch feeds the particulate (PIPS/P) detector and the other branch feeds the particulate/iodine sampler (PI filter). The particulate detector sample passes through a filter paper which collects particulates. The PIS sample passes through a particulate filter and then an iodine collector. Both may be withdrawn and analyzed in the laboratory for isotopic content. Both samples are merged upstream of the sample pump and are pumped back to containment. The pump discharge line has ports from which a grab sample can be withdrawn. The particulate detector utilizes beta detection and employs background subtraction to reduce interference from gamma radiation.

The noble gas monitor skid passes the sample through a particulate/iodine sampler (PI filter) and then the 300 cc noble gas sample chamber. The particulate filter and iodine collector may be withdrawn and analyzed in the laboratory for isotopic content. The pump discharge line has ports from which a grab sample can be withdrawn. The 300 cc sample chamber has lead shielding arranged in a 4π geometry to prevent interference from background radiation.

Each monitor skid has its own dedicated pump. This ensures that failure of a single pump can not disable both skids.

All sample tubing is stainless steel. Sample flow through the system is indicated by independent flow meters. The pumps can be operated locally or remotely in the control room through the ratemeters.

Refer to Table 12.2-2 for system design characteristics.

Each channel operates from its own integral high and low power supply. Each channel provides three alarm modes; Fail, Alert and High. Alert and High alarms are adjustable over the full span of the ratemeter.

Two operational checks have been designed into each channel. An electronic pulse generator provides a check for the input of the ratemeter. This signal when activated will simulate an indication of approximately 6×10^4 cpm. A radioactive check source is activated by a switch located on the front panel of each ratemeter for operational check of the detector.

Power sources, indicating devices and recorders are all located on the same control room radiation panel. All radiation channels are indicated, recorded and annunciated on the control room radiation monitoring panel. Abnormal radiation levels are indicated both visually and audibly.

The sensitivity of the containment atmosphere monitoring system for detecting reactor coolant leakage is discussed in Section 5.2.4.

12.2.4.2 Plant Vent Radiation Monitoring System

The plant vent radiation monitoring system is designed to representatively sample, monitor, indicate and record the radioactivity level in the plant effluent gases being discharged from the vent pipe. It provides a continuous indication in the control room of the activity levels of radioactive materials released to the environs so that determination of the total amount of activity release is possible.

The vent pipe radiation monitoring system continuously samples and/or monitors the vent pipe exhaust for particulates, iodine and gases at the point of release to the atmosphere. Its function is to confirm that releases of radioactivity do not exceed the predetermined limits set by 10 CFR 20.

The sample flow is withdrawn from the vent pipe through nozzles located at a minimum of 8 vent diameters from the last point of radio activity entry. The nozzle orifice size and sample flow rate were originally selected to provide an isokinetic sample velocity. However as discussed in Safety Evaluation PSL-ENG-SENS-00-108, an isokinetic sample velocity is not necessary to achieve representative sampling in this application.

The plant vent radiation monitor is further described in Section 11.4.2.7.

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12.2.4.3 Drumming Facilities Radiation Monitoring System

A review has been conducted to determine the propriety of adding additional airborne radioactivity monitors capable of detecting activity from small undetected leaks and spills within the reactor auxiliary building. Facility health physics personnel provide the primary means for the timely detection of such leakage through (1) daily surveys of the RAB, and (2) surveys of intended work areas prior to the issuance of a work permit. It is likely that during certain maintenance operations radioactive leakage may be initiated. At the discretion of the plant Radiation Protection Manager, a portable self-contained monitor would be installed in the work area during such operations.

It is felt that the health physics surveys and the use of a portable monitor does provide effective monitoring for the protection of plant personnel and the installation of fixed monitors in the RAB ventilation system is not necessary. Therefore, originally designed process (effluent) radiation monitoring system has been subsequently abandoned.

PAGE 12.2-14 HAS BEEN DELETED.

PAGE 12.2-15 HAS BEEN DELETED.

12.2.5 OPERATING PROCEDURES

Stationary and portable detectors are provided for monitoring airborne activity in operating and maintenance locations susceptible to radioactive contamination. Operating procedure lists and survey programs are described in Section 12.3.

Radiation Work Permits (RWPs) issued for a job will explicitly specify the use of health physics respiratory equipment when required during performance of maintenance and repair involving potential activity releases.

Detailed written procedures are prepared and approved as discussed in Technical Specification Section 6. It is intended that all written procedures conform to 10 CFR 20, or exceed these requirements, in minimizing personnel inhalation exposures.

12.2.6 ESTIMATES OF INHALATION DOSES (Historical Information)

Expected airborne activity concentrations for various auxiliary building areas are provided in Table 12.2-1 in terms of ratio of MPC'S. Also provided for reference are the maximum whole body doses from internal exposure which would be realized if an individual spent his entire 40 hour work week, 50 weeks per year, in any of the designated areas. In practice this cannot and will not occur because the total whole body dose from both external and internal sources is maintained well below the occupational exposure limits as specified by 10 CFR 20.1201. Areas with high concentrations of airborne activity are so marked and access controlled.

The estimated yearly man-Rem dose for plant personnel from the effect of inhalation is 19.2 among about 38 persons in maintenance, chemistry and health physics, and 2.5 among ten nuclear operators, totaling about 22 man-Rem per year. These rates are based on the operating procedures as described in Sections 12.1.5 and 12.2.5.

12.2.7 ESTIMATES OF RADIATION EXPOSURE TO UNIT 2 CONSTRUCTION WORKERS DUE TO UNIT 1 OPERATION (Historical Information)

Radiation fields in the general construction area, due to direct radiation from all Unit 1 sources are estimated to be below 0.003 mr/hr.

The major sources of activity near a potential area of construction for Unit 2 are the drumming facility, the holdup tanks, and the boric acid and waste concentrators. These sources are housed in cubicles alongside the southern wall of the Unit 1 auxiliary building. This wall has been sized so that the dose rate at the "hot spots" in the wall, i.e., diametrically opposite the sources of radiation, would be less than 0.25 mr/hr under the worst radiological conditions of I percent failed fuel and presence of the maximum expected level of corrosion product activity. Since average radiological conditions (0.1 percent of failed fuel) are more representative of the actual fields-during the construction phase of Unit 2, the contact dose rates at the wall are approximately 0.025 mr/hr. This dose is further attenuated as one moves progressively away from the Unit 1 outside wall. The level roughly halving for every 10 feet of distance.

Thus at a distance of 40 feet, the closest distance between the Unit 1 auxiliary building and the Unit 2 reactor building, the dose rate is approximately 0.003 mr/hr from these sources. Contributions from other sources in the auxiliary, fuel handling and reactor building of Unit 1 are negligible.

Contributions from outside tanks to the general construction area have been evaluated on the basis of maximum nuclide activities stored therein, listed in Table 12.1-3A, and found to be negligible under the assumption that construction workers would not spend time in the vicinity other than that required to go to and from their assigned work.

The contribution to the dose rate at the construction site from the Unit 1 turbine building has been estimated using a total activity inventory in the turbine, condenser, etc. of 0.15 Ci (1 Mev equivalent) and has been found to be much less than the 0.003 mr/hr contribution of the auxiliary building sources.

For a construction worker working every fourth 8 hour shift the maximum annual radiation dose from direct sources was estimated to be 6.6 mr (0.003 mr/hr x 2190 hrs).

In addition to this direct dose there is a small contribution from airborne radionuclides released from Unit 1. The whole body dose from inhalation of airborne radionuclides is negligible in comparison to external exposure and thus has not been considered here. The doses due to airborne releases were calculated assuming the release rates tabulated in Table 11.3-4 of the SAR adjusted for 0.1 percent failed fuel.

The χ/Q values used in estimating exposures in the Unit 2 construction area have been calculated using the effective relative concentration equation of question 6.23. The equation parameters, tabulated below, were obtained from onsite data in the following manner:

The annual average percent frequency of wind directions in the 9 sectors north of the reactor building (202.5 degree arc extending from west clockwise to east, including 9/16 of the calm wind conditions) is 49.6 percent. The average annual wind speeds for these 9 wind directions is 8.57 miles per hour or 3.83 mps. See Table 2.3-6. The application of these meteorological statistics to the daytime workers is considered conservative because they include late night to early morning low wind speed conditions.

The equation parameters obtained are:

N = Annual average time period,

O = 0.24 (40 hours per week, 52 weeks per year divided by 8760 hours per year),

F = 0.496 Fraction of time the wind direction occurred in the 9 wind directions,

U = 3.83 m/sec - Wind speed, and

all other terms as previously defined. The effective χ/Q calculated in this manner for releases from the containment is 4.32×10^{-5} sec/m^3.

The χ/Q value for releases from the turbine building was obtained from the above value for the containment building by adjusting for the difference in wake factors. Using this procedure the effective χ/Q for the turbine building releases is found to be 7.23×10^{-5} sec/m^3.

Utilizing the above assumptions the resultant maximum dose to construction workers from airborne release of radionuclides from Unit 1 is 1.9 mr/year. Therefore the maximum total annual dose to a construction worker from both the above sources, direct and submersion components, is 8.5 mr/year.

The estimated integrated dose (man-rem) to the construction population during the construction period from all sources was expected to be less than 10.3 man-rem/year.* This value is extremely conservative since it assumes all the construction workers are continually located 40 ft from the highest direct radiation source.

* The actual construction population has peaked above the projections used here for dose estimation.

REFERENCES TO SECTION 12.2

1. NUREG-0737-Clarification of TMI Action Plan Requirements.

2. Calculation PSL-1FSN-96-001, Rev. 0.

TABLE 12.2-0

APPLICATION OF INDUSTRY STANDARDS-TO VENTILATION DESIGN OBJECTIVES

The following illustrates how the guidance provided by ORNL-NSIC-65 was applied.

The construction of ductwork, filter housings, type and location of exhaust fans, use of butterfly valves and dampers in ductwork conforms to ORNL-NSIC-65 recommendations. In addition to ORNL-NSIC-65, NRC Regulatory Guide 1.52 and the latest editions of the following standards at the time of system design and construction have been considered in the design, construction and testing of air cleaning and ventilation systems;

a) USNRC Health and Safety Bulletin for Filter Unit Inspection and Testing Service for current year.

b) ASTM D1056, Sponge and Cellular Rubber Products.

c) ASTM A165, Electrodeposited Coatings of Cadmium on Steel.

d) Underwriter's Laboratories Standard UL-586, High Efficiency Air Filter Units.

e) ASTM A366, Cold-Rolled Carbon Steel Sheets, Commercial Quality; ASTM Structural Steel Specification A36.

f) AISC Code of Standard Practice.

g) AWS DI.0-69 Code for Welding in Building Construction.

h) Military Specification, MIL-F-51068, Filter Particulate, High Efficiency, Fire Resistant (latest edition)

i) Military Specification, MIL-F-51079, Filter Medium, Fire Resistant, High Efficiency.

j) U S Department of Commerce Commercial Standard CS-132. Hardware Cloth, (latest edition).

k) USNRC Report DP-1082, Standardized Nondestructive Test of Carbon Beds for Reactor Confinement Applications. (Savannah River Laboratory)

1) ASTM A240, Stainless Steel Plate, Sheet and Strip.

m) ANSI N-510-1975)Standard for Testing Nuclear Air Cleaning Systems.

n) RDT M-16-IT, Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds.

Each pertinent section of ORNL-NSIC-65 is addressed below and the relevant HVAC Approach for the St. Lucie design is cited.

2. CHAPTER 2 - PROBLEM AREAS

2.2 Air cleaning systems have been designed for normal and accident operation. Cleaning systems are provided with injection and sampling ports where necessary and adequate space has been provided for testing equipment.

2.2.2 Prefilters are provided before HEPA filters. Replacement of HEPA filters is based on a pressure drop of 3.00 in. wg, and fans are selected accordingly.

2.2.3 Screens, filters and/or louvers are provided in all air supply systems.

2.2.4 Prefilters in most cases and occasionally demisters in place of prefilters are provided in all air cleaning systems.

2.2.5 Air cleaning systems have been designed for a maximum pressure drop of 3.00 in. wg for HEPA filters and 1.00 in. wg for clean ones.

2.3 Air cleaning systems have been designed to provide satisfactory working conditions for personnel and to prevent the release of radioactive substances to the atmosphere.

2.3.1 Ventilation rates are based principally on cooling requirements and the inhalation hazard or potential inhalation hazards, of substances present in the air of the controlled spaces. Some examples follow:

Containment Building

a) The building has been designed to prevent the dispersal of activity to the environment in the event of an accident.

b) Shield building ventilation system has been designed to maintain a negative pressure of -1.0 in. wg. All air is exhausted through a safety feature filtration system.

c) Air in the building flows from areas of least contamination to areas of increasing contamination.

d) There is no recirculation of air from the central exhaust system.

Air Handling System

a) 100 percent redundancy is provided for all safety feature air cleaning systems.

b) All exhaust cleaning systems are exhausted through prefilter and REPA filters.

c) Does not apply.

d) HEPA filters will be tested in-place using DOP and have a minimum test efficiency of 99.0 percent.

e) Release of I-131 shall be within established limits.

f) The adsorber system shall be tested in place in accordance with the guidance provided by NRC Regulatory Guide 1.52.

2.3.3 Demisters are provided where heavy concentration of water, mist or steam can be expected under either normal or accident conditions.

2.3.5 Decay heat of oxidation has been considered in the design of material of filters.

2.3.5 Galvanized sheet metal has been used for the construction of ductwork and filter housing. Stainless steel charcoal adsorber trays are used. Filter racks are made of carbon steel and painted for corrosion protection.

2.4 Consideration to damage of the filter system from shock, vibration or fire; to the design and arrangement of ductwork and filter housing have been considered. Smoke detectors are provided to detect smoke and fire. Adequate fire protection is provided as indicated in UFSAR Section 9.5.1.

2.5 Cleaning filters have been designed for ease of maintenance, accessibility and simplicity of maintenance. Ducts and housing have been laid out with a minimum of ledges, protrusion, and crevices that can collect dust and moisture. Access doors are provided at accessible location.

Cleaning systems required to operate under post accident conditions have been designed to stand pressure surges. Centrifugal fans are used for all air cleaning systems.

2.6 The ventilating systems have been designed for proper consideration in the location of air intake louvers, filter housing and stacks. Centrifugal fans are used for all cleaning systems. Proper fans and motors are selected and due consideration is given for filter resistance when loaded up with dust.

2.7 Fans are located in the downstream side of the filters and are located very close to the vent stack. Effect of duct connection to fan on fan performance has been considered.

2.8 Ductwork for ventilating and air cleaning system has been designed in accordance with industry standard practice and Table 2.9 and 2.10. Welded joints construction and bolted and gasketed joints are provided.

2.9 Adequate controls are provided for control room operation of engineered safeguard air cleaning systems with high differential pressure alarm across HEPA filters, temperature recording of charcoal adsorber with annunciation in control room, low flow or fan failure alarm in control room. Control room operation of butterfly valves and dampers with indication light and automatic operation of engineered safeguard system in the event of accidents.

3. CHAPTER 3 - COMPONENTS

Prefilters, deminster, HEPA filter and charcoal adsorber have been designated in accordance with ORNL-NSIC-65 and applicable standards for efficiency, resistance, air flow capacity, dust holding capacity,filter media, separator, gasket, fire resistance and maintenance considerations. Prefilters are bag-type American Air filter Dri-Pak, medium efficiency 50-55 percent efficiency or high efficiency, 90-95 percent efficiency. Each cell is 24W x 24H x 36D/18D. Deminsters are Otto H. York Co. - 24 x 24 x 2 5/8" made of stainless steel. Each HEPA filter cell and V-bank carbon adsorber cell is 24W x 24H x 11 1/2 and rated for 1000 cfm. Each charcoal adsorber tray consists of 24W x 8H x 30D and rated for 333 cfm.

4. CHAPTER 4 - MULTIPLE FILTER SYSTEMS

The air cleaning systems are made of bank system mounting frames for demister, HEPA filters and charcoal adsorbers are made of carbon steel structural plate and is all welded construction and cadmium coated for corrosion protection. The filter units are clamped to the filter frame and sealed with ASTM D1056 neoprene sponge gasket. The filters are mounted from the down-stream side and are 3 high. Service grating has been provided where the filters are higher than 3 high. Adequate space and clearance for construction and maintenance work has been provided. Filter housings have been provided with access doors. A separate drain is provided for each chamber of filter housing. All engineered safeguard air cleaning systems are provided with vapor proof and explosion proof light in each chamber for maintenance purpose and visual inspection.

7. CHAPTER 7 - TESTING

In-place testing of HEPA filter and charcoal adsorber will be performed to ensure leaktightness of construction and acceptability of components furnished. In-place testing of the HEPA filter and charcoal adsorber will be done in accordance with approved procedures. No other testing will be performed.

TABLE 12.2-1
(Historical)

AREAS WITH AIRBORNE RADIOACTIVITY SOURCES

Area[1]	Air Flow (cfm)	Leak Rate (gpd)	Source[3] Strength Factor	ΣC/MPC	Maximum Whole Body Dose from Internal Exposure[2] (rem/year)
AUXILIARY BUILDING, ELEVATION -0.5'					
Reactor Drain Pumps	10,000	1.0	1.0	4.17(-2)	2.09(-1)
Charging Pumps	6,000	5.0	1.0	3.48(-1)	1.74
Spent Resin Tank	225	0.01	50.0	9.26(-1)	4.63
Boric Acid Holding Tank and Pumps[4]	1,025	0.01	50.0	2.03(-1)	1.02
Boric Acid Condensate Tanks[4]	1,500	0.01	0.01	2.70(-5)	1.35(-4)
Boric Acid Preconcentrator Filter	300	0.01	0.1	1.39(-2)	6.95(-2)
Holdup Drain Pumps	1,500	0.1	50.0	1.39	6.95
Chemical Drain Tank	1,200	5.0	0.1	1.74(-1)	8.68(-1)
Remaining Areas	6,410	1.0	0.1	6.50(-3)	3.25(-2)
AUXILIARY BUILDING, ELEVATION +19.5'					
Radio Chemistry Lab. and Counting Room	1,000	0.1	1.0	4.17(-2)	2.09(-1)
Sample HX and H_2 Analyzer	1,525	1.0	0.01	2.73(-2)	1.36(-1)
Boronometer[4]	475	0.1	1.0	8.77(-2)	4.38(-1)
Purification Filter, Flash Tank and Flash Tank Pumps	2,200	0.1	1.0	1.89(-2)	9.45(-2)
Fuel Pool Ion Exch., Boric Acid Condensate Ion Exch.[4] Deborating Exch.	800	0.1	50.0	2.606	13.0
Drumming Station	2,250	2.0	10.0	3.706	18.5
Boric Acid and Waste Concentrator[4]	1,800	0.1	50.0	1.158	5.78
Boric Acid Makeup Tank and Pumps	2,100	0.1	50.0	9.92 (-1)	4.96
Holdup Tanks	2,800	0.1	50.0	7.44 (-1)	3.72

(1) Refer to Figures 12.1-3, 12.1-4 and 12.1-5 for radiation zone classification for access.
(2) Assumes continuous exposure for a 40 hr week, 50 weeks per year.
(3) Ratio of nuclide radioactivity concentrations of leak source to those of the reactor coolant.
(4) These components have been abandoned in place.

TABLE 12.2-2

DESIGN CHARACTERISTICS - CONTAINMENT RADIATION M0NITORING SYSTEM

1. Air Particulate Channel

a. Detector

Type	Passivated Implanted Planar Silicon (PIPS)
Sensitivity to Y^{90} cpm per µCi on filter	1.89×10^5
Dynamic range, µCi on filter	4.2×10^{-5} to 1.5×10^1
Operating pressure, psia	12.3 to 15.1
Operating temperature range, °F	32 to 113 (120 max sample temp)
Amperage, voltage, frequency, phase	8.6 amps, 120V ac, 60 Hz, single phase
Check source*	Cs^{137} - 100 µCi

b. Ratemeter

Type	log or linear (selectable)
Accuracy, percent	± 2 FS
Range, cpm	L.T.10 to G.T.10^6
Voltage, frequency, power	120V/230V ac, 50/60 Hz, 50W

c. Moving Filter Paper**

Absorption efficiency, percent	99 for particulates 0.3 microns in diameter

2. Gaseous Radiation Monitoring Channel

a. Detector

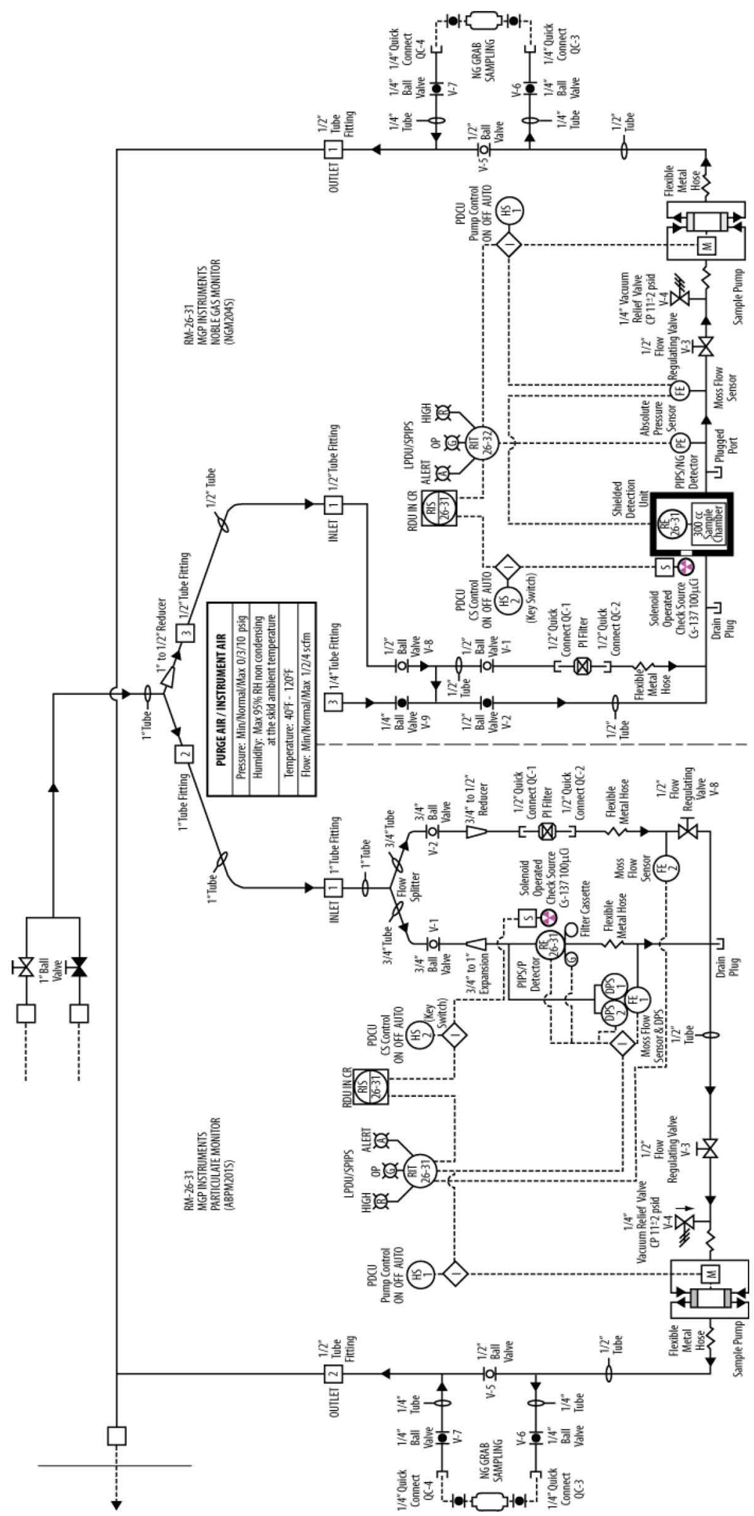
Type	Passivated Implanted Planar Silicon (PIPS)
Sensitivity to Xe^{133} cpm per µCi/cc	2.6×10^6
Dynamic range, µCi/cc	2.6×10^{-7} to 1.0×10^0
Operating pressure, psia	12.3 to 15.1
Operating temperature range, °F	32 to 113 (120 max sample temp)
Check source*	Cs^{137} - 100µCi
Shielding	2 inches 4π geometry lead
Amperage, voltage, frequency, phase	8.6 amps, 120V ac, 60 Hz, single phase

b. Ratemeter (same as air particulate channel)

3. Iodine Charcoal Filter Holder Channel

Type	Activated charcoal filter
Efficiency, percent	95

*Check source is activated from ratemeter located in control room.
**Filter paper movement may be selected for manual or programmed step advance.



FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 1**
CONTAINMENT RADIATION MONITORING SYSTEM
FIGURE 12.2-1

12.3 HEALTH PHYSICS PROGRAM

12.3.1 PROGRAM OBJECTIVES AND ORGANIZATION

The program objectives are:

a) To implement a radiation protection program for protecting the health and safety of plant personnel and the public.

b) To establish and maintain a comprehensive record system which will demonstrate the adequacy of the radiation protection program to ensure that occupational exposures are as low as reasonably achievable and which complies with all applicable Federal and State regulations.

The Radiation Protection Manager has the specific responsibility and authority for ensuring that the radiation protection program maintains exposures as low as reasonably achievable. He reports to the Plant General Manager. He has direct access to both the Site Vice President and Plant General Manager on matters related to radiological health and safety of employees and the public. He fulfills the responsibilities of the radiation protection manager as defined in the Technical Specifications.

All individuals in the Health Physics Department will meet the qualification criteria as outlined in the Health Physics instructions on qualification and training of Health Physics personnel before they are delegated responsibility in the radiation protection program. Exposure reduction training and Job application are covered in detail in the Radiation Protection Training program. Supervisors are given extensive training plus continuing communication with Health Physics in the form of written information or instructions and meetings so that foremen and supervisors are fully able to cope with specific jobs in such a way so as to minimize radiation exposure. Radiation Protection Training for all other employees is outlined in Section 13.2.

12.3.2 FACILITIES AND EQUIPMENT

The design of facilities including restricted areas or equipment for use in restricted areas is periodically reviewed by the health physics staff to ensure that provisions have been included to achieve as low as practicable exposures during maintenance, inservice inspection, refueling, and nonroutine operations. The specific provisions are listed in Section 12.1.1.1.

The Radiation Controlled Area is shown in Figures 1.2-2, 1.2-12 through 1.2-15, and 1.2-17. This area includes that in which radioactive materials and radiation above 2 mRem/hr may be present. Traffic is routed into and out of the Radiation Controlled Area in a path that will minimize exposure and the spread of contamination.

The portable gamma survey instruments are calibrated utilizing a gamma calibration unit. The dose rates on the surface of the calibrator do not exceed 2.0 mr/hr, while the dose rates inside the calibrator range from 2 mr/hr to 600 R/hr. Additionally, the use of a pulse generator may be used for count rate instrument calibration.

To perform the sampling analysis delineated in Safety Guide 21, the Applicant uses the following laboratory equipment.

Quantity	Description
1	Gas Flow Proportional Counter, For Gross Beta-Gamma, and Alpha Counting
1	Phosphor Scintillator. for Gross Beta-Gamma and Alpha Counting
2	Pulse Height Analyzers with NAI(TI) Detectors for Body Burden Analysis
1	Liquid Scintillator, For Tritium (Service provided by Chem. Dept.)
2	Pulse Height Analyzer System with GeLi Detector, For Isotopic Analysis

The general arrangement of the locker room facilities in the reactor auxiliary building is designed to provide adequate personnel decontamination and change areas as shown in Figure 1.2-13. The cold locker room is used as a change and storage area for clothing and personal items not required or allowed in the radiation controlled areas. The hot locker room is employed as a change area and storage area for potentially contaminated clothing. Personnel monitors are located at the access point(s). All personnel will survey themselves on leaving the Radiation Controlled Area. Showers, sinks and necessary monitoring equipment also are provided in the hot locker room to aid in the decontamination of personnel.

The health physics staff prepares written procedures and letters of instruction to implement the Radiation Protection Program.

Subjects that are addressed in health physics procedures are listed in Table 12.3-1. Time-radiation dose schedules are developed and followed to formalize operations to be executed in unusually high gamma radiation fields in such a manner that exposure to operating and maintenance personnel is minimized.

Health Physics personnel periodically observe jobs in progress in the Radiation Controlled Area and make daily radiation surveys to ensure that exposure to radiation and contamination levels are kept as low as reasonably achievable.

Administrative exposure guidelines are designed to prevent an individual from exceeding an annual exposure limit. When personnel are assigned to a job or a location where there exists the possibility that administrative guidelines may be exceeded, Health Physics investigates the exposure records of the personnel involved and initiates an extension authorization accordingly. This authorization can be given by Health Physics only after review of the individual's exposure history and the amount the guidelines will be exceeded.

Routine surveys for radiation and airborne contamination are divided into two categories. The daily surveys consist of center-of-room type surveys and air samples of frequently occupied areas to determine general trends and detect any significant increases. The weekly surveys consist of thorough radiation and airborne surveys of designated areas. Different areas are surveyed each day of the week and are repeated each week.

The events or incidents which necessitate special radiation surveys include, but are not limited to the following:

a) spills involving radioactive materials.
b) a significant increase in radiation levels as noted on area monitors, portable radiation detection instruments, or air monitors.
c) radiation work permit request in areas where the potential change of radiation and contamination levels are high.

All personnel entering contaminated areas are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available includes shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel such as plastic or rubber suits, face shields, safety glasses and respirators are also available. Health physics-trained personnel evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Appropriate written procedures govern the proper use of protective clothing where and how it is to be worn and removed, and how the change room and decontamination facilities for personnel equipment, and plant areas are to be used.

Provisions are made for decontamination of work areas throughout the plant. The spent fuel cask wash area has facilities to handle decontamination of large equipment. This area contains a decontamination enclosure and service facilities. A decontamination room external to the machine shop in the reactor auxiliary building is used for the decontamination of hand tools and small equipment.

The procedure for radiation work permits calls for consultation and pre-job planning between the supervisor in charge of the job and Health Physics personnel prior to issuance of a Radiation Work Permit. Process or other engineering controls (e.g., containment or ventilation) will be used to the extent practicable to control the concentrations of radioactive material in air. Respiratory protection devices are used when an evaluation of the working environment indicates that it is impractical to apply process or other engineering controls and that wearing the device is consistent with the goal of maintaining Total Effective Dose Equivalent (TEDE) ALARA.

Respiratory devices available for use include:

a) Supplied air hoods
b) full-face respirator (filter, charcoal canister, or supplied air)
C) self-contained breathing apparatus

Self-contained breathing apparatus is used for emergency conditions or oxygen deficient atmospheres.

Respiratory protection equipment is selected to provide a protection factor (given in 10 CFR 20 for the various devices) 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 a selection of a respiratory protection device with a protection factor greater than the peak concentration is inconsistent with the goal of keeping TEDE ALARA, respiratory protection equipment with a lower protection factor may be selected only if such a selection would result in keeping the TEDE ALARA.

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

A health physics office and counting laboratory are located in the reactor auxiliary building (el. 19.5'). These facilities include both laboratory and shielded counting rooms. These are equipped to analyze routine air samples and contamination swipe surveys. Portable radiation survey instruments, respiratory protection equipment and contamination control supplies are stored in the reactor auxiliary building.

Portable personnel radiation monitors are located in the following Entrance Control Points:

a) Auxiliary Building
b) Reactor Containment (at initial core loading; during shutdown)
c) Craft Control Point
d) Remote RCA (when required)

A portal monitor is located in the main guard station at the exit from the plant and in the Craft guard station and provides a final radiation survey of all personnel leaving the generating station area.

The types and minimum quantities of portable radiation survey instruments available for routine monitoring functions are listed in Table 12.3-2.

Portable beta-gamma air monitors with a calibrated air sampling pump and recorder, capable of continuous operation, is available for supplemental airborne activity measurement at areas specified by the health physics personnel for use where work is performed in potentially air contaminating operations. A second portable unit of like specifications is available for use as backup when maintenance is required on the first unit. These units supplement, also, the fixed air monitor units in the vent pipe and in the waste handling area and the daily radiation survey in the radiation controlled area made by health physics personnel. Additional Table 12.3-2 data is:

Response to Co^{60} gammas - 350 cpm/mr/hr

Range 0 - 50,000 cpm

Response time - 20 sec to 90% of reading

Survey instruments are calibrated periodically and maintenance records are maintained for each instrument.

In order to protect personnel from access to high radiation areas that may exist temporarily as a result of plant operations and maintenance, warning sign, barricades, and locked doors are used as necessary.

The requirements for controlling entry into high radiation areas are contained in the Technical Specifications. These requirements are implemented by plant procedures.

Contaminated areas and equipment are decontaminated as soon as practicable after detection or use by personnel trained in decontamination procedures. Prior to decontamination, each contaminated area in which the contamination levels are equal to or greater than the plant administrative limits are barricaded and conspicuously posted as a contaminated area. Decontamination is performed under the direction of health physics personnel.

12.3.3 PERSONNEL DOSIMETRY

Records of radiation exposure history and current occupational exposure are maintained for each employee for whom personnel monitors are issued. In addition to the review of TLD data, a periodic review of airborne exposure, beta-gama area radiation and contamination level data is made by Health Physics to clarify exposure trends.

12.3.3.1 External Exposure 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 or digital alarming dosimeters provide day-by-day indication of external radiation exposure. The types and minimum quantities of direct reading dosimeters available for use are listed in Table 12.3-3.

All persons subject to occupational radiation exposure are issued beta-gamma and neutron sensitive TLD's and are required to wear them when within the radiation control area. The TLD's are processed on a routine basis by a NVLAP accredited processor. The TLD of any individual is processed by special handling whenever it appears that an overexposure may have occurred.

Self reading or digital alarming dosimeters are issued, in addition to the TLD, to all personnel within the radiation control area. Dosimeters are read, recorded and rezeroed regularly. Dosimeter records furnish the exposure data for the administrative control of radiation exposure.

Special or additional personnel monitors are issued as required under unusual conditions. These devices are issued at the discretion of health physics personnel.

Visitors to the controlled access areas are escorted by qualified personnel and are issued appropriate personnel monitoring devices. An escort is not required for those who have received the necessary radiation protection training when this arrangement is approved by the Radiation Protection Manager and authorized by the Plant Manager.

12.3.3.2 Internal Exposure Monitoring

An initial bioassay sample or a whole body count is analyzed for radiation workers to determine a base-line for further reference. Anytime there is an incident which produces unknown concentrations of airborne radioactivity or when it is suspected that personnel have taken radionuclides into their body by some other means, a bio-assay sample or whole body count of those persons involved is analyzed for uptake of radionuclides. If a significant increase of radionuclides present in the body is reported, a second bio-assay sample and/or whole body count may be required.

A bio-assay sample or whole body count is analyzed for radiation workers who are terminating employment. Whenever necessary a service is contracted with a reliable vendor to provide a complete bioassay analysis program and submit a report after the sample is analyzed. A whole body counting facility is available at the St. Lucie Plant.

TABLE 12.3-1

HEALTH PHYSICS PROCEDURES SUBJECTS

Radiation Protection Manual
Respiratory Protection Program
Radiation Work Permits
Protective Clothing Requirements
Personnel Monitoring
Movement of Material and Equipment
Decontamination
Portable Shielding
Scheduling
Training
Portable Survey Instruments
Internal Exposure Monitoring
Pocket Dosimeter Source and Drift Check
Calibration and Operation of Counting Equipment
Calibration and Testing of Portal and Hand and Foot Monitors
Air Sampling
Shipping and Receiving Radioactive Materials
Leak Testing Sealed Radioactive Sources
Count Rate Instrument Calibration
Qualification and Training of Health Physics Personnel
Emergency Equipment Inventory and Checking
Personnel Dosimetry Records
Radiation Incidents
Storage of Radioactive Waste
Radiation and Contamination Surveys
HP Emergency Procedures

TABLE 12.3-2

PORTABLE RADIATION SURVEY INSTRUMENTS

Minimum Quantity	Range or Capacity	Type	Efficiencies or Energies Detected
10	0-5,000 mr/hr	Air Ion Chamber Low Range for β^- & γ	12 Kev - 7 Mev
7	0-50,000 mr/hr	Air Ion Chamber Medium Range for β^- & γ	12 Kev - 7 Mev
10	0-2,000 mr/hr	G. M. Detector Low Range for β^- & γ	40 Kev - 1.25 Mev
33	0-50,000 cpm	Beta Sensitive G. M. Detector	β^- > 40 Kev
1	0-100,000 cpm	γ scintillation Detector	γ > 60 Kev
7	0-1,000 R/hr	G. M. Detector β^- & γ	γ > 50 Kev
4	0-5,000 mr/hr	BF_3 Detector neutrons	0.025 ev - 10 Mev
2	0-25,000 mr/hr	Air Ion Chamber α, β^-, γ	α > 3. 5 Mev β > 70 Kev γ > 7 Kev
1	0.025 R, 0.25 R, 1.0 R, 10 R, 100 R	Air Ion Chamber Secondary Standard	γ > 50 Kev

TABLE 12.3-2 (Cont'd)

Quality	Range or Capacity	Type	Efficiencies or Energies Detected
2	0-500,000 cpm	G. M. Detector β^-	β^- > 40 Kev
1	0-10 R/hr	BF Detector neutron	0.025 ev - 10 Mev
1	0-10^6 R	Ion Chamber	γ >70 Kev
19	0-8 SCFM	Air Samplers	99.97% Particulate 85% Iodines
2	0-5 SCFM	3 Channel Continual Air Monitors β^- & γ	I_2 - 5.0 E -10 µCi/cc Part. - 1.0 E -10 µCi/cc Gas - 1.0 E -07 µCi/cc
6	0-100 liters/min.	Air Particulate Monitors - β^-	Part. - 1.0 E-10 µCi/cc
8	0-2 mr/hr	Portal Monitors G. M. Detectors	γ > 80 Kev
10	0-1,000 R/hr	Ion Chamber Low & High Range	γ - 6 Kev - 1.3 Mev

TABLE 12.3-3

DIRECT READING DOSIMETERS

Quantity	Range	Type	Energy Range
500	0-500 mr	Medium Range	γ-80 Kev - 2 Mev
50	0-1 R	Medium Range	γ -80 Kev - 2 Mev
50	0-5 R	High Range	γ -80 Kev - 2 Mev
10	0-10 R	High Range	γ -80 Kev - 2 Mev
10	0-200 R	High Range	γ -80 Kev - 2 Mev
300	1 mrem-999 Rem	Low to High Range digital Alarming Dosimeter	γ -80 Kev - 2 Mev

12.4 RADIOACTIVE MATERIALS SAFETY

12.4.1 MATERIALS SAFETY PROGRAM

Procedures, facilities and equipment for handling and processing of radioactive liquid, gaseous and solid wastes are described in Chapter 11 and in Sections 12.1 through 12.3. Procedures, facilities and equipment for the safe handling and storage of new fuel assemblies and spent fuel assemblies are described in Section 9.1.

Various radioactive sources are employed to calibrate the process and effluent radiation monitors, described in Section 11.4; the area radiation monitors, described in Section 12.2; and the portable and laboratory radiation detectors are described in Section 12.3. Check sources that are integral to the area, process and effluent monitors consist of exempt quantities of by-product material isotopes and do not require special handling, storage or use procedures for radiation protection purposes. The same consideration applies to radionuclide sources of exempt quantities which are used to calibrate or check the portable and laboratory radiation measurement instruments.

Sealed sources purchased or prepared by the Chemistry Department or under the direction of the Chemistry Manager for the calibration, testing or standardization of laboratory counting equipment is stored in the radiochemistry laboratory.

Sealed sources purchased by the Health Physics Department for the calibration, testing or standardization of laboratory counting equipment are stored in the health physics counting room or in a designated storage area.

If for any reason the sources become unusable, they will be disposed of according to the radioactive waste disposal procedures, Section 11.5. Accountability and documentation for all sources will rest with the Radiation Protection Manager. Records are maintained which include, but are not necessarily limited to, purchase order, purchase date, date received, supplier, isotope, quantity, date of any subdivision, activity of the subdivided source, remaining activity of the original source and date of ultimate disposal or consumption of the source at which time it is removed from the inventory list. A new record is established and maintained for each subdivided source and said source is treated as an additional source for accountability and documentation purposes. All documentation accompanying the purchase of any source becomes a permanent part of the Health Physics Department records.

Sealed radionuclide sources having activities greater than the quantities of radionuclides defined in Appendix C to 10 CFR 20 and Schedule B of 10 CFR 30 are subject to material controls for radiological protection. These controls include:

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

b) Labeling of each source with the radiation symbol, stating the activity, isotope and source identification number.

c) Inventorying and monitoring of each source for removable surface contamination at six-month intervals.

d) Storage of each source that is not installed in an instrument or other piece of equipment in a locked area.

e) Maintenance of records on the results of inventories, leakage tests, use, location, condition, principle user and the receipt and final disposition dates for all sources.

The sources are handled and used in accordance with the procedures listed in Table 12.3-1. In the event of an inventory discrepancy the Radiation Protection Manager investigates, determines the reason for the discrepancy and if the discrepancy resulted in an uncontrolled release or exposure to employees or visitors to the nuclear station.

Recognized methods for the safe handling of radioactive materials, such as those recommended by the National Council on Radiation Protection and Measurements, have been implemented to maintain potential external and internal doses at levels that are as low as practicable. The materials safety program is defined by written policies in the Radiation Protection Manual.

12.4.2 FACILITIES AND EQUIPMENT

The laboratory facilities and equipment contained therein for handling radioactive materials is described in detail in Section 12.3. Equipment and facilities for the sampling of radioactive liquids and gases are described in detail in Section 9.3.2. The area radiation monitoring and the process and effluent monitoring systems are detailed in Sections 12.2 and 11.4, respectively. Radiation Protection instrumentation is described in Section 12.3.2.

12.4.3 PERSONNEL AND PROCEDURES

Key personnel responsible for handling and monitoring the materials are the Chemistry Manager and Radiation Protection Manager whose experience and qualifications are listed in Technical Specification Section 6.0.

The radiation safety instructions to working personnel appropriate to the handling and use of radioactive sources are listed in Table 12.3-1. These procedures collectively cover all the material required to inform employees how to safely handle radioactive materials.

Radioactive sources that are subject to the material controls described in Section 12.4.1 are only used or handled by or under the direction of chemical and radiation protection personnel. Each individual using these sources is familiar with the radiological restrictions and limitations placed on their use. These limitations protect both the user and the source. Radiation work permits, described in Section 12.3.2, provide detailed instructions for all work in high radiation, and airborne radioactivity areas. The qualifications and training programs for personnel who are responsible for handling and monitoring radioactive materials are described in plant procedures and Technical Specification Section 6.0.

Considerable time and effort are devoted to assure that employees understand radiation and radiation protection as it applies to their work. Supervisors are responsible for ensuring that their employees receive adequate 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. Training continues with detailed discussions of the specific radiological hazards associated with work assignments. In the course of their work, employees receive additional training in radiation protection practices from supervisors, senior co-workers and chemical and radiation protection personnel.

All new or temporary employees shall receive a radiation protection orientation and pass a proficiency examination prior to assignment of work in the radiation controlled area. Orientation includes instruction in the Radiation Protection Manual. Employees and visitors that have not completed the radiation protection orientation and passed the proficiency examination are escorted within the radiation controlled area.

12.4.4 REQUIRED MATERIALS

A listing of isotopes, quantities, forms and uses for all by-product source and special nuclear materials is given in Table 12.4-1. Instrumentation check and calibration sources having less than 100 mCi beta/gamma activity or 100 milligrams of source or special nuclear material have been excluded from this listing.

TABLE 12.4-1

BYPRODUCT, SOURCE AND SPECIAL NUCLEAR MATERIALS; RADIOACTIVE SOURCES LISTING

Isotope	Quantity*	Form	Use
U-235	See Sections 4.1 and 4.2	See Sections 4.1 and 4.2	Reactor fuel
U-238	See Sections 4.1 and 4.2	See Sections 4.1 and 4.2	Reactor fuel
Pu-Be	2 @ = 20 Ci ea	Each source contains two capsules inserted between Sb-Be pellets, all sealed in a stainless steel tube	Neutron source for reactor fuel loading, refueling, shut-down and approach to criticality
Pu-238-Be	1 @ 1 ci	Sealed, solid	Calibration and check source for health physics neutron survey instruments
Cs-137	1 @ 0.1 ci	Encapsulated in welded, stainless steel	Calibrator for area radiation monitors
Cs-137	1 @ 200 Ci	Sealed, solid consisting of eight separate capsules	Calibrator for health physics gamma survey instruments
Pu-239-Be	1 @ 1 ci	Sealed, encapsulated	Boronometer source

* Source Strength and Source Quantity represent historical information

APPENDIX 12A

ROCKWELL METHODOLOGY OF DETERMINING DOSE RATES
OUTSIDE SHIELD WALLS

This appendix gives sample calculations using Rockwell's methodology to assess the dose rate outside a shield wall.

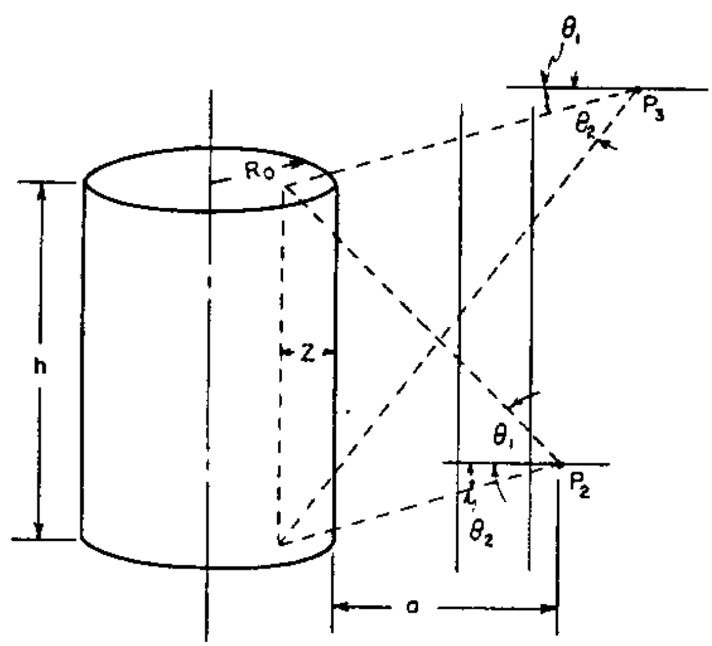
Two examples are given. The first is a dose rate calculation outside the shielding wall of the cubicle housing the pressurizer of a threeloop system. The second example is a dose rate calculation outside the wall shielding the upper portion of a steam generator for the same threeloop plant. Refer to Figure 12 A-1 for sketches of the geometry involved.

The predicted dose rates using the Rockwell methodology are compared below to measured values from H. B. Robinson Unit No. 2, which is a plant of identical geometry.

Included in the following pages are the quantitative source description for a typical (i.e., cylindrical) source. The calculations of the dose rates outside the shield walls discussed above are also included. The calculations include all assumptions and physics data that were used in making the calculation.

12A.1 QUANTITATIVE DESCRIPTION OF CYLINDRICAL SOURCE

12A.1.1 Exterior on Side (μ_s Z Curves)



At P2

$$\phi = \frac{BS_V R_\phi^2}{4(a+z)} [F(\theta_1, b_2) + F(\theta_2, b_2)] \qquad \theta_1 \neq \theta_2$$

At P2

$$\phi = \frac{BS_V R_\phi^2}{4(a+z)} [F(\theta, b_2) \qquad \theta_1 = \theta_2 = \theta \qquad if\ h = \infty, \theta = \pi/2$$

At P3

$$\phi = \frac{BS_V R_\phi^2}{4(a+z)} [F(\theta_2, b_2) - F(\theta_2, b_2)]$$

or

EFFECT OF GEOMETRY OF RADIATION SOURCE

$$\phi_u = \frac{S_n}{2\mu_s} \sum_{i=0}^{n} \frac{(-1)^i\, n!}{(\mu_s)^1\,(n-1)!} \{(h+c\}^{n-i} \left[E_{2+1}(b_1) - \frac{E_{2+1}(b_1 \sec\theta)}{(\sec\theta)1+1} \right] - c^{n-i} \left[E_{2+i}(b_3) - \frac{E_{2+i}(b_{1\sec}\,\theta)}{(\sec\theta)i+1} \right] \}$$

If the source strength per unit volume is given by a power series

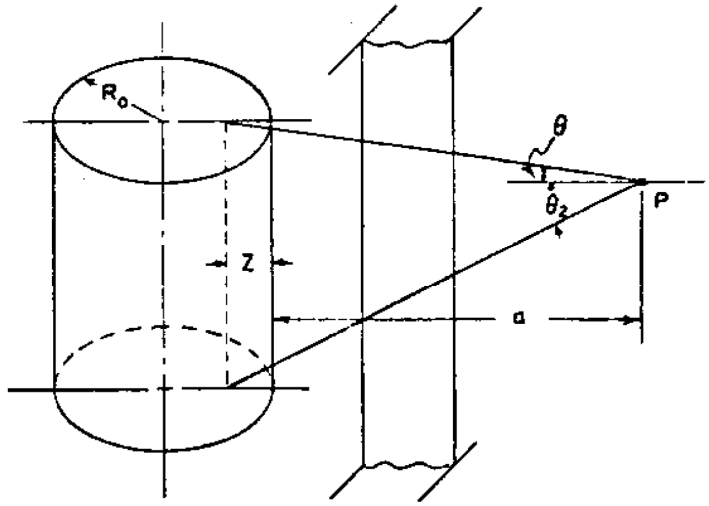
$$S_V(X) = \sum_{n=0}^{N} S_n (x+c)^n$$

then the flux is a linear sum of terms like that above.

If, as a special case of the above, $S_V(x) = S_V$ = constant, then it follows directly from the above equation that

$$\varphi_u = \frac{S_v}{2\mu_s} [E_2(b_1) - E_2(b_3) + \frac{E_2(b_3 \sec\theta)}{\sec\theta} - \frac{E_2(b_1 \sec\theta}{\sec\theta}$$

12.A.1.2 <u>Exterior on Side</u>



A right-circular cylindrical source with constant strength S_V per unit volume can be approximated by a line source having strength per unit length

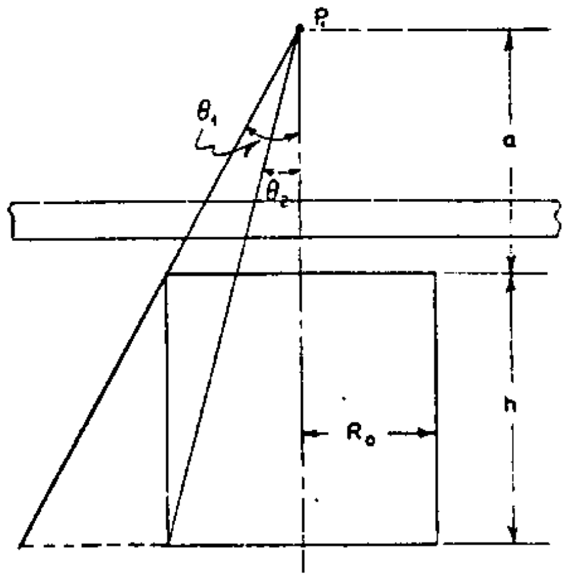
$$S_L = (\pi R_0^2) S_V$$

provided the line source is placed within the cylinder to correctly account for the self-absorption of the cylinder. There is no simple relation of the form $Z = Z(R_0,a,b_1)$ governing the placement of the equivalent line source. However, by empirically fitting the approximate method to the exact calculations of Report WAPD-RM-213, it has been found possible to reduce the many pages of curves in that report to three curves for Z plus the $F(\Theta,b)$ curves.

For values of $a/R_o \geq 10$ the curve of $\mu_s Z$ vs. $\mu_s R_o$ (which originally appeared in the Project Handbook[2]) has been found to yield an error in the flux of less than 10 per cent. For certain values of $a/R_o < 10$, this curve will yield fluxes which are off by more than a factor of 10.

For values of $a/R_o < 10$ the other two graphs should be used in conjunction with each other to obtain $\mu_s Z$. Knowing R_o,a, and μs, find m from the first graph; knowing a/R_o and b_l, find μ Z/m from the second graph; then multiply these together to obtain $\mu_s Z^s$ In all cases the error in the flux will be equal to, or less than, +40 per cent and -5 per cent. The maximum positive errors tend to appear with large values of b_l and small values of a/R_o; the maximum negative errors appear with large values of a/R_o.

12-A.1.3 Exterior on End



P_1 is a point on the axis of a right-circular cylinder of radius R_o and height h.

An upper limit to the uncollided flux that reaches Pl can be obtained by replacing the cylinder with a truncated cone, the apex of which is at P_1, The radius of that face of the truncated cone nearest the point P_l is taken to be R_o. Defining

4

$$\theta_1 \equiv \tan^{-1}\left(\frac{R_0}{a}\right)$$

then, from the equations for the truncated cone source

$$\theta_u = \frac{S_v}{2\mu_s}\left[E_2(b_1) - E_2(b_3) + \frac{E_2(b_3 \sec\theta_1)}{\sec\theta_1} - \frac{E_2(b_1 \sec\theta_1)}{\sec\theta_1}\right]$$

A lower limit can be obtained by replacing the cylinder with a truncated cone having apex angle

$$\theta_u \equiv \tan^{-1}\left(\frac{R_0}{a+h}\right)$$

12A.2 SAMPLE DOSE CALCULATION FOR PRESSURIZER

The activities in the pressurizer used in the calculation are listed in Table 12A-2. The activities are combined into energy groups.

12A.2.1 Vapor Phase Dose Contribution

The contribution of the vapor phase begins 6 feet above floor elevation 286. The calculation assumes that the concentrations vary little in the horizontal plane.

Assume

t_s = 2'0"

$R_o^2 = 1.37 \times 10^4$

$\pi R^2 = 4.34 \times 10^4$

Ignore the thickness of vessel

E(Mev)	S_L	μ_c	b	K	B	F(θ, b)	Dose Rate (mr/hr)
0.1	8×10^{11}	0.28	17.1	1.6(-4)	70	10^{-8}	0.053
0.4	5.9×10^{9}	0.215	13.0	8.2(-4)	44	7.4×10^{-7}	0.0926
0.8	9.9×10^{10}	0.16	9.76	1.6(-3)	24	2×10^{-5}	44.8
2.5	3.2×10^{9}	0.092	5.62	3.8(-3)	4.6	1.5×10^{-3}	49.5/94.4 mr/hr

$$a = (3.85+3+2)(30.5) = 270 \text{ cm} \quad \theta = \tan^{-1}\frac{6.5}{8875} = 36^\circ$$

t_s = 3'0" a = 300 cm

E(Mev)	b	B	F(θ, b)	DR (mr/hr)
0.1 }				
0.4 } negligible compared to other two				
0.8	14.6	44	1.4×10^{-7}	0.52
2.5	8.4	6.6	8.2×10^{-5}	3.50
				4.02 mr/hr

12A.2.2 Liquid Phase Dose Contribution

The contribution of the liquid phase at the same point is:

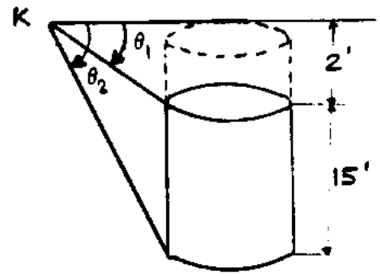
$$DR = \frac{K\,S_v R^2\,B}{4(a+z)} \frac{\lfloor F(b_{2,}\theta_2) - F(b_1,\theta_1)\rfloor}{}$$

where:

$a/R_o = 1.25$

$a+R_o = 270$ cm

$a = 150$ cm



$$\theta_1 = \tan^{-1} \frac{61}{a+Z} \qquad \theta_2 = \tan^{-1} \frac{518}{a+Z} - 45^0$$

$R_o^2 = 1.37 \times 10^4$

E Mev	μ_s	μ_c	B	K	$\mu_s(a+R_o)$	m	b_1	$1/m\mu_sZ$	μ_sZ	Z	Θ_1 ()
0.4	0.105	0.215	44	8.2(-4)	28.3	2.15	13.0	1.70	3.65	36.0	18.2^0
0.8	0.08	0.16	24	1.6(-3)	21.6	1.88	9.76	1.54	2.90	36.3	18.2^0
1.3	0.064	0.13	11.7	2.4(-3)	17.3	1.70	7.95	1.47	2.50	39.1	18.2^0
1.7	0.054	0.11	7.8	2.9(-3)	14.6	1.60	6.75	1.45	2.32	43.0	18.2^0
2.2	0.046	0.098	5.4	3.5(-3)	12.4	1.50	5.97	1.50	2.25	51.2	16.9^0
6.2*	0.028	0.06	12.4	7.4(-3)	7.6	1.20	3.72	1.52	1.82	65.0	15.7^0

E (Mev)	b_2	$F(\Theta_2,b_2)$	$F(\Theta,b_2)$	Difference	dose rate (mr/hr)
0.4	16.65	1.9(-8)	1.5(-8)	4.0(-9)	negligible
0.8	12.66	1.0(-6)	8.4(-7)	1.6(-7)	0.0028
1.3	10.45	1.05(-5)	7.5(-6)	3.0(-6)	0.0018
1.7	9.07	4.5(-5)	3.0(-5)	1.5(-5)	0.00026
2.2	8.22	1.1(-4)	6.6(-5)	4.4(-5)	0.0262
6.2*	5.52.	1.9(-3)	1.0(-3)	9.0(-4)	2.43

Normal Operating dose rate = 2.46 mr/hr
Shutdown dose rate = 0.030

*During normal operation only.

The dose rate from the liquid phase at elevations below 286' is simply twice that given by using $F(\Theta_2, b_2)$ instead of the difference of the two functions. Thus, for t_e = 2'0" we have:

	E (Mev)	dose rate (mr/hr)
	0.4	negligible
	0.8	0.218
	1.3	0.0126
	1.7	0.00156
	2.2	0.131
*	6.2	10.24

Normal = 10.42 $\frac{mr}{hr}$ operating dose rate

Shutdown = 0.167 $\frac{mr}{hr}$ dose rate

with t_s = 3'0" a/R_0 = 1.56 $(a + R_0)$ = 300cm $2_2 >$ 45

E (Mev)	b_1	B	$\mu S(a+R_0)$	m	$\frac{1}{m}\mu_s Z$	$m_s Z$	Z	b_2	Θ_1
0.4	20.15	108	30.3	2.22	1.82	4.05	38.5	24.2	15.5^0
0.8	15.5	0.44	23.1	1.93	1.75	3.38	42.1	18.88	15.5^0
1.3	12.05	19.5	18.5	1.75	1.65	2.89	45.1	14.94	15.5^0
1.7	10.48	11.8	15.6	1.64	1.57	2.58	47.8	13.06	15.5^0
2.2	8.97	8.0	13.3	1.55	1.50	2.33	50.5	11.3	15.5^0
6.2*	5.58	3.0	8.14	1.26	1.45	1.825	65.0	7.40	14.5^0

E (Mev)	$F(\Theta_2, b_2)$	$F(\Theta_1 b_2)$	Diff.	dose rate(mr/hr)
0.8	1.8 (-9)	1.4 (-9)	4.0(-10)	
1.3	1.0 (-7)	7.0 (-8)	3.0(-8)	negligible
1.7	6.4 (-7)	4.8 (-7)	1.6(-7)	
2.2	4.2 (-6)	2.9 (-6)	1.3(-6)	
6.2	2.6 (-4)	1.5 (-4)	1.1(-4)	0.34
0.4		negligible		
				0.34 mr/hr

For the elevations below 286' with t_s = 3'0" during normal operation we have DR= 1.61 mr/hr.

12A.2.3 Dose Rates from Deposited Activities

The source geometry of deposited activity is assumed to be an infinite planar source with the strengths given in Table 12A-2. Thus:

$$DR = \frac{KBS_A}{2} E_1(b_1)$$

The contribution from deposited activities for the 1.7 Mev group (S_A 1.41 x 10^5) and a 2 cm wall of steel (b 1, = 0.72) at contact with the surface of the vessel is:

$$DR = \frac{2.9 \text{ x } 10^{-3}(1.41 \text{ x } 10^5)2.5(0.35)}{2} = 1.43 \text{ x } 10^2 \text{ mr/hr}$$

For a 2'-O concrete wall, however, the largest contribution will not exceed
(b_1 = 6.75) 0.0715 mr/hr.

The contribution to the dose rate from deposited activity is only important during shutdown, and outside a 2'-O" wall will be less than 0.2 mr/hr.

Figure 12A-2 plots dose rates for all three phases versus shield wall thickness.

12.A.3 STEAM GENERATOR DOSE CALCULATION

Pertinent data for the steam generator dose calculation is given below:

Reactor Coolant Water Volume	921 ft^3
Secondary Side Water Volume	1910 ft^3 at rated load
	3730 ft^3 at no load

Cold leg ID - 271/2"
Hot leg ID - 31 "

One must calculate the effective μ s for the steam generator bundle section.

The free volume of this section = 3100 ft^3
The water volume = 2831 ft^3
Volume of metal = 331 ft^3

Equivalent $$\mu_s = \frac{\mu_{H_20}\, \upsilon_{H_20} + \mu_{Fe}\, \upsilon_{Fe}}{\upsilon_{H_2} = \upsilon_{Fe}} = \frac{(0.028)(2831) + (0.24)(331)}{3100}$$

$$= \frac{79.3 + 79.5}{3100} = \frac{158.8}{3100} = 0.051$$

From Figure 7.1 of Westinghouse's source term manual the (activity of N-16 at the steam generator inlet is:

2.22 x 10^6 γ/cc-sec (this is the activity of the hot leg)

The γ activity of the steam generator outlet is:

1.575×10^6 γ/cc-sec (middle leg)

Average N-16 activity = 1.897×10^6 γ/cc-sec

a) El 261 and above contribution for normal operation

Try t_w = 2'-0" a = 198cm Ro = 157cm a/Ro = 1.26
a+Ro = 355cm μ_s = 0.051

E	S:	K	B	μ_s(a+Ro)	m	b_1	1/mμ_sZ	μ_sZ
6.2	1.575(6)	7.4(-3)	2.4	18.1	1.71	6.12	1.63	2.79

Z	b_2	a+Z	2	F(Θ_1,b_2)	D R (mr/hr)
54.6	8.91	252.6	45	5.3×10^{-5}	78.0

With t_w = 3'-0" a = 228.5 a/Ro = 1.53 a+Ro = 385.50 Θ>450

E	B	μ_s(a+Ro)	m	b_1	1/m	μ_sZ	$\mu_s Z_2$	Z_2	b_2	a+Z
6.2	3.0	19.65	1.80	7.98	1.45		2.61	51.0	10.59	279.5

F(Θ_1 b_2) = 9.0×10^{-6} $\therefore$ DR = 13.0 (mr/hr)

Both the above are conservative since the activities of the steam generator shell is probably more than 4", and the emergent γ's will not have an energy of 6.2 Mev but rather a spectrum of energies with max energy = 6.2 Mev.

Use t_e = 3'3"

b) El 286 and above contribution for normal power operation

Calculate the contribution above the 286 floor at 6', where the tube bundles end. From preceding calculations, the contribution is obviously just 1/2 times that calculated for elevations below El 286, hence for a

2' shield - 39 mr/hr
3' shield - 7 mr/hr

c) El 261 and above during shutdown

Max activity conditions during shutdown are the same as those for the RHR system and the letdown system, i.e.

<u>Energy</u> (Mev)	<u>Specific Source Strength γ/cc-sec</u>
0.4	1.25×10^6
0.8	3.14×10^5
1.3	1.38×10^5
1.7	7.65×10^4
2.2	9.55×10^4
2.5	7.6×10^4
3.5	6.0×10^3

In addition after a given length of operating time corrosion product activity will be deposited on the tube bundle surfaces. The maximum deposited activity (36 mo. operation) is:

Mn-54 = 7.4×10^4 γ/cm^2 sec
Mn-56 = 1.22×10^5 γ/cm^2 sec
Co-58 = 4.07×10^5 γ/cm^2 sec
Fe-59 = 1.11×10^5 γ/cm^2 sec
Co-60 = 1.3×10^5 γ/cm^2 sec

No. of U-tubes = 4674 Reactor Water Coolant Vol - 921 ft 3
O.D of U-tubes = 0.75 in.
Average U tube thickness = 0.043 in. Length = 72.5 ft.
I.D of U tubes = 0.707 in. Inside Surface Area = 5.85×10^6 cm^2

To calculate dose rate at shutdown from the deposited corrosion product, assume that they can be treated as a volumetric source distributed in the volume of the tube bundles, i.e.

E(Mev)
0.84 Mn-54 7.4×10^4 γ/cm^{2}sec (5.85×10^6 cm^2)/(2.61×10^7cm^3)
=1.66×10^4(/cc-sec

2.1 Mn-56 2.73×10^4 γ/cc-sec

1.60 Co-58 9.1×10^4 γ/cc-sec

1.30 Fe-59 2.48×10^4 γ/cc-sec

1.25 Co-60 2.92×10^4 γ/cc-sec

2'-0" concrete

E	μ_s*	b_{Fe}	b_c	B	K	b_1	μ_s (a+Ro)	m	1/mμ_sZ
0.4	0.170	7.0	13.0	44.0	8.2(-4)	20.0	60.0	3.0	1.95
0.8	0.125	5.2	9.76	24.0	1.6(-3)	14.96	44.4	2.7	1.85
1.3	0.097	4.2	7.95	11.7	2.4(-3)	12.15	34.4	2.3	1.78
1.7	0.825	3.6	6.75	7.8	2.9(-3)	10.35	29.3	2.1	1.72
2.2	0.076	3.2	5.97	5.4	3.5(-3)	9.17	27.0	2.02	1.68
2.5	0.0705	3.0	5.61	4.6	3.8(-3)	8.61	25.0	1.95	1.67
3.5	0.061	2.62	4.75	3.3	4.8(-3)	7.37	21.6	1.82	1.64

*

$$s = \frac{^{\mu}H_2 0}{} \frac{^{\mu}H_2 0}{^{\mu}total} + \frac{^{\mu}Fe\ ^{\mu}Fe}{}$$

a=198 Ro = 157 a/Ro = 1.26

a+Ro = Ro^2 = 2.48 x 10^4

E	$\mu_s Z$	Z	b_2	$F(\theta 1, b_2)$	(a+Z)	DR (mr/hr)
0.4	5.85	34.4	25.85	1.3 x 10^{-12}	232.4	3.3 x 10^{-6-}
0.8	5.0	40.0	19.96	5.5 x 10^{-10}	238.0	3.45 x 10^{-4}
1.3	4.09	42.2	16.24	2.6 x 10^{-8}	240.2	7.9 x 10^{-3}
1.7	3.62	43.8	13.97	2.7 x 10^{-7}	241.8	5.25 x 10^{-2}
2.2	3.40	44.8	12.57	1.15 x 10^{-}	242.8	1.35 x 10^{-1}
2.5	3.26	46.1	11.87	2.3 x 10^{-6}	244.1	1.55 x 10^{-1}
3.5	2.98	49.0	10.35	1.5 x 10^{-}	247.0	5.7 x 10^{-2}
						0.407 mr/hr

Dose rate due to fission and corrosion products in reactor water and deposited activity.

		2.67 x 10^{-3}	1.3 Mev
		2.85 x 10^{-2}	1.7 Mev
		3.01 x 10^{-2}	2.2 Mev
Dose rate due to deposited activity alone	=	0.0613 mr/hr	

Obviously the thickness of concrete required for normal operation will be more than sufficient for shutdown purposes.

A plot of dose rate versus shield thickness is given in Figure 12A-3.

12A.4 DATA USED IN EXAMPLE CALCULATION

The data used in calculating the dose rates for the pressurizer and steam generator are given in Figure 12A-4 through 12A-8.

TABLE 12A-1

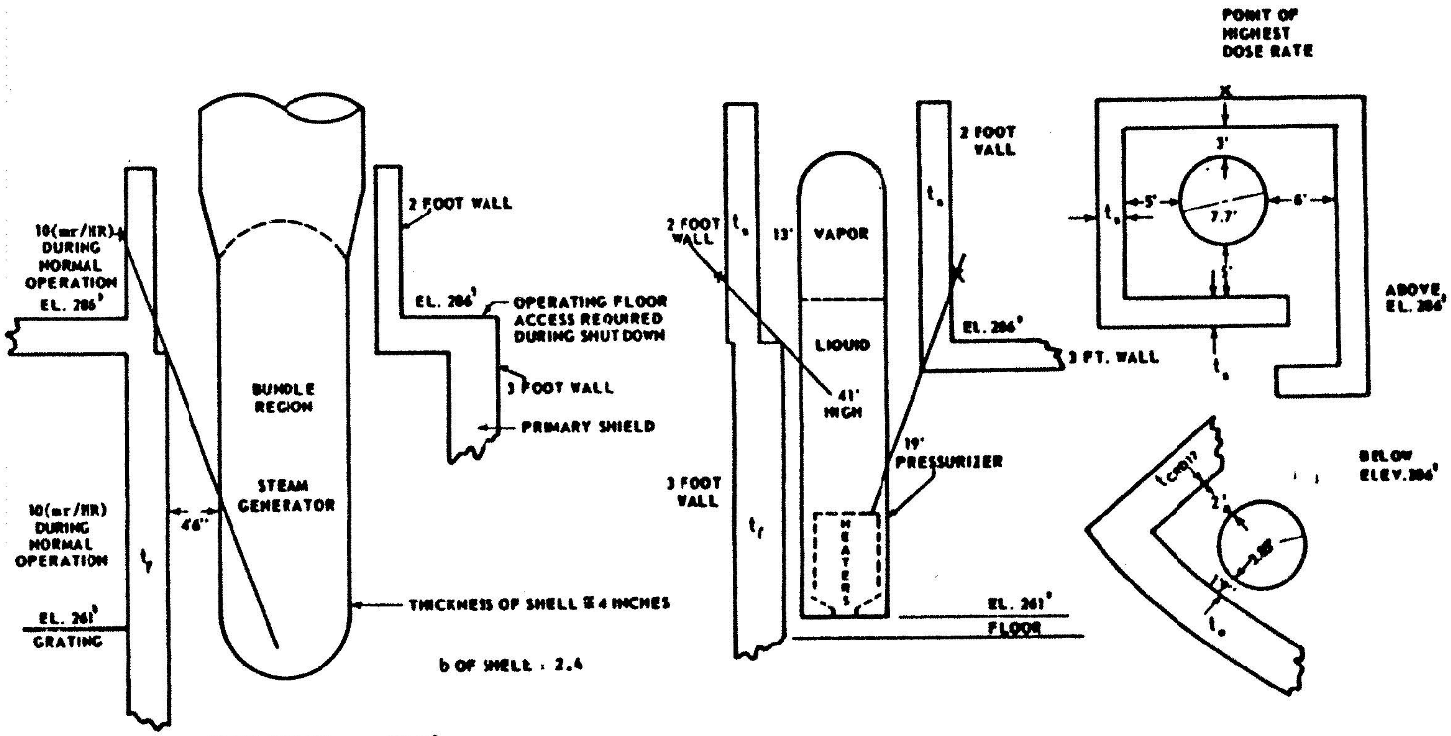
COMPARISON OF PREDICTED AND MEASURED DOSE RATES

Location	Predicted Value	Measured Value
Outside 2 foot thick pressurizer shield wall	(@ 100 percent power) 25 mrem/hr	(@ 90 percent power) 25 mrem/hr
Outside 2 foot thick upper portion of steam generator shield wall	39 mrem/hr	10 mrem/hr

TABLE 12A-2

COMBINED ENERGY GROUPS OF PRESSURIZER ACTIVITIES

E (MEV)	#/cc-sec
Liquid Phase	
0.4	4.8×10^4
0.8	7.75×10^4
1.3	3.7×10^3
1.7	2.03×10^4
2.2	5.83×10^3
6.2	3.0×10^4
Vapor Phase	
0.1	1.85×10^7
0.4	1.36×10^5
0.8	2.29×10^6
2.5	$7.4 \times l0^4$
Deposited Activity	
	#/cm^2-sec
0.4	3.63×10^3
1.3	1.66×10^4
1.7	1.41×10^5
2.2	8.13×10^2



LIQUID

LIQUID VOLUME	840 ft.³
STEAM VOLUME	560 ft.³
TOTAL PU VOLUME	1400 ft.³
HEATER VOLUME	380 ft.³
TOTAL VOLUME	1780 ft.³

Amendment 15, (1/97)

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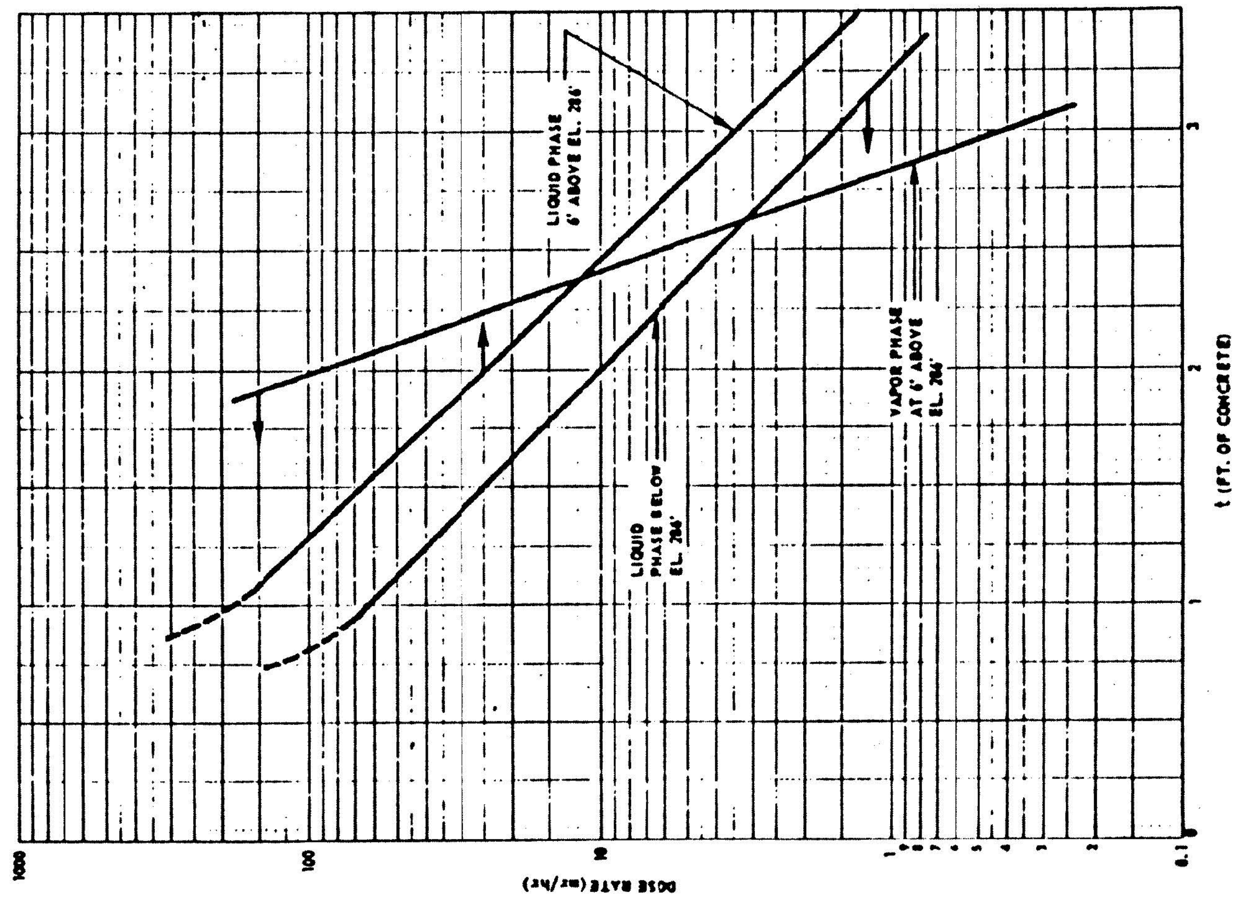
Example Pressurizer and Steam Generator Geometry
FIGURE 12A-1



Amendment 15, (1/97)

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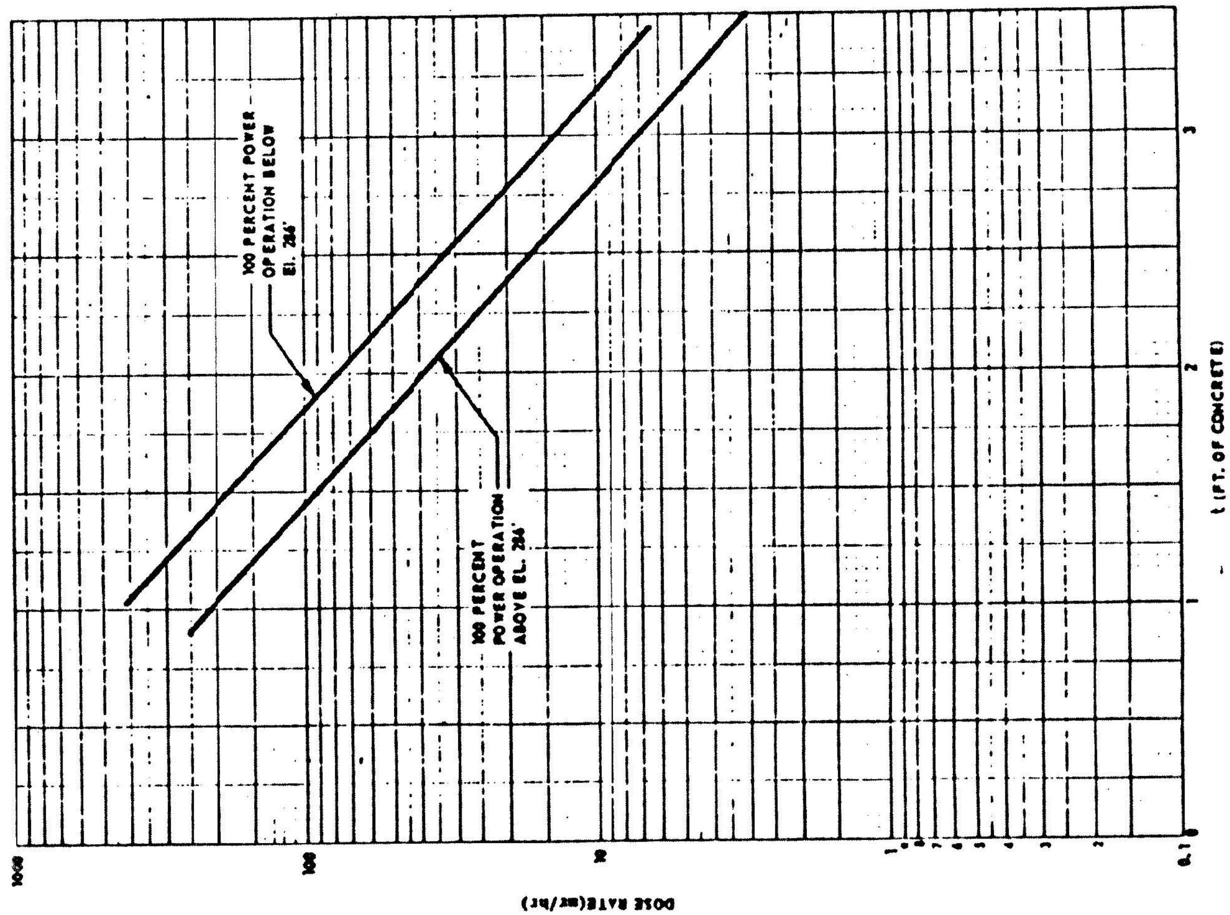
Example Pressurizer
Dose Rates
FIGURE 12A-2



Amendment 15, (1/97)

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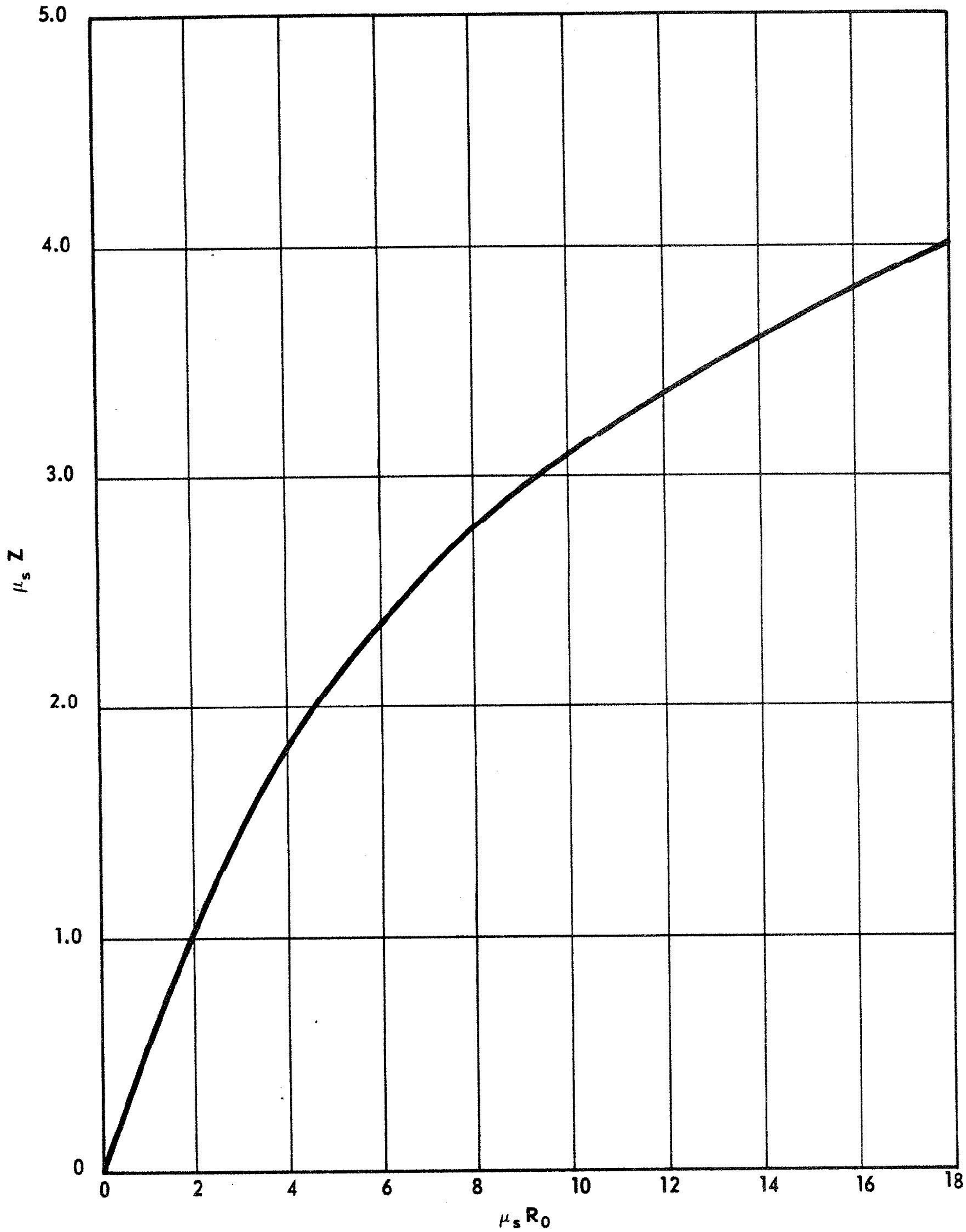
Example Steam Generator
Dose Rates
FIGURE 12A-3



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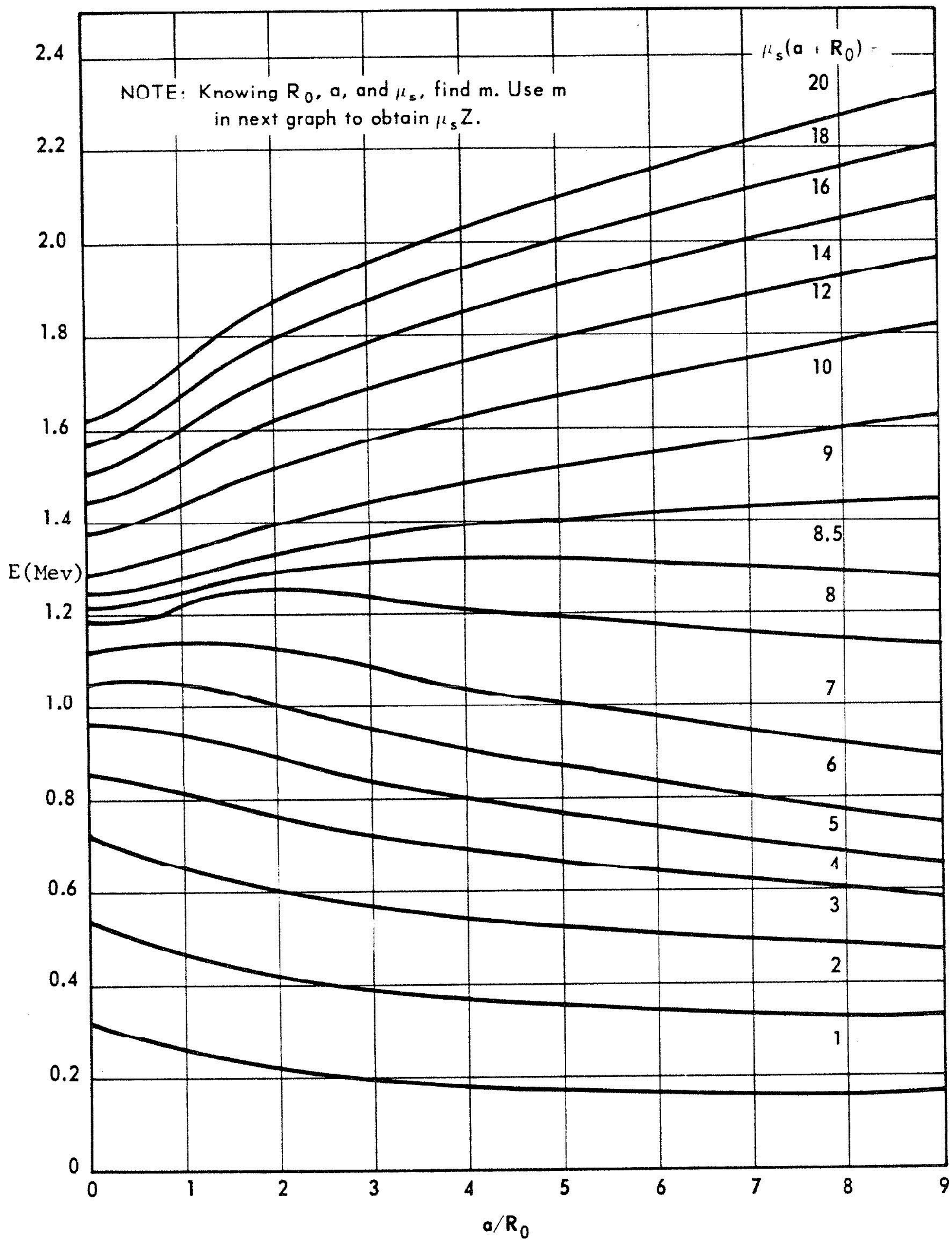
SELF-ABSORPTION DISTANCE, Z,
OF A CYLINDER
AS A FUNCTION OF CYLINDER DIAMETER, R_0,
FOR $a/Ro>10$

FIGURE 12A-4



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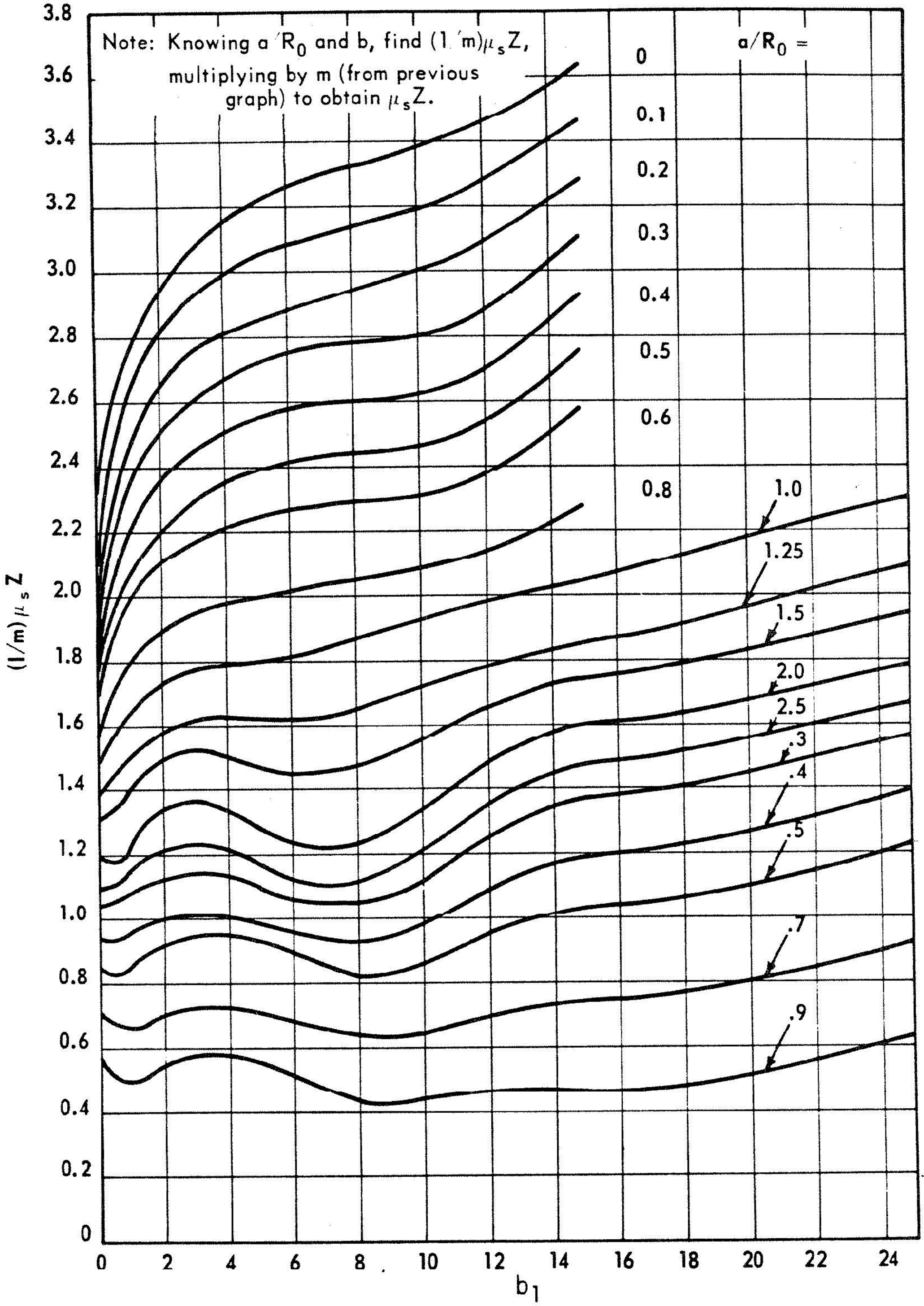
SELF-ABSORPTION DISTANCE, Z, OF A CYLINDER AS A FUNCTION OF CYLINDER DIAMETER, R_0, FOR $a/R_0 < 10$

FIGURE 12 A-5



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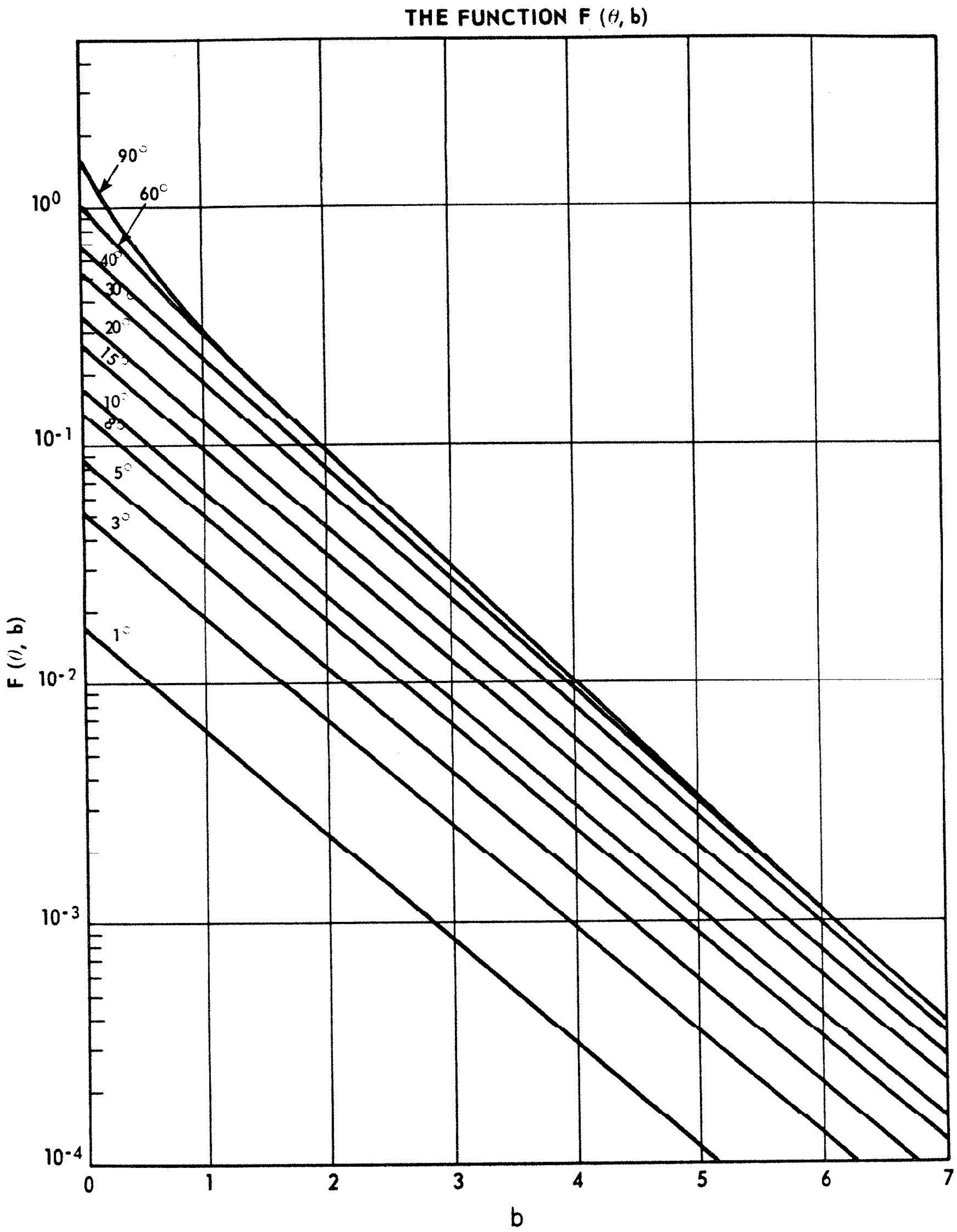
SELF-ABSORPTION DISTANCE, Z, OF A CYLINDER AS A FUNCTION OF CYLINDER DIAMETER, R_0, FOR a R_0 10

FIGURE 12 A–6



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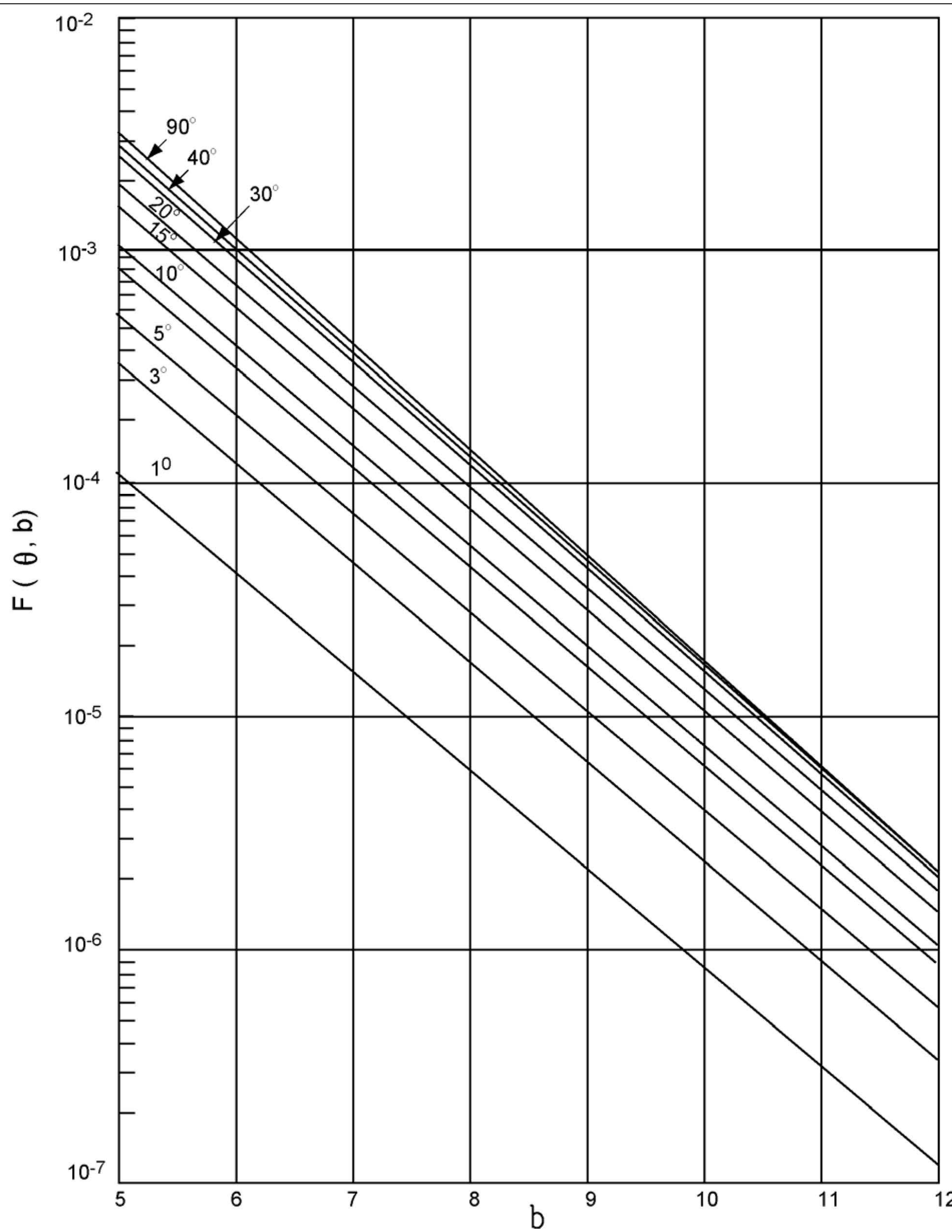
THE FUNCTION F (θ, b)
F vs. b
For F = 10^{-4} to 10; b = 0 to 7; θ = 1 to 90
FIGURE 12A-7a



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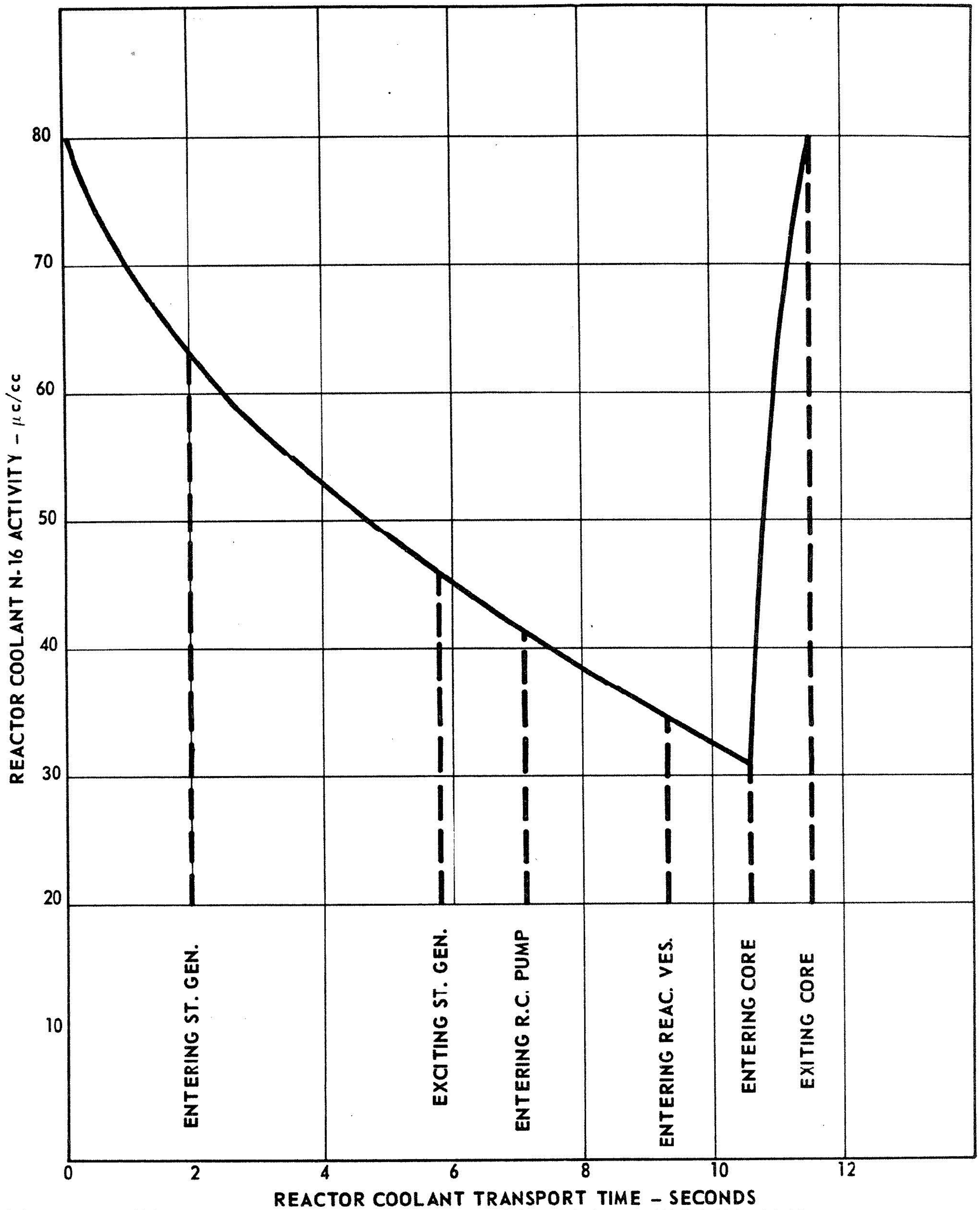
THE FUNCTION F (θ,b)

F vs. b

For F = 10^{-7} to 10^{-2}; b = 5 to 12; θ = 1 to 90

FIGURE 12A-7b



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H.B. ROBINSON No. 2 – 2295 MWt
REACTOR COOLANT N-16 ACTIVITY
FIGURE 12 A–8

APPENDIX 12B

EXAMPLE CALCULATION OF SCATTERED

RADIATION CONTRIBUTION TO DOSE RATES THROUGH PARTIAL SHIELD WALLS

As an example of calculations of scattered radiation contribution to dose rates through partial height walls, the following discusses the procedure used to evaluate the scattered dose rate at two points in the equipment and chemical drain tank cubicle area, one inside the pump cubicle, and one in the corridor.

Since the mean free path in air for gamma radiation of the spectrum listed below is very long (i.e., 200-300 ft for the lowest energy) the major scattered contribution will be due to radiation scattering off walls and ceilings. For purposes of conservatism it is assumed that both tanks are filled with fluid having the maximum expected activity of the equipment drain tanks).

This maximum activity can be broken into the following groups of maximum energy.

Gamma Energy	Activity (MeV/cc sec)
0.0 - 0.4	1.66×10^{3}
0.4 - 0.8	2.77×10^{3}
0.8 - 1.3	2.40×10^{2}
1.3 - 1.7	3.61×10^{1}
1.7 - 2.2	2.37×10^{1}
2.2 - 2.5	1.11×10^{0}
2.5 - 3.5	1.33×10^{-1}

The geometry of the drain tanks, walls, etc. and the location of the two points are shown in Figure 12B-1.

The radiation sources are approximated by a point source. The scattering off the walls and ceilings of the radiation from the equivalent point source is then analyzed as follows (see Figure 12B-2):

$r_l = d_l/\cos\theta_o$ $\qquad$ $r_2 = d_2/\cos\theta_1$

$$\therefore \theta_0 = \cos^{-1}\left(\frac{d_1}{r_1}\right) \qquad \theta_1 = \cos^{-1}\left(\frac{d_2}{I_2}\right)$$

where

$$r_1 = \left(x^2 + y_2 + d_1^2\right)^{1/2} \quad \text{and} \quad r^2 = ((x_3 - x)^2 + \left(y_3 - y\right)^2 + d^2)^{1/2}$$

Similarly

$$\phi_1 = \tan^{-1}\left(\frac{x_3 - x}{y_3 - y}\right)$$

$$\phi_0 = \tan^{-1}\left(\frac{x}{y}\right)$$

$$\phi = \phi_1 + \phi_o$$

The angle of scattering is given by

$\cos\theta_2 = \sin\theta_o \cos\Phi \sin\theta_1 - \cos\theta_0 \cos\theta_1$

The Klein-Nishina x-sect is given by

$$K(\theta_s) = \frac{(2.818 \ x\,10^{-13})^2}{2} \quad p^2 \left(1 + p^2 - p(1 - \cos^2\theta_s)\right)$$

Where

$$P \equiv (1 + \frac{E_0}{.511}(1 \ \cos \ \theta_s))^{-1} \qquad E_0 \ \textit{(source energy in MeV)}$$

If D_l is the dose rate at a unit distance from our hypothetical point source in air, then

$$dD = \frac{D_1 \alpha \cos\theta_0 \, dx \, dy}{r_1^2 \ r_2^2}$$

is the differential dose rate at the detection point due to scattering
from an elemental area dx dy

$$\alpha = \frac{(C\,K(\theta_s)\,10^{26} + C')\cos\theta_1}{\cos\theta_1 + \cos\theta_0}$$

Herein C and C' are the Chilton-Huddleston* coefficients.

The total scattered dose rate is of course obtained by integration over
the entire scattering surface

$$D = D_1 \int_{x2}^{x1} dx \int_{y2}^{y1} dy \frac{(C\,K(\theta_5)10^{26} + C')\cos\theta_1 \, \cos^3\theta_0}{(\cos\theta_1 + \cos\theta_0) \ d_1^2 \ \left((x^3 - x)^2 + r_2^2 (y^3 - y)^2 + d_2^2\right)}$$

The above integral is solved numerically by use of Gaussian quadrature.

First the limits are changed:

$$\int_a^b f(x)\,dx = \int_{-1}^{1} f\left(\frac{b-a}{2}y + \frac{b+a}{2}\right)\left(\frac{b-a}{2}\right)dy = \frac{b-a}{2}\sum_{n=1}^{N}\omega_1 f(x_1)$$

*Radiation Shielding, TR-40, Nov. 1966

where T are the Christoffel's numbers and x are the zeroes of the Legendre polynominals and

$$x_I \equiv \frac{b - a}{2} y_i + \frac{b + a}{2}$$

Since each tank contains 1000 gallons (≈3785 liters = 3.785 x 10^6 cc) the activity inventory of each tank is given by the sums of the activity of all energy groups.

4150 #/cc sec	
3460	
185	
21.3	
10.7	
7827.0	7830 #/cc sec x 3.785 x 10^6 cc = 2.96 x 10^{10} #/sec

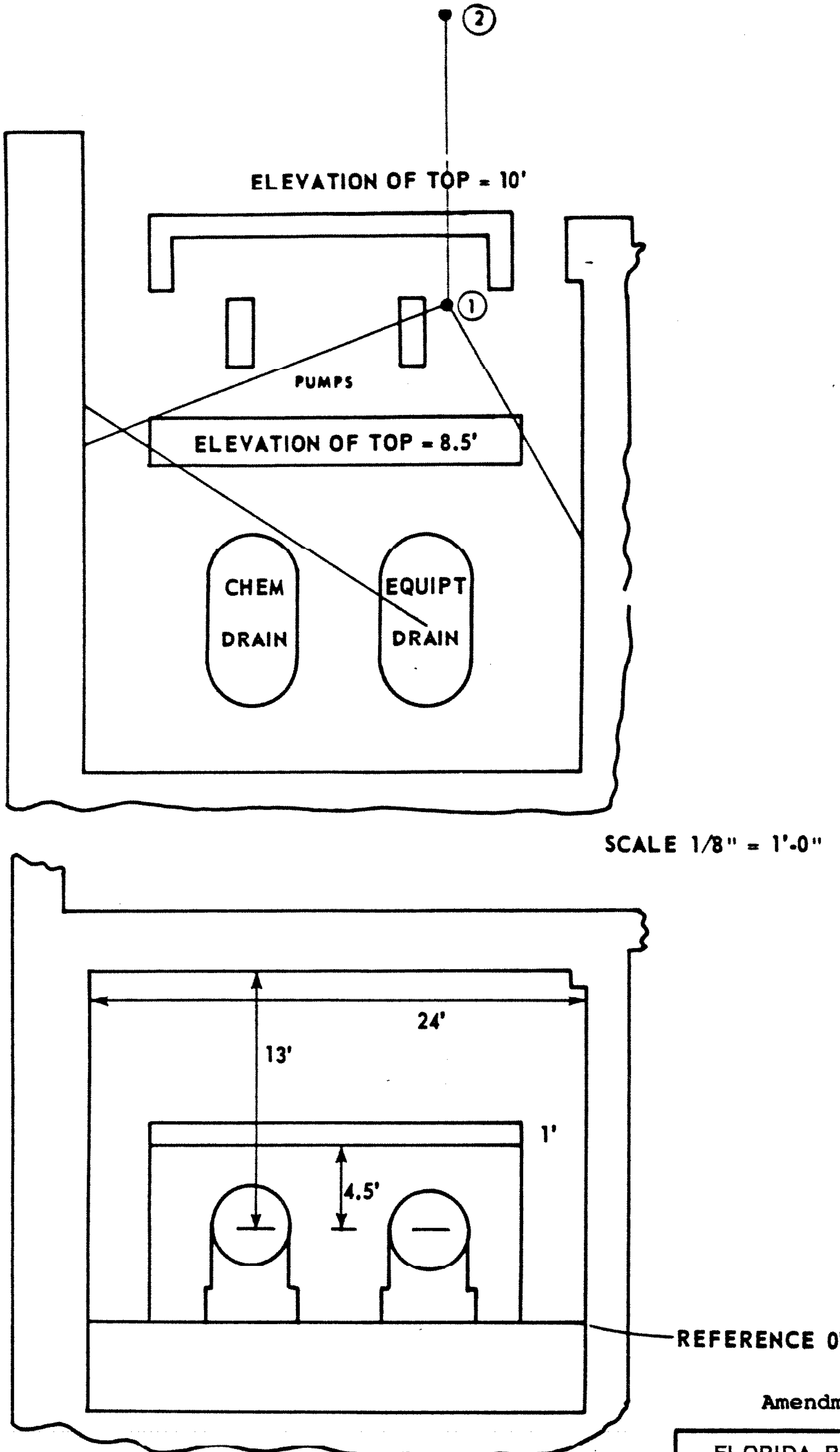
Maximum activity in one tank = 0.8 Ci (assumed at 1.25 Mev)

Neglecting all self-shielding, wall attenuation, etc., and concentrating all of this activity into one point source, then the dose rate at a unit distance away from that source in air is given by:

$$DR = \frac{{}_s Ke^{-\mu}}{4\pi} = \frac{2.69 \text{ x } 10^{10}\ 2.3 \text{ x }}{4\pi} \frac{10^{-3}}{} = 5.41 \text{ x } 10^6 \ mR/hr$$

Hence, using this very conservative dose rate at a unit distance away from the point source, the scattered contributions at points one and two are, assuming both tanks at maximum activity;

at point one -≤1.0 mr/hr
at point two -≤0.2 mr/hr



Amendment 15, (1/97)

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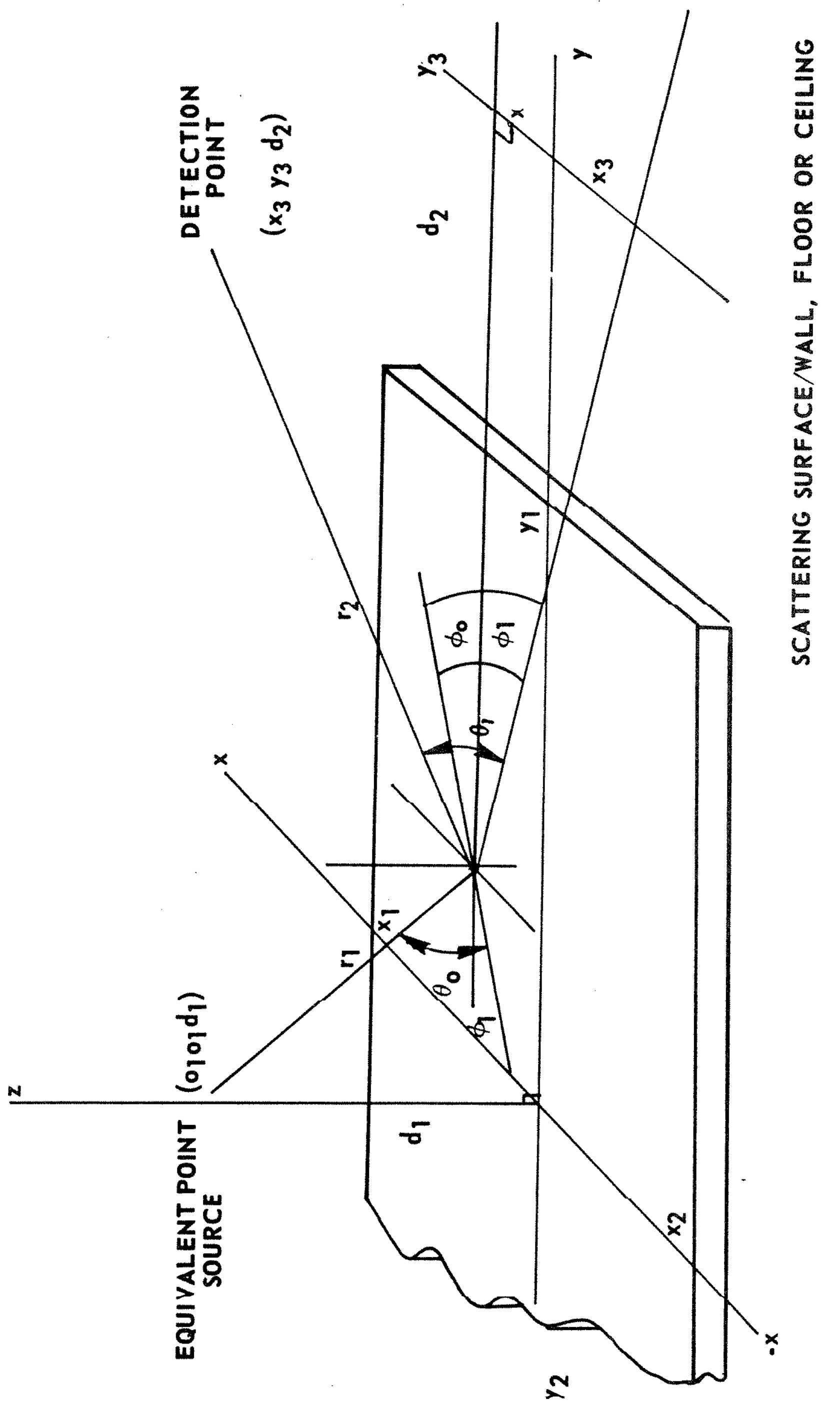
RADIATION SCATTERING DIAGRAM FOR EQUIPMENT AND CHEMICAL DRAIN TANK AREA

FIGURE 12B-1



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GEOMETRIC REPRESENTATION OF RADIATION SCATTERING

FIGURE 12B-2