CHAPTER 7

RADIATION PROTECTION

7.0 RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE ALARA

7.1.1 POLICY CONSIDERATIONS

The Carolina Power & Light Company (CP&L) corporate and facility (H. B. Robinson) health physics policies, described in Section 12.1.1 of Reference 7.1, are applicable to the Independent Spent Fuel Storage Installation (ISFSI). Carolina Power & Light Company is committed to a program of keeping occupational radiation exposure as low as reasonably achievable (ALARA). The Company follows the general guidance of Regulatory Guides 1.8, 8.8, 8.10, and publications that deal with ALARA concepts and practices, including Title 10, Code of Federal Regulations, Part 20.

The goals and objectives of the health physics programs are to maintain the annual dose to individual facility personnel to as low as reasonably achievable and to maintain the annual integrated dose to facility personnel; i.e., the sum of annual doses (expressed in man-rem) to all facility personnel, as low as reasonably achievable. The health physics programs identify the organizations participating in the programs, the positions involved, and the responsibilities and functions of the various positions in conducting the programs.

Adequate trained personnel are provided to develop and conduct all necessary health physics programs. The health physics personnel possess the necessary training and expertise to carry out the health physics programs in an efficient manner to assure that Company and regulatory requirements are met.

Appropriate training programs in the fundamentals of radiation protection and facility exposure control procedures are established to provide instructions to all facility personnel including contractors whose duties require working in radiation areas. Training programs for health physics personnel are provided to improve their performance in the health physics programs.

7.1.2 DESIGN CONSIDERATIONS

The designs of the dry shielded canister (DSC) and horizontal storage module (HSM) comply with 10 CFR 72 concerning ALARA considerations. Specific considerations that are directed toward ensuring ALARA are:

a) Thick concrete walls on the HSM to reduce the surface dose to an average of less than 20 mrem/hr.

b) Lead shield plugs on the ends of the canister to reduce the dose to workers performing the drying and sealing operations.

c) Use of a shielded transfer cask for handling and transportation operations involving loaded DSCs.

d) Fuel loading procedures which follow accepted practice and build on existing experience.

e) Recess in HSM front for cask docking to reduce scattered radiation during transfer.

f) Double seal welds on each end of DSC to provide redundant radioactive material containment.

g) Placing demineralized water in the cask annulus and DSC and sealing the DSC-cask gap to minimize contamination of the DSC exterior during loading.

h) Placing external shielding blocks over the HSM air outlets to reduce direct and streaming doses.

i) Passive system design that requires minimum maintenance.

j) Insertion of internal shielding blocks around air inlets to reduce direct and streaming doses.

k) Development of shipping procedures based upon previously used procedures and experience to control contamination during handling and transfer of fuel.

l) Use of additional shielding in front access cover plates.

7.1.3 OPERATIONAL CONSIDERATIONS

Operational considerations at HBR2 that promote the ALARA philosophy include the determination of the origins of radiation exposures, the proper training of personnel, the preparation of radiation protection procedures, the development of conditions for implementing these procedures, and the formation of a review system to assess the effectiveness of the ALARA philosophy.

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

Procedures for the ISFSI were integrated into existing HBR2 procedures and incorporate the same ALARA philosophy.

The ISFSI is considered a radiation control area. Therefore, a fence with a locked gate is located partially around the ISFSI. The HBR2 Radiation Control unit controls the key to this area.

7.2 RADIATION SOURCES

7.2.1 CHARACTERIZATION OF SOURCES

Table 3.1-2 lists the radiation sources to be stored at the Robinson ISFSI. Due to the lower initial enrichment and the longer burnup, the Robinson fuel has slightly different neutron and gamma sources compared to the NUTECH Topical Report (Reference 7.2). The neutron source is slightly larger and the gamma source is slightly smaller. However, based on the analysis described in Section 7.3, the shielding of the DSC and the HSM are sufficient to ensure that the dose limits are not exceeded.

7.2.2 AIRBORNE SOURCES

Loading the irradiated fuel into the DSC is carried out under water and is identical to the existing HBR2 loading procedures for the IF-300 cask. Airborne sources are processed by the existing fuel building ventilation system.

The sealing and drying of the DSC are performed under procedures to prevent any gaseous leakage. All vent lines (liquid and gaseous) are routed to the existing HBR2 radioactive waste processing systems. Once the DSC is dried and seal welded, there are no design basis accidents which can cause breaching of the DSC and the airborne release of radioactivity. However, to demonstrate ultimate safety, a postulated leak is analyzed in Section 8.2.8 of Reference 7.2.

At the storage location, the only potential for airborne radioactivity is from any potential contamination of the DSC exterior. However, as described in Section 5.1 of the NUHOMS Topical Report, procedures are used to minimize this contamination and should limit any possible DSC surface contamination to below the limits specified in the Technical Specifications. There is essentially no onsite release of radioactivity from the DSC exterior and therefore does not present any health hazard.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 INSTALLATION DESIGN FEATURES

The ISFSI is a passive outdoor storage system. Each HSM is capable of providing sufficient ventilation to ensure adequate cooling of the DSC and its contents. The convective cooling system is completely passive and requires no filtration system.

7.3.2 SHIELDING

7.3.2.1 Radiation Shielding Design Features

Radiation shielding is an integral part of both the DSC and HSM designs. The features described in this section assure that doses to personnel and the public are ALARA.

The DSC body is a section of 0.5 inch thick, 36 inch inside diameter stainless steel pipe. Two lead-filled end plugs and three steel plates provide shielding at the ends of the DSC. During handling operations, shielding in the radial direction is provided by the IF-300 shipping cask.

Two penetrations in the top lead plug allow water draining, vacuum drying and helium backfilling of the DSC. The penetrations are located away from fuel assemblies and contain sharp, non-coplanar bends to reduce radiation streaming. Table 7.3-1 lists relevant dimensions of the shielding materials present at the ends of the canister.

The HSM provides shielding in both the radial and axial directions during the storage phase. Forty-two inch thick, portland-cement concrete walls and roofs provide the shielding. The module's front access is covered by a two-inch thick composite plate which has additional shielding.

Four penetrations in the module allow convective air cooling of the DSC and module internals. Two identical intake vents at the bottom of the front HSM wall draw air into a shielded box inside the module. The exit vents are placed at both ends of the module roof. Openings to the HSM interior are placed above the end shield regions and not directly over the active fuel region. Sharp duct bends and precast concrete shielding caps over the exhaust exits assure that radiation streaming is reduced to a minimum. Figure 4.2-2 shows details of the module penetrations.

Further details of the radiation shielding design features are presented in Section 7.3.2 of Reference 7.2.

7.3.2.2 Shielding Analysis

The shielding analysis methodology for the NUHOMS generic design (Section 7.3 of Reference 7.2) is applicable to the HBR ISFSI as described in this section. However, due to the increased burnup and decreased enrichment of the HBR fuel, the neutron and gamma source terms are slightly different from those used for the generic design. Source terms are 11.4% lower for gamma rays and 17.2% higher for neutrons. This causes the dose rates calculated in Reference 7.2 to require scaling for use in this SAR. Also, because of the revised structure criteria and the necessity of fitting into an existing shipping cask, some steel and lead plate thickness on the DSC have changed. The HSM also has a rear ram access. These design changes required complete analyses of some of the HBR ISFSI shielding.

Figure 7.3-1 shows the locations at which dose rates are presented. Table 7.3-2 shows the resulting surface doses for HBR fuel. HBR ISFSI-unique shielding calculations have been performed for points shown in Figure 7.3-1. Dose rates at other points were scaled from the values in the Topical Report. The following paragraphs provide a brief description of the analysis at each point that was reanalyzed.

Due to the different design of the bottom shield plug on the DSC the shielding analysis of the HSM front air outlet was redone (points 2.1 and 3.1). This analysis used the DOT code to calculate the neutron and gamma dose rates at point 3.1. Hand calculation was then used to calculate the attenuation by the shield cap of the radiation beam coming out of the air outlet slot. The results are shown in Table 7.3-2.

Points 4, 5, and 7 provide estimates of the dose rate at the front of the module with the door open or closed. The dose rate at point 5 was obtained with the DOT analysis of the front half of the module (same analysis used for point 3.1). Dose rates of point 7 (4.5 feet away from the open door) and point 4 (surface of closed door) were obtained by hand calculations.

Because the dose at point 6 is primarily due to radiation leaving the canister longitudinal surface (and the design for the topical and the HBR canisters are identical for the canister body), the dose rate here was obtained by scaling the topical dose rates by the factors of the source strengths. The dose rates at points 8 and 9.1 were obtained by using the dose rate on the top cover plate surface of the DSC from the DOT analysis and a hand calculation to determine the attenuation due to the 3 1/2 feet distance to the outside HSM surface and the attenuation of the above plate (point 8). The dose rate at points 8 and 9.2 were obtained from a DOT analysis of the rear of the HSM with the canister half way in the HSM (a scoping study was done by hand albedo methods to determine that this was the case for the worse dose at the uncovered rear ram exit.

The dose rates for all points on the DSC top and transfer cask were calculated using the QADMOD (gamma only), DOT and the MORSE (Monte Carlo) programs. The three different codes were used to assure that the radiation streaming through the cask-canister gap was adequately modeled and estimated. The dose rates from all three methods yielded similar results (reduction factors of 10^{-3}). The QAD and DOT analysis methodologies were described in the Topical Report. The MORSE calculations are described below.

MORSE, a three-dimensional Monte Carlo shielding code (Reference 7.5), was used to assess the severity of the neutron and gamma ray streaming which occurs when the loaded DSC is inside the transport cask. The shielding calculations were performed in 22 neutron groups and 18 gamma ray groups with a P_3 expansion of the angular distributions using a coupled cross section set. Thus, secondary gamma rays are included when primary neutrons are taken as the source.

The MORSE model was constructed in three dimensions using the MARS geometry package. The PICTURE code was used to verify the model. One octant of the cask/canister system was modeled with reflective boundaries at symmetric planes. Fuel assemblies were modeled discretely as two-zone regions containing an outer layer of (homogenized) 0.125 inch thick Boral and a homogenized interior composed of irradiated fuel and cladding. The upper two spacer disks were modeled discretely due to their significant effect of local dose rates in the area of interest. Additional spacer disks were omitted to reduce computational time and ensure conservatism. The lead region in the top shield plug was modeled as a disk with reduced diameter to account for the steel siphon line region. The cask/canister system was modeled with the nominal 0.25 inch annular gap. A boundary crossing routine was employed to determine the average dose rate in the annular gap containing air only. The choice of boundary crossing, rather than point detector collision estimating, was made in order to allow octant modeling and to improve the statistical deviation.

The source terms were obtained in a fashion similar to the NUHOMS Topical Report and will not be discussed here. Russian Roulette, path length stretching, and source energy biasing were all used to minimize statistical deviation in the area of interest. Russian Roulette weighting parameters were established based on the number of mean free paths from significant sources to the cask/canister gap. Path length stretching was in the forward direction. Adjoint XSDRNPM-S (Reference 7.5) calculations were used to determine source energy biasing parameters.

Dose rates were calculated from the fluxes by using the Snyder-Neufeld factors for neutrons and the Henderson factors for gamma rays (Reference 7.8).

The application of MORSE to the cask/canister streaming problem represents a refinement in the albedo technique used in the NUHOMS Topical Report. Sixty-four thousand, eight hundred neutron histories and 1,920,000 primary gamma ray histories were executed to obtain streaming dose rates. The neutron dose rate (which includes a negligible contribution due to secondary gamma rays) was calculated to a one-sigma deviation of 6.6%. The primary gamma ray dose rate was calculated to a one-sigma deviation of 11.9%

The dose rates reported for the DSC in the cask include numerous combinations of the presence of water in the DSC and DSC-cask gap (the gap hereafter). These combinations are present due to the operational procedures. The various cases are described below.

Point 1.1, lead plug on DSC with water in the DSC and gap is the condition during the welding of the lead plug assembly to the DSC. After welding the DSC will be drained, dried and backfilled with He. The calculations for point 1.2 reflect the estimates for the dose rate after the DSC is drained. However, it should be noted that personnel will not be required to be directly over the canister during this time.

All operations will be done from the side of the cask and only the forearms and hands of the personnel will be over the cask for a short time during the connecting and disconnecting (with "quick connect" Swagelok fittings) of water, air and He lines. The dose rate in the gap during this initial welding and drying is shown for point 3.1. Point 2 gives the dose rates on the top cover plate during the welding of that plate to the DSC body. The dose rate in the gap is given for point 3.2.

After the top cover plate is welded on the DSC, the cask lid will be lowered into position, bolted on, and then the water drained from the gaps. Point 5 gives the estimated dose rate for the top of cask lid.

Point 4 gives the dose rate at the cask side where the operating personnel will be working.

Point 6 gives the dose rate at the cask collar side during the insertion of the DSC. While the DSC is being seal welded, the lead plug is next to the inner surface of the cask collar and, hence, the dose at the outside of the collar is small. The large dose rate shown for point 6 is only present during the loading of the DSC into the HSM. This dose rate is present over a 6-inch wide ring of the collar which is not inserted inside the HSM cask docking recess. Although no personnel will be within 20 feet of this area during loading, it would be prudent, from an ALARA standpoint, to use portable lead/ polyethelyne shielding to reduce the dose rate in the vicinity of the HSM/cask interface.

7.3.3 VENTILATION

The HSM has a ventilation system to provide for natural draft cooling of the DSC. However, no off-gas or filtered treatment system is required due to the low exterior contamination level of the DSC.

The ISFSI is designed for essentially no release of radioactive material during normal storage of the DSC in the HSM. Therefore, additional design work and equipment would not result in a reduction of radioactive materials released. Furthermore, no credible site accident will result in a radioactive leak because of the integrity of the double seal welds at each end of the DSC, the passive nature of the system, the operational limits and controls used during handling and the integrity of the stainless steel body of the canister.

7.3.4 RADIATION MONITORING INSTRUMENTATION

The operation of the ISFSI will be monitored under the HBR2 radioactivity monitoring program. No additional radiation monitoring instrumentation is required.

TABLE 7.3-1

DSC END SHIELDING MATERIAL THICKNESSES

DSC Top End[1]

Shields[1]	
Lead Shield Plug	0.50 in. Steel + 3.5 in. Lead + 1.75 in. Steel
Cover Plate	1.25 in. Steel

DSC Bottom End

Shields[1]	
Pressure Plate	2.00 in. Steel
Lead Shield Plug	0.25 in. Steel + 4.75 in. Lead

1

1 "Top" and "Bottom" refer to the top and bottom ends of the irradiated fuel assemblies.

TABLE 7.3-2

SHIELDING ANALYSIS RESULTS

ISFSI Location	Nominal Surface Dose Rates (mrem/hr) Neutron	Gamma	Total
DSC in HSM			
1. HSM Wall or Roof	0.03	2.5	2.5
2. HSM Air Outlet Shielding Cap	0.03	10	10
2.1 Front HSM Shield Cap	0.06	103	103
2.2 Rear HSM Shield Cap	0.06	10	10
3. HSM Air Outlet (no shielding cap)			
3.1 Front HSM Air Outlet (no shielding cap)	3.0	4450	4450
3.2 Rear HSM Air Outlet (no shielding cap)	1.6	440	440
4. Center of Door	35	81	116
5. Center of Door Opening	50	378	428
6. Center of Air Inlets	27	29	56
7. 4.5 Ft. from HSM Door	7.1	17	24
8. Ram Opening with Access Plate (fully inserted DSC)	2.9	0.37	3.3
9. Ram Opening Without Access Plate			
9.1 Fully Inserted DSC	3.8	0.92	4.7
9.2 Half Inserted DSC	1.2	605	606
DSC in Cask			
1. Centerline Top of DSC Plug			
1.1 No Water in DSC, Water in Gap	296	92	390
1.2 Water in DSC and Gap	0.83	36	37
2. Centerline Top of DSC Cover Plate (no water in DSC, water in gap)	252	62	314
3. Cask/Canister Annular Gap			
3.1 Water in Gap and DSC	0.34	390	390
3.2 Water in Gap (no water in DSC)	190	365	555
4. Transfer Cask Side Surface (no water in DSC)	3.3	2.8	6.1

TABLE 7.3-2 (Continued)

5. Transfer Cask Top Cover Surface	194	24	218
6. Cask Collar During DSC Transfer from Cask to HSM	37	620	657

Note:

1. These values for worst case situation where no water is present in the gap. For ALARA purposes, water should be present in the gap which will reduce streaming exposures by approximately 1/20.

7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

Estimated doses for the fuel loading, drying, sealing and transfer are provided in Section 9.1.2 of the HBR2 Updated FSAR (Reference 7.1) and Section 7.4.1 of the NUHOMS Topical Report (Reference 7.2). Onsite radiation dose rates due to the storage of the fuel in the first three HSMs are shown in Figure 7.4-1. If all eight modules are built and filled, the dose rate will be 8/3 times the values listed in Figure 7.4-1. The resulting dose for a person located at the ISFSI fence for eight hours a day for 250 days per year would be approximately 10 mrem. For a person located at the offices, the yearly dose would be less than 1 mrem. (Actual dose inside the office is less due to shielding from the building). If an additional 176 modules are built and filled, the onsite radiation dose rates due to the storage of fuel in the added modules are shown in Figure 7.4-1a.

7.4.1 OPERATIONAL DOSE ASSESSMENT

This section establishes the expected cumulative dose delivered to site personnel during the fuel handling and transfer activities associated with one ISFSI module. Chapter 5 describes in detail the ISFSI operational procedures, a number of which involve radiation exposure to personnel.

A summary of the operational procedures which result in radiation exposure to personnel is given in Table 7.4-1. The cumulative dose is calculated by a four step process. First, the number of personnel required to perform the task is estimated. Then the time required to perform the task is estimated based on the operational guidelines presented in Chapter 1, engineering judgement, and previous experience with similar or identical operations. An ambient dose rate is obtained for the operation. It is based on an estimated distance from the individual's trunk to the most significant radiation source. The dose rate is conservatively calculated by modeling radiation sources as line, cylinder, or plane sources, as is appropriate to the particular geometry of the operation. With the number of personnel, the time required, and the local dose rate, individual and collective exposures may be calculated.

7.4.2 STORAGE TERM DOSE ASSESSMENT

Figure 7.4-2 is a graph of the dose rate versus distance from the face of a filled storage array of three ISFSI modules. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces, are considered. Air-scattered dose rates are determined with the computer code SKYSHINE-II (Reference 7.3). Initial loading of all modules with the NUHOMS Topical Report design basis five-year post irradiated fuel is assumed. If five additional modules are added, the dose rate will be 8/3 times that shown in Figure 7.4-2.

Estimates of cumulative doses to site personnel from three filled ISFSI modules are given in Table 7.4-2. The dose rates are based on data shown in Figure 7.4-2. Occupancy information for number of personnel, location, and time is estimated based on the five-year plan for facilities' layout at the HBR2 site. Because of the very rapid decrease of dose rate with distance from the storage facility, a maximum distance of 600 feet was used in these analyses. No credit is taken for shielding of personnel by buildings, or for radioactive decay.

TABLE 7.4-1

SUMMARY OF ESTIMATED ONSITE DOSES DURING FUEL HANDLING OPERATIONS

	Number of	Time	Average Distance From Source Surface	Dose Rate	Dose Per Person	Total Personnel Dose

Operation	Personnel	(Hours)	(Feet)	(mrem/hr)	(mrem)	(mrem)
Location: Fuel Pool						
Load fuel into canister	2	8.0	-	5	40	80
Bolt lid assembly onto cask	2	0.5	1.5	22.3	11.2	22.3
Location: Cask Handling Area						
Decontaminate outer surface of cask	3	8.0	-	6.1(2)	48.8	146.4
Place scaffolding around cask	4	0.8	4.0	9.5	7.6	30.4
Unbolt lid, remove lid and spacer	2	0.75	1.5	22.3	16.73	33.5
Remove approximately 15 gal. of water from DSC and lower water level in canister cask gap	2	0.5	1.5	22.3	11.2	22.3
Weld lead plug to canister	2	3.0	4.0	3.5	10.5	21.0
Hydrotest canister	2	2.0	1.5	22.3	44.6	89.2
Remove water from canister(1) cavity	2	2.3	4.0	29.8	68.5	137.0
Seal weld prefabricated plug to siphon tube connection	2	0.5	1.5	210	105	210
Vacuum dry canister and(1) backfill with Helium	2	4.0	4.0	29.5	118.0	236
Helium leak test weld	2	1.0	1.5	210	210	420
Seal weld prefabricated plug to vent tube	2	0.5	1.5	210	105	210
Perform NDE (PT)	1	1.0	1.5	210	210	210
Install end cap	2	0.5	1.5	210	105	210
Weld end cap to canister	2	2.3	4.0	29.5	67.9	135.8
Install cask lid and bolt into place	2	0.5	1.5	118.3	59.2	118.3

TABLE 7.4-1 (Continued)

Remove scaffolding from around cask	4	0.8	4.0	9.5	7.6	30.4
Transport cask to skid and trailer	2	0.5	-	2	1	2
Location: Trailer						
Attach skid tie down to cask	2	0.5	1.0	6.1(2)	3.1	6.1
Transport cask to HSM	5	0.5	5	6.1(2)	3.1	15.5
Remove cask lid	2	0.5	1.5	118.3	59.2	118.3
Install cask jacking system and align cask with HSM	4	1.5	3.0	6.1(2)	9.3	37.2
Transfer canister from cask to HSM	4	0.5	3.0	6.1(2)	3.1	12.4
Install steel plate over front access of HSM	2	0.5	3.0	2.5(2)	1.3	2.6
Tack weld front access door	2	0.5	1.5	70.2	35.1	70.2
Install seismic retainer assembly	2	0.5	1.0	4.7(2)	2.4	4.8
Install cover plate to rear access	2	0.5	1.5	2.5(2)	1.65	3.3
TOTAL		42.95			1366	2635

(1) Monitoring operation - personnel could leave radiation field.
(2) Conservatively assumed to be surface dose rate.

TABLE 7.4-2

ESTIMATED ANNUAL ONSITE DOSES DURING STORAGE PHASE

Area	Number of Personnel	Time (Hours/yr)	Average Distance From Facility (Feet)	Dose Rate (mrem/hr)	Dose Per Person (mrem/yr)	Total Personnel Dose (mrem/yr)
E&RC Building	30[1]	2080	560	1.8×10^{-3}	3.7	110
Operations and Maintenance (O&M) Building		225[1]	2080	360	4.3×10^{-3}	8.9 2000
Yard Work Area[12]	13[1]	2080	120	4.9×10^{-2}	100.0	1300
Yard Work Area[2]	12[1]	2080	350	4.4×10^{-3}	9.2	110
TSC/EOF Building	25[1]	2080	580	1.6×10^{-3}	3.3	83
Primary Access Point	10[2]	8760	290	7.5×10^{-3}	66.0	660
Bulk Storage Warehouse	25[1]	2080	360	4.3×10^{-3}	8.9	220
Daily Visual Inspection	1	60	4	26	1560.0	1560
HSM Interior Inspection	1	0.5	1.5	4.0	2.0	6045

[1]One 8-hour shift per day, 5 days per week.

[2]Continuous occupancy (44 people at one 8-hour shift per day, 5 days per week, 50 weeks per year).

7.5 HEALTH PHYSICS PROGRAM

Appropriate health physics programs are established for all Company operations which deal with radiation. The programs are consistent with the corporate health physics policy and all applicable regulations. The Nuclear Assessment Section periodically evaluates the various health physics programs and other Company activities which have impacts on the programs and reports to senior management regarding the effectiveness and adequacy of the programs. The Nuclear Assessment Section makes recommendations to senior management, as necessary, to maintain effective overall health physics programs.

The existing HBR2 health physics program is applicable to the ISFSI. The HBR2 program has been established to provide an effective means of radiation protection for plant personnel, visitors, and the general public. Section 12.5 of the UFSAR describes the HBR2 health physics program.

7.5.1 ORGANIZATION

In the HBR2 organization, the Superintendent - Radiation Control reports to the Plant General Manager. This organization allows the Plant General Manager to be involved in the review and approval of specific ALARA goals and objectives as well as review of data and dissemination of information related to the ALARA program.

The organization also provides the ALARA Analyst, who is normally free from routine health physics activities, to implement the plant's ALARA program. This individual is primarily responsible for coordination of plant ALARA activities and routinely interfaces with first line supervision in radiation work planning and post-job review.

7.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The ISFSI will utilize the equipment, instrumentation, and facilities of HBR2 as necessary. Section 12.5 of the UFSAR describes the HBR2 equipment, instrumentation, and facilities.

7.5.3 PROCEDURES

The ISFSI will utilize existing HBR2 health physics procedures. These procedures are discussed in Section 12.5 of the UFSAR. Health physics procedures specific to the ISFSI were incorporated into the existing HBR2 procedures. In particular, radiation surveys of the ISFSI will be conducted on an annual basis. Additional health physics/radiation protection procedures required for use during the ISFSI test program will be developed as needed and will comply with the existing HBR2 program.

7.6 ESTIMATED OFFSITE COLLECTIVE DOSE ASSESSMENT

Because the ISFSI provides containment yielding essentially no radioactive gaseous or liquid effluents, assessment of offsite collective dose is limited to one of direct and reflected radiation to the nearest residence.

7.6.1 EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

The ISFSI is located within the protected area of HBR2. The HBR2 environmental program is described in the Offsite Dose Calculation Manual, Section 4.0 (Reference 7.4).

7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

An analysis of multiple contribution was performed in order to determine the radiological impact the ISFSI will impose on the population surrounding the H. B. Robinson plant. This impact added to contributions made by other uranium cycle facilities were compared to the natural background radiation and the regulatory requirements of 40CFR190.

The maximally exposed member of the public would receive approximately 1.6 mrem per year from an ISFSI made up of a three-unit HSM (reference Figure 7.6.1). An ISFSI consisting of an eight-unit HSM would contribute approximately 4.3 mrem per year. This is a result of external radiation only; there are no gaseous, particulate, or liquid effluents associated with the normal operation of the ISFSI. It can be concluded that the actual exposure contribution from the ISFSI along with the total of all other uranium fuel cycle activities is within the regulatory limits set forth in 40CFR190.

7.6.3 ESTIMATED DOSE EQUIVALENTS

During the normal operation of the ISFSI, there are essentially no effluents released.

The cumulative dose for the population (due to the ISFSI) integrated over 10 regions out to 50 miles radially from the ISFSI, is less than 2.0 man-rem per year. The currently accepted national average background radiation level is approximately 300 mrem/year per person which includes 200 mrem/year per person for radon.

REFERENCES: CHAPTER 7

7.1 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

7.2 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

7.3 C. M. Lampley, "The SKYSHINE-II Procedure: Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air," NUREG/CR-0781, RRA-T7901, USNRC, 1979.

7.4 Carolina Power & Light Company, H. B. Robinson Steam Electric Plant, Unit No. 2 Offsite Dose Calculation Manual (ODCM),@ Docket No. 50-261.

7.5 Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/NUREG/CSD-3.

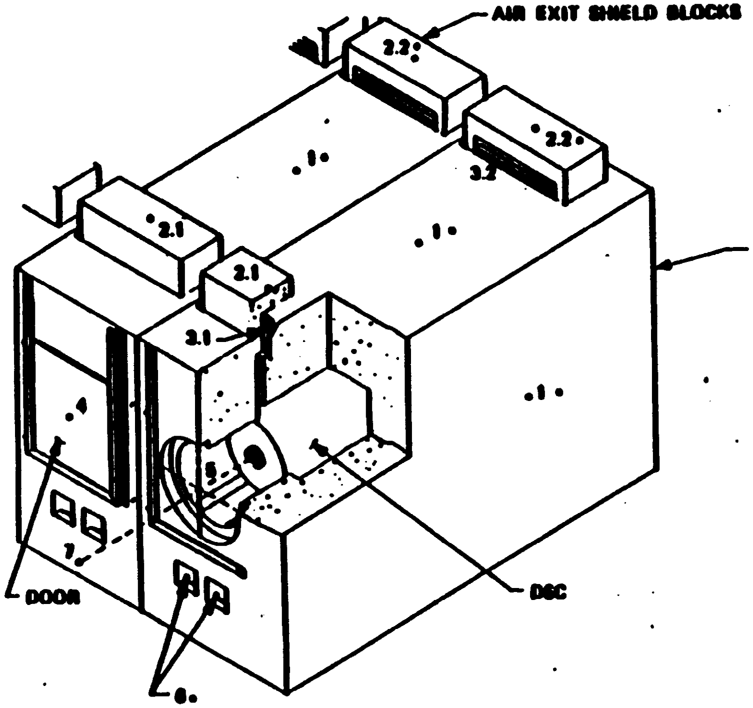
7.6 Oak Ridge National Laboratory, "ANISN-ORNL Multigroup One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," RSIC CCC-254, 1973.

7.7 J. H. Price, W.G.M. Blattner, "Utilization Instructions for QADMOD-G," RRA-N7914, RSIC CCC-396, 1979.

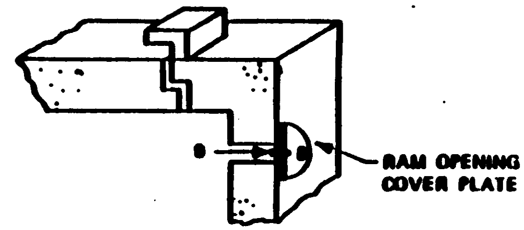
7.8 Oak Ridge National Laboratory, "Cask 40 Group Coupled Neutron and Gamma-Ray Cross-Section Data," RSIC DLC-23, 1978.

7.9 Grove Engineering, "MICRO SKYSHINE User's Manual," July 15, 1987, Grove Engineering, Inc., Rockville, MD.

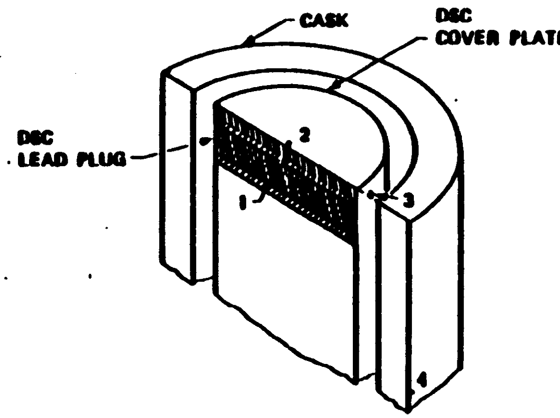
7.10 Grove Engineering, "MICROSHIELD User's Manual," Version 3, August 1988, Grove Engineering, Inc., Rockville, MD.



HSM DOSE RATE LOCATIONS



RAM OPENING DOSE RATE LOCATIONS



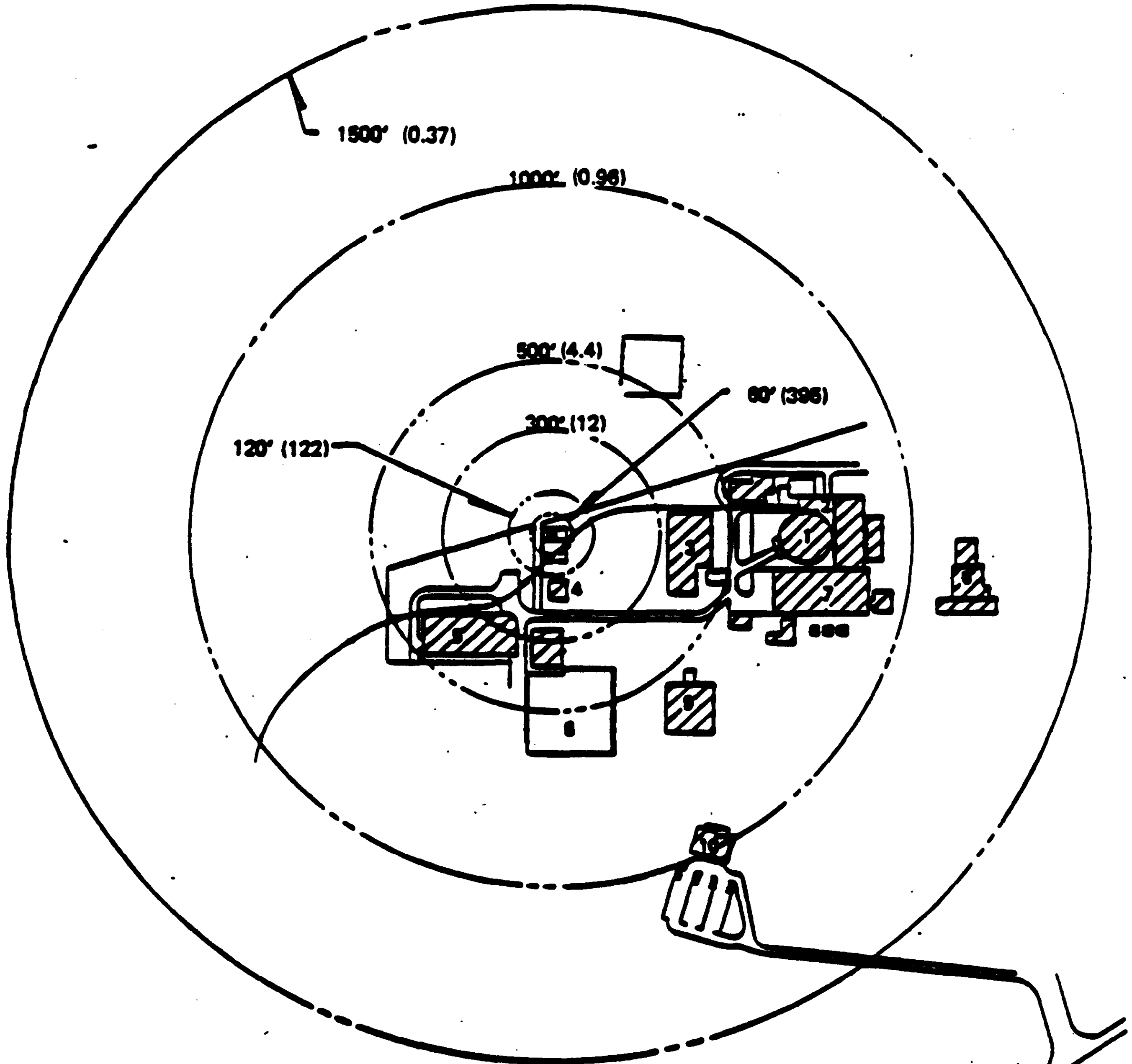
DSC-CASK DOSE RATE LOCATIONS

* INDICATES LOCATION WHERE TOPICAL REPORT VALVES HAVE BEEN SCALED TO PRODUCE REPORTED RESULTS.

H.B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

LOCATIONS OF REPORTED
DOSE RATES (TABLE 7.3-2)

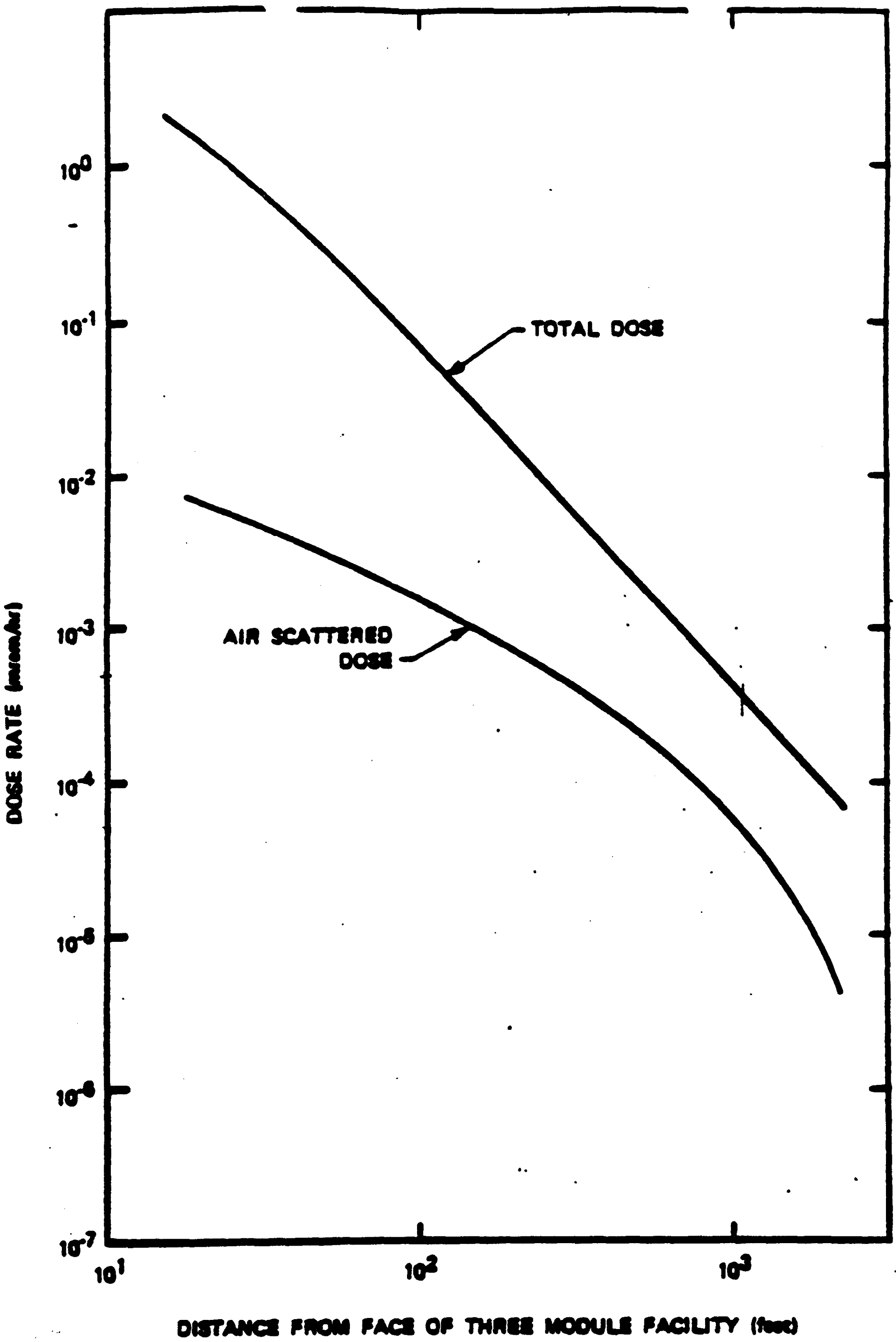
Figure 7.3-1



1. REACTOR CONTAINMENT BLDG
2. FUEL BLDG
3. OPERATIONS & MAINTAINANCE BLDG
4. STEAM GENERATOR STORAGE BLDG
5. BULK WAREHOUSE
6. UNIT ONE COAL PLANT
7. TURBINE BAY
8. PARKING
9. NEW TEC/EOF BLDG
10. VISITORS CENTER

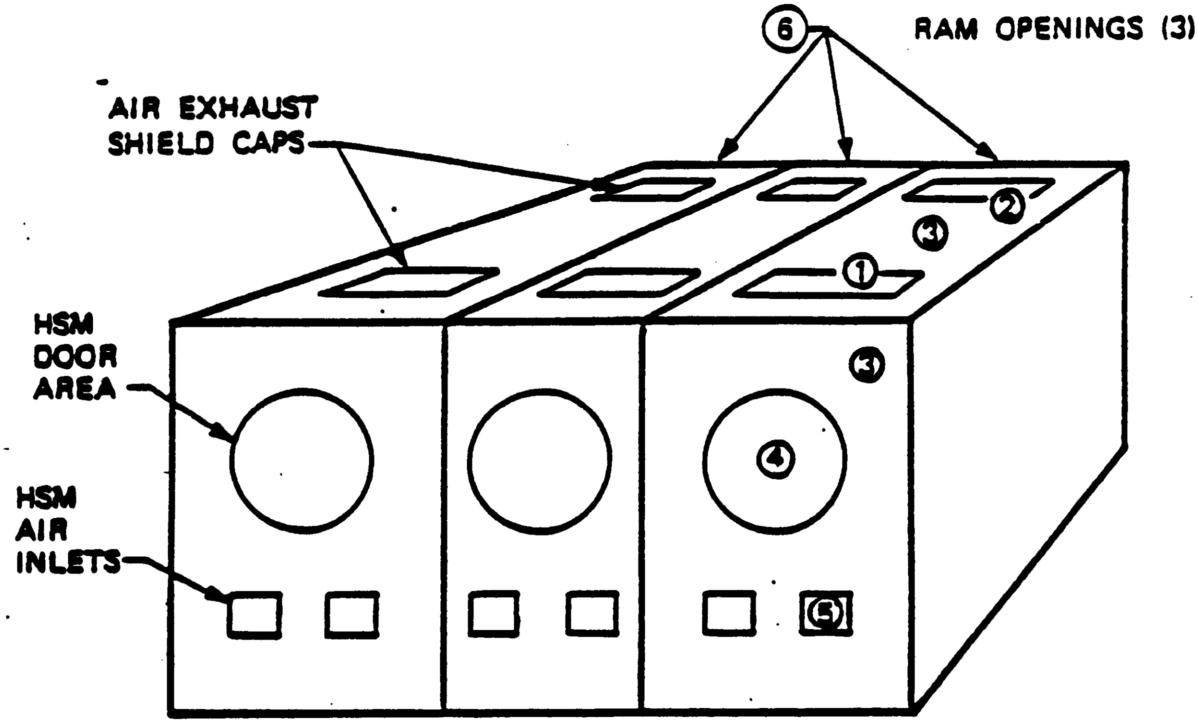
H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

ANNUAL DOSE (mrem/yr) FROM
3 HSMs (Assuming 2080 hours/yr)
Figure 7.4-1



H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

DOSE RATE VS. DISTANCE FROM
SURFACE OF HSM (Assuming 3 Modules)
Figure 7.4-2

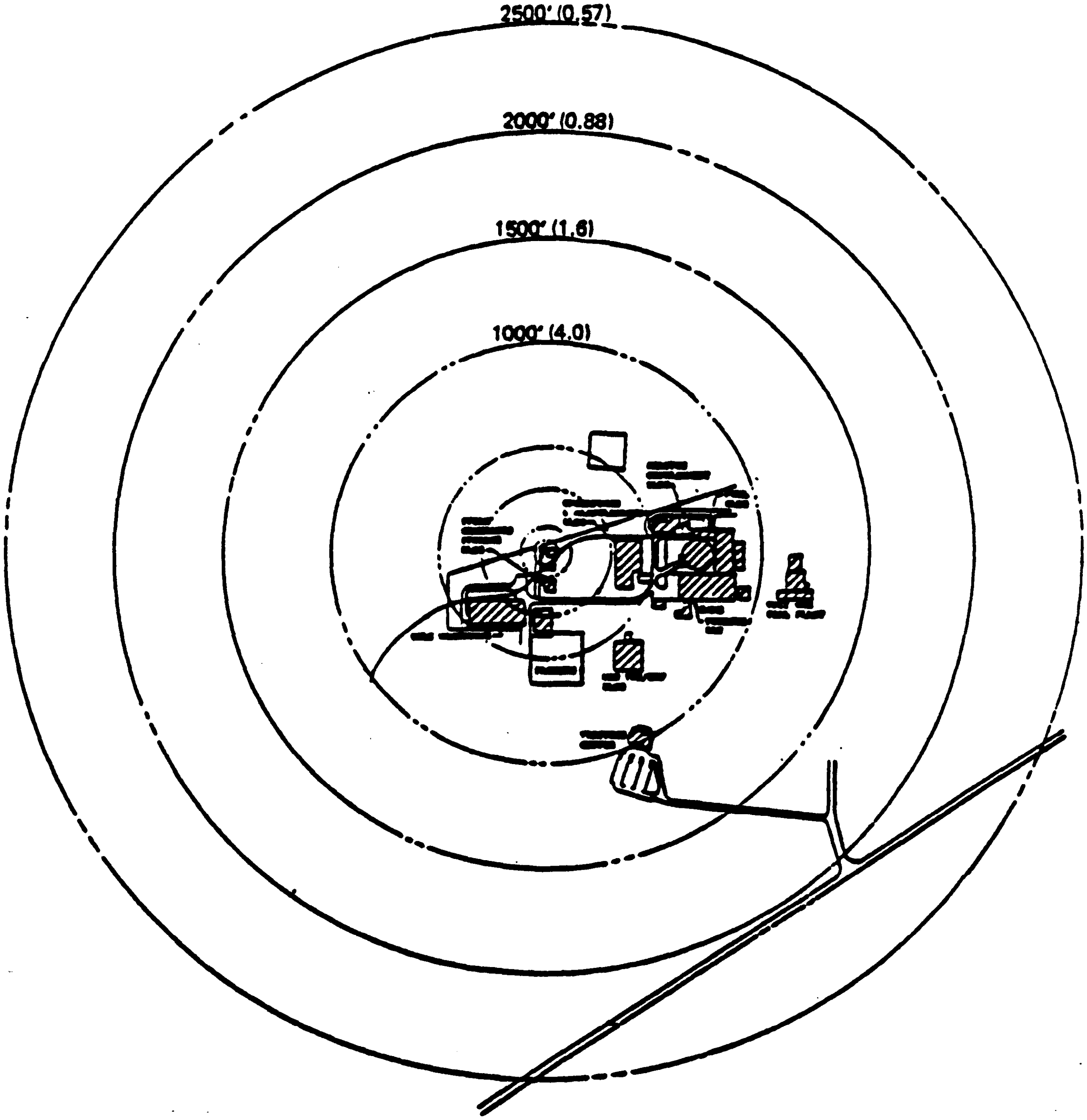


LOCATION		AREA (Ft^2)	DOSE RATE (mrem/hr) NEUTRON	GAMMA
ROOF	1	21.5	0.06	103
	2	21.5	0.06	10
	3	501.5	0.03	2.5
		AREA WEIGHTED AVE.	0.032	6.76
FRONT	3	278.5	0.03	2.5
	4	24.9	36	81
	5	6.0	27	29
	6	3.2	2.9	0.37
		AREA WEIGHTED AVE.	3.37	9.33

H.B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

RADIATION ZONE MAP OF
MODULE SURFACE DOSE RATES

Figure 7.4-3



Amendment No. 1

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

ANNUAL OFFSITE DOSE (mrem per year) FROM 3 HSMS (based on 24 hours a day, 365 days a year)
Figure 7.6-1