5.5.5.6 Radwaste System Evaporator

The new Radwaste System Evaporator is electrically heated. It is accessed through the PuNp Load Out Area on the 48' elevation in the Process Building. Evaporator bottoms are periodically transferred to steel barrels and stored in the Evaporator Bottoms Room or Canyon for subsequent shipment for treatment and subsequent burial. Steam vapor from the evaporator is demisted and routed to the air tunnel, then through the sand filter to the stack.

5.5.5.7 Ventilation Supply Room

Blowers and associated equipment to supply main building ventilation air are located in a room, 39 ft. x 21 ft. in plan, on the top floor of the south gallery area of the main building (81 ft. reference elevation). Personnel access is from the Control Room by way of the Computer Room and emergency power room with additional access from the Instrument Gallery. Two hooded air intake openings are provided in the reinforced concrete roof (elevation approximately 93 ft.). Air conditioning system coolers also are located on the south gallery roof.

5.5.5.8 Basin Pump Room Addition (BPRA)

In 1980 an addition was made to the original basin pump room (BPR) as shown in Figure 1-4 to house chemical decontamination equipment for basin water cooling system decontamination, and equipment to utilize heat from basin water as an energy source to heat the Main Building, including the fuel storage area. Because of its isolation from main building areas, the BPR and BPRA are cooled by a separate air conditioner.

a. <u>Basin Pump Room Addition (BPRA) Building</u>: The BRPA is located near the west wall of the existing pump room (BPR). The addition is a prefabricated steel building built on a concrete slab with outside dimensions of about 20 ft. by 30 ft. in plan. A space of about 4 feet separates the BPRA from the BPR wall except for an enclosed walkway connecting the BPR to the BPRA. A concrete pad extends along the north wall of the BPRA and a double door is located in the center of this wall. An air conditioner compressor mounting pad is located outside the north side of the BPRA.

   An above grade reinforced concrete vault housing a basin water-to-freon heat exchanger is located in the southwest corner of the BPRA. The vault drains to a sump which may be emptied by pumping collected water to the Radwaste System. Piping between the BPR and BPRA is routed overhead, passing through the enclosed walkway and connecting to existing piping systems in the BPR.

b. <u>Systems and Equipment</u>: A new pump was installed in the existing BPR to circulate basin water though the heat exchanger located in the heat exchanger vault. Four GE heat pumps are mounted on a steel rack adjacent to the heat exchanger vault. Freon is circulated from the heat exchanger and heat pumps to existing heating and cooling units located in the ventilation room of the Main Building. These units were modified to adapt to the new system. The heat pump system is reversible to provide either heating or cooling of fresh air entering the Main Building ventilation system.

A 600-gallon stainless steel tank is located in the BPRA and serves as the collection point for basin area low-level radwaste streams. A pump, adjacent to the tank, transfers liquid from this vessel to the low-level radwaste evaporator system.

The BPRA and existing BPR are air cooled by a system located in the addition. The compressor for this system is mounted outdoors on the pad at the west end of the BPRA.

5.5.5.9 Basin Chiller Room

In 2000 an addition was made to the basin pump room addition (BPRA) as shown in Figure 1-4 to house heat exchangers for the basin water cooling system.

a. Basin Chiller Room (BCR) Building: The BCR is attached to the west wall of the existing pump room addition (BPRA). The room is a prefabricated steel building built on a concrete slab with outside dimensions of about 18 ft. by 20 ft. in plan. The access door to the chiller room is in the west wall of the BPRA.

   An above grade reinforced concrete vault housing 2 basin water-to-freon heat exchangers is located in the northeast corner of the BWCR. The vault drains to a sump, which may be emptied by pumping collected water to the Radwaste System. Piping between the basin and the chiller heat exchanger is routed overhead, passing through the BPR and BPRA, connecting to existing piping systems in the BPR.

b. Systems and Equipment: Two new pumps were installed in the existing BPR to circulate basin water though the heat exchangers located in the heat exchanger vault. Two 100-ton air cooled heat pumps are mounted outside, on concrete piers to the west of the chiller room. One of these is enough to maintain basin water temperature. The second unit is a back-up. Freon is circulated from the heat pumps to the heat exchangers to chill the basin water.

   The BCR is air cooled by a system located in the BPRA.

5.6 WASTE VAULTS

Three below-grade vaults were constructed as part of the MFRP:

a. Low Activity Waste (LAW) Vault - originally provided for on-site interim storage of low-level wastes from aqueous processes.

As of July 1994, all additions to the LAW Vault were terminated. Waste streams are now processed by the new radwaste system (see Appendix B.23). As of October 1996, the LAW Vault is empty and dry, but still contains radioactive material as contamination adhering to the vault walls and floor. The LAW Vault connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the LAW Vault.

b. Cladding Vault - originally provided for interim storage of compacted, leached hulls and other contaminated metal scrap from fuel reprocessing operations. This vault has been emptied and cleaned. CRA and CSF drains, which previously went to the Cladding Vault have been capped. Stack drain has been routed to the stack condensate system. This vault is not being used but is being held available on a contingency basis.

c. Dry Chemical Vault (DCV) - provided for interim storage of contaminated dry process chemicals of low activity level[21]. This vault was emptied in 1993. The DCV connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the DCV.

Local hydrology (e.g., drainage patterns to water courses and soil ion exchange capacity) is not of major significance in ensuring safety of fuel storage operations.

Subsurface water conditions encountered during MFRP construction were more severe than expected. Therefore, concrete density and monitoring and control equipment were designed to handle these conditions. No significant difficulties with this equipment have occurred.

Storage vaults were designed and constructed to provide high integrity confinement of contained materials and include systems for detecting leakage into or out of these tanks. The systems permit detection of radioactive material in highly diluted samples (caused by water intrusions) and provide pump-out capability to collect and dispose of intrusion water as well as any leakage from stored material.

5.6.1 Cladding Vault

A below-grade cylindrical vault, 45 ft. in diameter and 72 ft. deep was provided for underwater storage of leached cladding hulls and other metallic scrap.

5.6.2.1 Cladding Vault Construction

The cladding vault is constructed of reinforced concrete about 2 feet thick and is lined with stainless steel. The top of its 2.5 ft. thick reinforced concrete cover is at 41.5 ft. elevation. The vault is located adjacent to the LAW vault on the south side of the main building (Figure 1-4). It is connected to the mechanical cell in the canyon by a reinforced concrete waste disposal cart tunnel (top about at grade level) which extends across the top of the vault to a 235 sq. ft. cart equipment pit. The pit roof has two access openings with shield plugs. The vault is equipped with leak detection and sampling systems similar to those for the fuel storage basins, with level recorder and unit alarm in the control room (SAS) and local control in the mechanical cell operating area.

Intrusion water around the vault is pumped to the Radwaste System.

5.6.2.2 Cladding Vault Description

a. Elevation: The circular floor of the cladding vault is 80.5 ft. below grade level and the interior height of the structure is 72 ft. The floor of the waste disposal cart tunnel which connects the cladding vault and the mechanical cell in the main building canyon area is approximately at the same level as the underside of the vault roof (8.5 ft. below grade) which is about 1 ft. above the maximum liquid level in the vault. The floor of the equipment pit located adjacent to the vault is 14 ft. below grade level. The top of the cladding cart tunnel and equipment pit roof is 0.75 ft. above grade and that of the vault proper is 6 ft. below grade.

b. Construction: The cylindrical vault structure is reinforced concrete lined with stainless steel. Excavation extended roughly 82 ft. into the underlying bedrock, which was sufficiently sound to provide clean vertical surfaces for 2 ft. thick concrete walls to be poured against, using conventional interior forming. The reinforced concrete floor is approximately 4 ft. thick. The equipment pit and the cart tunnel also are of reinforced concrete and tied to the vault structure. The roof of the cart tunnel, which extends across the vault and the cover of the equipment pit is approximately 4 ft. thick. The remainder of the vault top cover is 2.5 ft. thick reinforced concrete.

c. Vault Liner: The cylindrical vault structure is completely lined with 0.125 in. thick (11 gauge) 304L stainless steel sheets placed flush against the concrete walls and floor. As in the storage basins, the sheets are welded continuously along each edge to a gridwork of stainless-steel angles and plates embedded in the concrete. At the floor-to-wall joint, the sheets are welded to a stainless-steel angle. Quality control and verification procedures parallel those applied to the storage basins.

d. Leak Collection, Monitoring and Pump-Out Provisions: Drain slots are provided in the concrete walls and floor, between the liner and concrete. These lead to a perimeter collection header behind the floor-to-wall junction. The perimeter header is sloped to a low point, which is connected to a single leak collection sump. The sump consists of a 6 in. diameter vertical stainless-steel pipe embedded in the vault wall, which extends from the top of the vault to approximately 1 foot below the vault floor level. It contains a liquid level detector line and necessary piping for a 5 gpm (nominal) pump-out system. Auxiliaries for the level detection and pump out systems, including a monitoring sample station, are located in the mechanical cell gallery of the main building. Water from the pump-out system is routed to the Radwaste System.

5.7 SUPPORT FACILITIES

Support facilities are described in the following sections. As in previous sections, those functions related exclusively to fuel reprocessing are omitted or discussed only briefly.

5.7.1 Utility and Service Building

On the north side of the main building is located the single-story high-bay utility and service building (Figure 1-4). It is 71 ft. by 50 ft. in plan and is of conventional steel frame, insulated siding and roof construction on a grade level concrete foundation. The building is divided into a utility section which houses the demineralized water system; primary electrical switchgear, training room, operations ready room, and first aid area; and a personnel section containing change room, lunchroom, and office areas. The arrangement takes into account the normal industrial safety requirements for major electrical equipment. Consideration also is given to isolation of normal industrial functions and equipment from all potential sources of radioactive contamination. Utility services are not critical to safety of fuel storage operations. Interruption of these services for short periods of time, up to several months, would have no off-site impact as long as basin water level is maintained. Principal features are described in the following paragraphs.

5.7.1.1 Utility Section

The 1,700 sq. ft. utility section of the building is divided into two major rooms, the larger of which houses water demineralization and three smaller room partitions for training, an operations ready room, and a first aid area. The demineralizer system consists of ion exchange resin provided by a contract service. It is capable of treating 25 gpm continuously. The pump required for operation and distribution is located nearby.

A separate 300 sq. ft. room in the utility section houses the primary electrical distribution switchgear for the plant. Incoming power from the CECo distribution system is reduced to 480 volts prior to entry into the utility building.

5.7.1.2 Outside Facilities

The following facilities are directly associated with utility system operations (Figure 1-4):

a. A chain link fence surrounds a rectangular area 62 ft. by 30 ft. in size located on the east side of the building and encloses the terminal structure of two 34,000 volt incoming overhead transmission lines and two CECo owned 1,500 kVA transformers which reduce the incoming supply to 480V. The fenced area is locked to preclude accidental access to high voltage equipment.

5.8 UTILITY SYSTEMS

Water, electric service, and sewage systems are described in the following sections.

5.8.1 Water Supply

Water to meet potable, utility and fire-fighting requirements is obtained principally from a 788 ft. deep, 12 in. diameter well located within the OCA, southeast of the administration building (Figure 1-4). A submersible, 100 gpm vertical turbine pump is provided, capable of developing 100 ft. of head. This pump is connected to the emergency power distribution system. The pump discharges through filters to a 50,000-gallon elevated water sphere, located near the well. Tests have confirmed a continuous pumping rate of 250 gpm from this well.

An electric water heater in the well house is used to prevent freezing of water in the sphere.

Water is rendered potable by filtration and chlorination before delivery via underground lines to various personnel occupancy areas. Process-related requirements are supplied from the utility water system.

5.8.1.1 Utility Water Supply

Underground piping is provided to distribute utility water from the elevated storage tank to the utility building for supplying the demineralizer system, and to various points in the main building for uses not requiring demineralized or potable water.

5.8.1.2 Demineralized Water Supply

Demineralized water is used for fuel storage basin supply. This water is supplied from the series cation-anion demineralizer located in the utility building, which is capable of treating 25 gpm continuously from the utility water supply system. Distribution to points of use is via a pump-pressurized header system. There is a 1 in. line to the basin to furnish make-up water. Basin water level is maintained under manual control of the basin operator, who would normally add water when basin water level dropped 2 in., which is low enough to affect basin cleanup system operation. A back-up-low-level alarm in the CAS/SAS activates if basin water level drops 6 in. below normal.

5.8.1.3 Fire-Fighting Water Supply

Potable and utility water usage is limited by location of outlet piping to the topmost 8,000 gal. of water sphere capacity, with the remaining 42,000 gallons reserved for fire protection. Distribution is via a standard underground piping system located beneath historical frost penetration in accordance with underwriter and building codes.

5.8.1.4 Backup Water Supplies

Parallel fuel storage basin pumps and heat exchangers reduce the likelihood of complete loss of basin cooling capability. In the highly unlikely event that cooling system capability could not be restored within 50 days[23] (or more, depending on circumstances), makeup can be provided from demineralized or utility water storage or from other emergency sources, including water pumped from the DNPS cooling lake, or even from the river[21]. Emergency pumping equipment could be brought to the site and placed in operation within the 50-day period with no impact on public health or safety from stored fuel.

5.8.2 Electrical Supply

GEH-MO fuel storage activities require an electrical peak load capacity of 725 kW, with an average load requirement of 500 kW. Principal load requirements come from crane operation, ventilation system requirements, control and instrumentation requirements, and operation of auxiliary systems (e.g., air, and water).

Although interruption of any of these functions would not result in an unsafe condition, secondary power sources (originally intended for fuel reprocessing requirements) are provided, which ensure continuing operation of equipment and services, including security systems, important to plant operation.

5.8.2.1 Normal Electrical Power Source

The normal source of electric power for GEH-MO is the CECo distribution system. Supply is via two separate 34,000V pole-mounted lines from the DNPS Switchyard to GEH-MO power terminal facilities located adjacent to the utility building. Each of these lines serves one of two CECo owned 1,500 kVA transformers. A current limiting bus connects the 480V power terminals of each transformer to a bus system in the load center switchgear located in the utility building.

The substation type load center consists of metal-enclosed, high current capacity, manually and electrically operated air circuit breakers and bus bar systems for distribution of power to seven motor control centers and an essential services load center which feeds two motor control centers.

Bus sections and associated circuit breakers are provided with protective relays, which de-energize appropriate portions (or all) of the system in the event of overload or short circuit conditions.

5.8.2.2 Essential Services Power Facilities

The loss of electrical power, even for many hours, would not result in a situation presenting a hazard to employees or the public because of stored irradiated fuel. However, a diesel generator is available. All electrical loads, which contribute directly to plant capability under abnormal conditions are supplied from an essential services distribution system. This system consists of metal-enclosed, high current capacity load center type switchgear through which 480V, three phase power is supplied to one motor control center in the EEB and one motor control center located in the main building. The 400kVA diesel driven standby generator located in the EEB generator room is provided with appropriate controls so it can automatically supply power to the essential services load center in the event both utility incoming power sources are lost. Interlocks are provided within the load center switchgear that prevent the diesel driven generator from being connected in parallel with the incoming utility power system.

Special electrical subsystems are provided to meet particular power needs such as those for instrument operation and system control functions. Control power of 24 VDC is supplied from two rectifiers. The demand is such that one rectifier can carry normal plant load as well as keep batteries charged. Rectifiers convert 480 VAC power from the essential services power distribution system and are located in the same room as the rectifiers. Power is routed from the subsystem location in the gallery area electrical equipment room to a distribution network within the main building control panel and to control relay cabinets located directly behind the main control panel, in the BPR, in the utility building and in the EEB.

5.8.2.3 Distribution System

Industrial type motor control centers provide power to each individual use point. These control units utilize local or remotely operated magnetic contactors sized for the particular load requirements being served. Distribution systems throughout the plant utilize commercial electrical cabling of specified capability. Routing between buildings is via underground concrete-encased conduit. Power distribution cables are routed in standard electrical cable trays and conduit. Within the main building, the bulk of power supply cabling and wiring for instrumentation and control functions are carried in separate wiring trays with appropriate protection against unwanted interactions, fire damage, etc.

5.8.2.4 Operating Characteristics

Electrical power required for normal fuel storage operations can be supplied by either of two incoming power lines from the CECo distribution system. Upon loss of either line, a manual, two-of-three circuit breaker system can be actuated to switch load to the single operating line. The bus-tie breaker cannot be actuated unless one incoming line breaker is open. To restore normal operation after the supply outage, the bus-tie breaker is opened, and incoming line circuit breakers are closed. Some distribution system circuit breakers as well as control system lockout switches and relays must be manually reset.

The essential services power distribution system is normally fed from the No. 1 bus bar. If power to this bus bar section is lost, the power supply for the essential services power distribution system automatically transfers to the No. 2 bus.

In the unlikely event that power from both incoming supply lines is disrupted, the following sequence of automatic operations will take place:

a. The standby diesel generator will start.

b. The essential services load center will separate from the normal supply source.

c. For load shedding purposes, some circuit breakers in the two essential services motor control centers will open.

d. After the diesel-driven generator is up to speed, the circuit breaker connecting the generator to the essential service power distribution system will close and restore power to some lighting systems, basin cooling water pump(s) and other important loads.

e. With power available to the essential service power distribution system, preselected loads will be automatically and sequentially restarted (e.g., one air compressor to maintain instrument and process air, supply and one ventilation exhaust fan to maintain minimum air pressure differentials).

   An ammeter in the CAS/SAS indicates output of the diesel driven generator. Lights on the main control panel indicate status of the two utility power sources. Separate annunciators on the main control panel are provided to alert the SAS/CAS operator to a malfunction in the diesel generator system, 24 VDC system and utility supply system.

5.8.3 Site Natural Gas Supply

All use of natural gas was discontinued in 2002 and natural gas service is no longer available on site.

5.8.4 Sewer Systems

At GEH-MO, industrial and sanitary sewage system are combined and discharged to sanitary lagoons and a holding basin with no direct discharge of any process or sanitary liquid effluent to local waterways. The systems meet requirements of the State of Illinois, and appropriate permits for operation have been issued.

5.8.5 Rail Transportation Facilities

Rail service to the site is provided by a spur track from the DNPS siding, approximately 0.5 mile north of the plant site, which connects to the Canadian National right-of-way serving the general area. The spur track is designed to carry heavy cask carloads at low speed (ASCE 100 lb. rails, appropriately limited curves and grades). After crossing the county road, the track is divided into three spurs and enters the OCA. All tracks have been cut and sections removed to prevent any rail movement.

The eastern spur enters the cask receiving area in the main building, terminating in a car bumper set in a heavy concrete block to protect the decontamination and basin areas from involvement in a rail accident. The spur is sloped to the north, and a manual derail is located north of the receiving area to stop a runaway car. The center spur serves the cask service facility. The western spur is a storage track, terminating in a standard car bumper, with capacity to store four cars.

5.9 ITEMS REQUIRING FURTHER DEVELOPMENT

GEH-MO fuel storage activities have been underway since January 1972, and, except for a continuing program of improvements based on operating experience, no specific equipment or facility item is now known to require further development.

5.10 REFERENCES

1. Fuel storage basins are designated Basin 1 and Basin 2. Basin 2 was originally the high-level waste storage basin, converted to fuel storage under Materials License No. SNM-1265, Docket 70-1308, December 1975.

2. K. J. Eger, Operating Experience - Irradiated Fuel Storage - Morris Operation, Morris, Illinois, General Electric Company, NEDO-20969B.

3. Non-contaminated waste is accumulated in dumpsters, which are mechanically emptied into a commercial garbage truck for disposal at a licensed land fill site. Trash is monitored before leaving the site to assure no radioactive material is included in uncontaminated waste.

4. Refer to Section 5.5 for discussion of reinforced concrete design bases common the main building and associated structures, including the cask unloading basin.

5. When the fuel storage basin was almost full, storage racks were installed in the high activity waste basin - now Basin 2 - on an interim basis (see letter dated April 6, 1973, requesting amendment to License No. SNM-1265).

6. Densities expressed in metric tons of uranium and abbreviated TeU.

7. B. F. Warner, the Storage in Water of Irradiated Oxide Fuel Elements, British Nuclear Fuels, Ltd.

8. A. B. Johnson, Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, Battelle Pacific Northwest Laboratories, September 1977 (BNWL-2256).

9. P. R. C. Winter, Battelle Pacific Northwest Laboratories, telex to H. A. Klepfer, General Electric, September 28, 1977.

10. Electrofilm, Inc., North Hollywood, California, 91605.

11. The heat transfer calculations have not been changed from the old basis. It is doubtful that boiling would ever occur under credible conditions.

12. Site survey and foundation report by Dames & Moore, Park Ridge, Illinois, see Appendix A.

13. This method was selected as an alternative to dye-penetrate checking.

14. Process photographs of actual operations (typical Figure 1-13) were made through up to 50 ft. of basin water.

15. Proprietary product of Norton Co.

16. L. L. Denio, D. E. Knowlton, and E. E. Voiland, Control of Nuclear Fuel Storage Basin Water Quality by Use of Powdered Ion Exchange Resins and Zeolites, June 1977, (ASME 77-JPGC-NE-15).

17. The capacities shown for the cooling systems are based on basin water at 120 °F, ambient air at 95 °F.

18. Also referred to as "emergency generator," a term originating from the original design as a reprocessing facility. Loss of electric power at the fuel storage facility would not constitute an emergency.

19. Except LAW vault intrusion water; piped to process water.

20. This vault contained natural or depleted uranium, fluoride salts, and other materials used during MFRP testing. This vault is currently empty.

21. Loss of cooling is discussed in Chapter 8, "Accident Analysis."

22. "Durability of Spent Nuclear Fuels and Facility Components in Wet Storage," International Atomic Energy Agency Report IAEA-TECDOC-1012

23. "Fuel Basin Water Evaluation: Conductivity Change and Evaporation Rate," T. D. Maikoff, August 2004.

6.0 WASTE MANAGEMENT

Waste management practices at GEH-MO have included underground vault storage, metal melt, burial and incineration by contracted services and on-site volume-reduction by evaporation of liquid waste. Also included is disposal of basin water filter media via HIC disposal.

6.1 UNDERGROUND WASTE VAULTS

6.1.1 Dry Chemical (DCV) Vault

As of October 1993, the DCV is empty, containing only residual radioactivity in the form of radioactive contamination on the walls and floor. The DCV vault connecting piping has been removed or capped, and the vault is laid away, with no current plans for use.

6.1.2 Low Activity Waste (LAW) Vault

As of October 1996, the LAW Vault is empty, containing only residual radioactivity in the form of radioactive contamination on the walls and floor. The LAW vault connecting piping has been removed or capped, and the vault is laid away with no current plans for use.

6.1.3 Cladding Vault

As of October 1996, the Cladding Vault is empty and is held available on a contingency basis.

6.2 RADWASTE SYSTEM

Concurrent with the decision to eliminate use of the LAW Vault was an immediate need for an alternate means to treat and reduce the volume of low-level liquid waste. In 1993, a system was designed, installed and is in operation. See Appendix B.23 for details of operation.

6.3 SOLID RADIOACTIVE WASTE

Accumulated low-level radioactive waste is disposed of by metal melt, incineration and/or burial. On-site storage of radioactive LSA waste is an option but is not favored or planned.

6.4 NONRADIOACTIVE WASTE

Nonradioactive, conventional solid wastes (trash) are disposed of via commercial trash pickup. No other effluents of consequence are released to the environment.

7.0 RADIATION PROTECTION

7.1 INTRODUCTION

This section describes the GEH-MO radiation protection program and provides estimated and actual occupational radiation exposures to operating personnel during fuel storage operations. Information is provided on facility and equipment design, planning and procedures, programs, and techniques and practices employed in meeting requirements for protection against radiation as specified in 10 CFR Part 20.

7.2 MAINTAINING OCCUPATIONAL RADIATION EXPOSURES AS LOW AS REASONABLY ACHIEVABLE (ALARA)

GEH-MO requires exposure of personnel to ionizing radiation be kept **A**s **L**ow **A**s **R**easonably **A**chievable (ALARA). This is a requirement of, and is implemented through, the health physics program described in this section.

7.3 RADIATION SOURCES

This section describes sources of radiation that are bases for radiation protection design and which are used as input to shield design calculations.

7.3.1 Irradiated Fuel

General characteristics of irradiated fuel are given in Section 4. However, for purposes of estimating dose rates, calculations are based on parameters that more realistically reflect fuel in storage. Although most fuel currently in storage has cooled much longer than a year (33 to 50 years as of April 2020), and has an average exposure of under 20,000 MWd/TeU, it is conservatively assumed that all fuel in the basin has the following characteristics for radiation protection calculations under normal operation:

a. Exposure - 24,000 MWd/TeU
b. Specific power - 40 kW/kgU
c. Cooling time - 12 months

For calculation purposes, fission product activity in fuel with assumed conservative characteristics is given in Table 7-1 and resulting gamma spectrum is given in Table 7-2. Assumptions for the basin radiation source calculations include:

a. The radiation source is approximated by a uniformly distributed source within a volume of 21,000 cu. ft. (1,500 sq. ft. floor area x 14 ft. length of fuel).

b. The source volume is 14.5 ft. below the pool surface (approximate depth to top of fuel bundles in storage).

c. Credit is taken for self-shielding in the source volume, assuming that the source medium is water only (i.e., no credit taken for shielding from fuel or stainless steel, etc.).

Calculations are performed in Section 7.4.2.1.

7.3.2 Storage Basin Water

The radioactive material concentration in the storage basin water results from a balance between the addition rate from stored fuel and the basin cleanup system removal rate. Operating experience gained in storage of irradiated fuel at GEH-MO since early 1972 demonstrates that radioactive material concentration in the basin water can be reliably maintained at personnel exposures that are ALARA.

The values presented in Table 7-1 and 7-2 are conservative relative to the current cooling times of the actual fuel stored at GEH-MO and are still applicable for radiation protection calculations.

Table 7-1
FISSION PRODUCT ACTIVITY
(24,000 MWd/TeU, 40 kW/kgU)

			Activity (Ci/TeU)			
Isotope	Half-Life		1 Year	5 Years	10 Years	20 Years
Kr-85	10.701	y	7.62E+03	5.88E+03	4.25E+03	2.22E+03
Rb-86	18.82	d	6.93E-04	0	0	0
Sr-89	50.55	d	9.41E+03	1.88E-05	0	0
Sr-90	28.82	y	6.47E+04	5.88E+04	5.21E+04	4.10E+04
Y-90	64.06	h	6.48E+04	5.87E-04	5.21E+04	4.10E+04
Y-91	59	d	2.08E+04	6.33E-04	2.54E-13	0
Zr-93	1.53E+06	y	2.31	2.31	2.31	2.31
Zr-95	63.98	d	4.15E+04	5.55E-03	1.42E-11	0
Nb-95m	86.6	h	5.27E+02	0	0	0
Nb-95	34.97	d	8.78E+04	2.33E-08	0	0
Ru-103	39.35	d	2.68E+03	1.78E-08	0	0
Ru-106	366.5	d	1.72E+05	1.09E+04	3.43E+02	3.45E-01
Rh-103m	56.116	m	2.68E+03	0	0	0
Rh-106	29.8	s	1.72E+05	1.09E+04	3.43E+02	3.45E-01

Ag-110m	252.2	d	1.23E+04	2.22E+02	1.47	6.48E-05
Ag-111	7.5	d	8.53E-11	0	0	0
Cd-115m	44.8	d	3.49	5.32E-10	0	0
Sn-119m	250	d	2.64E+01	4.60E-01	2.91E-03	1.17E-07
Sn-121m	55	y	9.12E-02	8.67E-02	8.14E-02	7.18E-02
Sn-123	129	d	6.13E+02	2.39E-02	1.31E-05	0
Sn-125	9.625	d	3.93E-08	0	0	0
Sb-124	60.2	d	3.23	1.61E-07	0	0
Sb-125	2.71	y	4.84E+03	1.74E+03	4.84E+02	3.75E+01
Sb-126	12.4	d	4.74E-02	0	0	0
Te-125m	58	d	1.18E+03	3.08E+05	1.02E-14	0
Te-127m	109	d	1.32E+03	1.22E-01	1.10E-06	0
Te-127	9.35	h	1.30E+03	0	0	0
Te-129m	33.52	d	4.31E+01	3.00E-12	0	0
Te-129	69.5	m	2.74E+01	1.90E-12	0	0
I-129	1.57E+07	y	2.10E-02	2.10E-02	2.10E-02	2.10E-02
I-131	8.04	d	2.37E-08	0	0	0
Xe-131m	11.77	d	1.59E-05	0	0	0
Xe-133	5.245	d	4.71E-15	0	0	0
Cs-134	2.062	y	8.89E+04	2.32E+04	4.32E+03	1.50E+02
Cs-136	13	d	1.48E-04	0	0	0
Cs-137	30.174	y	7.79E+04	7.11E+04	6.34E+04	5.04E+04
Ba-137m	2.5513	m	7.34E+04	6.72E+04	6.00E+04	4.78E+04
Ba-140	12.789	d	5.13E-03	0	0	0
La-140	40.27	h	5.90E-03	0	0	0
Ce-141	32.55	d	8.00E+02	2.46E-11	0	0
Ce-144	284.5	d	5.30E+05	1.51E+04	1.76E+02	2.42E-02
Pr-143	13.59	d	1.56E-02	0	0	0
Pr-144	17.3	m	5.30E+05	1.51E+04	1.76E+02	2.42E-02
Nd-147	10.98	d	7.69E-05	0	0	0
Pm-147	2.62344	y	1.04E+05	3.61E+04	9.65E+03	6.88E+02
Pm-148m	41.29	d	9.45E+01	2.11E-09	0	0
Pm-148	5.37	d	6.51	1.48E-10	0	0
Sm-151	87	y	9.36E+02	9.07E+02	8.71E+02	8.04E+02
Eu-154	8.59	y	4.39E+03	3.17E+03	2.11E+03	9.42E+02
Eu-155	4.96	y	1.02E+03	5.83E+02	2.90E+02	7.17E+01
Eu-156	15.11	d	6.64E-03	0	0	0

Tb-160	72.1	d	1.66E+01	1.32E-05	3.13E-13	0
All Remaining Fission Products			1.18E+01	1.17E+01	1.16E+01	1.14E+01

Table 7-2

GAMMA ENERGY SPECTRUM (E) FOR FUEL IN STORAGE - VOLUMETRIC SOURCE (S_v)
(24,000 MWd/TeU, 40 kW/kgU, 12 mo. Cooling)

Energy Group	E (MeV)	S_v (MeV/cm^3 sec)
1	1.75 to 2.25	2.2156×10^8
2	1.25 to 1.75	1.342×10^8
3	0.75 to 1.25	1.1078×10^{10}
4	0.25 to 0.75	2.1151×10^9

Because of passive storage conditions, if any defects occur during storage, they would likely be minor perforations (or "pin holes") in the fuel cladding rather than gross cladding failure.

Radioactive material in basin water consists of corrosion product surface contamination and fission product nuclides escaping through minor perforations in the clad. A reported value of the escape rate coefficient of 10^{-13} per second indicates diffusion rates within fuel are so low that major releases from the fuel matrix will not occur[1].

7.3.2.1 History of Radioactive Material Concentration

The history of radioactive material concentration in basin water is shown graphically in Figure 7-1[2]. The general trend is a gradual increase in concentration with increasing fuel loading and time, culminating in plateaus and abrupt decreases. Plateaus may be caused by a reduction in the source or establishment of a steady-state condition between radioactive material addition and removal. Decreases are due to accelerated removal of radiocesium and radiocobalt by use of filtration, a special ion exchange material in the basin water filter, and radioactive decay.

7.3.2.2 Contaminants

The principal dissolved radioactive contaminant in basin water is Cesium-137 with concentrations (typically now 7.3×10^{-4} µCi/ml) ranging up to 2.1×10^{-3} µCi/ml. A means of cesium removal has been found that makes reduction and control of this contaminant relatively simple. For example, over a 10-week period in 1974, radiocesium concentration was reduced to one-third of that at the beginning of the period. The basin water inventory was correspondingly reduced from about 29 Ci to 11 Ci. In 1975, during a 4-week period, the radiocesium concentration was reduced by a factor of six and the basin water inventory reduced from 14 Ci to 2.3 Ci. At the end of the latter period, the radiocesium concentration was 9×10^{-4} µCi/ml.

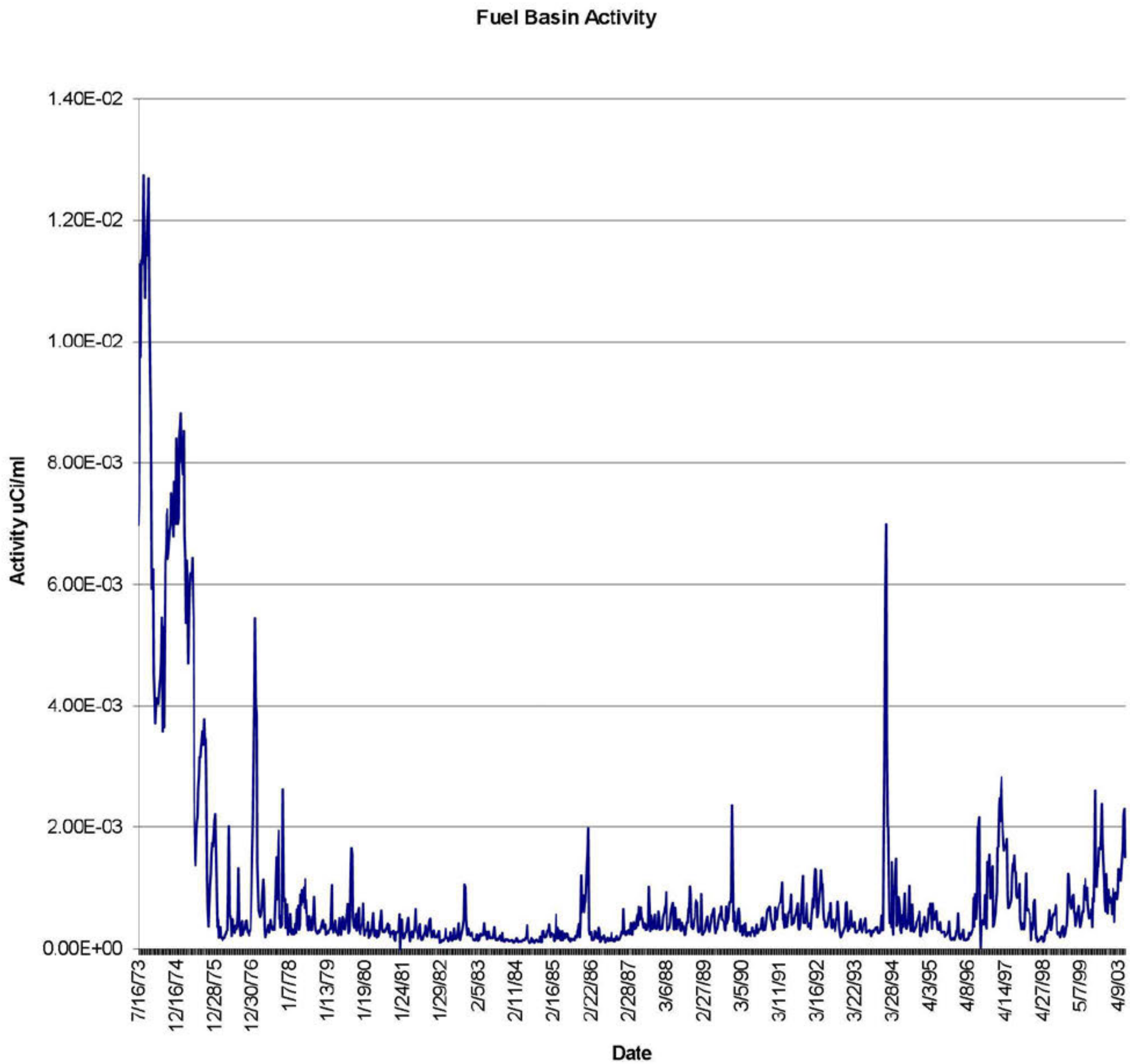

Figure 7-1. History of Morris Operation Basin Water Activity.

An inorganic molecular sieve medium, Zeolon[3], is used to selectively remove cesium. Tests demonstrate that Zeolon-100 removes about two-thirds of the radiocesium per Powdex charge. By routinely using Zeolon and adjusting Powdex replacement frequency, concentrations are effectively controlled.

In addition to radiocesium, the radionuclide contributing most significantly to basin water contamination is cobalt-60. Concentrations in basin water (typically now 2.1×10^{-6} µCi/ml) are attributed to corrosion products on fuel bundle surfaces released to water. Normal filtration and ion exchange reduce cobalt concentrations without special effort.

Fuel in the basins is currently about 714 TeU (1/99). While the basin is near capacity (98%), the radiocesium source term has not significantly increased (about one curie per week as measured in fourth quarter of 1993)[4]. Ability to control basin water radionuclide content ALARA is not compromised.

7.3.2.3 Basin Chiller Decontamination

After a period of operation, contaminants accumulate on the inner surfaces of chiller piping, tubes, and headers. A peroxide chemical decontamination process was installed to reduce exposure rates around the chiller heat exchangers acceptable levels.

7.3.3 Airborne Radioactive Material Sources

Four potential sources could release radioactive material to ventilation air. Most of this material would be captured by the sand filter and some fraction would be exhausted to the stack. These potential sources are:

a. Effluent from the Radwaste System evaporator

b. Off-gas from defective fuel rods in the basin

c. Decontamination activities

d. Uranium used in MFRP testing

Although release of radioactive material in the demisted effluent from the evaporator is possible, such occurrences would be rare and the amount released would be very small. In more than 40 years of fuel storage experience, there has been no apparent leaking of gases from stored fuel. Incidental airborne contamination from decontamination activities could occur. Use of special enclosures ("greenhouses") and other techniques limit such releases to very small amounts, and these activities are infrequent. Natural uranium was used in MFRP testing and some contamination is present within the canyon that could become dislodged and subsequently exhausted via the air tunnel.

Actual measurements of particulate radioactive materials in air exhausted via the stack are made routinely at GEH-MO. In 2019, 2.34×10^{-6} Ci of beta emitting nuclides were released. The resulting dose to the public was 1.3×10^{-6} mRem.

There has been no measurable stack release of a noble gas. Average concentrations of airborne beta activity within fuel storage areas are a small fraction of DAC values.

Annual basin air samples indicate fuel basin Kr-85 source term is about 0.8 Ci per year[5].

7.4 RADIATION PROTECTION DESIGN FEATURES

7.4.1 Facility Design Features

Layout and arrangement drawings of the fuel storage facility, showing locations of all radiation sources (fuel storage areas) described in Section 7.3, are contained in Appendix A-14. Design features related to radiation protection include basin leak detection, collection and control systems, water make-up capabilities, fuel and basket lifting tools that preclude raising fuel too close to the pool surface, water clean-up capability, and other features as discussed in this section.

GEH-MO has provisions for controlling personnel access to areas of the plant having actual or potential levels of radiation or radioactive contamination that exceed levels specified in plant procedures. There is little potential for high radiation dose rates or contamination levels in most areas.

Radiation measurement equipment is provided at various locations throughout the fuel storage area. This equipment includes area radiation monitoring, criticality monitoring, portable survey meters, and portable and fixed air sampler-monitors.

Basic procedures and criteria for controlling personnel exposures are specified in the GEH-Morris Operation Instructions (MOI's) and Special Procedures (MOSP's). Programs adopted for controlling radiation exposure are the result of previous experience at other installations. The program uses modern equipment and techniques proven effective for control of exposures. Such an approach has effectively maintained exposures well below 10 CFR 20 limits.

7.4.2 Shielding

The main building design, originally intended for fuel reprocessing, has maintained personnel exposure to well within 10 CFR 20 limits.

Direct radiation from fuel in storage is shielded by basin water. Concrete shields are used where appropriate, such as for the basin filter.

7.4.2.1 Direct Radiation from Fuel in Storage

The direct radiation dose rate from fuel in storage is from actual measurements of dose rates obtained in July 2004. A dose rate of 3 mRem/hr was found to be constant with depth when measured below the surface of the pool to within 7 ft. of the top of the fuel bundle upper tie plate. The 3 mRem/hr dose rate is due to radioactive contamination in the pool water. Underwater, about 4 ft. above fuel bundles, dose rates were 12 mRem/hr to 290 mRem/hr.

Routine dose rate measurements are taken throughout the storage basin area. For example, during the year 2016, the average dose rate on the basin crane was about 1.0 mRem/hr as measured by OSL.

7.4.3 Ventilation

The main building ventilation system (Figure 1-17) performs the functions of fresh air supply, personnel comfort control and radioactive contamination control within the plant. To accomplish these functions, a single inlet-single exhaust system is provided in which incoming air is filtered and heated for cleanliness and personnel comfort and then distributed to various main building zones at pressures controlled to assure air flow is always from zones of slight (or no) contamination towards zones of potentially higher contamination. Exhaust air is collected in the air tunnel and drawn through the sand filter or HEPA filters and discharged through the stack.

7.4.3.1 Primary Safety Considerations

a. If airborne radioactive materials escape from waste treatment systems, the material is confined within the main building ventilation system under all credible operating conditions.

b. Spread of airborne radioactive material from contaminated areas is prevented under normal and abnormal operating conditions.

c. Radioactive material released from the plant must be held ALARA.

7.4.3.2 Principal Mechanisms for Ensuring Safety

a. Confinement of mobile radioactive material is ensured by:

   (1) Providing high integrity ventilation exhaust ducts, filters, fans and auxiliary equipment required for system operation, with protection against earthquake and tornado effects.

   (2) Using the building structure to provide secondary confinement barriers of structural strength and leak tightness appropriate to potential contamination, potential internal pressures, and exterior forces that could exist.

b. Protection against spread of airborne radioactive material within the building is ensured by:

   (1) Maintaining ventilation airflow in series patterns from zones of low (or no) contamination potential towards those of successively higher potential levels.

   (2) Providing a single ventilation exhaust path and means for automatic pressure balancing to prevent crossflow between ventilation subsystems.

   (3) Locating the ventilation air intake point to minimize potential to recycle stack effluents.

c. Discharge of radioactive material from the plant stack is held ALARA by:

   (1) Providing effective demisting of vapor from the Radwaste System.

   (2) Passing all potentially contaminated ventilation air and Radwaste System evaporator gaseous effluent through the sand filter.

7.4.4 Airborne Effluent Monitoring Instrumentation

7.4.4.1 Functional Description

Multiple samplers collect samples of air discharged from the plant. Sufficient detailed information is obtained to calculate the integrated total quantity of radioactive material released from the stack to the atmosphere.

7.4.4.2 Major Components and Operating Characteristics

a. Sand Filter Inlet Sampler: A sample of the air stream entering the sand filter is passed through a particulate filter that is analyzed weekly for alpha and beta-gamma activities.

b. Ventilation Exhaust Sampling: Provisions are made for taking parallel samples of the ventilation exhaust air stream, downstream of exhaust fans, for continuous sampling of stack effluent release. Sample streams are filtered to collect particulates for periodic counting. Parallel sample streams are combined downstream of their individual blowers.

7.4.4.3 Sampling Considerations

Effluent samples withdrawn for monitoring and analysis must be representative of sampled streams, unbiased and sufficiently sensitive to ensure radionuclides released to the environs are adequately assessed.

a. Representative air sampling is ensured by:

   (1) Utilizing dual stack sampling points designed to provide an accurate cross section of effluent radioactive materials.

   (2) Collecting samples by means of isokinetic probes designed so that particulate concentrations collected are representative of the air stream sampled.

   (3) Calibrating sampler equipment to establish sample volume relationships to requisite accuracies.

   (4) Providing sample lines as short and straight as possible, with no abrupt turns, to minimize line effects.

   (5) Providing sample line heating to prevent condensation in lines.

b. Continuous sampling of effluent is ensured by:

   (1) Providing two redundant sampling systems to determine alpha and beta-gamma particulate. I-131 monitoring is provided in the remarkable event of a criticality incident in the basin.

   (2) Employing sample pumps of high reliability.

   (3) Controlling air-flow through sampling systems equipped with low flow alarms to indicate pump failure or filter blockage.

   (4) An effective system maintenance program.

7.4.5 Radiation Monitors

7.4.5.1 Functional Description

A number of different radiation monitoring systems are provided throughout the fuel storage areas to detect radiation associated with normal operations, and to detect and warn personnel of abnormal levels.

7.4.5.2 Major Components and Operating Characteristics

a. Area Radiation Monitors (ARMs): ARMs are located in various occupancy zones to provide continuous indication of gamma radiation levels. These monitors employ Geiger-Müeller tube sensor-converters equipped with auxiliary units to provide local indication of radiation levels as well as local audible and visual alarms. Output from the basin area units (criticality monitors) is routed to an indicator-alarm-trip unit and to the Site Instrument Monitoring System (SIMS) in the CAS/SAS. ARMs have a range of 0.1 to 1000 mR/hr, with the criticality monitors, specifically, having a range of 0.1 – 10,000 mR/hr.

Each monitor is equipped with two adjustable set-point trip units - one to alarm on high readings and the other to warn of instrument malfunctions as evidenced by abnormally low readings.

b. Air Sampling and Monitoring: A combination of portable air samplers and fixed air monitoring stations are utilized to determine concentration of airborne radioactive material in fuel storage areas and to provide warning of approach to levels requiring corrective action. A sampler consists of an electrically powered vacuum pump, flow control system and filter. After an appropriate sampling period, the filter is removed for counting. Air monitors, consisting of sampling systems equipped to detect buildup of activity on filters, are provided in areas of personnel occupancy in which airborne concentrations may exceed 10 CFR 20 limits.

c. Criticality Monitors: A detection system (two ARMs) is provided in the fuel storage pool enclosure to warn personnel in the highly unlikely event of a criticality incident and to initiate evacuation to staging areas. Detectors are set at a trip point high enough to lessen potential for false alarms. Two detectors ensure monitoring continuity. Criticality alarms are unique, intermittent klaxons so situated that they can be heard throughout the main building, auxiliary buildings, and in outside areas.

d. General: Portable survey instruments and hand-foot counters are in use. Optically stimulated luminescent dosimeters (OSL's) are posted throughout the site. An ionization chamber is mounted in the basin water treatment filter cell to provide information regarding filter radiation level.

7.4.5.3 Radiation Monitor Considerations

Typical locations of basin area radiation monitors are depicted in Figure 7-2 (See Appendix A-17 for updated locations). ARMs are designed to be fail-safe in that they alarm both audibly and visually in the event of an upscale reading. On a downscale reading a warning light will signify instrument malfunction.

[redacted]

Figure 7-2. Radiation monitor typical locations (See App A-17 for update).

a. Adequacy of protection system coverage is ensured by:

   (1) Providing gamma radiation monitors in selected personnel access areas.

   (2) Locating air samplers to provide measurements representative of breathing zone concentrations.

   (3) Use of portable instruments to monitor specific activities.

b. Assurance of clear indications of abnormal conditions is provided by:

(1) Equipping monitors with local alarms to assure adequate warning of personnel in the vicinity of the monitor.

(2) Providing distinctive alarm signals designed to be clearly heard over background noise levels and readily recognized as to meaning.

(3) Including signal recognition and interpretation in operating training requirements and in operating procedures and instructions.

(4) Providing alarm monitoring in the CAS/SAS for the basin criticality monitors.

c. Reliability of personnel protection system functions is assured by:

(1) Providing capability for checking operability and accuracy of all monitors with calibrated radioactive sources.

(2) Providing redundant power supply systems.

(3) Utilizing system components of demonstrated capability and proven reliability

(4) Source check portable instruments before use

(5) Use of self-reading pencils

(6) Use of Optically Stimulated Luminescent dosimetry (OSL) badges

(7) An effective system maintenance program

Calibration of ARMs, air monitors, and other radiation detection equipment is checked periodically. In addition to these requirements, all radiation monitors are calibrated periodically.

7.5 PERSONNEL EXPOSURE ASSESSMENT

Radiation levels at GEH-MO are controlled ALARA. Personnel exposures are determined primarily by background radiation levels and are a function of the total man-hours of occupancy in the basin area and activities under way during an exposure occurrence period.

Management controls include operating limits for radioactive material concentration in basin water requiring special corrective actions. The gross β concentration value is 0.02 μCi/ml. If the rate is exceeded, a cleanup campaign is initiated.

7.6 HEALTH PHYSICS PROGRAM

The primary purpose of the health physics program is minimization of occupational exposure. Personnel protection is accomplished through use of monitoring equipment (described in Section 7.4.5) and by monitoring the radiological status of hazardous areas within the facility. Trained personnel make frequent surveys to appraise conditions and specify protective measures needed for work. Personnel also monitor or inspect activities to keep plant personnel informed of area radiation and contamination status.

Daily exposures during routine operation are maintained ALARA. The personnel monitoring program includes OSL badges and self-indicating pencil dosimeters.

Workers who require access to contamination control areas participate in the bioassay program on an annual basis and at other times as determined by the RSO or Safety Committee. Internal exposures are estimated through reviews of air sample data, and whole body counting. Urinalysis is performed as deemed appropriate by the Radiation Safety Officer[8].

The radiation safety program is conducted according to MOI's and MOSP's. Plant operations are conducted under procedures consistent with site instructions. Operations and maintenance procedures, which have safety significance, are reviewed by the Safety Committee.

7.7 ESTIMATED MAN-REM OFF-SITE DOSE ASSESSMENT

GEH-MO fuel storage activity produces no significant radioactive effluent. The environmental monitoring program is one of effluent sampling and radiation monitoring.

7.7.1 Effluent and Environmental Monitoring Program

Environmental radiation monitoring near the GEH-MO site has been performed since1958.

Monitoring program results from 1968 to 1994 confirm the absence of detectable off-site radioactive contamination. Off-site exposure resulting from fuel storage is a very small fraction of regulatory limits. In addition, Illinois Department of Public Health measured radiation dose rates in populated areas around the DNPS/GEH-MO sites and in 13 central Illinois counties from 1971 to 1976 indicate no significant difference in radiation exposure between the two areas even though the joint site consists of two reactors and a fuel storage facility[9].

Specifications for the current environs monitoring program are depicted in Table 7-3 and locations of sampling points are documented in Figures 7-3 through 7-5. Samples are collected at points on the GEH-MO property boundaries. Reference samples provide a background, which enable GEH-MO to distinguish significant radioactive material introduced into the environment by GEH-MO operation from that introduced by nuclear weapons testing and other sources.

Particulate radioactive material in air consists of residual radioactive fallout from weapons testing and other man-made events plus cosmic and natural sources. Cosmic and background sources result in a dose rate of 2 to 3 mRem/week. River water concentrations show a natural background of about 1×10^{-8} µCi/ml due to natural radium, uranium and radiopotassium.

The program meets USNRC requirements.

Table 7-3
MORRIS OPERATION RADIOLOGICAL MONITORING PROGRAM

Particulates in Air

No routine particulate environmental air samples are collected due to operation of the GEH-MO. Air samples are collected at the site boundary in the event one of the following occurs:

1. The stack air monitoring system and back-up system fails or is out of service for a time period greater than 24 hours.
2. License specification 4.1.1 "Effluent Air" gross beta activity exceeds 4×10^{-8} µCi/ml.
3. An airborne activity release alert is declared as defined by the GEH-MO Emergency Plan.

SAMPLE MEDIUM	COLLECTION SITE	ANALYSIS	FREQUENCY
Exposure by OSL	Duplicate OSL's are placed at the 15-acre site boundary in positions approximately corresponding to eight points of the compass.	Gamma radiation analysis	Quarterly
Water	a. Sanitary Lagoons	Gross β H-3	Monthly
	b. North Drainage Ditch (If water in ditch)	Gross β H-3	Monthly
	c. Eight site monitoring wells	Gross β H-3	Quarterly

[redacted]

Figure 7-3. OSL Sampling Locations

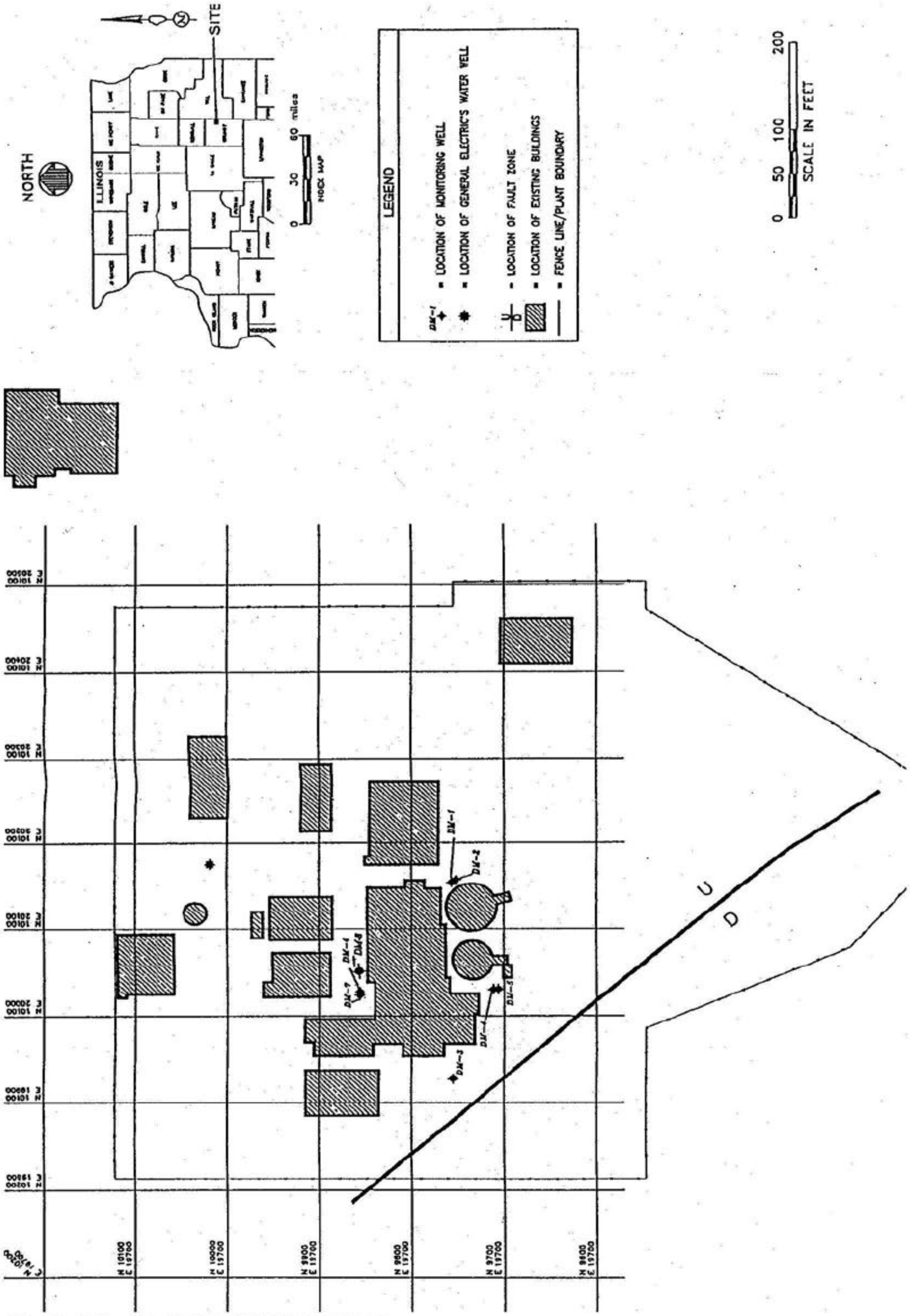

Figure 7-4. Monitoring Well Locations

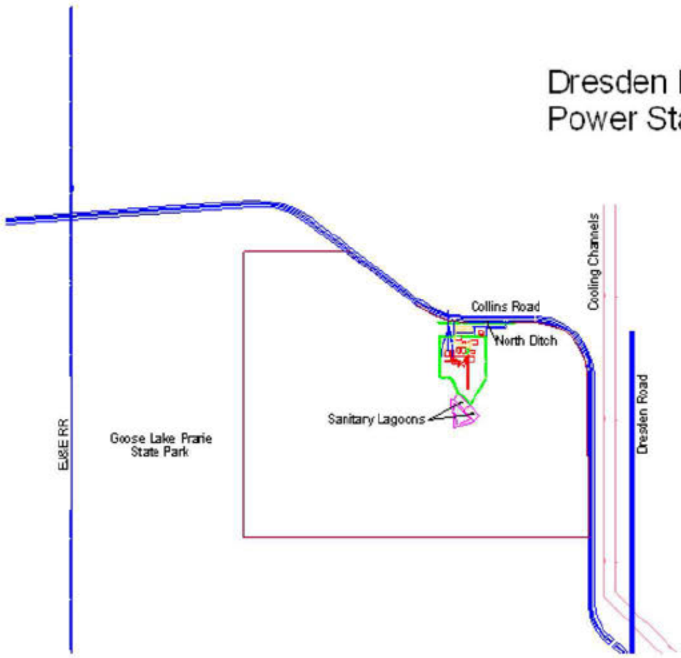

Figure 7-5. Environmental Water Sample Locations at the sanitary lagoons and north ditch

7.7.2 Estimated Exposures

Exposure from radioactive material released in stack effluents is estimated using COMPLY (an EPA software program approved by the NRC). Atmospheric diffusion characteristics, including joint stability-frequency and wind speed data, method and conditions used in calculations of ground-level radiation doses, are discussed in Section 3. Population distribution around the plant is included.

7.7.2.1 External Exposure

Calculations have been made of external exposure from gamma emitters in the stack plume, beta exposure from immersion in the plume and from ground deposition of gamma-emitting particulate activity. Immersion in the plume, and gamma dose from the overhead plume are the only significant contributors. Kr-85 contributes essentially all the exposure from immersion since it is the only radioactive noble gas present after decay of other short half-life noble gases. Kr-85 is a beta emitter with a gamma photon abundance of only 0.4%. Therefore, exposure to Kr-85 results in primarily a beta dose to exposed skin and is of less radiological significance than penetrating whole-body exposure. Shielding provided by clothing will eliminate most skin dose from exposure to β radiation.

For purposes of this analysis, conditions described in Section 7.3.2 were used, with equations and conversion factors for skin doses taken from DOE/EH 0070[10]. Skin dose calculations indicate a maximum off-site dose (about 800 meters from the main stack) of about 0.0045 mRem/yr.

7.7.2.2 Internal Exposure from Inhalation

GEH Morris Operation recognizes the constraint of 10 mRem per year to the general public as suggested by Regulatory Guide 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors", December 1996. Adoption of Reg. Guide 4.20 requires that GEH-MO modify the calculation methodology formerly used. GEH-MO uses "COMPLY" (EPA software program) at screening level 4 to demonstrate compliance and to derive the air emission dose to the public.

7.7.2.3 Man-Rem Calculations

Man-Rem calculations were estimated for annual whole-body exposure due to inhalation of released beta emitters and skin dose due to release of Kr-85. Averages of exposures were calculated for concentric circles with radii of multiples of 10 miles. These average values were multiplied by the population within each area which gives an average annual whole-body man-Rem. The sum of these values for each area out to a radius of 50 miles gives a total of less than 2×10^{-6} man-Rem/yr whole body and less than 0.12 man-Rem skin dose for the period from 1970 to the year 2100.For comparison, the population exposure from normal background radiation (taken at 100 mRem/yr) in the same area is approximately 665,000 man-Rem for 1970, to 750,000 man-Rem for the year 2000. Therefore, the radiological impact from the GEH-MO fuel storage operations is relatively insignificant.

7.7.3 Liquid Releases

There are no planned releases of liquid wastes from the site boundaries.

7.8 REFERENCES

1. RESSAR-41 Reference Safety Analysis Report, Vol. 6, Westinghouse, December 1973 and Amendments.

2. K. J. Eger, Operating Experience - Irradiated Fuel Storage - Morris Operation, Morris, Illinois, General Electric Company (NEDO-20969B).

3. A proprietary product of the Norton Co.

4. See "Annual Report" to Region III, USNRC dated February 14, 1994.

5. See NEDG-249122-1, "In-Plant Test Measurements For Spent fuel Storage At Morris Operation,, " May 1981.

6. T. Rockwell, Reactor Shielding Design Manual, VanNostrand, 1956.

7. M. J. Bell, Origen - The ORNL Generation and Depletion Code, ORNL-4628.

8. Teledyne Isotope, Northbrook, IL. Provides environmental sampling and analysis (especially for radionuclides) and bioassay services.

9. State of Illinois, Department of Public health, Monitoring and Regulation of Nuclear Facilities in Illinois, Springfield, Illinois (1977). The report shows slightly higher radiation levels in the control counties.

10. DOE/EH 0070, "External Dose Rate Conversion Factors from Calculation of Dose to the Public", July, 1988.

11. DOE/EH 0071, "Internal Dose Rate Conversion Factors for Calculation of Dose to the Public", July 1988.

8.0 ACCIDENT SAFETY ANALYSIS

8.1 INTRODUCTION

This section contains an analysis of postulated accidents in terms of the causes of such events, the consequences, and the ability of the GEH-Morris Operation (GEH-MO) organization to cope with each situation.

The function of GEH-MO is to store, and ship irradiated nuclear fuel. A primary requirement of these operations is to protect the public and employees from excessive exposure to ionizing radiation, as specified by the requirements of 10 CFR 72.106. Specifically, any individual at or beyond the controlled area boundary shall not receive a dose greater than 5 Rem to the whole body or any organ from any design basis accident (i.e., those accidents described in this section).

8.1.1 Release Pathways

Exposure of the public and employees might result from postulated accidents, by direct radiation from the fuel, by airborne release of radioactive material, or by release of radioactive material to groundwater. These postulated events are discussed in this section. None of these potential releases have off-site impacts, which exceed the limitations of 10 CFR 72.104.

8.1.1.1 Direct Radiation

Exposure of the public and employees could be postulated to result from direct radiation from fuel in storage or by release of radioactive material to the environs. Direct radiation from the fuel would occur only if the water level in the storage basin became too low to provide adequate shielding. This would pose a hazard to persons only if they were in relatively close proximity to the basin. Loss of water could result from postulated drainage or evaporation of the basin water, but only when basin make-up water supply quantity or rate is not sufficient to keep up with the water loss. Sudden draining of water from the basin is not credible.

8.1.1.2 Airborne Release

Airborne release of radioactive material could result from fuel being mechanically damaged sufficiently to release fission gases from the plena of fuel rods. Of the gases released, only Kr-85 and I-129 would be of concern.

No mechanism exists in the fuel storage environment to cause an airborne release of particulate radioactive material in quantities sufficient to result in exposures approaching limits specified in 10 CFR 72.104. During certain cask operations (e.g., decontamination and venting) particulate releases might occur but in very small quantities, even under the most severe conditions that can be postulated. These quantities would be much too small for an off-site impact. A criticality incident could result from the dropping of a basket in such a way that all the fuel falls out of the basket and comes to rest in a critical array, or by the deformation of fuel baskets into a critical array by a tornado-generated missile. In reality, however, the above events have an extremely low probability of occurring and the impact of either would be substantially less than the limits of Part 72.104.

8.1.1.3 Waterborne Release

Vault intrusion water is normally disposed of in the sanitary lagoons, so that an off-site release would not be likely even in the unlikely event the water is contaminated.

Water from the storage basins can be released due to a leak in the basin structure, permitting water to escape to the surrounding rock.

8.1.2 Accident Description/Discussion

The following sections contain discussion of various postulated accidents and estimates of the quantity of radioactive material release and projected consequences. A summary of events resulting in postulated radiation exposures to the public is shown in Figure 8-1. No combination of normal and credible accident events has been developed that would result in an off-site release or direct radiation exposure that would exceed the regulatory limits for an accident (10 CFR 72.106).

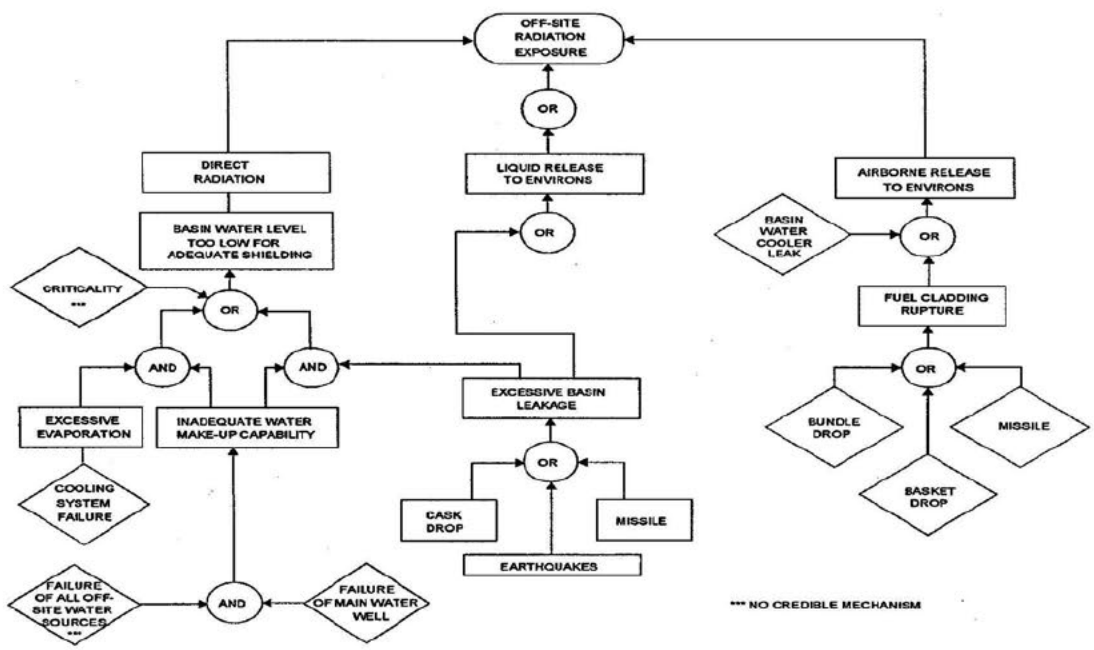

Figure 8-1. Event Diagram of Postulated Accidents

A release of noble gases and halogens from DNPS, similar to or greater than at TMI-2, would not affect fuel storage safety at Morris. The location and construction of the GEH-MO control room, the availability of respiratory protective masks and systems, the availability of protective clothing, and other radiological emergency preparations at Morris would minimize the impact on GEH-MO of any release from DNPS[1]. Even if it should become necessary to temporarily evacuate GEH-MO, the slow loss of basin water by evaporation and the ease of replacement negates possible detrimental effects and protects the public health and safety.

8.1.3 Exposure Paths

Of the possible exposure paths, only whole-body exposure from external radiation and internal exposure through inhalation are considered credible at any off-site location. No mechanism has been identified that will cause radioactive contamination of farmlands, feed lots, or other sensitive areas, that could result in an ingestion dose greater than a small fraction of regulatory limits.

8.2 LOSS OF FUEL BASIN COOLING

The basin cooling system is not critical to safety. When the cooling system is not in service, the water make-up system can be used to replace water lost by evaporation. Even if the water make-up system is out of service, there is adequate time to repair or replace both cooling and make-up systems or to provide make-up water from alternate on-site or off-site sources. (The water make-up system includes the water well and all equipment related to the normal make-up water supply to the basin.)

The time available to provide make-up water if the cooling and water make-up systems are out of service has been determined by measurement of evaporative losses with the fuel in storage as of June 2004. Based on actual measurement of basin heat-up rate, the time available to provide make-up water before reaching the technical specification (Section 10.3.1) limit of 9 feet of water above the top of the fuel bundle upper tie plate is more than 60 days.

8.2.1 Basin Water Temperature

Maximum basin water temperature as measured in June 2004 after 60 days of operation with no cooling or makeup water was 123° F and more than 319,263 gallons of water would have to evaporate before the top of the fuel bundles upper tie plate would be exposed. This would require approximately 150 days.

The probability of excessively high radiation dose rates resulting from loss of fuel basin cooling is clearly quite small given ample time for repairs and water replacement.

8.3 DRAINAGE OF FUEL BASINS

There are no piping penetrations, which could drain the fuel storage basins, and there are no paths for siphoning water from the basin. Therefore, to inadvertently drain water from the basin, the basin structure must be penetrated. Since the basin structure is below grade and given low permeability of surrounding rock (except for the overburden) and high level of upper strata groundwater, leakage (even if it were undetected) would not uncover the fuel (Appendix A.13).

8.3.1 Basin Liner rupture Experience

An accident occurred in June 1972 that resulted in the rupture of the basin liner and demonstrated the ability of GE-MO to withstand and recover from such an incident. No measurable exposure to ionizing radiation was experienced by site personnel or the general public as a result of the incident, and no groundwater contamination above background levels was detected.

8.4 CASK DROP INTO THE CASK UNLOADING BASIN

A postulated means of damaging the basin floor structure is dropping a shipping cask on either the cask unloading pit set off shelf or the floor.

The cask unloading pit set off shelf is protected by an energy absorbing pad designed to accommodate the impact of a cask. Detailed design analysis of the pad is given in Appendix A. Included in that appendix is an analysis of an impact on the corner of the shelf and an impact on the floor of the cask unloading pit. In each case, it is shown the integrity of the structure is not breached and in neither case is basin water released to the environs. Rapid recovery from a breach in the liner caused by a cask incident is discussed in Section 8.3.1.

8.5 FUEL DROP ACCIDENTS

Accidents could occur during fuel handling that might result in mechanical damage to the fuel and subsequent release of fission gases. Such accidents could happen during transfer of fuel from a storage basket to a cask, or during transfer of storage basket from basin to unloading pit. In any case, the postulated accident is assumed to occur in the fuel unloading pit since the fuel is lifted to greater height than in the storage basins.

During cask handling operations, there is no movement of a cask over fuel. The design of the fuel storage facility is such that a cask cannot be moved over the fuel storage basins. Further, administrative controls prevent cask movement when fuel is present in the unloading pit.

The following discussion addresses the fission gas inventory in the fuel, water decontamination factors, and assumptions that pertain to both fuel drop, and basket drop analyses.

a. Fission Gas Inventory in the Fuel

Fission gas inventory in the fuel is dependent primarily on the total fuel exposure. Of the radioisotopes present in the fission gas inventory, Kr-85 and I-129 represent the greatest curie inventory in fuel that has cooled 1 year or more. Figure 8-2 depicts the Kr-85 inventory as a function of cooling times for different fuel exposure levels. Amounts of I-129 in the fuel range from about 0.008 Ci/TeU for 8,000 MWd/TeU exposure to 0.04 Ci/TeU for exposure of 44,000 MWd/TeU and remain essentially constant with time.

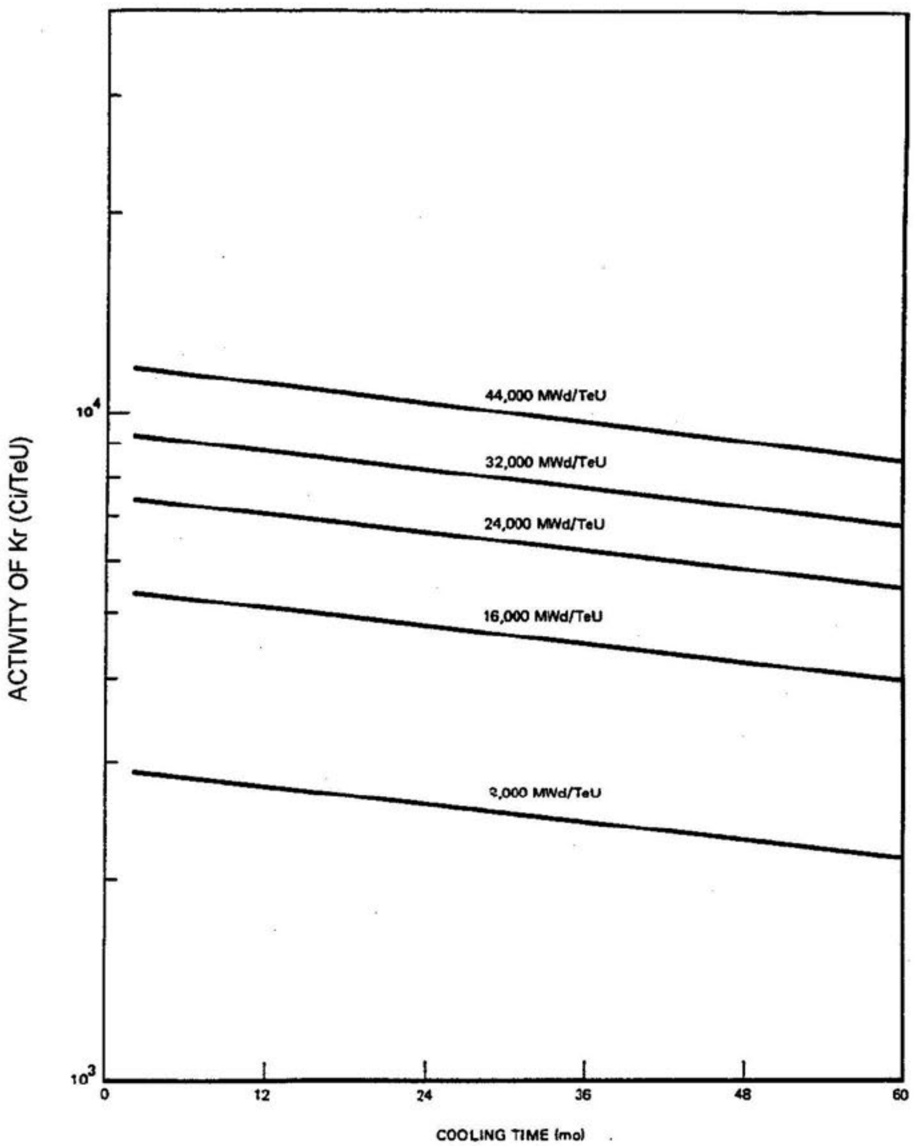

Figure 8-2. Kr-85 Activity as function of cooling time for different fuel exposures. (Total inventory in fuel rod.)

Other fission gases, including I-131, Xe-131m and Xe-133, decay relatively quickly. After one year of cooling time, all three are decayed to insignificant levels as shown in Figure 8-3. The total fission gas inventory for a 1-year cooling time is given in Table 4-1, Section 4.

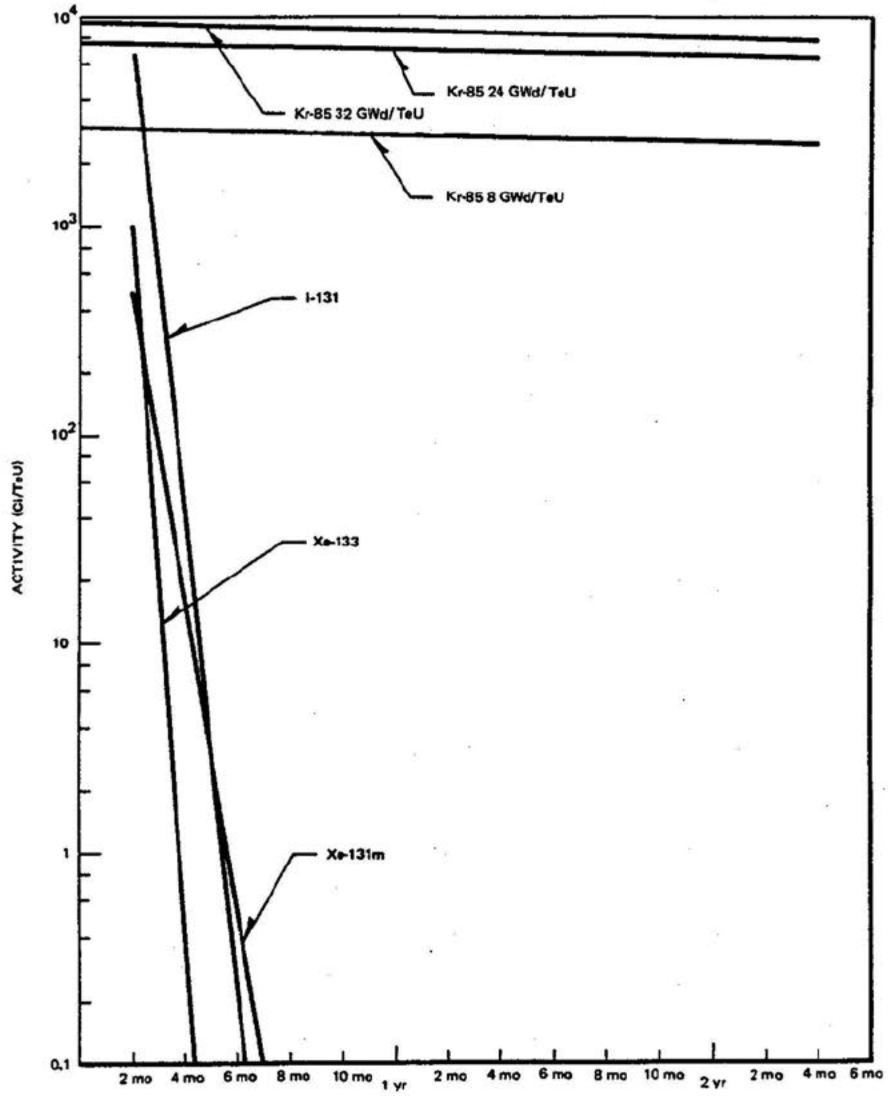

Figure 8-3. Iodine, Krypton and Xenon Decay

The amount of fission gas released from UO^2 fuel and accumulated in the plenum of each rod is dependent on the specific power (fuel temperature) during operation. At higher specific power, a greater fraction of gas will be released to the plenum. Calculations of fission gas inventory result in a release fraction that ranges from 20% to 45% depending on the irradiation history of the fuel rods. For example, a Westinghouse safety analysis report states that approximately 2.5% of Xe and approximately 3% of iodine are found in the gas plenum (Docket 50-295, "Zion Nuclear Power Station," Commonwealth Edison Co.).

GEH uses plenum percentages for radioisotopes that are based on fission product release data from defective fuel experiments[2]. A comparison of these values with the NRC Regulatory Guide and the values used in the fuel drop analysis for GEH-MO is shown below:

	GEH Fuel Drop Analyses for Reactors	NRC Regulatory Guide	GEH Fuel Drop Analyses for Morris Op
	PERCENT OF RADIOISOTOPES(S) IN PLENUM		
Radioiodine			
I-131	1.2	10	2
Kr-85	30	30	30
All other noble gases		10	
Xe-131m	3.9		
Xe-133	2.5		

These values are considered realistic values based on the analytical and experimental data contained in the references cited above. The value for radioiodine is also recommended by Appendix VIII, WASH-1400. The Kr value agrees with that in Regulatory Guide (RG) 1.25.

b. Water Decontamination Factor

Not all iodine released from a fuel rod would be released from the basin water. Being highly soluble, much of the iodine would dissolve and remain in the water. RG 1.25 recommends a factor of 100 for pool decontamination of iodine.

In analysis of a fuel handling accident, Westinghouse based decontamination factors on iodine tests conducted to determine the mass transfer from the gas phase to surrounding liquid[3]. That work resulted in the formulation of a mathematical expression for the iodine decontamination factor in terms of bubble size and bubble rise time. The equation is:

$$\text{Decontamination Factor} = (7.3)\exp[0.313\ t/d]$$

where t = rise time, and
d = effective bubble diameter.

Evaluating the decontamination factor for iodine released from a fuel bundle, a minimum factor of 760 is calculated for a water depth of 26 ft. However, for their "conservative analysis" the factor was reduced to 500.

For a fuel bundle drop at GEH-MO, the worst-case accident occurs in the cask unloading pit. Minimum water depth in that pit is about 32 feet. Therefore, a decontamination factor of 500 is sufficiently conservative.

c. Assumptions

The following assumptions are made for the safety analysis:

1. The fuel bundle or basket drop occurs in the fuel unloading pit.

2. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are released.

3. The overall effective decontamination factor for iodine is 500. Because water has a negligible effect on removal of the noble gases, the decontamination factor is 1.

4. Ventilation air flow exhaust rate from the basin areas is 7,600 scfm via the air tunnel, sand filter and the main stack. Duration of release is 2 hours.

5. Worst case $\overline{X}/Q$ is 2.8×10^{-5} sec/m^3. (See Appendix A.5, Section A.5.1b, Short-Term (Accident) Diffusion Estimates.)

6. Fuel characteristics are 44,000 MWd/TeU exposure, 1-year cooling.

7. Dose conversion factors are:

Species	Whole Body $\frac{\text{mRem}}{\mu\text{Ci}} - \frac{\text{m}^3}{\text{sec}}$	Thyroid $\frac{\text{mRem}}{\mu\text{Ci}} - \frac{\text{m}^3}{\text{sec}}$
Noble Gas	4.75×10^{-7}	-
Halogen	8.72×10^{-5}	4.472×10^{-1}

8.5.1 Fuel Bundle Drop Accident

a. It is highly unlikely fuel rods would be ruptured in a fuel drop accident. However, to establish an upper boundary in the consequence analysis, it is assumed all rods in the bundle have ruptured releasing all fission gases present in the plena to the basin. The following release is calculated:

Species	Amounts Released (Ci) BWR	PWR
Noble Gases	684	1530
Iodine	3.3E-7	0.48E-7

It is assumed that all of the fission gases are expelled from the basin and passed through the sand filter and released from the main stack.

Using the assumed values for atmospheric diffusion and dose conversion factors, the maximum off-site dose rates are:

Body Organ	Maximum Dose Rate (mRem/hr) BWR	PWR
Whole Body	4.5E-3	1.0E-2
Thyroid	1.8E-6	4.0E-6

If an individual off-site were exposed at the maximum dose rate for the duration of the accident (2 hr.), the maximum doses are estimated to be about 0.02 mRem whole body and 8.0 x 10^{-6} mRem thyroid. Such doses are clearly insignificant and well below the Part 72 guideline of 5 Rem for whole body or any organ.

If this accident were to occur with the ventilation system inoperable, the basin enclosure would contain the fission product gasses and act as a radiation source. Using Microshield v5.05 a Grove Engineering software program for estimating exposure from gamma radiation, the exposure from this source would be (mR/hr):

	BWR	PWR
Off-Site dose	.12	.26
Dose at Basin enclosure boundary	14.2	31.7

b. Actual Bundle Drop Experience

In actual fuel drops, some fuel bundles suffered minor damage, but in all cases, no major deformation of the fuel bundles occurred. For example, during the winter of 1973-1974 the Pilgrim Nuclear Power Station was down for a scheduled refueling and maintenance outage. During transfer of irradiated fuel from the core, a fuel bundle was accidentally dropped from the fuel grapple to the fuel pool floor. The bundle was carefully inspected. There was no indication of major fuel rod failure or distortion nor was there a measurable release of airborne activity as a result of this drop.

In the fall of 1974 during a scheduled outage of the Millstone Nuclear Power Station, an irradiated fuel bundle was dropped to the floor while being transferred from the fuel preparation machine to the fuel storage rack. Consequences of that drop included fracture of all the tie rods, separation at the upper tie plate, and minor permanent deformation at the upper tieplate. Although the fuel bundle appeared to be slightly bent and twisted, no major dislocation of rods, rod segments, or fuel pellets was indicated.

Early in the operation of the Garigliano reactor in Italy, a fuel drop occurred during transfer of fuel to the operating floor. A fuel rack containing five unirradiated fuel bundles dropped on a concrete floor, a distance of about 70 ft. in air. As a result, the rack was badly bent and twisted. Approximately 20% of the 36 fuel rods in each bundle split. Although some fuel pellets were expelled, most of the pellets remained within the fractured rods. Damage to each fuel bundle was confined to the lower one-third of the rods, the lower tieplates and spacers. The upper portion of the bundles remained intact with no apparent damage.

In another case, a fuel bundle was dropped more than 15 ft. and landed on a fuel rack. Consequences of that accident were damage to the nosepiece of the lower tieplate and a slight twist of the assembly. No deformation of the fuel rods or other bundle components was found.

8.5.2 Fuel Basket Drop Accident

After the cask is unloaded and the fuel placed in a storage basket, the basket is transferred to a fuel storage basin (Basin 1 or 2). During this transfer, the basket is less than 3 ft. above the basin floor. When in the cask unloading pit, the maximum height is about 22.5 ft. (equivalent drop height in air is about 12.6 ft.) above the cask unloading pit floor.

In the unlikely event that a basket is dropped in the cask unloading pit, there could be damage to the basin liner, the basket, and the fuel it contains. Damage to the basin liner would be less extensive than that analyzed for a cask drop accident. (See Section 8.4). The criticality aspect of a postulated basket drop accident is discussed in Section 8.9.

The fuel rods within a fuel bundle most likely would not break in a postulated basket drop accident. It has been concluded that fuel bundles in a shipping cask retain their integrity in a 30 ft. cask drop[4]. Since the pipe construction of the fuel basket offers support and protection for the fuel, the postulated drop should cause minor, if any, damage to the fuel.

Comparing actual fuel drops (see discussion in Section 8.6.1) with a postulated basket drop accident at GEH-MO, conditions in the actual cases discussed were more severe in that drop heights were greater than the maximum drop height in the GEH-MO cask unloading pit (12.6 ft. equivalent in air). Many of the actual drops involved fuel bundles that were unsupported and not as well contained as fuel would be in the GEH-MO fuel storage basket.

A structure is installed in front of the entrance of the fuel storage basin (Figure 1-15) to restrain a basket in the event it is somehow dropped at the entrance and the top of the basket tips toward the cask unloading pit. The restraint prevents the basket from tipping in such a way as to disgorge the fuel it may contain.

To transfer a basket from the cask unloading pit, the basket is moved directly under the cask unloading pit doorway guard (Section 5.4.3.3) and lifted through the bottom of the structure. Then the basket is moved laterally into the fuel storage basin. Therefore, the orientation of the basket involved in a postulated drop accident is vertical (i.e., a side drop is not possible and is not analyzed).

8.5.2.1 Accident Analysis

In addition to the assumptions listed in Section 8.6.c, it is assumed the storage basket is full of fuel at the time the accident is postulated. It is unlikely any of the fuel rods would be damaged in such a drop. However, to conservatively evaluate consequences, all the rods in all the bundles are assumed to have ruptured and all the plenum fission gases are assumed to be released to the basin water.

a. The amount of fission gases released to the basin area is calculated to be:

	Amount Released to Basin Area (Ci)	
Species	BWR	PWR
Noble Gases	6156	6120
Iodine	3.01E-6	2.99E-6

b. The maximum off-site dose rates for 2 hr. release duration were calculated to be:

	Maximum Dose Rate (mRem/hr)	
Body Organ	BWR	PWR
Whole Body	4.05E-2	4.0E-2
Thyroid	1.62E-5	1.6E-5

An individual off-site who received the maximum exposure for the 2-hour period would receive less than 0.08 mRem to the whole body and 3.25E-5 mRem to the thyroid. Such an exposure is insignificant compared to the Part 72 guideline value of 5 Rem to the whole body or any organ.

If this accident were to occur with the ventilation system inoperable, the basin enclosure would contain the fission product gasses and act as a radiation source. Using Microshield v5.05 a Grove Engineering software program for estimating exposure from gamma radiation, the exposure from this source would be (mR/hr):

	BWR	PWR
Off-Site dose	1.05	1.04
Dose at Basin enclosure boundary	127.5	126.8

8.5.3 Recovery Practice

Specific procedures for recovering from a basket or bundle accident cannot be described because of the many variables involved (arrangement of bundles on the unloading pit floor, etc.). In general, however, recovery would involve picking up each bundle using appropriate grapples and inspecting each bundle for damage before inserting into a basket. Damaged bundles would be handled (canned or as otherwise appropriate) in much the same manner as for damaged incoming fuel.

8.6 TORNADO-GENERATED MISSILE ACCIDENT

An accident is postulated in which a tornado-generated missile is hurled into the fuel storage basin. Because the building covering the basins is not designed to withstand the forces of a tornado, it is assumed that the building has been blown away, leaving the fuel basins exposed.

The impact of a missile could cause damage to the basin liner or fuel, but not both concurrently. As indicated in the discussion of potential missiles in Appendix A-15, a missile would not have sufficient energy to damage both fuel and basin liner after striking one or the other.

Criticality aspects of this accident are discussed in Section 8.9. The analysis below concerns the consequences of a missile damaging the fuel. In the missile analysis given in Appendix A-15, two missiles were analyzed. One was a 12 in. diameter by 20 ft. long section of a telephone pole weighing 630 lb. The other missile was a small automobile, 5 ft. by 5 ft. by 8 ft. in dimensions and weighing 1,800 lb. The spectrum of missiles has been expanded to include those listed in Table 8-1. The impact velocity given in the table is defined as that when the missile enters the water of the storage basin.

8.6.1 Accident Analysis

Each missile that was analyzed is listed in Table 8-1. The approximate velocities and kinetic energies at depths of 14 ft. and 21 ft. are given in Table 8-2. These values are those the missile could have if it entered the storage basin water in a vertical orientation. If the missiles entered the water in a horizontal orientation the drag force is greater in many cases and its velocity and kinetic energy would be less. Therefore, the values shown in Table 8-2 are "worst-case" values.

Postulated missile damage depends principally on the cross-sectional (or impact) area, its weight, and the amount of energy it could transfer to the fuel bundle. As indicated in Table 8-2, Missile F has the greatest amount of energy at a depth of 14 ft, which is the depth to the top of the fuel storage baskets. Because of its weight and frontal area (approximately 143 sq. in.), it could potentially cause the most damage. Yet, there is a limit to the number of fuel bundles such a missile could damage.

If the missile entered vertically into the pool, it could potentially strike as many as six BWR bundles or four PWR bundles. The storage basket would move under the impact and the pipes that make up the basket would probably break free. This action would likely absorb all the energy delivered by the missile.

Other missiles, mostly various sizes of pipe, could cause fuel rupture. However, the damage would be confined to one or two fuel bundles, except for Missile E, the 12 in. diameter pipe. This missile could potentially damage as many as six BWR or four PWR fuel bundles, which is comparable to that estimated for the utility pole, Missile F.

Table 8-1
LIST OF TORNADO-GENERATED MISSILES

Missile	Dimensions	Weight (lb)	Impact Velocity as Fraction of Tornado Velocity*
A-Wood Plank	4 in. x 12 in. 12 ft.	200	0.8
B-Steel Pipe	3 in. diam, 10 ft. long, Sched 40	78	0.4
C-Steel Rod	1 in. diam x 3 ft. long	8	0.6
D-Steel Pipe	6 in. diam, 15 ft. long, Sched 40	285	0.4
E-Steel Pipe	12 in. diam, 15 ft. long, Sched 40	743	0.4
F-Utility Pole	13.5 in. diam x 35 ft. long	1,490	0.4
G-Automobile	20 $ft.^2$ frontal area	4,000	0.2

- Defined as rotational plus translational velocity.

Table 8-2

VELOCITIES AND KINETIC ENERGIES OF MISSILES IN WATER
WHEN ENTERING FUEL POOL IN A VERTICAL POSITION

	14 ft. Depth		21 ft. Depth	
Missile	Velocity (ft./sec.)	Kinetic Energy (ft.-lb.)	Velocity (ft./sec.)	Kinetic Energy (ft.-lb.)
A	196	1.2×10^5	124	4.8×10^4
B	195	4.6×10^4	188	4.3×10^4
C	236	7.0×10^3	202	5.0×10^3
D	200	2.0×10^5	196	1.8×10^5
E	200	4.6×10^5	195	4.4×10^5
F	159	6.0×10^5	136	4.3×10^5
G	13	1×10^4	13	1×10^4

Missile G, the automobile, reaches a terminal velocity of about 13 ft./sec. within a depth of about 7 ft. It would then settle to the top of the fuel or to the floor. If it hit the fuel, the energy (one of the least of all the missiles) that it could transfer to the fuel is distributed over a 20 sq. ft. area. No fuel is expected to fail as a result of impact from this missile.

8.6.2 Assumptions

Assumptions used in the safety analysis include the following

a. All the fuel rods in six BWR bundles or four PWR bundles are ruptured. The impact of only one basket is considered.

b. The accident takes place in the fuel storage basin.

c. An average of 30% of the total Kr-85 and 2% of the I-129 activity is in the fuel rod plena and available for release.

d. No solid fission products are released (negligible particulate radioactive material is present in the fuel plena).

e. The overall effective decontamination factor is assumed to be 1 (the accident is assumed to occur in the fuel storage basin).

f. Fuel characteristics are 24,000 MWd/TeU exposure, specific power of 40 kW/kgU and one-year cooling.

g. The storage basin is open (i.e., the sheet-metal building over the basin is assumed to have been blown away by the postulated tornado).

h. A maximum X/Q value is 4.0×10^{-4} sec/m^3 is taken from Appendix A.5, Section A.5.1 for a short-term ground level release.

8.6.3 Dose Rate Calculations

Using the above assumptions, the amount of fission gases released was calculated to be:

	Amount Released (Ci)	
Species	BWR	PWR
Noble Gas	2.5E+3	3.7E+3
Iodine	1.2E-6	1.8E-6

Assuming an individual was present during the entire period during which the cloud passed, his maximum exposure is calculated to be approximately:

	Dose (mRem)	
Body Organ	BWR	PWR
Whole Body	0.5	0.8
Thyroid	2.3E-4	2.4E-4

Comparing these values with the Part 72 guideline values of 5 Rem to the whole body or any organ, they are clearly insignificant.

8.7 CHILLER SYSTEM LEAK

A water to freon heat exchanger system replaced the fin-fan coolers in 2000, and basin water no longer is piped outside the building to the original fin-fan coolers. The release of radioactive material into the atmosphere because of a leak in the basin chiller system - specifically, a leak in a water-to-freon heat exchanger is not possible. The operating pressure of the freon is greater than the basin water, so freon would leak into the basin water and not the reverse.

If the leakage occurred in the heat exchanger structure, the water would be channeled to a sump and automatically pumped to the Rad Waste System.

8.8 CRITICALITY ACCIDENT

The safety margin against an accidental criticality could potentially be reduced by receiving fuel that is more reactive than assumed in the design analyses or by mechanical damage to the storage basket or fuel sufficient to cause the stored fuel bundles to be forced into a critical configuration.

8.8.1 Fuel Handling Procedures

Nuclear safety in the cask unloading pit is maintained, in part, by handling one fuel bundle or one fuel basket at a time in accordance with approved procedures. However, fuel baskets are not limited to one fuel bundle when being transferred to storage: each basket can hold as many as four PWR fuel bundles or nine BWR fuel bundles.

The baskets are designed to rest in a grid installed in the fuel storage basins. A single grid section is installed in the cask unloading pit to hold a maximum of three baskets in line.

Fuel bundles are transferred, one at a time, from the shipping cask to the storage baskets. (See Section 1.) The baskets are removed from the cask unloading pit, one basket at a time, and placed in the fuel storage basin. Prior to moving the cask, all fuel must be removed from the cask unloading pit; either moved to storage in Basins 1 or 2, or loaded into the cask for transfer.

8.8.2 Reactivity Calculations

KENO calculations were performed by BNWL for a square array of four PWR bundles having 3/16-inch stainless steel plate between the bundles and around the array. For fuel having an enrichment of 1.575% U-235 and a K_∞ of 1.1996 the k_{eff} values for the array were as follows:

Bundle Pitch (in.)	k_{eff}
8.675	0.930 ± 0.004
9.250	0.923 ± 0.004
9.732	0.890 ± 0.005

The results calculated with the GE codes are about 5% more conservative than those calculated with the KENO code. Fuel characteristics for these calculations were as follows:

Rod Pitch:	0.604 in.
Rod o.d.:	0.448 in.
Pellet diameter	0.400 in.
Cladding Material	Zirconium
Rod Array:	14 x 14

PWR fuel having an initial k_∞ of 1.35 (2.8% U-235) and having undergone one cycle of irradiation (10,000 MWd/TeU) would have a post-irradiation k_∞ based on BNWL calculations using the LEOPARD code, of approximately 1.19. Calculations of uniform arrays of PWR fuel were made by GE personnel using proprietary reactor design codes, to describe the relationships between k_∞ spacing and K_{eff}. These calculations did not include the poisoning effect of the stainless steel in the baskets, which BNWL calculations indicated would reduce k_{eff} by 2.5%. Figure 8-4 depicts the relationship between k_∞ and K_{eff} for PWR fuel bundle arrays with 2 in. separation. A 2.5% reduction in k_{eff} is included for the effect of stainless steel. The data shows that k_∞ would have to exceed 1.21 for the array to be critical.

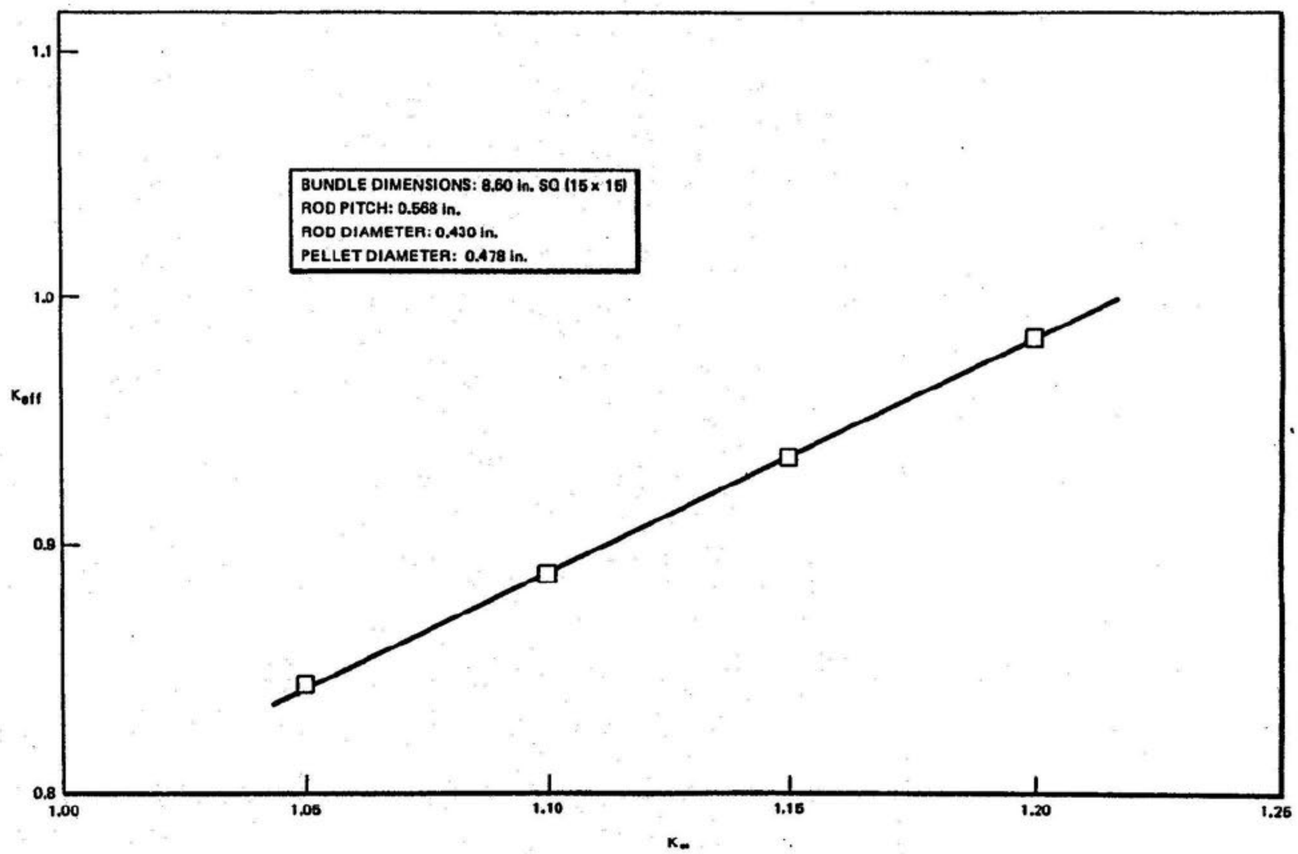

Figure 8-4. PWR fuel bundle array at 2-inch separation.

8.8.3 Missile Impact

The close-packed, pipe sleeve construction of the fuel baskets makes it highly improbable that a missile could cause sufficient compaction of the fuel baskets to cause a criticality accident since the baskets would have to be compressed along two axes simultaneously. Conceivably, a single basket could be driven diagonally into a corner, causing the inner corners of two fuel bundles to be driven together at the top, while the inner corners of the other two elements would at least maintain the designed separation or tend to be spread apart.

Accurate predictions of the effects of the impact of a tornado-borne missile on a system as complex as an array of the fuel storage baskets, would be difficult to make or to prove. To provide insight into the potential increase in neutron multiplication that could arise from reduced spacing, an analysis of three PWR bundles in a "T" configuration, closely spaced over their entire length, was done to estimate the effect of driving three assemblies into a corner. Since this example does not provide consideration of the fourth bundle in the basket, an example of reduced spacing involving four PWR bundles is provided. Such a condition represents an extremely improbable event since the fuel would have to be compacted into a corner from two directions 90° apart over a substantial portion of its length. Because such a compaction would result in separation of the fuel in the compacted array by more than 10 inches of water from the fuel in the closest baskets, the four-bundle array can be considered isolated. The results of calculations performed by GE personnel for a water-reflected, close-packed, square array of four PWR fuel bundles are shown in Figure 8-5.

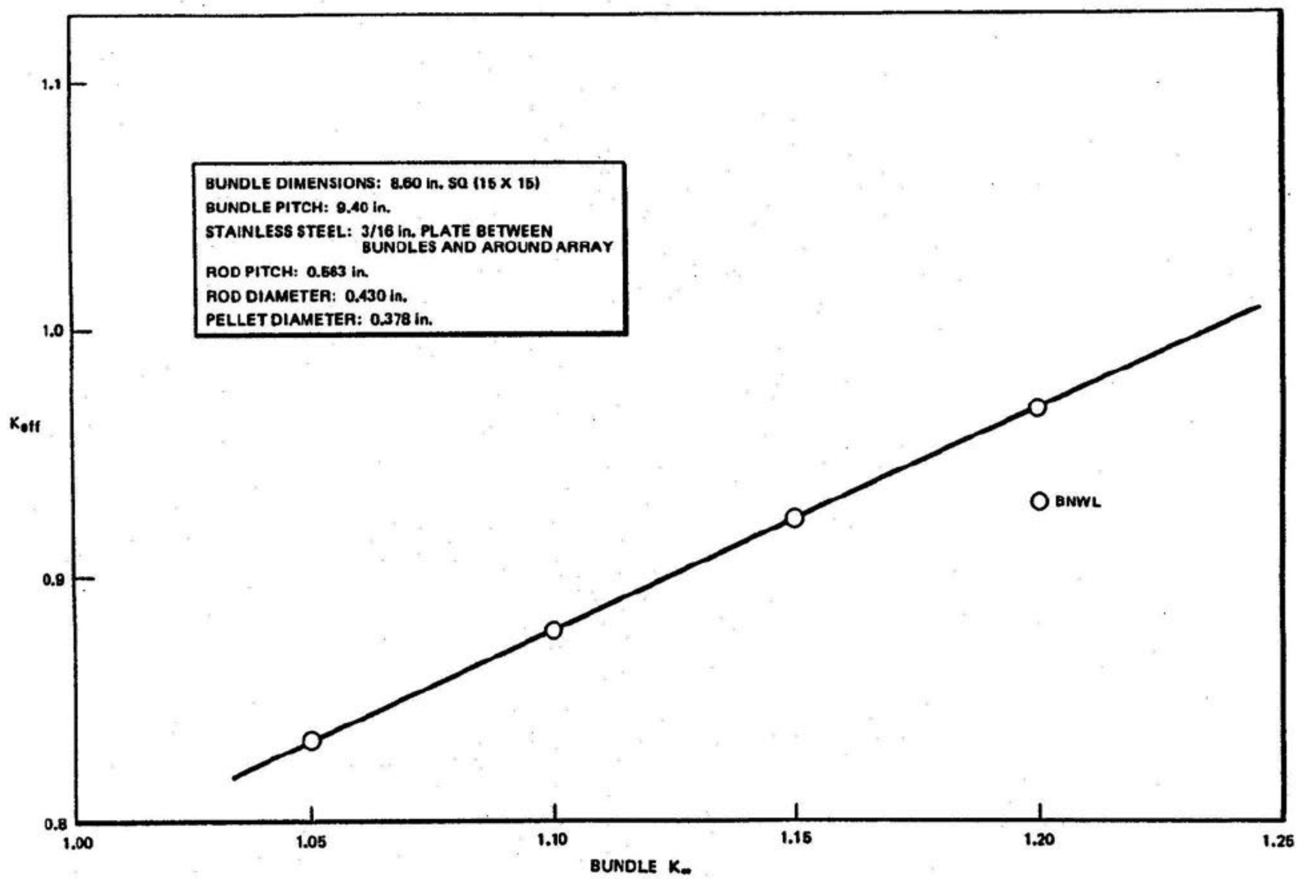

Figure 8-5. Close-Packed array of four PWR bundles.

For such a four-bundle array to become critical, the infinite multiplication factor must average at least 1.23. (Reactivity calculations are discussed in Section 8.9.2)

8.8.4 Consequences of a Criticality Accident

No criticality accidents have occurred in low enriched LWR bundle systems. Accidents have occurred in chemical reprocessing or critical assemblies involving plutonium or highly enriched uranium. Historical criticality incidents in nuclear separation facilities have had fission magnitudes estimated at 1.3×10^{17} to 4×10^{19} fissions. In no case has the reaction been of an explosive nature.

The accidents have either displaced the critical mass such that it was no longer in a critical geometry and thereby terminating the criticality, or the critical mass pulsed in and out of critical geometry.

A criticality accident in the fuel storage basin of GEH-MO is precluded by many factors, some of which include:

a. Geometric constraints imposed by the fuel bundles, storage baskets and holding grid

b. Design and operation of the storage system

c. Administrative procedures for fuel receiving and storage

d. Lower content of fissile material in the fuel bundle than assumed in calculations

e. Neutron poison content in the fuel not assumed in calculations

Nevertheless, a hypothetical criticality is postulated to provide a basis for evaluating the consequences of such an accident. Recovery from a hypothetical criticality would be much the same as from a basket or bundle drop (Section 8.5.2.1), except that a suitable tool suspended from the crane would be used to separate the critical assembly, stopping the reaction. Radiation levels at the pool surface would be low (up to 15 mRem/hr) so that no special protective measures would be required.

8.8.4.1 Assumptions

Primary assumptions used to evaluate a criticality accident include:

a. a point source is assumed at a depth of 16 feet; and

b. Fission gases released to the pool atmosphere as a result of the criticality are negligible. Release of fission gases due to the missile impact is covered by Section 8.7.

Since no reasonable mechanism exists for a criticality accident in GEH-MO fuel storage pools, no meaningful values for characteristics such as reactivity insertion rates, specific power, etc., can be defined. However, a range of 10^{18} to 10^{20} fissions has been evaluated and adequately covers the range of total fissions for such a system.

A depth of 16 ft. was assumed because about 90% of the active fuel is below the 16 ft. level. The top of the active fuel is 14.5 ft. below the water surface.

It is assumed that all the fission products, including fission gases, would be contained within the UO_2 fuel matrix. Temperatures would not be sufficient to drive the fission products from that matrix. Any products that migrate from the fuel matrix would be contained within the fuel void spaces inside the fuel rod.

The gamma flux at the surface of the pool is approximated by the equation for a point source:

$$(\phi)=\left(\frac{BS}{4\pi t^2}\right)(\exp(-ut))$$

where

ϕ = scalar flux (MeV/cm^2-sec);
B = build-up factor;
S = source strength (MeV/sec);
t = distance from source to pool surface (487.68 cm); and
μ = macroscopic cross section for shield material, water (cm^{-1})

Gamma-ray spectra for prompt fission photons are given in Table 8-3. Table data were found in Reactor Physics Contents, ANL-5800, Section 8. The four-group Spectrum B that is given in Table 8-3 was used to calculate the gamma flux. Values for the buildup factors were found in Rockwell's Reactor Shielding Design Manual, page 435.

The dose rate is:

$D' = \phi / c$

where

D' = dose rate mR/hr

c = flux to dose conversion factor $\frac{MeV/cm^2 - sec}{mR/hr}$

Values for c for each energy group are:

$c_1 = 5.2 \times 10^2$

$c_2 = 6.2 \times 10^2$ $\frac{MeV/cm^2 - sec}{mR/hr}$

$C_3 = 7.8 \times 10^2$

$C_4 = 8.6 \times 10^2$

The dose rate in terms of mR/fission is given by:

$$\frac{BM(E)e^{-ut}}{4\pi t^2 c(3600)}$$

where

M(E) = energy/fission, or MeV/fission

Table 8-3

PROMPT FISSION GAMMA-RAY SPECTRA

	Spectrum A		Spectrum B	
E (MeV)	N(E) (γ/fission)	M(E) (MeV/fission)	E (MeV)	M(E) (MeV/fission)
0.5	3.1	1.55	-	-
1.0	1.9	1.90	1.0	3.451
1.5	0.84	1.26	-	-
2.0	0.55	1.10	2.0	3.085
2.5	0.29	0.725	-	-
3.0	0.15	0.450	-	-
3.5	0.062	0.217	-	-
4.0	0.065	0.260	4.0	1.035
4.5	0.024	0.108	-	-
5.0	0.019	0.095	-	-
5.5	0.017	0.094	-	-
6.0	0.007	0.042	6.0	0.256
6.5	0.004	0.026	-	-
	7.028	7.827		7.827

Values of M(E) are given in Table 8-3 for Spectrum B. The calculated doses in terms of mR/fission at the surface of the water in a storage basin are given in Table 8-4. The calculated doses at the surface of a basin from 10^{18} fissions, 10^{19} fissions, and 10^{20} fissions are 0.413 mR, 4.13 mR, and 41.3 mR, respectively. These doses are obviously not of serious consequence.

For comparison, extrapolation of actual measurements from an experiment produced a gamma-ray tissue dose rate of 0.18 mRad/hr. These data were taken from Figure 8.8 in Section 8, ANL-5800, showing plots of centerline attenuation data for water measured in the Bulk Shielding Facility at ORNL.[5]

The curves in Figure 8.9 of ANL-5800 also give data for fast neutron dose rate and thermal neutron flux. These data are given as a function of watts for the source, which is a reactor in this case. As indicated, the thermal neutron flux for 16 ft. (approximately 488 cm) is 5×10^{-8} n/sq cm - watt. The fast neutron tissue dose curve drops sharply and ends at a value of 2×10^{-7} erg/gm - hr watt for approximately 175 cm. The fast neutron dose at a distance of about 488 cm is negligible.

Table 8-4

DOSE, mR, PER FISSION,
AT BASIN SURFACE

Group	Dose: mR/fission
1	2.118×10^{-25}
2	6.780×10^{-22}
3	1.391×10^{-19}
4	2.736×10^{-19}

A criticality of 10^{18} fissions produces about 8.9 kWh of energy. If it is assumed the event lasts 3 hours, the power level for those 3 hours is about 3 kW. The thermal neutron flux was determined to be approximately ($1.5 \times 10^{-4\ n/sq.\ cm.}$) - sec at the surface of the pool. The corresponding dose rate is about 6.2×10^{-7} mRem/hr.

The consequences of a postulated criticality in the storage basin are no more serious than the short-term operation of a low-power, swimming-pool type nuclear reactor commonly used at some universities.

8.9 REFERENCES

1. According to recent studies in the U.S. and abroad, significant evidence indicates that consequences of a hypothetical fuel melting accident may be less than currently predicted by at least one or two orders of magnitude, see appendices E, F, and G, Report of the President's Commission on the Accident at Three Mile Island.

2. N. R. Horton, W. A. Williams, and J. W. Holtzelaw, Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor, March 1969 (APED-5756).

3. RESSAR-41, April 1974.

4. See "IF-300 Shipping Cask Consolidated Safety Analysis Report," NEDO-10084-2, Chapter V.

5. Attenuation in Water of Radiation from the Bulk Shielding Reactor: Measurements of the Gamma-Ray Dose Rate, Fast-Neutron Dose Rate and Thermal Neutron Flux, July 8, 1958 (ORNL-2518).

9.0 CONDUCT OF OPERATIONS

9.1 INTRODUCTION

General Electric Hitachi Nuclear Energy has established a GEH-MO organization such that administrative controls are in place to ensure decisions are made at the proper level of responsibility, with appropriate technical advice, and in a timely manner. The record of safety and regulatory compliance established by GEH-MO throughout its operation has been excellent.

9.2 CORPORATE ORGANIZATION

Principal organizational levels of General Electric Company in effect as of December 2018 are shown in Figure 9-1.

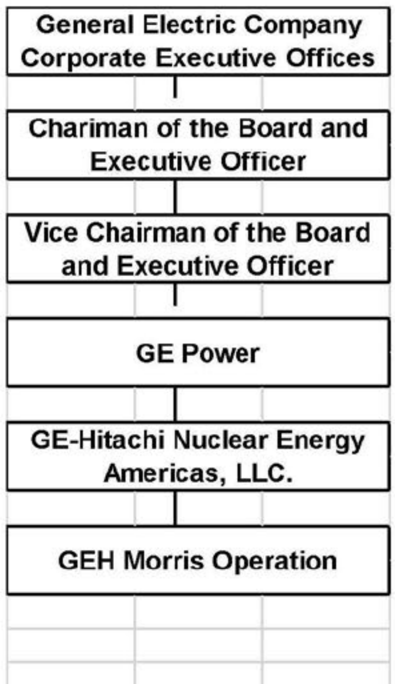

Figure 9-1. GEH Morris Operation relationship to the GE Corporate Offices.

9.2.1 Organization Functions, Responsibilities, and Authorities

Formal policies are established at Corporate, Sector, Operations, Division and Operation levels of GE's organization to ensure safety and quality of products and services and compliance with requirements of government agencies. These policies are applicable to GEH-MO as summarized in the following paragraphs.

9.2.1.1 Company Policies

Formal, Company-level policies are documented in two forms: Company Policy Statements and Company Management Policies. These company policies are a definition of common purposes for organization components of the Company as a whole where it is desirable to foster a uniform course of action.

9.2.1.2 Nuclear Energy Policies

GEH Nuclear Energy (GEHNE) uses a system of documented policy guides and instructions to establish requirements and implement Company policies regarding safety and quality as related to nuclear energy business activities.

9.2.1.3 Operation Policies

Morris Operation (GEH-MO) focuses Company and GEHNE policies to specifically address the Operation's requirements.

9.2.1.4 Irradiation Processing Operation

GEHNE and MO activities are governed by procedures and instructions established in accordance with Company and Operations policy requirements.

9.2.2 GENE Components

Morris Operation is a sub-section of the GEHNE Advanced Programs.

9.2.2.1 Morris Operation

The GEH-MO sub-section is responsible for operation of GEH-MO as an Independent Spent Fuel Storage Installation (ISFSI). This organization and its function are discussed in Section 9.2.3.

9.2.2.2 Regulatory Compliance

GEH-MO Regulatory Compliance is responsible for directing and coordinating activities related to obtaining and support of licenses and permits including developing practices and procedures and compliance verification in accordance with applicable Company and Government requirements.

9.2.3 Morris Operation Organization

The GEH-MO organization (Figure 9-2) is designed to be relatively self-sufficient in ensuring public, personnel, and facility safety. Senior positions and responsibilities within the organization are described in the following paragraphs:

Morris Operation Organization Chart

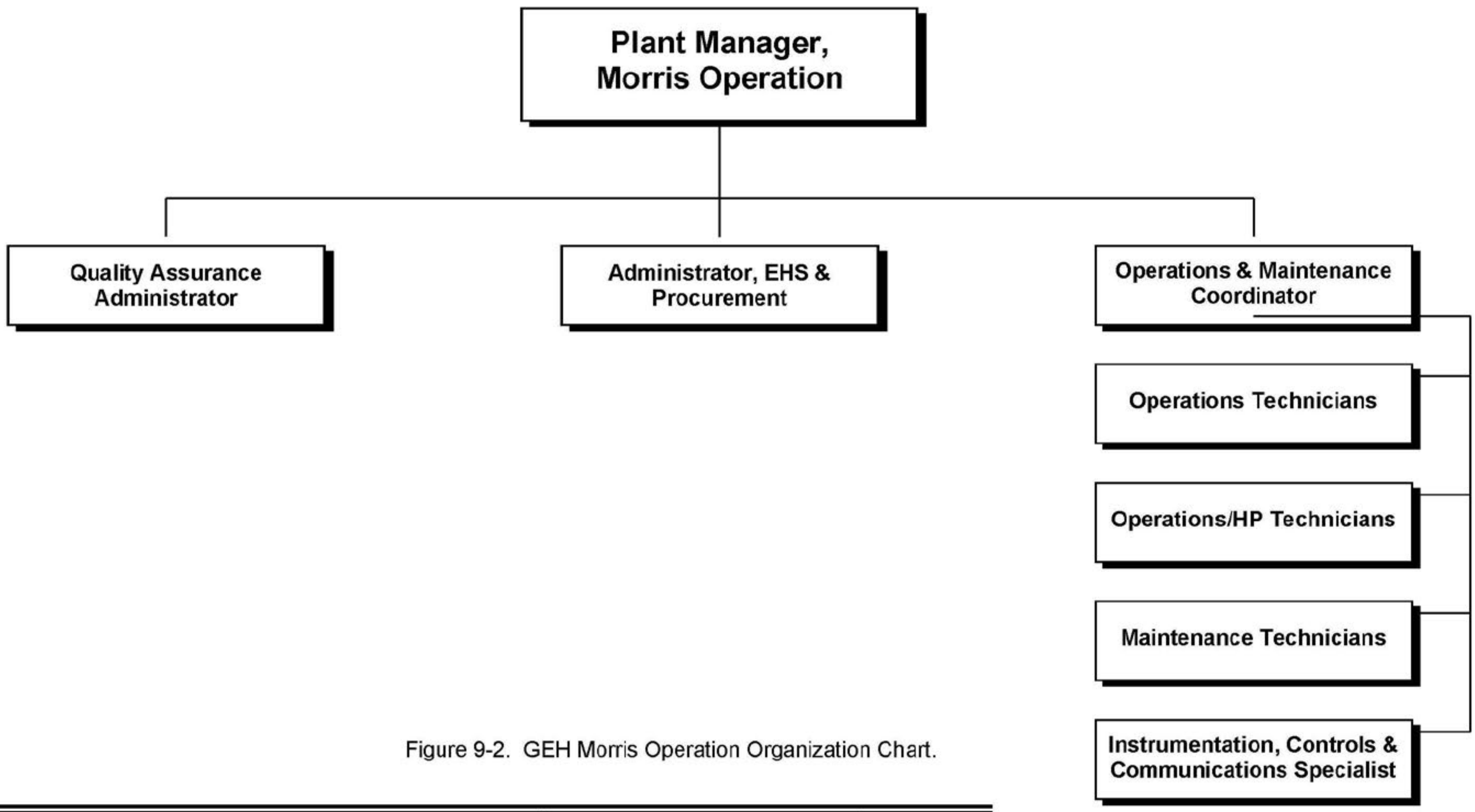

Figure 9-2. GEH Morris Operation Organization Chart.

9.2.3.1 Manager - Morris Operation

The Plant Manager - MO is responsible for safe operation and maintenance of facilities, including compliance with license conditions and applicable Federal, State, and local regulations to ensure protection of health and safety of public and plant personnel. The Plant Manager is also responsible for licensing compliance activities including special nuclear material accountability and plant physical security. In addition, the PM is responsible for providing industrial and radiological safety support, coordinating site regulatory matters with local, State, and Federal regulatory agencies, and directing site environmental activities. The PM reviews facility and operating procedure changes to determine need for nuclear safety review and reviews fuel data to ensure conformance with criteria for fuel storage.

9.2.3.2 Operations and Maintenance Coordinator

The O&MC is responsible to the Manager - MO for maintaining plant facilities and equipment in safe and operable condition and conducting site operations in compliance with established safety and license requirements and operating procedures.

9.2.3.3 QA Manager

The QA Manager is responsible for preparing and maintaining QA Plan, and supporting documents, and implanting all QA/QC operations.

9.2.4 Safety Committee

In addition to the organization shown in Figure 9-2, a facility Safety Committee (SC) is established within GEH-MO. The SC will consist of members as determined by the Manager - Morris Operation and described in a SC operating procedure. Three members must be present to conduct business. Other individuals may participate in SC meetings. The Manager - Morris Operation serves as committee chairperson when items of particular significance are being considered (e.g., in the evaluation of major operational safety matters, and development of recommended changes in facilities or procedures affecting safety margins).

The SC exercises jurisdiction over those matters having radiological or nuclear safety implications, with review and approval authority.

9.3 TRAINING PROGRAMS

To provide and maintain a flexible, well-qualified work force for safe and efficient operation, a comprehensive training program has been implemented. Training includes:

a. Orientation and Indoctrination

b. Radiation and Industrial Safety

c. Security/Safeguards

d. Emergency Response

e. Quality Assurance

f. Basic Plant Facilities and organization

g. Fuel Shipping and Storage Operations

h. Utilities and Operating Systems

The amount of training and retraining each individual receives is directly related to his function. Personnel are provided general orientation that includes description of GEH-MO and its functions, facility safety, security, emergency plans and general procedures.

9.3.1 Operator Qualification, Training, and Certification

Personnel assigned duties involving operation of systems and equipment directly related to cask movement or loading, movement of fuel, operation of basin water cooling or cleanup systems, radioactive waste management operations, and other activities in the cask-handling and fuel storage areas are trained, tested, and certified as qualified to perform specified duties.

9.3.2 Trained and Certified Personnel

GEH-MO maintains an adequate complement of trained and certified personnel to operate the facility.

9.4 NORMAL OPERATIONS

9.4.1 Facility Procedures

Facility procedures are discussed by category in following paragraphs. Systems and equipment requiring certified personnel may be operated by noncertified personnel only if under direct visual direction of an individual trained and certified for the specific operation.

9.4.1.1 Morris Operation Instructions (MOIs)

MOIs are a system of task specific written instructions that provide guidance and direction for performance of GEH-MO activities. The instructions provide for proper safety, quality, and functional considerations in planning and implementation of plant activities, including administration, licensing, engineering and maintenance, materials, operations, quality assurance, safeguards, safety field services and transportation.

9.4.1.2 Standard Operating Procedures (SOPs)

Operation of GEH-MO facilities is directed by a system of SOPs that provide detailed guidance and control for anticipated conditions. Individual procedures are prepared by Operations and Maintenance and approved by the SC before being implemented. Operations personnel are authorized to modify standard procedures on an interim basis as required to cover specific conditions arising during operations. SOPs are modified only after due consideration of safety implications of the change. Operating activities are monitored on a shift-by-shift basis by supervisory staff for compliance with SOPs.

9.4.1.3 Environmental Health and Safety Plan (EHSP)

Control of work involving ionization radiation and radioactive materials is provided by a system of radiation protection and standards developed and documented in the Environmental Health and Safety Plan (EHSP). Deviation from established requirements may be required from time to time either on a planned basis under special operating conditions or by emergencies. Planned deviations must have prior approval. Emergency deviations must be reported promptly to the Operations Technician on duty who, in turn, notifies the Plant Manager and the O&MC.

9.4.1.4 Special Work Permits (SWPs)

Special Work Permits (SWPs) address activities involving nonstandard conditions not addressed by routine implementing procedures. They are prepared for interim use on a controlled basis and are based on specific evaluation of safety implications. Definite time limits are set for SWPs during which off-standard conditions are to be corrected or established requirements revised. SWPs are approved by SC Members.

9.4.1.5 Regulated Work Permits (RWPs)

Regulated Work Permits (RWPs) are essentially time extended SWPs that address safety requirements for mundane facility activities in potentially hazardous areas. The RWP system is designed to ensure that such work is accomplished in accordance with standards and requirements required by the EHSP.

Responsibility for the procedural system is assigned to the O&MC including provisions for shift-by-shift monitoring of activities for compliance with control requirements and maintenance of necessary records of such activities. RWPs are approved by the SC and reviewed annually.

9.4.1.6 Equipment Maintenance Programs

A Work Request (WR) system is employed at GEH-MO for initiating requests for maintenance, repairs, modifications, alterations and new installations. WRs are reviewed by the Operations & Maintenance Coordinator (O&MC) or delegate, and Quality Assurance for conformance to facility procedures and instructions. Equipment maintenance is performed in accordance with manufacturers' recommended practices and operating experience. Overall responsibility for equipment maintenance is assigned to the O&MC. Assistance is provided by other components, as required, to ensure safety and operability criteria are correctly interpreted and performance capability maintained.

9.4.2 Records and Reports

Files of activities relating to safety are maintained to demonstrate adequacy of design safety considerations and to ensure consistent application of safety principles and objectives to plant operation and maintenance.

9.4.2.1 Record Retention

Documented records of facility activities are maintained to demonstrate control requirements have been met, including procedural system documentation and compliance records noted in preceding paragraphs; environmental monitoring program reports; personnel exposure data and regulatory activity files.

9.4.3 Facility Modification

GEH-MO employs a formal design review program in accordance with QA requirements. Minor modifications and tests and experiments may be performed under provisions of Section 9.4.4.

9.4.3.1 Project Design Activity

Design activity includes establishing functional classifications, specifications, drawings, and other documentation - all subject to safety committee review. Independent overview is required for design verification. Design activities are performed in accordance with QA program requirements.

9.4.3.2 Licensing Activity

It is the responsibility of the PM to determine if a facility modification requires a formal safety analysis review. A "Changes, Tests, and Experiments" (10 CFR 72.48) report is written with guidance from the PM and approved by the SC. Licensing action is initiated by the PM. Other GEH-MO personnel may be enlisted to provide licensing activity support.

9.4.3.3 Project Implementation

The Manager, MO, may at his discretion, designate a Project Manager who is assigned responsibility for construction, installation, testing, startup, and related activities. The Manager - MO retains full responsibility for project safety and normal concurrent activities involving operation of the facility during modification.

Responsibility for liaison with regulatory bodies remains with GEH-MO - usually the Manager, MO. Project management personnel coordinate with the safety committee during project execution to achieve stated project and operation goals. Procedures for the new facility or function are developed and implemented as described in Section 9.4.

Upon completion of startup and turnover operations, project documentation is completed and filed, and responsibility for operations of the new facility or function is assumed by GEH-MO.

9.4.3.4 Audits and Reviews

Policies and resulting requirements established for GEH-MO require periodic audit and review of various aspects of fuel storage activities. General topics for audit include:

Design and Maintenance
Nuclear criticality safety
Radiation protection
Physical security
Emergency plan
Environmental protection
Quality Assurance
Facility Operation

Internal audits are conducted by GEH-MO Management. Formal audits and reviews are conducted by other GEHNE components in accordance with established policies and procedures.

9.4.4 Changes, Tests, and Experiments

Facility alterations, personnel changes, and methods and procedures are changed/revised without prior U.S. Nuclear Regulatory Commission (NRC) approval if the SC deems no lessening of safety or security shall occur. This policy is consistent with 10 CFR 72.48 requirements.

Implementation of such changes, tests, and experiments is accomplished as directed by applicable procedures. In general, implementing procedures requires appropriate analysis and evaluation, with concurrence and license amendment activity when appropriate.

9.4.4.1 Unreviewed Safety and Environmental Issues Criteria

Changes in facility or procedures described in this report and tests and experiments (hereafter referred to as "action") are reviewed for safety and environmental issues previously unreviewed by the NRC under the following criteria:

a. Proposed action shall be deemed to involve an unreviewed safety issue if the probability or consequences of an accident or malfunction of equipment important to safety would exceed technical specification limits or other conditions of the facility license, established by regulations, or if a significant possibility of an accident or malfunction of a type different than previously evaluated would be created.

b. Proposed action shall be deemed to involve an unresolved safety issue if the margin of safety defined in any Technical Specification is significantly reduced.

c. Proposed action shall be deemed to involve an unreviewed safety issue if occupational radiation exposure, either individually or collectively, is significantly increased over that experienced in routine operations involving receipt, storage, and transfer of spent fuel.

d. Proposed action shall be deemed to involve an unreviewed environmental issue if the impact of that action would have a significant environmental effect not considered previously.

9.4.4.2 Records and Reports for Changes, Tests and Experiments

The following special records and reports are required regarding changes, tests and experiments:

a. Records of facility changes shall be made and maintained until termination of license, and shall include bases for determining that changes did not involve unreviewed safety and environmental issues. Changes of a long-term or permanent nature will be recorded by issuing revisions to appropriate sections of this report.

b. Records of temporary facility changes, tests and experiments shall be prepared and maintained until termination of license. These records shall include safety evaluations to document bases for determining that subject changes, tests and experiments did not involve unreviewed safety and environmental issues.

c. An annual report of actions under Section 9.4.4 shall be furnished to the NRC regional office and other addresses required by applicable regulations. The annual report shall contain a brief description of changes, tests and experiments and include a summary of the safety and environmental evaluation of each action.

9.5 EMERGENCY PLAN

9.5.1 Purpose and Scope

An emergency plan is established and personnel are trained in emergency procedures so effective actions can be taken under the stress of emergency conditions.

GEH-MO emergency planning is related to overall emergency planning of GEHNE, and to applicable regulatory requirements. Emergency assistance arrangements are established with law enforcement, medical, and other local agencies and services.

9.5.2 Responsibilities

Establishment of an emergency plan is the responsibility of the Manager - MO. Responsibility for preparation of emergency procedures and instructions has been delegated to the O&MC. Assistance and concurrence of engineering and operation components of GEH-MO are required in developing and approving emergency procedures. Independent review for adequacy and effectiveness is included in SC review activities previously described. Implementation of emergency response procedures is the responsibility of the Emergency Coordinator (EC).

Responsibilities for training, equipping, testing and other preparatory activities necessary to ensure maximum effectiveness when an actual emergency occurs are assigned to appropriate line organization positions.

9.5.3 Action Procedures

An emergency is defined as any set of conditions which requires immediate corrective actions beyond those specified in facility procedures and authorized supplementary instructions to protect health and safety of public and plant personnel.

9.5.3.1 Emergency Classification

Classes of emergencies for which specific action procedures are prepared include:

a. Criticality Incidents: Defined as existence of a local neutron multiplication factor greater than 1.0 anywhere in the plant.

b. Contamination Accidents: Defined as unanticipated appearance of significant quantities of radioactive materials beyond prescribed bounds. Radiation monitors and air samplers are provided in areas of potential contamination to provide continuous assessment of conditions. Local and CAS/SAS alarm systems are provided for strategically located monitors in fuel storage areas.

c. Fire: Detection and alarm systems are provided for areas of concern and are supplemented by manual alarm provision and response procedures.

d. Major Equipment Failures or Operational Accidents: Defined as any component failure or malfunction having significant potential for personnel injury or major damage to plant facilities. Detection systems are provided for certain conditions; detection of others will be by direct observation or by indication that operating characteristics have changed. All such incidents are reported immediately to the EC on duty for prompt assessment and initiation of corrective procedures.

e. Other: Specific action plans exist for external conditions having potential to affect GEH-MO safety such as earthquake, windstorm, accidents at adjacent facilities, etc. Where applicable, provisions are made for advance warning of such conditions so actions can be taken to minimize potential effects (e.g., evacuation of vulnerable areas when a tornado is imminent).

9.5.4 Activation of Emergency Organization

The GEH-MO emergency organization is activated by the EC to the extent appropriate to the emergency. Details are documented in NEDO-31955, "Morris Operation Emergency Plan".

9.5.4.1 Communication Methods

Activation of on-site and off-site emergency personnel, organizations, and support functions depends upon normal communication channels. The facility is equipped with telephone and public address systems and the emergency alarm system. These systems are augmented by radio communications established between GEH-MO and selected law enforcement, fire fighting, and other emergency services.

9.5.4.2 Notification of Off-Site Agencies

The EC shall (without prior management approval) request off-site agency response to an emergency situation. This includes fire department, local law enforcement and hospital/ambulance services. Procedures are established to provide direction for obtaining emergency assistance.

Notification to other agencies is made in accordance with assistance agreements, appropriate governmental regulations, and established GE company policies and operating instructions.

9.6 DECOMMISSIONING

During GEH-MO design and construction, specific attention was directed to control and confinement of radioactive materials and to provide features that would facilitate decontamination and decommissioning. A decontamination and decommissioning plan is contained in Appendix A.7.

10.0 OPERATION SPECIFICATIONS

10.1 INTRODUCTION

These technical specifications govern safe possession, storage and transfer of irradiated fuel from light-water reactors at Morris Operation.

10.1.1 DEFINITIONS

The following definitions apply for the purpose of these technical specifications:

a. Administrative Controls: Provisions relating to organization and management procedures, recordkeeping, review and audit, and reporting necessary to assure that operations involving storage of spent fuel at Morris Operation are performed in a safe manner.

b. Design Features: Features of the facility associated with basic design such as construction materials, geometric arrangements, dimensions, etc., which, if altered or modified, could have a significant effect on safety.

c. Functional and Operating Limits: Limits on fuel handling and storage conditions necessary to protect the integrity of the stored fuel; to protect employees against occupational exposures; and to guard against the uncontrolled release of radioactive materials.

d. Fuel Bundle: Unit of nuclear fuel in the form used in the core of a light-water reactor (LWR). Normally, will consist of a rectangular arrangement of fuel rods held together by end fittings, spacers and tie rods. The BWR fuel bundle does not include the reusable fuel channel which is not shipped with the fuel bundles.

e. Limiting Conditions: The lowest functional capabilities or performance levels of equipment required for safe operation of the facility.

f. Surveillance Requirements: Surveillance requirements include:(i) inspection and monitoring of spent fuel in storage; (ii) inspection, test and calibration activities necessary to ensure the integrity of required systems and components and the stored spent fuel is maintained; (iii) confirmation that facility operation is within required functional and operating limits; (iv) a confirmation that limiting conditions required for safe storage are met.

g. <u>Tonne (Te):</u> One metric ton, equivalent to 1000 kg or 2204.6 lb. Fuel quantity is expressed in terms of the fuel heavy metal content, measured in metric tons, and written TeU.

10.2 FUNCTIONAL AND OPERATING LIMITS

10.2.1 AUTHORIZED MATERIALS

10.2.1.1 Specification

a. Light-water reactor nuclear fuel stored at GEH-MO has previously met specific requirements detailed in earlier revisions of this document. Fuel currently in storage has been at GEH-MO since 1989, the basins are essentially full. No new fuel will be received and storage is limited to the current inventory as shown below.

Station	Type	Cladding	Bundle Array	1st Bundle Received	Last Bundle Received	Total Bundles
Connecticut Yankee	PWR	SS	15x15	01-13-72	08-05-87	82
Cooper	BWR	Zircalloy	7x7 & 8x8	08-24-84	01-27-89	1054
Dresden	BWR	Zircalloy	7x7	09-05-75	03-31-77	753
Monticello	BWR	Zircalloy	8x8	11-21-84	04-24-87	1058
San Onofre	PWR	SS	14x14	03-27-72	09-07-80	270

b. Tools and equipment incidental to the conduct of GE Hitachi Nuclear Energy and nuclear related business, which have become radioactively contaminated may be possessed, stored, repaired and decontaminated. The total contamination of all tools and equipment shall not exceed 10 Ci as determined by external exposure from the items as received. Items containing smearable contamination shall be packaged for storage.

c. Tools and equipment specifically related to the conduct of fuel storage operations, such as shipping cask internals, contaminated with radioactive materials may be possessed, repaired and/or decontaminated.

10.2.1.2 Basis

The design criteria and subsequent safety analysis of GEH-MO assumed certain characteristics and limitations for fuels that have been received and are currently stored. Specification 10.2.1.1a assures these bases remain valid by defining the authorized stored fuel inventory.

The design bases for criticality analyses were selected from detailed analytical studies based on the physical parameters of specific fuel designs (See Table A.10-1, Appendix A.10). The largest bundle cross-sectional area and infinite bundle length were assumed in the calculations. These limits were based on unirradiated clean fuel and include allowance for the poisoning effect of the stainless-steel baskets. Fuel centerline locations and other orientations were assumed to be those giving the maximum system reactivity.

Specification 10.2.1.1b provides for storage of tools and equipment incidental to the conduct of GE Hitachi Nuclear Energy businesses while awaiting decontamination, reuse, or ultimate disposal. Activity will be calculated from exposure rate measurements from a package, assuming the radiation originates from a uniform volumetric source having approximately the same dimensions as the package. Unless otherwise determined, gamma emissions of 1 MeV/disintegration will be assumed.

Specification 10.2.1.1c provides for storage of tools and equipment specifically related to the conduct of General Electric fuel storage operations, such as cask internals and yokes while awaiting decontamination, reuse, or ultimate disposal. These tools and equipment may be contaminated with Co-60, Cs-137, or other isotopes as encountered in fuel handling and storage activities.

10.2.2 FUEL STORAGE PROVISIONS

10.2.2.1 Specification

Irradiated fuel bundles shall be stored in authorized fuel storage baskets, mounted in a support grid, under water in a fuel storage basin.

10.2.2.2 Basis

The design criteria and subsequent analysis for GEH-MO assume irradiated fuel is stored under water in fuel storage baskets, mounted in a support grid in a fuel storage basin. Specification 10.2.2.1 assures that these assumptions remain valid. The fuel storage baskets and support grid are those described in Section 5.

10.3 LIMITING CONDITIONS

10.3.1 LIMITING CONDITION – WATER SHIELD

10.3.1.1 Specification

The depth of water between the top of the fuel bundle upper tie plate and the surface of the basin water shall be a minimum of nine (9) feet.

10.3.1.2 Basis

This specification establishes a minimum water shielding depth to limit radiation dose rate in the basin area. This specification applies to all fuel in storage or being transferred from storage to cask (also, see Section 10.5.2).

Tests have shown the water surface dose rate does not increase above background until the water depth is decreased to about 7 feet. A conservative water shield depth of 9 feet has been chosen to provide an increased margin of safety.

10.3.2 LIMITING CONDITION – CRITICALITY

10.3.2.1 Specification

A structure (unloading pit doorway guard: Figure 5-5)[1] shall be used at the doorway between the unloading basin and storage Basin No. 1 to prevent a basket from tipping in a manner such that its contents may be emptied into the unloading basin.

10.3.2.2 Basis

The analysis of a fuel basket drop accident (Section 8) indicates that a basket dropped or tipped over in Basin No. 1, near the doorway to the cask unloading basin, could empty its contents into the unloading basin. It is assumed that the fuel might fall into a critical configuration in the bottom of the unloading basin. The unloading pit doorway guard assures that a basket cannot empty its fuel into the unloading basin.

10.4 SURVEILLANCE REQUIREMENTS

There is no credible event, planned discharge or design basis accident identified at GEH-MO that would expose a member of the public to radiation in excess of the limits specified in 10 CFR 72.104 or 10 CFR 72.106. However, surveillance of various radiation levels, water levels, and other physical quantities, as well as inspections and other periodic activities are contained in this Section to provide assurance that these limits are met. These requirements are summarized in Table 4-1 from details contained in 10.4.1 through 10.4.

Table 4-1 Surveillance Requirements Summary

Section	Quantity or Item	Period	Value
4.1.1	Effluent air	Weekly	ß: 4×10^{-8} µCi/ml
4.2.1	Water-evaporation pond and sanitary lagoons	Monthly	ß: 10^{-5} µCi/ml α: 5×10^{-6} µCi/ml
4.3.1	Sealed sources ß, γ, n, ∝ Sealed sources - α	Semiannual Quarterly	α or ß: 0.005 µCi α: 0.005 µCi
4.4.1	Instruments	(see Table 4-2)	

4.5.1	Basin water	Monthly	Conductivity: <1.35 μMho/cm
4.6.1	Basin water	Monthly	<0.02 μCi/ml

10.4.1 EFFLUENT AIR

10.4.1.1 Specification

Effluent air shall be continuously sampled for particulates at a location between the main stack and the sand filter. Samples shall be analyzed weekly for gross bets (β) activity. The maximum values shall not exceed a weekly average of $4x10^{-8}$ μCi/ml.

10.4.1.2 Basis

This specification requires sampling of ventilation air leaving the sand filter to demonstrate that offsite concentrations do not exceed 10 CFR 20 limits. The GEH-MO sampling and analysis program provides data for estimating the amounts of radioactive material released to the environment during routine or accident conditions.

10.4.2 HOLDING BASINS

10.4.2.1 Specification

Water in the sanitary holding basin and evaporative pond shall be sampled at least once each month, and analyzed for gross alpha and gross beta radiation. The maximum concentrations shall not exceed 10^{-5} μCi/ml beta and $5x10^{-5}$ μCi/ml alpha radiation. If either pond is dry [2] no sampling of that pond is required.

10.4.2.2 Basis

Morris Operation is designed to preclude the release of radioactive materials in normal liquid effluents. As a precautionary measure the sanitary lagoons, which receive and retain plant sewage and some ground water runoff, are periodically sampled to detect inadvertent contamination by radioactive materials.

10.4.3 SEALED SOURCES

10.4.3.1 Specification

Each licensed sealed source (not irradiated fuel) containing radioactive material in excess of

100 µCi of beta-gamma emitting material or 10 µCi of alpha-emitting material shall be tested for leakage at least once every 6 months, except that each source designed for the purpose of emitting alpha particles shall be tested at intervals not to exceed 3 months. The maximum level of removable (non-fixed) contamination shall be less than 0.005 µCi total for each source, using dry-wipe testing techniques.

10.4.3.2 Basis

Surface contamination is measured to determine that a sealed source has not developed a leak. The limitations on removable contamination are based on 10CFR 70.39(c) limits for plutonium, but other provisions of this reference are not applicable.

10.4.4 INSTRUMENTATION

10.4.4.1 Specification

Systems and equipment shall be tested for operability and calibrated at least once during the intervals specified in Table 4-2. Calibration shall be performed in accordance with manufacturer's recommendations, specific GEH-MO approved procedures, and operational tests shall be performed to check alarm functions and demonstrate other operational features of the system or equipment

Table 4-2 Summary Requirements System and Equipment Test Calibration

System or Equipment	Operability Test	Calibration
Basin Leak Detection System	Weekly	Monthly
Area Radiation Monitors	Quarterly	Quarterly
Criticality Monitors	Annual	Quarterly

10.4.4.2 Basis

Bases for these test and calibration requirements are as follows:

a. Basin Leak Detection System: Operation of this system ensures that a leak in the basin liner will be promptly detected so that corrective action can be initiated. Since the operation of the system is related to the level of water in the detection system, the level alarm set point is checked and instruments receive periodic calibration.

a. Area Radiation Monitors: The audible alarm system for these monitors is tested (operated), and the alarm set point calibrated periodically to provide assurance of reliable operation within equipment specifications, to avert personnel to radiation above preset levels.

b. Criticality Monitors: The audible alarm systems for these monitors, which warn personnel of a criticality, are tested (operated) and the alarm set point calibrated periodically to provide assurance of reliable operation within equipment specifications.

10.4.5 BASIN WATER CHEMICAL CHARACTERISTICS

10.4.5.1 Specification

Basin water chemistry shall be maintained as follows:

Item	Acceptable Analysis
Conductivity	less than 1.35 μMho/cm (equivalent to pH of 5.5 to 8.0 in demineralized water)

10.4.5.2 Basis

Basin water chemical characteristics are selected to maintain a benign environment for fuel and equipment stored in the basin water.

10.4.6 BASIN WATER RADIOACTIVE CONTAMINANTS

10.4.6.1 Specification

Additional basin water cleanup measures shall be initiated if the concentration of radioactive materials in the water exceeds 0.02 μCi/ml beta.

10.4.6.2 Basis

Periodic sampling of basin water is required to assure that concentration of radioactive materials remain as low as reasonably achievable. The values selected are consistent with current decontamination practices.

10.5 DESIGN FEATURES

10.5.1 FUEL STORAGE BASIN

The energy-absorbing pad on the cask set-off shelf shall not be altered without appropriate safety review and documentation as required by 10 CFR 72.48.

10.5.1.1 Basis

The cask drop accident was analyzed for the IF-300 cask with the energy-absorbing pad in place (Section 8).

10.5.2 FUEL STORAGE SYSTEM

The following pieces of equipment employ favorable geometry, specific materials, and methods of construction to assure nuclear criticality safety and radiation protection and are considered important to safety. Modifications to the design in dimensions, construction materials or construction methods shall not be made without appropriate safety review and documentation in accordance with 10 CFR 72.48.

a. Fuel Storage Basin - concrete walls, floors, and expansion gate are principal elements in protection of stored fuel, and in isolation of basin water from the environment.

b. Fuel Storage Basin - stainless steel liner forms a second element in fuel protection and basin water isolation, facilitating decontamination.

c. Fuel Storage System, including baskets and supporting grids is a principal element in protection of stored fuel.

d. Unloading Pit Doorway Guard - is designed to prevent a loaded fuel basket from being tipped so that fuel bundles could fall into the cask unloading pit. The unloading pit doorway guard is an element in protection of fuel during movement of a loaded basket.

e. Filter Cell Structure - the concrete cell part of the basin pump room area provides radiation shielding to reduce occupational exposure.

f. Fuel Storage Basin building – the steel structure that surrounds/protects the fuel Basins.

g. Fuel Basket Grapple – Used to remove the fuel baskets from their storage location in the fuel basin support grid.

h. Fuel Grapple – Used to remove the fuel bundles from the fuel baskets when they are in the unloading pit.

i. Fuel Basin Crane – Crane utilized to move the full fuel baskets to the unloading pit.

j. Fuel Handling Crane – Crane used to remove the fuel bundles from the fuel storage baskets and place into a cask.

k. Cask Crane – 125 Ton overhead crane used to lift a fully loaded cask from the unloading pit and place cask onto transport vehicle.

l. Spent Fuel Cladding – Fuel in Morris Operation basins are clad with SS or zircalloy.

10.6 ADMINISTRATIVE CONTROLS

10.6.1 RESPONSIBILITY

The Manager, Morris Operation shall be responsible for overall facility operation in accordance with these specifications and applicable government regulations and shall delegate in writing the succession of this responsibility during his absence. Operations involving licensed materials shall be performed by, or under the supervision of individuals designated by the Manager, Morris Operation, or his delegate.

10.6.2 ORGANIZATION

10.6.2.1 The facility staff organization is shown in the CSAR, Figure 9-2 and senior positions and responsibilities within the organization are described in CSAR 9.2.3.

10.6.3 PLANS AND PROCEDURES

Plans and procedures shall be established and implemented to assure compliance with these technical specifications and applicable governmental regulations.

10.6.3.1 Changes to Plans and Procedures

All changes or revisions of established plans or procedures required by this section shall be made in accordance with the GEH-MO modification control practices described in Section 9.

10.6.3.2 Plans and Procedures – Minimum Requirements

Plans and procedures required by this section shall include:

a. A safety manual defining responsibilities and specifying actions to protect the health and safety of employees and others while onsite, appropriate safety training programs, and other measures to maintain exposures as low as reasonably achievable.

b. Requirements for analysis of cask drop accident consequences prior to handling spent nuclear fuel shipping casks not previously handled at GEH-MO per 10 CFR 72.48.

c. Procedures for the conduct of routine fuel storage operations.

d. A Preventive maintenance system for structures, systems and components important to site radiological and criticality safety.

e. Arrangements for providing makeup water to the storage basins under normal and emergency conditions.

10.6.4 REVIEW AND AUDIT

10.6.4.1 Safety Committee

Plans, procedures and operations carried out under established plans and procedures involving elements of radiological safety shall be reviewed and approved by a Safety Committee. Three members must be present to conduct business. Other individuals may participate in Safety Committee meetings. This committee will consist of five members as determined by the Manager, Morris Operation and described in Safety Committee operating procedure and CSAR Section 9.0, Figure 9-2.

The Safety Committee shall normally meet on a monthly basis, but at no greater than 45-day intervals. The Manager, Morris Operation shall establish appropriate procedures and practices for the conduct of Safety Committee responsibilities.

10.6.4.2 Audits

Morris Operation activities shall be audited to ascertain the degree of compliance with specifications, standards and procedures. Audits shall be conducted by organizations and persons, at such times as designated by GE Hitachi Nuclear Energy Management. Audits and audit response shall be performed in accordance with General Electric procedures.

10.6.5 ACTION REQUIRED FOR SPECIFICATION NONCOMPLIANCE

10.6.5.1 Functional and Operating Limits

The following actions shall be taken if a functional or operating limit is exceeded:

a. Prompt action shall be taken to assure timely return of operations to specification compliance.

b. The Safety Committee shall be promptly notified of the noncompliance.

c. NRC Operations Center shall be notified as specified in 10 CFR 72.75, advising them of events that resulted in a noncompliance condition.

d. A review of the incident shall be made by the Safety Committee to establish the cause and to define means to prevent reoccurrence in accordance with the GEH-MO Reporting of Defects and Non-compliances program as documented in NEDE-31559 and implementing procedures.

10.6.5.2 Limiting Conditions

The following actions shall be taken if a limiting condition is exceeded:

a. Prompt corrective action shall be taken to assure timely return of operations to specification compliance.

b. The Safety Committee shall be advised of the noncompliance within 24 hours.

c. A report shall be sent to the NRC Operations Center as specified in 10 CFR 72.75 to advise them of events resulting in limiting conditions being exceeded.

d. A review of the incident shall be made by the Safety Committee to establish the cause and to define means to prevent reoccurrence in accordance with the GEH-MO Reporting of Defects and Non-compliances program as documented in NEDE-31559 and implementing procedures.

10.6.5.3 Surveillance Requirements

The following actions shall be taken if surveillance requirements are not satisfied:

a. The Manager, Morris Operation, or his delegate, shall take such action as may be required to assure future compliance with surveillance requirements and, if necessary, to assure return of operations to specification compliance in minimum time.

b. The Safety Committee shall be advised of any event, or sequence of events, involving surveillance requirements that involve systems directly related to radiological safety. The Committee shall investigate such events and recommend corrective action.

10.6.5.4 Design Features

Design features shall only be changed in accordance with specification 6.3.1, Chapter 9, and 10 CFR 72.48. Unauthorized modifications of specified design features, or unauthorized introduction of unapproved tools, fixtures or other equipment shall require action as specified for functional and operating conditions in Specification 6.5.1.[3]

10.6.6 LOGS, RECORDS AND REPORTS

10.6.6.1 Logs and Records

a. A shift log shall be maintained to record nonroutine and significant events that may occur during a shift.

b. Minutes of the Safety Committee shall be documented, including copies of reports required in Section 6.5.1, and other actions of the Safety Committee.

c. Records of facility changes, and changes in procedures described in the CSAR shall be maintained throughout the lifetime of the facility.

d. Records of tests or experiments conducted under provisions of Section 9 and 10 CFR 72.48 shall be maintained throughout the facility lifetime and shall include written safety evaluations that provide the bases for determining if the test or experiment did not involve unreviewed safety or environmental questions.

10.7 REFERENCES AND NOTES

1. The use of the unloading pit doorway guard is described in CSAR Chapters 1 and 5.

2. Dry to the extent that water samples cannot be obtained in the usual manner.

3. Authorized modifications and approved tools, fixtures, or other equipment are those processed under the provisions of CSAR Section 9.

10.8 ENVIRONMENTAL MONITORING PROGRAM

10.8.1 Specification

The licensee will maintain the effectiveness of the environmental monitoring program detailed in specific GEH-MO Compliance and Operability Test procedures and in CSAR Section 7.7.1.

10.8.2 Basis

The environmental monitoring program results from over 20 years of Morris Operation environmental monitoring experience. These years of operational experience with the monitoring program provide a sound basis for evaluating the programs effectiveness.

10.9 ANNUAL ENVIRONMENTAL REPORT

10.9.1 Specification

An annual report will be submitted to the NRC IAW 10 CFR 72.44(d)(3), within 60 days after January 1 of each year, specifying the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents during the previous 12 months of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent release and direct radiation at the site property protection area.

10.9.2 Basis

The report of Specification 7.3.1 is required pursuant to 10 CFR 72.44(d)(3).

11.0 QUALITY ASSURANCE

11.1 INTRODUCTION

Activities at Morris Operation (GEH-MO) are conducted in accordance with a quality assurance plan reviewed and accepted by the USNRC and implemented by instructions and procedures at GE-MO.

11.2 QUALITY ASSURANCE (QA) HISTORY

QA program requirements during initial design and construction of GEH-MO, as a fuel reprocessing plant, were developed by GE. During construction, the USAEC -- then the regulatory agency -- increased emphasis on specific methods of achieving quality assurance, proposing amendment of 10 CFR 50 to include Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

Prior to promulgation of Appendix B, GE had incorporated quality assurance provisions into the over-all safety assurance program for the reprocessing plant. Except for specific requirements related to documented record accumulation, key elements required by the proposed amendment (as applicable to fuel reprocessing facilities) had been included in the GE program. The program was documented in Supplement 3 to the "Design and Analysis Report - Midwest Fuel Recovery Plant." Construction of the facility was completed under this program.

GE curtailed operation of the facility in late 1974. At that time, GE proposed installation of a new fuel storage system. This system was licensed by the USNRC in December 1975. Design, fabrication and installation were performed under the current quality assurance plan, in accordance with applicable requirements of 10 CFR 72 Subpart G.

11.3 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

No credible event, planned discharge, or design basis accident at GEH-MO is identified that would expose a member of the public to radiation in excess of limits specified in 10 CFR 72.104 or 10 CFR 72.106.

It is, therefore, the position of GEH-MO that the term "basic components" in the sense defined by 10 CFR 21.3(1)(i)(c) and 10 CFR 21.3 (2) is not applicable to GEH-MO.

However, "structures, systems and components important to safety" as promulgated in 10 CFR 72.122, "Overall Requirements" are identified below.

a. Fuel storage basin (FSB) - concrete walls, floors, and expansion gate are principal elements in protection of stored fuel, and in isolation of basin water from the environment.

b. Fuel storage basin - stainless steel liner forms a second element in fuel protection and basin water isolation, facilitating decontamination.

c. Fuel storage system, including baskets and supporting grids is a principal element in protection of stored fuel.

d. Steel expansion gate – Identified as Gate #4, along the south east corner of fuel basin II. The gate is constructed of reinforced concrete with a thickness of 8" and height and width dimensions of 29'-6" and 5'-0", respectively. The water side of the gate is lined with 16-gauge stainless steel to prevent the reinforced concrete from coming into contact with the water in the basin.

e. Unloading pit doorway guard - is designed to prevent a loaded fuel basket from being tipped so that fuel bundles could fall into the cask unloading pit. The unloading pit doorway guard is an element in protection of fuel during movement of a loaded basket.

f. Filter cell structure (FCS) - the concrete cell part of the basin pump room area provides radiation shielding to reduce occupational exposure.

g. Fuel Storage Basin building – the steel structure that surrounds/protects the fuel Basins.

h. Fuel Basket Grapple – Used to remove the fuel baskets from their storage location in the fuel basin support grid.

i. Fuel Grapple – Used to remove the fuel bundles from the fuel baskets when they are in the unloading pit.

j. Fuel Basin Crane – Crane utilized to move the full fuel baskets to the unloading pit.

k. Fuel Handling Crane – Crane used to remove the fuel bundles from the fuel storage baskets and place into a cask.

l. Cask Crane – 125 Ton overhead crane used to lift a fully loaded cask from the unloading pit and place cask onto transport vehicle.

m. Spent Fuel Cladding – Fuel in GEH-MO basins are clad with SS or zircalloy cladding.

APPENDIX A INDEX

A.1 LICENSING ACTION HISTORY
MIDWEST FUEL REPROCESSING PLANT AND MORRIS OPERATION

Action	License No.	Date
Provisional Construction Permit for the MFRP 12/28/67 to 7/l/70	CPCSF-3	Dec. 28, 1967
State of Illinois, Sanitary Water Board, Industrial Wastewater Containment & Discharge Permit (Evaporation Pond)	1968-EA-626	Sept. 23, 1968
State of Illinois, Sanitary Water Board, Sanitary Sewage Treatment Facilities, Waste Stabilization Lagoons and Chlorination Permit	1968-EA-627	Sept. 23, 1968
Order Extending Provisional Construction Permit Completion Date from 7/1//70 to 7/1/71	CPCSF-3	June 10, 1970
Order Extending Provisional Construction Permit Completion Date from 7/1/71 to 4/1/72	CPCSF-3	June 17, 1971
Registration Radiation Installation, State of Illinois Dept. of Public Health		Aug. 6, 1971
USAEC, Source Material License	SNM-1281	Dec. 27, 1971
USAEC, Special Nuclear Materials License	SNM-1265	Dec. 27, 1971
Order Extending Provisional Construction Permit Completion Date 4/1/72 to 4/1/73	CPCSF-3	March 28, 1972
State of Illinois, Environmental Protection Agency, Div. of Water Pollution Control, Evaporation Pond Permit	1973-EA-53-OP	Jan. 4, 1973
State of Illinois, Environmental Protection Agency, Div. of Water Pollution Control, (Process Sewer System)	1973-EA-248-OP	Feb. 13, 1973
USAEC, Materials License Removal, Expiration Date 3/31/74	SNM-1265	March 9, 1973
Order Extending Provisional Construction Permit Completion	CPCSF-3	March 30, 1973 to April 1, 1974
State of Illinois, Environmental Protection Agency, Div. of Air Pollution Control	063-806-AAC NEDM-21845	June 12, 1973 to April 18, 1981
State of Illinois, Pollution Control Board, NO_x Variance	PCB73-512	March 7, 1974
Illinois Environmental Protection Agency, Water Pollution Control Permit, Evaporation Pond Permit	1974-EA-665-OP	April 18, 1974 to April 18, 1979
USNRC, Materials License Revision & Reissued, Expiration date 8/31/79	SNM-1265	Aug. 23, 1974 to Aug. 31, 1979
Facility License for Possession Only (Terminates CPCSF-3), Expiration Date 5/22/75	CSF-2	Aug. 23, 1974
Order Authorizing Dismantling of Facility to Render Inoperable	CSF-2	Nov. 21, 1974
Order Terminating Facility License for Possession Only	CSF-2	Nov. 26, 1974
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Material License	IL-00329-01	July 28, 1975 to August 31, 1979

Action	License No.	Date
USNRC, Materials License Revised & Reissued for Increased Capacity of Facility Expiration date 8/31/79	SNM-1265	Dec. 3, 1975
USNRC, License Reissued, Expiration date 8/31/79	SNM-1281	Dec. 3, 1975
U.S. Environmental Protection Agency, Region V, National Pollutant Discharge, Elimination System (NPDES) Permit	IL-000-2887	June 2, 1976 (Terminated by request 1/31/77)
Illinois Environmental Protection Agency Effluent Irrigation System Permit	1976-EB-408-1	Sept. 17, 1976
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Materials License to August 31, 1977	IL-00329-01	July 26, 1976
Illinois Environmental Protection Agency, Effluent Irrigation System Permit to Construct, Own and Operate	1976-EB-408-1 (supersedes 1976-EB-408)	Sept. 17, 1976
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Material License, to August 31, 1984	IL-00329-01	Aug. 14, 1980
USNRC, Consolidated Safety Analysis accepted	SNM-1265	Apr. 29, 1977
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Materials License to August 31, 1978	IL-00329-01	Aug. 29, 1977
Illinois Environmental Protection Agency, Air Pollution Control Permit Gaseous Effluent to April 18, 1981	063-806-AAE	April 26, 1978
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Material License to August 31, 1984	IL-00329-01	Aug. 14, 1980
USNRC, Authorizes LaCrosse fuel	SNM-1265	Nov. 17, 1978
USNRC, Application for renewal	SNM-1265	Feb. 27, 1979
USNRC, Terminated SNM-1281 by combining with SNM-1265, and recognizes Operation Specifications	SNM-1265	Apr. 10, 1979
Illinois Environmental Protection Agency Water Pollution Control Permit, Evaporation Pond to May 1984	1979-EO-4660 (supersedes 1974-EA-665-OP)	May 11, 1979
Illinois Dept. of Public Health, Division of Radiological Health, Radioactive Materials License to August 31, 1980	IL-000329-01	Aug. 10, 1979
Illinois Environmental Protection, Agency Air Pollution Control Permit, Gaseous Effluent to April 17, 1986 to April 17, 1986	063-806-AAE	Jan. 14, 1981
USNRC, Renewal of SNM-1265 under Part 72 (see Application for Renewal, February 27, 1979) to May 31, 2002	SNM-2500	May 4, 1982
Illinois Department of Nuclear Safety, Division of Nuclear Safety Radioactive Materials License. Replaces IL 00329-01	IL-01427	August 9, 1989

Action	License No.	Date
U.S. Environmental Protection Agency, Region V, National Pollutant Discharge, Elimination System (NPDES) Permit IL0002887 terminated.	IL-000-2887	March 31, 1997
Illinois Department of Nuclear Safety, Division of Nuclear Safety Radioactive Materials License. Terminates IL-01427 by removing all IDNS licensed material from site.	IL-01427	December 24, 2001
USNRC, Renewal of SNM-2500 under Part 72 (see Application for Renewal, May 22, 2000) to May 31, 2022	SNM-2500	December 21, 2004

A.2 REFERENCE PUBLICATIONS

The following publications and documents have been previously submitted to USAEC and USNRC in licensing actions as noted below.

Docket Information[a]	Date[b]	Title or Subject	Docket
(GE document - no number)	11/66	Design and Analysis - Midwest Fuel Recovery Plant	50-268
NEDO-14503	4/71	MFRP Technical Specifications (Proposed)	50-268
NEDO-14504	6/71	Applicants Environmental Report - Midwest Fuel Recovery Plant	50-268
NEDO-14506	7/71	Midwest Fuel Recovery Plant Emergency Plan	50-268
NEDO-10178	12/70	Final Safety Analysis Report - Midwest Fuel Recovery Plant	50-268
NEDO-10178-1 through NEDO-10178-17	7/71	Response to USAEC questions and Amendments through 36	50-268
Letter: B. F. Judson (GE) to H. J. Larson, Director, Div. of Materials and Fuel Cycle Facility Licensing, USNRC	4/74	License SNM-1265, Docket 70-1308 Request to Increase Storage Capacity w/Preliminary Safety Evaluation Report	70-1308
NEDO-20825	3/75	Safety Evaluation Report for Morris Operation Fuel Storage Expansion	70-1308
P&RS 74766[1]	4/75	Fuel Storage System Design Report	70-1308
P&RS 74766[1] Supplement 1	5/75	Supplement 1 to Fuel Storage System Design Report	70-1308
(None)[2]	5/75	Criticality Safety Basis for the MFRP Project-1 Fuel Storage Baskets	70-1308
NEDO-20776	1/75	Fuel Recovery Operation Quality Assurance Plan	70-1308
NEDO-20969	8/75	Operating Experience - Irradiated Fuel Storage - GE Morris Operation	70-1308
NEDO-20825-1	9/75	Response to NRC Staff Questions	70-1308
NEDO-21326-1 NEDO-21326-2	1/77	Consolidated Safety Analysis Report (basic issue)[3]	70-1308
NEDO-21326-1A NEDO-21326-2A	4/77	Incorporate changes and correction re USNRC REVIEW	70-1308
NEDO-21326-2A1	4/77	Incorporate Proposed Operating Specifications	70-1308
NEDO-21326-1A2 NEDO-21326-2A2	8/77	Incorporate new geological information and appendices	70-1308

NEDO-21326-1a3 NEDO-213262a3	2/78	Minor changes, corrections	70-1308
NEDO-21326-2a4	1/79	Incorporate Operating Specifications	70-1308
NEDO-21326-c	1/79	General revision - license renewal application	70-1308

[a] General Electric publication number unless noted otherwise

[b] Month/year

REFERENCES

1. Programmed & Remote Systems Corp., St. Paul, Minn. (job Number)

2. Battelle Pacific Northwest Laboratories, Richland, Wash.

3. B-series revisions not related to existing plant.

A.3 ESTIMATION OF GROUND-LEVEL RADIATION DOSE RATES FOR STACK EMISSION OF RADIOACTIVE MATERIALS[1] (CONTINUOUS RELEASES)

ABSTRACT

A method of estimating ground-level radiation dose rates corresponding to given stack emission rates of radioactive materials is described. The method considers external dose from both beta and gamma sources, internal dose from inhalation of ground-level concentrations of the material and possible ingestion of agricultural products.

The method relates emission rate (in curies/sec, Ci/sec) to an average annual dose and is suited for application to standard tabulations of meteorological data on wind speed and wind direction frequencies.

The method assumes that the normal Gaussian diffusion equations describe the dispersion of the plume. Situations where topographic or nearby manmade structures could cause significant downwash of the plume are not considered. Special calculations should be used for such situations.

1.0 INTRODUCTION

Continuous emission of radioactive airborne material, as from a stack, is commonly controlled on the basis of not exceeding a stipulated annual average dose to any person in the plant environs. Operationally, control is on the emission rate. Therefore, it is of interest to know what factors apply to convert emission rate (usually in Ci/sec) to annual dose (usually in mRem). Following is a method calculating this relationship.

2.0 METEOROLOGICAL FACTORS

The most significant meteorological factor in determining annual average dose is how often the plume is transported in any given direction; i.e., the wind direction frequency, or wind rose. Long-term average wind data (climatology) are usually tabulated in terms of direction sectors rather than point-by-point directions. That is, the sixteen (sometimes eight) standard compass directions encompassing an angular direction of 22-1/2 degrees each are used. This method of calculating dose rate from a continuous stack emission (plume) is suited to application of this normal climatological summary of wind frequency.

The air concentration per unit amount released at any point (x, y, z) in the cloud at any instant is given by Watson and Gamertsfelder[2] as

$$(X) = \left(\frac{Q_0}{2\pi\sigma_y\sigma_z\overline{\mu}_h}\right)\exp\left(-\left(\frac{y^2}{2\sigma_y^2}\right)-\left(\frac{z^2}{2\sigma_z^2}\right)\right)\left(\frac{Q}{Q_0}\right) \qquad \text{(C-1)}$$

where:

(X) = Average air concentration (Ci/m^3 or µCi/cc);

Q_0 = Release rate (Ci/sec);

$\overline{\mu}_h$ = Average wind speed at height of emission (m/sec);

σ = Standard deviation of cloud width in horizontal y-direction and vertical z-direction (m);

t = Time after release (seconds) and is equal to the downwind distance divided by the average wind speed $\left(x/\left(\overline{\mu}_h\right)\right)$;

(Q/Q_0) = Correction for cloud depletion due to deposition and is the fraction of initial amount released which is present at downwind distance X (= $\left(\overline{\mu}_h\right)(t)$)

$$= \exp\left[-\left(\frac{V_d}{\overline{\mu}_h}\right)\left(\sqrt{\frac{2}{\pi}}\right)\left(\frac{\overline{\mu}_0}{\overline{\mu}_h}\right)\int_0^t \overline{\mu}_h\left(\frac{\exp\left(-\frac{z^2}{\left(2\sigma_z^2\right)}\right)dt}{\sigma_z}\right)\right]$$

Vd = Deposition "velocity" (m/sec) (see Table C-1 for values of this parameter);

$\overline{\mu}_0$ = Average wind speed at ground level (m/sec); and

exp [ ] = Function which is a power of "e".

This equation does not take into account the depletion of the radioactive content of the cloud by radioactive decay of the isotope of concern. With this taken into account, the equation becomes:

$$(X)=\left(\frac{Q_0}{(2\pi)(\sigma_y)(\sigma_z)(\overline{\mu}_h)}\right)\exp\left[-\left(\frac{y^2}{2\sigma_y^2}\right)-\left(\frac{z^2}{2\sigma_z^2}\right)\right]\left[\frac{Q}{Q_0}\right]\exp[-\lambda t] \qquad \text{(C-2)}$$

where:

$\exp(-\lambda t)$ = Radioactive decay function.

Equation (C-2) describes air concentrations at locations sufficiently close to the point of elevated release that the plume has not reached ground level. Where air concentrations at ground level are of interest, this equation requires modifications of some kind. Specific instances of appropriate modifications for different varieties of dose are discussed later.

It is considered a reasonable approximation to assume that throughout the year all the plumes which travel anywhere within a given sector direction do not have a skewed frequency distribution within the sector. Then, the average cloud concentration in the sector is found by integrating Equation (C-2) in the cross-wind direction and dividing by the sector width:

$$(x)_{avg}=\left(\frac{\int_{-\infty}^{\infty}(x)dy}{\theta x}\right) \qquad \text{(C-3)}$$

where:

θx = Sector width.

Equation (C-3) cannot be integrated since the interrelationship between the variables σ_y, σ_z, and $\overline{\mu}_h$ with respect to their average values is not generally known. However, for <u>any specific combination</u> of wind speed and stability, at a given downwind distance all these variables are known and can be treated as constants. The integration can then be performed. Thus, the average concentration in the sector for all occurrences of <u>any specific condition</u> is given by:

$$(x)^i_{avg}=\left(\frac{Q_0}{(\sqrt{2\pi})(\theta x)(\sigma_z)(\overline{\mu}_h)}\right)\exp\left[-\left(\frac{z^2}{2\sigma_z^2}\right)\right]\left[\frac{Q}{Q_0}\right]\exp[-\lambda t] \qquad \text{(C-4)}$$

where:

$(x)^i_{avg}$ = Average concentration for the ith condition;

θ = Angle of sector = $\pi/8$ radians for 1/16 sector or 11-1/2°; and

x = Downwind distance and is $\overline{\mu}_h$.

Thus, the average cloud is seen to have a uniform concentration cross-wind or horizontally and a concentration distribution vertically which is of the Gaussian form. The standard deviation in the vertical direction is as described by Watson and Gamertsfelder:[2]

$$\sigma_z^2 = a\left[1-\exp\left(-k^2t^2\right)\right]+bt \qquad \text{(stable case)} \qquad \text{(C-5)}$$

$$\sigma_z^2 = \left(\frac{\left(C_z^2\right)\left(x^{(2-n)}\right)}{2}\right) \qquad \text{(neutral and unstable case)} \quad \text{(C-6)}$$

where:

a, b, k^2 = Diffusion constants; and

C_z = Suttons's vertical diffusion coefficient.

For values of the above constants, see Table C-2 and Figures C-1, C-2, C-3, and C-4.

Table C-1
DEPOSITION VELOCITIES

$\left(\frac{V_d}{\overline{\mu}_0}\right)$ (a)

Condition	Particulates	Halogens
Very Stable	1.5×10^{-4}	2.4×10^{-3}
Moderately Stable	2.2×10^{-4}	3.4×10^{-3}
Neutral	3×10^{-4}	4.6×10^{-3}
Unstable	6×10^{-4}	8×10^{-3}

a Ratio of deposition "velocity" to wind speed -- multiply by ground wind speed ($\overline{\mu}_0$) to obtain deposition "velocity."

Table C-2
DIFFUSION COEFFICIENTS

Constants	**Very Stable**	**Moderately Stable**	**Neutral**	**Unstable**
$a(m^2)$	34	97	--	--
$b(m^2/sec)$	0.025	0.33	--	--
$K^2(sec^{-2})$	8.8×10^{-4}	2.5×10^{-4}	--	--
β	0.016	0.016	--	--
m	1.6	1.6	--	--
Cz ($\bar{\mu}$ = 1 m/s)	--	--	0.15	0.30
Cz ($\bar{\mu}$ = 5 m/s)	--	--	0.12	0.26
Cz ($\bar{\mu}$ = 10 m/s)	--	--	0.11	0.24
n	--	--	0.25	0.20

The degree of atmospheric stability is defined here in terms of the standard dry adiabatic temperature lapse rate of -1 °C per 100-meter increase in elevation (-5.4 °F per 1000 feet). This is taken as a convenient reference point for defining the four classes of stability:

very stable	≥ + 1.5 °C
moderately stable	≥ - 0.5 °C but < + 1.5 °C
neutral	≥ - 1.5 °C but < - 0.5 °C
unstable	< - 1.5 °C

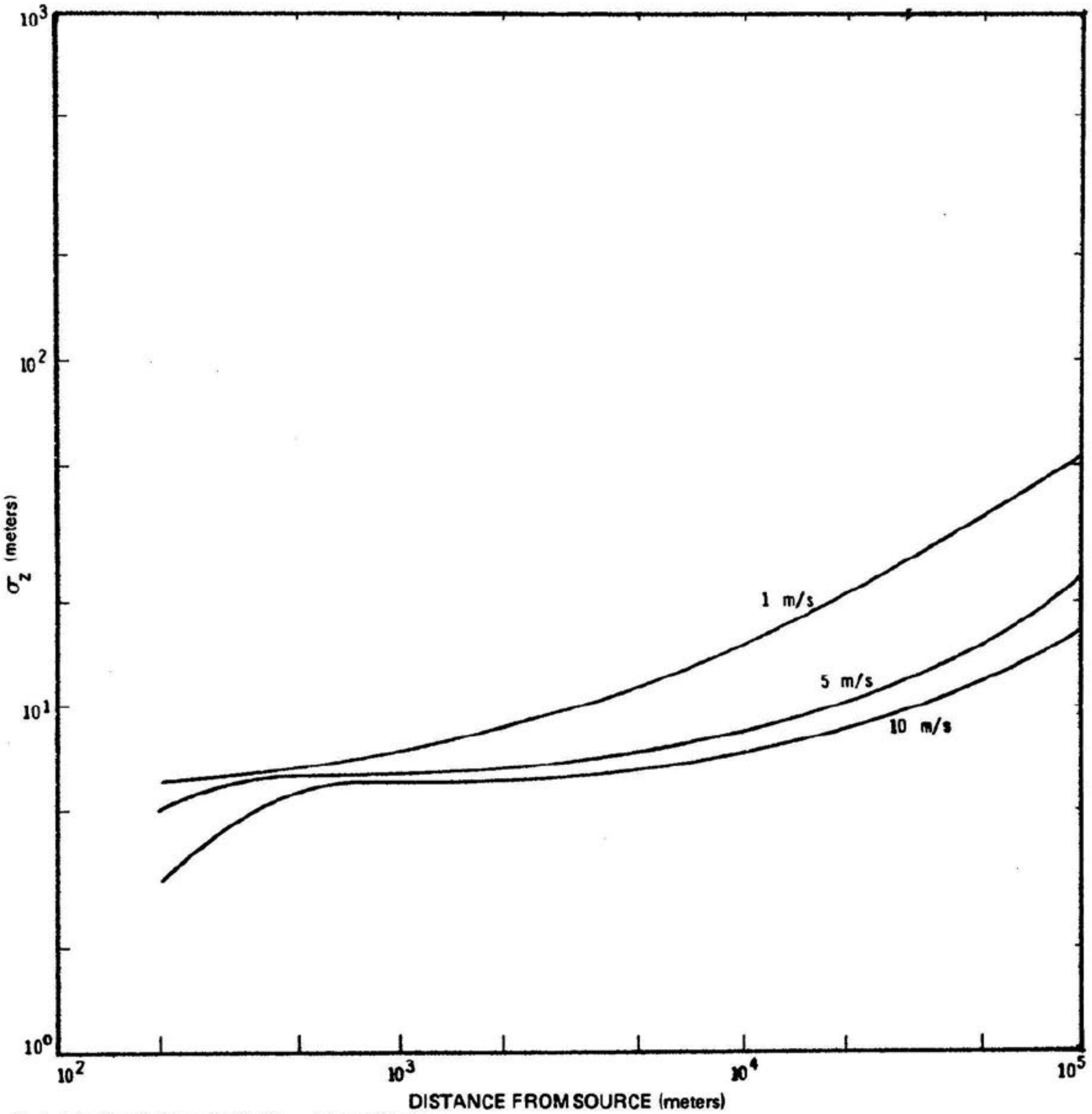

Figure C-1 Vertical Cloud Width – Very Stable

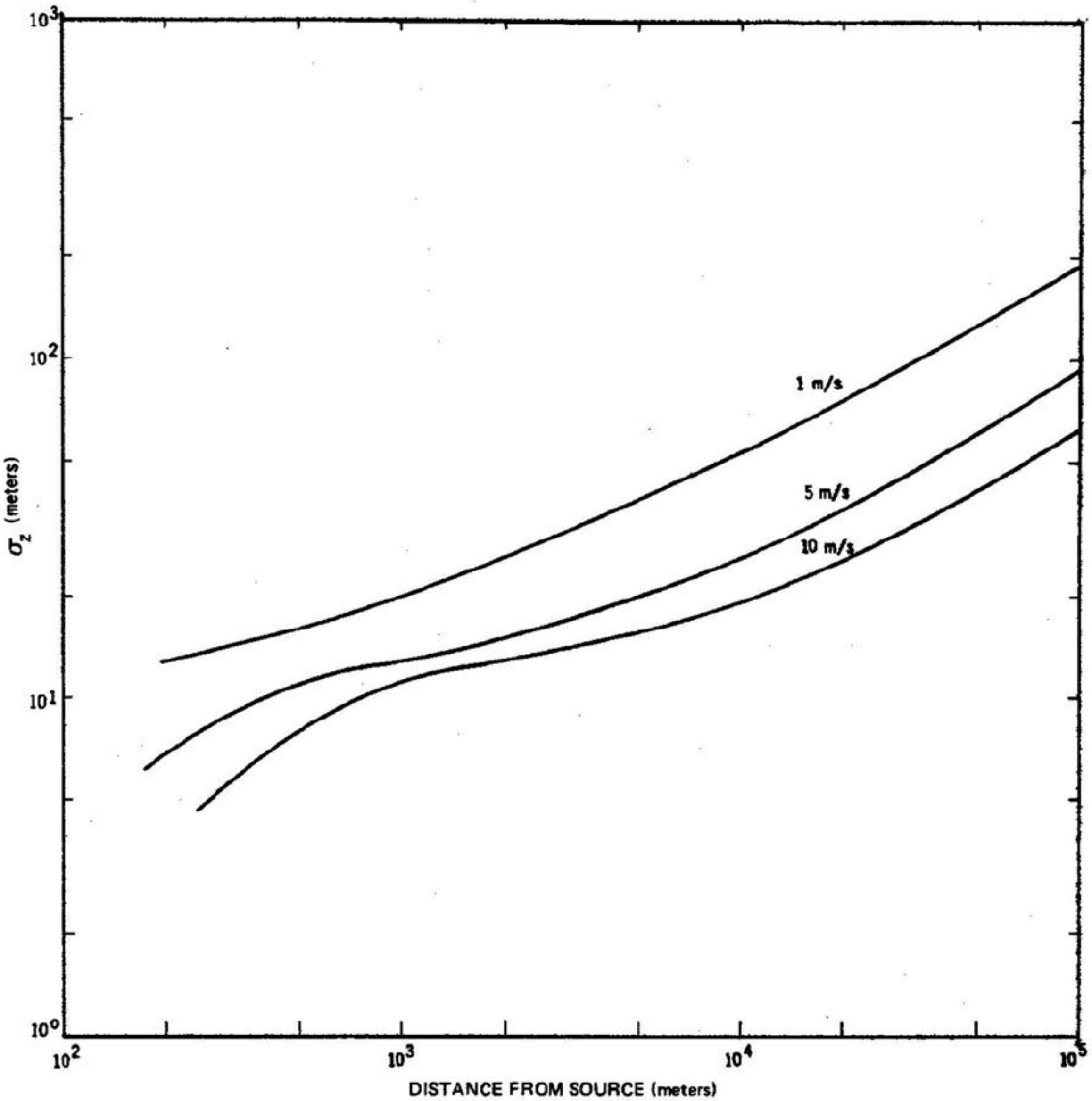

Figure C-2. Vertical Cloud Width - Stable

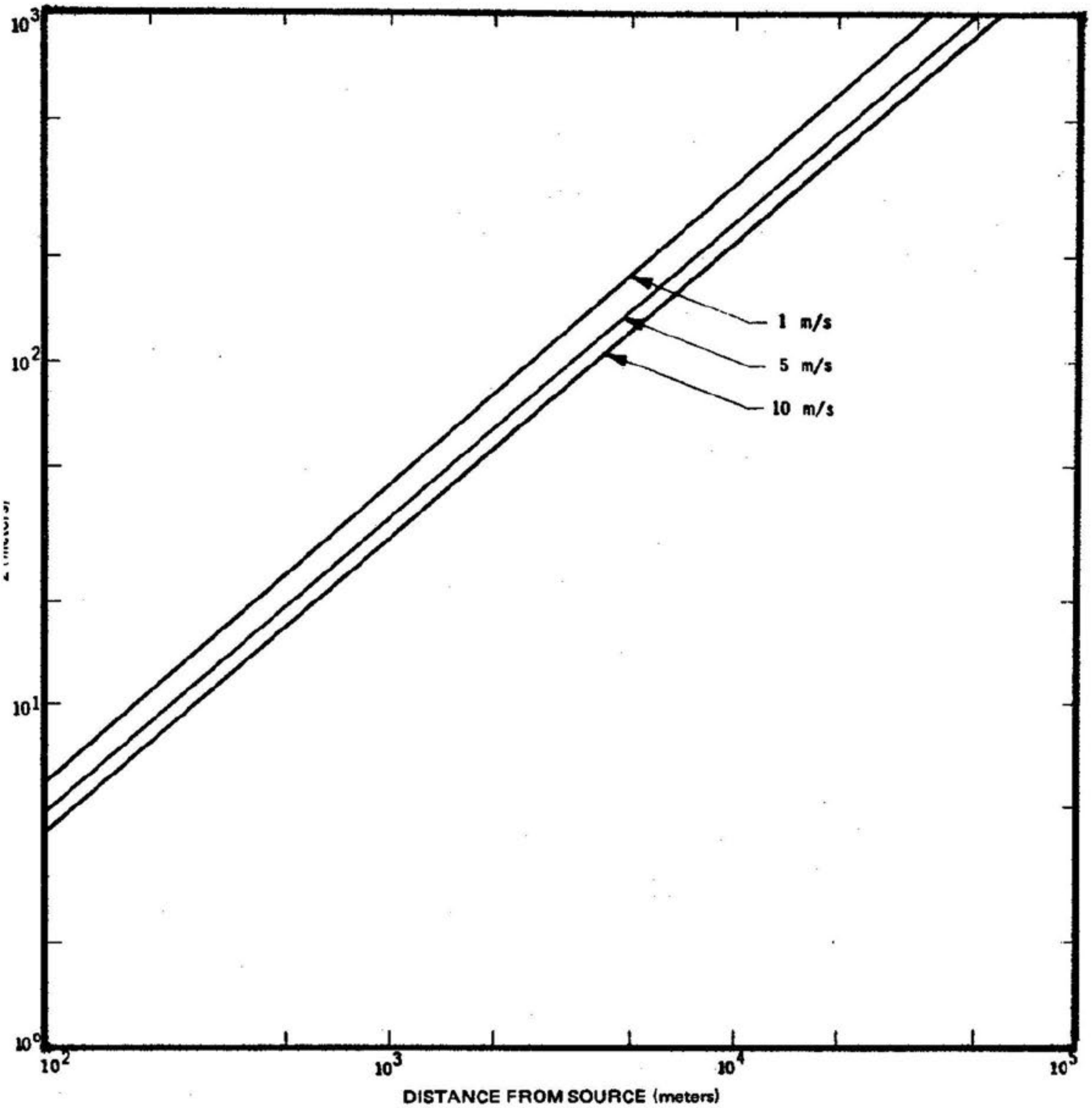

Figure C-3. Vertical Cloud Width – Neutral Stability.

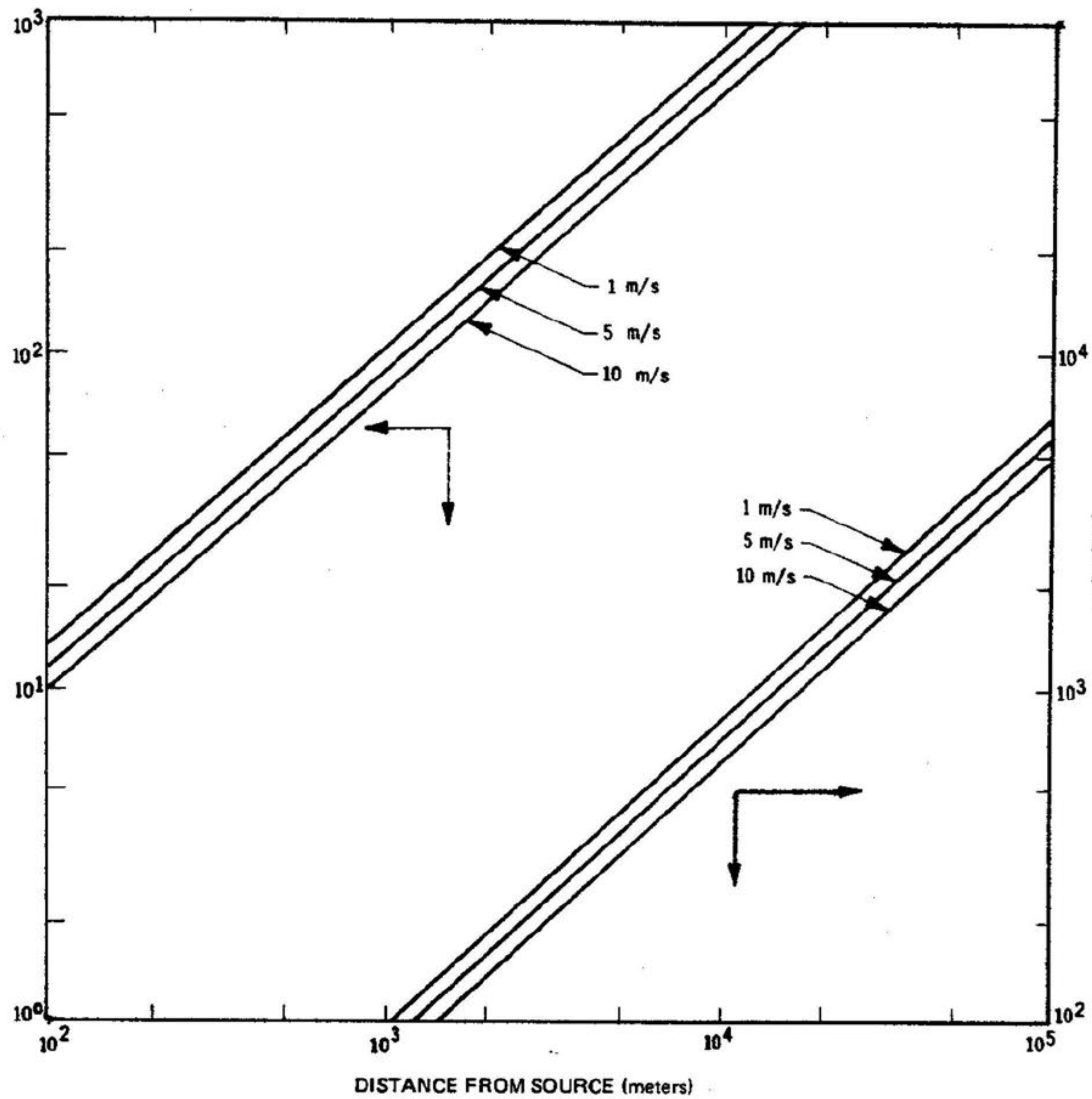

Figure C-4. Vertical Cloud Width - Unstable

3.0 RADIOLOGICAL FACTORS

Four different varieties of ground level radiation exposure are consequential to stack emission of radioactive materials. These are:

1. External radiation to persons on the ground who are not in the plume but who receive radiation (principally gamma radiation) from the plume. This is the case for persons located near the stack.
2. External radiation to persons on the ground and in the plume (gamma and beta radiation). This is pertinent only at distances where the plume has reached the ground.
3. Internal radiation exposure to persons in the plume as a consequence of inhalation.
4. Internal radiation exposure from ingestion of agricultural products affected by deposition of radioactive materials on vegetation.

Each type of exposure is considered separately below.

3.1 External Dose (Gamma)

The ground-level gamma dose rate from an elevated plume of radioactive materials having a distribution as given in Equation (C-4) may be considered as the sum of the dose rates from all the points in the plume. The source strength of each point is (X) dV and the total source is:

$$S = \int_{-\infty}^{\infty} (X) dV \qquad \text{(C-7)}$$

where:

dV = dxdydz and is an incremental volume of the cloud which may be considered as a point source. Since the integration is carried out to infinity in the z-direction, the entire cloud is included so that the "reflection" effect, if any, is accounted for in the calculation.

The flux from a point source considering buildup in the air is given by Glasstone[3]:

$$\theta = \left(\frac{(BS)\exp(-\mu T)}{4\pi T^2} \right) \quad \text{(photons/m}^2\text{/sec)} \qquad \text{(C-8)}$$

where:

B = Buildup factor = $1 + K\mu T$;

K = $\left(\frac{\mu - \mu_a}{\mu_a}\right)$, where μ is total absorption coefficient and μ_a is energy absorption coefficient;

T = Distance from source and is equal to $\sqrt{\left(x_1^2 + y_1^2 + z_1^2\right)}$ in the coordinate system used; and x_1, y_1, z_1, are coordinates of point at ground level relative to incremental volume of cloud.

The gamma dose rate from a flux of a given energy (E) from Glasstone is

$$(D.R.)\gamma = \left(5x10^5\right)\theta E\mu_a(mR/hr), \qquad \text{(C-9)}$$

so that the total dose rate from the plume at any point is found by combining Equations (C-7), (C-8), and (C-9):

$$(D.R.)\gamma = \left(\frac{\left(5x10^{-5}E\mu_a\right)}{4\pi}\right)\int_{-\infty}^{\infty}\left(\frac{B(X)\exp(-\mu T)dV}{T^2}\right) \text{ (mR/hr)} \qquad \text{(C-10)}$$

After substituting $(X)^i_{avg}$ for (X) the average dose rate for the ith meteorological condition can be found. This equation is the gamma dose rate either for a person immersed in the cloud at (x, y, z) or at some point outside the cloud.

Solution of Equation (C-10) requires use of numerical techniques. As Equation (C-10) is written it assumes a monoenergetic source. For a mixture of isotopes, it is proper to perform the calculation for each gamma energy present and considering its abundance. Since μ and μ_a are energy dependent and appear in an exponential term care must be exercised if an average energy is to be used.

The total gamma dose in the year at any point is found by determining the total dose at that point from all plumes traveling in all directions. That is, the dose at any point from plumes traveling in all sixteen directions are added to give a total dose from all plumes in all directions. At each step in the summation, the dose is calculated by multiplying dose rate (in mR/h) during any meteorological condition by the annual frequency (in hours) of that condition.

The dose so calculated may be taken as the annual average dose rate in air, milliroentgens per year. Conversion of this into annual doses absorbed by individuals requires that time of occupancy, local shielding (if any) and other factors be taken into account.

3.2 External Dose (Beta β)

The range of β particles in air is only a few meters. Hence, for β calculations, a cloud of material released via a stack and which expands to large dimensions at downwind distances where the cloud has reached ground level, is frequently considered an "infinite" cloud. In such a cloud, the air dose rate is calculated by assuming that the rate of energy release per unit volume in the cloud is equal to the rate of absorption in that volume (no buildup). The body is considered a small volume within the flux in the cloud and therefore causes no perturbation in the flux.

β flux incident on the human body comes from one direction only, so that the air dose rate at the surface of the body is only one-half of that in the air. In addition, the cloud is not infinite since the ground represents a boundary to the cloud, such that at the ground the cloud is a hemisphere of "infinite" radius but approaches the "infinite" cloud at some height above ground equal to the range of the β in air. Thus the dose rate varies across the body (vertically) and so an average value of 0.64 for the actual dose rate compared to the "infinite" cloud calculations is used[4]:

$$(D.R.)_{\beta} = 0.53x10^{6}(X)\overline{E} \quad \text{(mRad/hr)} \qquad \text{(C-11)}$$

After substituting $(X)^{i}_{avg}$ for (X) the average beta dose rate for the ith meteorological condition can be found. Since the range of betas in air is quite short, the annual total beta dose in a given direction is the sum of the dose rates (in mRad/h) during each ith condition accompanied by wind blowing in that direction weighted by the annual frequency (in hours) of occurrence. Conversion of this dose into a dose delivered to persons requires adjustments to take into account the shielding effect of clothing.

In the discussion of beta dose rates, the air concentration designated by Equation (C-4) is used. Equation (C-4) is not correct in describing the plume after it has diffused to ground level. The ground represents a barrier to vertical (downward) diffusion. Accordingly, some treat the ground as a perfect reflector, and estimate near-ground-level concentrations on the basis of doubling those otherwise calculated. Whether this is done, or some other factor or method is used to account for this boundary effect, Equation (C-4) needs an appropriate adjustment.

3.3 Internal Dose From Inhalation

Internal dose from inhalation may be related directly to an annual average ground-level air concentration. The average air concentration at ground level is as given in Equation (C-4) for any specific meteorological condition. The annual average concentration is the sum of the average during each meteorological condition weighted by its frequency of occurrence. This weighted concentration may then be compared with the value given in 10CFR20, Appendix B, Table II (which is equivalent to an annual dose limit in 10CFR20.1301) for the isotope of interest, or the value of the mixture, if several isotopes are examined.

Some isotopes, and their values, are not listed in 10CFR20. For these, the values can be calculated from ICRP[5].

In the discussion of internal dose rates, the air concentration designated by Equation (C-4) is used. Equation (C-4) is not correct in describing the plume after it has diffused to ground level. The ground represents a barrier to vertical (downward) diffusion. Accordingly, some treat the ground as a perfect reflector, and estimate near-ground-level concentrations on the basis of doubling those otherwise calculated. Whether this is done, or some other factor or method is used to account for this boundary effect. Equation (C-4) needs an appropriate adjustment.

3.4 Internal Dose From Ingestion

Radioactive materials, which deposit on vegetation and on the ground can cause radiation dose from consumption of agricultural products. For certain food chains, concentration effects exist. One such radioisotope is I-131; the appropriate chain is air-pasture-cow-milk-infant thyroid. On the other hand, the value for I-131 in air is based on exposure via the air-lung-thyroid route. The milk exposure mode is far more limiting. That is, the thyroid dose from breathing air of any given I-131 content is much less than the thyroid dose (to an infant) drinking milk solely from cows feeding from pastures exposed to the same air. This is a result of a brief deposition of iodine on pasture grass, concentration due to the large area of grass eaten by the cow, and relatively efficient transfer to the milk. This effect must be considered when relating an emission rate for iodine to an environmental dose where there are cows involved. Current U.S. practice, in context of USAEC licenses associated with stack emission, assigns a reconcentration factor of 700 to I-131. Thus, for example, the value for I-131 in 10CFR20 is 2×10^{-8} µCi/cc for inhalation considerations but is $2 \times 10^{-8}/700$ or 1.9×10^{-11} µCi/cc for ingestion consideration for a baby with an assumed 2-gram thyroid drinking 1 liter of milk per day.

Other isotopes besides I-131 are associated with food chain concentration effects, but less dramatic than those for I-131. In the case of those isotopes for which data are not available on the "reconcentration factor," an estimate of its value may be obtained by consideration of known differences between the isotope and iodine. Three factors may be distinguished:

(1) Effective radioactive half life on pasture relative to I-131. This determines the quantity existing on the pasture at equilibrium. That is, an isotope with an effective half life twice as long as I-131 would have twice as much on the pasture at equilibrium, all else being equal.

(2) Deposition rate. This determines the rate at which material is deposited on the pasture. This effect may be compared in terms of the deposition "velocity" - wind speed ratio given in Table C-1.

(3) Biological transferal. This accounts for the biological difference in terms of portion of material taken into the body, which reaches the critical body organ via the intake modes of inhalation and ingestion. That is, such a difference exists for almost all isotopes and is a part of the reconcentration effect; but this difference varies from one isotope to the other and will affect the "reconcentration factor" differently in each case. For example, from ICRP[4] a 0.3 fraction of I-131 reaches the thyroid (critical organ) if ingested compared to 0.23 if inhaled. This is a factor of 1.3. For Sr-89, a 0.21 fraction reaches bone (critical organ) via ingestion compared to 0.28 via inhalation. This is a factor of 0.75. Thus the "reconcentration factor" for Sr-89 should be 0.75/1/3 or 0.58 times that for I-131 as far as this effect is concerned. In the case of the milk chain, biological transfer within the cow must also be considered. Watson and Gamertsfelder[1] estimate the I-131/Sr-90 ratio to be 10 for transfer into the milk.

4.0 ENGINEERING FACTORS

From Equations (C-3) and (C-9) it is evident that the dose rate is significantly affected by the height of the plume above ground level. This height is made up of the physical stack height plus plume rise due to exit velocity and buoyancy. Many formulae are available to calculate the plume rise. The method used here is the Holland formula[3] as modified by Moses[6].

$$\Delta H = c\left(\frac{1.5(Vs)d + 4x10^{-5}Q_h}{\overline{\mu}_h}\right), \qquad \text{(C-13)}$$

where:

V_s	=	Exit velocity (meters/sec);
d	=	Stack diameter (Meters);
Q_h	=	Heat emission of effluent (cal/sec);
$\overline{\mu}_h$	=	Wind speed at stack exit (meters/sec); and
c	=	Correction factor from Moses.

In proposing the correction factor "c" in the plume rise formula, Moses used data from an experimental stack at Argonne with a diameter of about 1.5 feet and from a stack at Duisburg, Germany, which has a diameter of 3.5 meters. His conclusions are that a value of 3 for the correction factor is proper for large stacks with appreciable buoyancy whereas a factor of 2 is recommended for small stacks with modest buoyancy. In applying the Moses correction to individual situations, a linear interpolation is made from the actual stack diameter compared to those from which data were obtained (see Figure C-5).

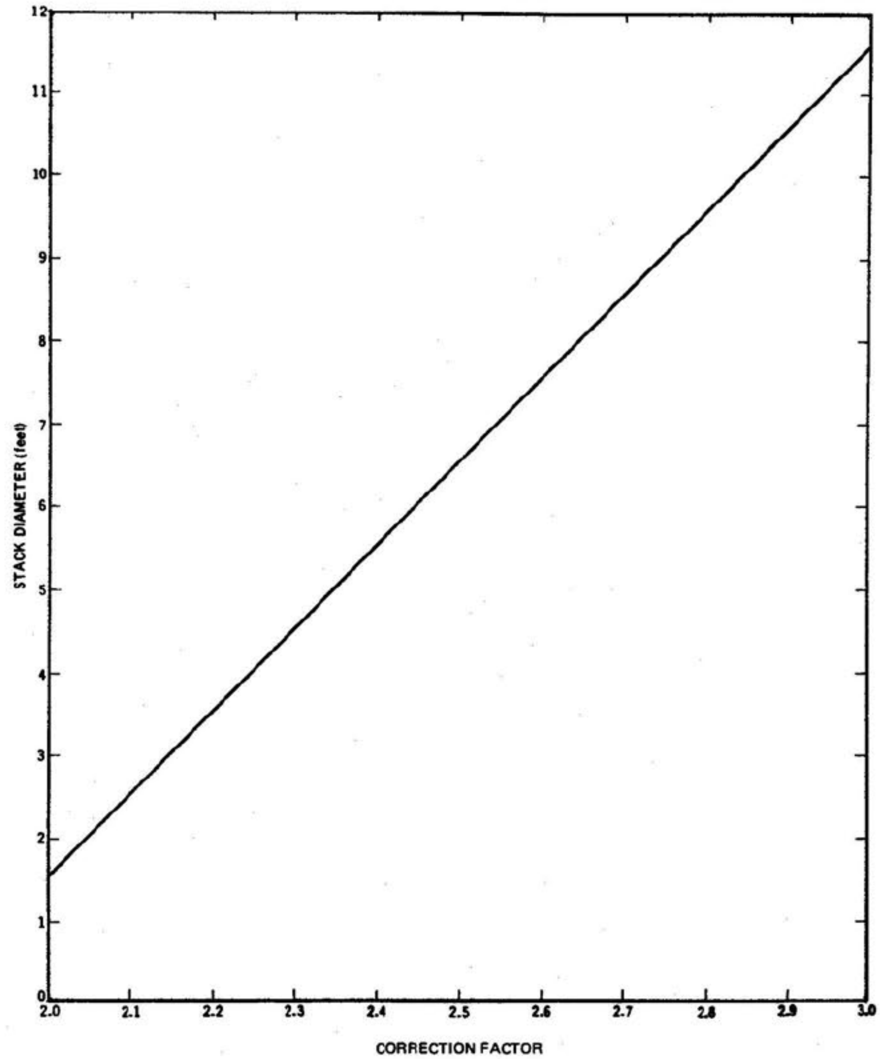

Figure C-5. Holland Plume Rise Formula Correction Factor

5.0 CONCLUSION

A method of estimating annual dose rates from a given continuous stack emission rate has been described. It has been assumed that the standard Gaussian diffusion equations describe the plume dispersion. Situations where topographic or nearby manmade structures could cause significant downwash of the plume were not considered. Special calculations should be used for such situations.

At locations where operation of a facility includes a need to estimate environmental effects of normal operation airborne releases, the method described here may be used. Generally, environmental monitoring is contemplated so that data provided therefrom, when measurable quantities are released, may be used to modify estimates appropriately.

REFERENCES

[1]Originally Appendix C, NEDO-10178, Safety Analysis Report, Midwest Fuel Recovery Plant, Morris, Illinois (Docket 50-268). Figure numbers, table numbers, and other identification within this appendix are those of the original document.

[2]Watson, E. C., and Gamertsfelder, C. C., "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria," HW-SA-2809 (February 1963).

[3]Glasstone, S., and Sesonski, A., Nuclear Reactor Engineering, D. van Nostrand Co. (1963)

[4]"Meteorology and Atomic Energy," AECU 3066.

[5]Report of Committee II (ICRP) on Permissible Dose for Internal Radiation (1959).

[6]Moses, H., Strom, G. H., and Carson, J. E., "Effects of Meteorological and Engineering Factors on Stack Plume Rise", Nuclear Safety, Vol. 6, No. 1, Fall (1964).

A.4 ESTIMATION OF GROUND-LEVEL RADIATION DOSES FROM RELEASE OF AIRBORNE RADIOACTIVE MATERIALS[1]

ABSTRACT

A method of estimating ground-level radiation doses corresponding to a release of airborne radioactive materials is described. The method considers external dose from both, beta and gamma sources, internal dose from inhalation of ground level concentrations of the material and external dose as a result of fallout from the cloud.

The method relates quantity released (in curies) to a dose for various meteorological conditions, types of materials released, and for short-term or prolonged release periods.

The method assumes that normal Gaussian diffusion equations describe the dispersion of the cloud. Situations where topographic or nearby manmade structures could have significant effects on the cloud are not considered. Special calculations should be used for such situations.

1.0 INTRODUCTION

The calculation of ground-level radiation doses from a cloud of airborne materials such as assumed in reactor accident analysis may be divided into two general parts. The first part involves the atmospheric transport and dilution of the cloud by the wind. This results in a calculated integrated air concentration[2] in the cloud at some dose point of interest. The second part of the analysis is the conversion of air concentration to radiation dose of interest.

The sources of radiation usually considered in reactor accident analysis are (a) the noble gases and their external whole-body dose effect, (b) the halogens and the resulting thyroid dose from inhalation, (c) volatile solids (cesium, rubidium, selenium, arsenic, antimony, molybdenum, and tellurium) resulting in lung dose from inhalation, and (d) bone dose from inhalation of the nonvolatile solids (all others). The whole-body dose from fallout of materials is also usually calculated.

Various meteorological conditions are generally examined in such analyses to give a spectrum of radiological effects during the poor diffusion conditions of inversion and the better diffusion conditions of lapse or unstable. For example, very stable and moderately stable, each at a wind speed of 1 m/sec (2 mph), neutral conditions at wind speeds of 1 and 5 m/sec (10 mph), and unstable conditions at wind speeds of 1 and 5 m/sec may be used.

2.0 ATMOSPHERIC DIFFUSION MODEL

In the calculation of the transport and dilution of an airborne cloud, the time period of release of the cloud is very significant. This is so, primarily because the wind does not tend to remain fixed direction-wise, but rather it meanders and fluctuates to a considerable extent. Thus, if a cloud is formed during a long release period, portions of it will tend to be transported in different directions. On the other hand, if the cloud is formed from an explosive release or "puff" it will all tend to be transported in the same direction. This variability of the wind refers principally to the horizontal changes as opposed to vertical changes, since the former is often very significant while the latter is much more subdued.

A means of describing dilution for a cloud released over a long period of time (say several hours) has been suggested by Simpson[3]. If the total release is viewed as successive shorter-term releases (but not puffs) during which the average wind direction is reasonably constant (although short-term fluctuations may exist) then the dilution of these shorter-term releases may be calculated with presently available methods. The net dilution at any given point would then be the sum of the dilution for each incremental cloud transported in the various average directions (some additional discussion is given on this point under Section 4.0, Application of Methods).

The calculation of the dilution or integrated air concentration in a cloud for a unit release of material transported in a given direction is usually described by the Gaussian equation[4]:

$$(X) = \left(\frac{Q_0}{(2\pi)(\sigma_y)(\sigma_z)(\overline{\mu}_h)} \right) \exp\left[-\left(\frac{z^2}{2\sigma_z^2} \right) - \left(\frac{y^2}{2\sigma_y^2} \right) \right] \left[\frac{Q}{Q_0} \right] \quad \text{(D-1)}$$

where:

(X)	=	Integrated air concentration (Ci-sec/m3 or µCi-sec/cc);
Q	=	Quantity released (Ci);
$\overline{\mu}_h$	=	Average wind speed at height of release or effective height if cloud rise occurs (m/sec);
σ	=	Standard deviation of cloud width in horizontal y-direction and vertical z-direction (m);
t	=	Time after release (sec) and is equal to the downward distance divided by the average wind speed $X \div \overline{\mu}_h$;
Q/Q_0	=	Correction for cloud depletion due to deposition and is the fraction of initial amount released which is present at downwind distance x (x = $\overline{\mu}_h t$);

$$= \qquad \exp\left[-\left(\frac{V_d}{\overline{\mu}_h}\right)\sqrt{\frac{2}{\pi}}\left(\frac{\mu_0}{\overline{\mu}_h}\right)\int_0^t \overline{\mu}_h \frac{\exp\left(-\frac{z}{2\sigma_z^2}\right)}{\sigma_z} dt\right];$$

V_d = Deposition "velocity" (m/sec) (see Table C-1) for values of this parameter);

$\overline{\mu}_0$ = Average wind speed at ground level (m/sec); and

exp [ ] = Function which is a power of "e"; and

y, z = Horizontal and vertical distance from cloud centerline; y = 0 and z = 0 gives cloud centerline concentration and z = h (height of release) gives ground level concentration. The cloud centerline is assumed transported downwind at the same height as the release height.

Equation (D-1) does not take into account the depletion of the radioactive content of the cloud by radioactive decay of the isotope of concern. With this taken into account, the equation becomes:

$$(X) = \left(\frac{Q_0}{2\pi\sigma_y\sigma_z\overline{\mu}_h}\right)\exp\left[-\frac{z^2}{2\sigma_z^2}-\frac{y^2}{2\sigma_y^2}\right]\left[\frac{Q}{Q_0}\right]\exp[-\lambda t] \qquad \text{(D-2)}$$

where:

$\exp[-\lambda t]$ = Radioactive decay function.

Equation (D-2) describes air concentration in a cloud which is not restrained in its expansion and dilution. This is the case for an elevated cloud which has not expanded enough to reach ground level. For cases where the cloud has reached ground level some modification of Equation (D-2) is needed. In the case of a ground-level release, Equation (D-2) is generally multiplied by two.

It can be seen from Equation (D-2) that the important parameters to be calculated are σ_y and σ_z. As indicated previously, the scale of horizontal wind variation changes considerably with time so that two methods of calculating σ_y are used, one for the puff release period and the other for the prolonged period. In the case of σ_z only one method of calculation is employed since the vertical wind fluctuations are not as strongly time dependent.

For the puff release case, the standard deviation of cloud width in the horizontal and vertical directions has been described[4] by Equations (D-3) and (D-4):

$$\sigma_y^2 = \frac{C_y^2 X^{2-n}}{2} \quad \text{, and} \tag{D-3}$$

$$\sigma_z^2 = a\left[1-\exp\left(-k^2t^2\right)\right]+bt \quad , \tag{D-4}$$

where:

a, k^2, b, n, C_y = Diffusion coefficients dependent on wind speed and atmospheric stability (see Table D-1 for recommended values).

TABLE D-1
VALUES FOR VARIABLES

Variable	Height of Release (meters)	Wind Speed (m/sec)	Atmospheric Stability[a] Very Stable	Moderately Stable	Neutral	Unstable
$\left(V_d/\overline{U}_0\right)$ Noble Gases	--	--	0	0	0	0
$\left(V_d/\overline{U}_0\right)$ Halogens	--	--	0.0024	0.0034	0.0046	0.0080
$\left(V_d/\overline{U}_0\right)$ Particulates	--	--	0.00015	0.00022	0.00030	0.00060
R	all	--	6.37	8.146	10.6	17.66
a	all	--	34	97	--	--
b	all	--	0.025	0.33	--	--
K^2	all	--	0.0088	0.00025	--	--
n	=0.0	--	0.3	0.3	0.25	0.20
n	>0.0	--	0.4	0.4	0.25	0.20
C_y	=0.0	1 - 3	0.18	0.18	0.21	0.35
C_y	=0.0	4 - 7	0.18	0.18	0.15	0.30
C_y	=0.0	> 7	0.18	0.18	0.14	0.28
C_y	>0.0	1 - 3	0.18	0.18	0.15	0.30
C_y	>0.0	4 - 7	0.18	0.18	0.12	0.26
C_y	>0.0	> 7	0.18	0.18	0.11	0.24
C_z	=0.0	1 - 3	--	--	0.17	0.35
C_z	=0.0	4 - 7	--	--	0.14	0.30
C_z	=0.0	> 7	--	--	0.13	0.28
C_z	>0.0	1 - 3	--	--	0.15	0.30
C_z	>0.0	4 - 7	--	--	0.12	0.26
C_z	>0.0	> 7	--	--	0.11	0.24

[a] The degree of atmospheric stability is defined here in terms of the standard dry adiabatic vertical temperature lapse rate of -1 °C per 100 meter increase in elevation (-5.4 °F per 1000 feet). This is taken as a convenient reference point for defining the four classes of stability:

very stable	≥ +1.5 °C
moderately stable	≥ -0.5 °C but < +1.5 °C
neutral	≥ -1.5 °C but < -0.5 °C
unstable	< -1.5 °C

In the case of the prolonged release, the vertical standard deviation is described by Equation (D-4) but the horizontal deviation is described[5] by Equation (D-5):

$$\sigma_y^2 = At - A\alpha\left[1 - \exp\left(-\frac{t}{\alpha}\right)\right], \qquad \text{(D-5)}$$

where :

A,α = Diffusion coefficients ;

and

A = 13 + 232.5 ($\sigma\theta\overline{\mu}_h$);

α = $\dfrac{A}{2\left(\sigma\theta\overline{\mu}_h\right)^2}$; and

$\sigma\theta$ = Standard deviation of horizontal wind direction variation during release.

The distinction between what is a puff release and what is a prolonged release is arbitrarily set at 30 minutes. That is, releases of less than 30-minute duration are considered puff releases and above that are prolonged releases.

3.0 RADIATION DOSE MODEL

Three different varieties of ground-level radiation exposure are consequential to a release of radioactive materials. These are:

1. External radiation to persons on the ground from the cloud as it passes by. (This may be gamma-only dose for an elevated cloud, or beta and gamma dose from a ground-level cloud.)

2. Internal radiation exposure to persons in the cloud as a consequence of inhalation.

3. External radiation to persons on the ground from fallout on the ground after passage of the cloud.

Each type of exposure is considered separately below.

3.1 External Passing Cloud Dose (Gamma)

The ground-level gamma dose rate from a cloud of radioactive materials having a distribution as given in Equation (D-2) may be considered as the sum of the dose rates from all the points in the cloud. The source strength of each point is (X)dV and the total source is

$$S = \int_{-\infty}^{\infty} (X)dV \quad , \qquad \text{(D-6)}$$

where:

dV = dx, dy, dz, and is an incremental volume of the cloud which may be considered as a point source. The integration is theoretically carried out to infinity to include the entire cloud.

The flux from a point source, considering buildup in the air is given by Glasstone[6]:

$$\phi = \frac{BS\exp(-\mu T)}{4\pi T^2} \qquad \text{(photons/m}^2\text{/sec)} \qquad \text{(D-7)}$$

where:

B = Buildup factor = 1 + KμT;

K = $\frac{\mu - \mu_a}{\mu_a}$ where μ is total absorption coefficient and μa is energy absorption coefficient (see Figure D-1)

T = Distance from source and is equal to $\sqrt{x_1^2 + y_1^2 + z_1^2}$ in the coordinate system used; and x_1, y_1, z_1, are coordinates of point at ground-level relative to incremental volume of cloud.

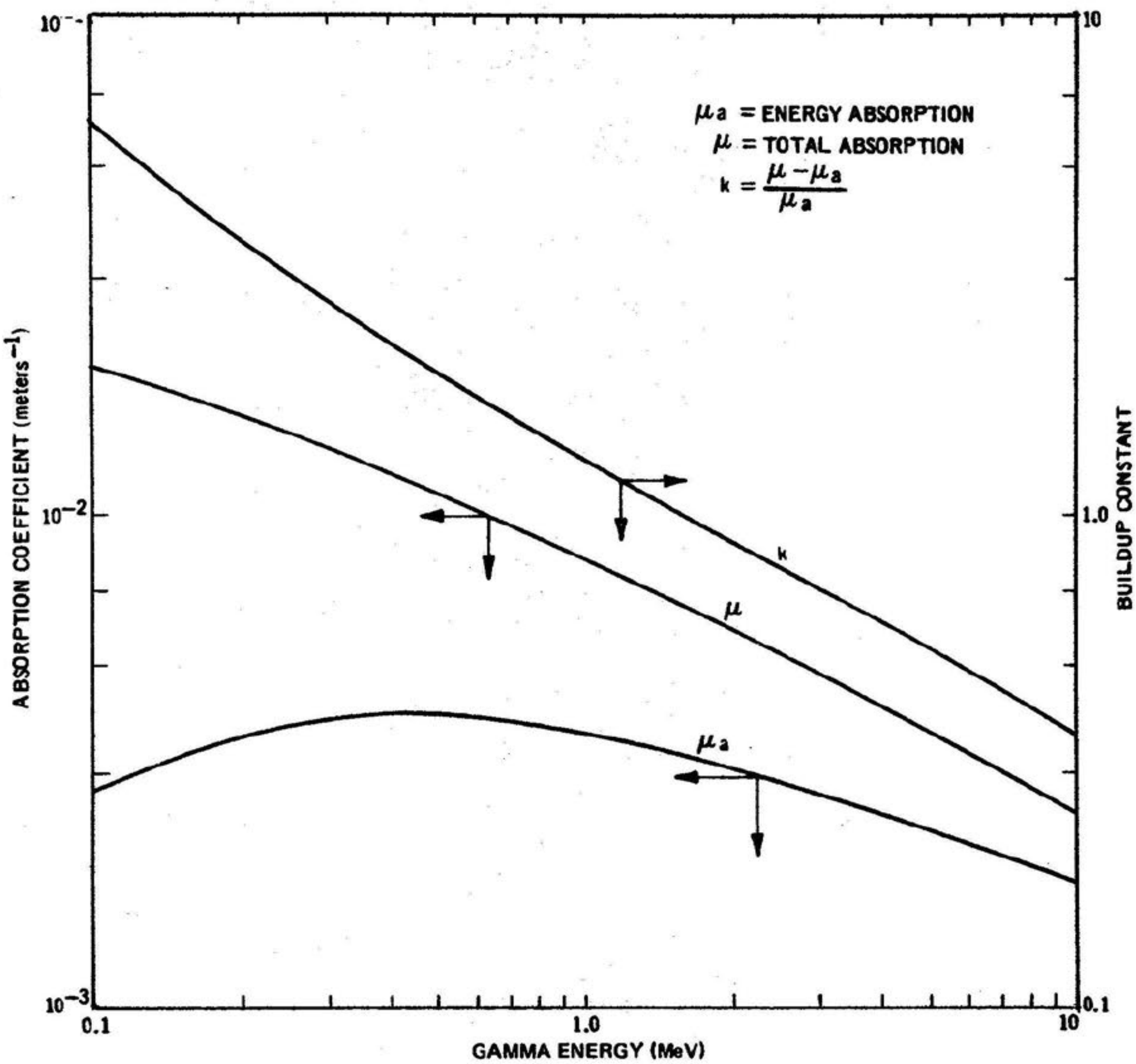

Figure D-1. Gamma Radiation Absorption Coefficients and Buildup Constants for Air (STP)

The gamma dose rate from a flux of a given energy (E) from Glasstone[6] is

$$(D.R.)_\gamma = 1.4 \times 10^{-11}\, \theta E \mu_a \text{ (rad/sec)}, \qquad \text{(D-8)}$$

so that the total dose from the cloud at any point is found by combining Equations (D-2), (D-7), and (D-8).

$$(D)\gamma = \frac{1.4x10^{-11} E\mu_a}{4\pi} \int_{-\infty}^{\infty} \frac{B(X)\exp(-\mu T)dV}{T^2} \text{ (rad)} \qquad \text{(D-9)}$$

Solution of Equation (D-9) requires use of numerical techniques. As the equation is written it assumes a monoenergetic source. For a mixture of isotopes, it is proper to perform the calculation for each gamma energy present considering its abundance. Since μ and μ_a are energy dependent and appear in an exponential term, care must be exercised if an average energy is to be used. See Table D-2 for the typical noble gases of interest in reactor accident analyses.

TABLE D-2
RADIOBIOLOGICAL FACTORS -- NOBLE GASES

Isotope		**Disintegration Gammas Emitted**	
Name	**Half-Life**	**Number**	**Energy**
Noble Gases			
Kr-83m	1.86 h	1	0.032
		2	0.009
Kr-85m	4.4 h	1	0.15
		2	0.305
Kr-85	10.76 y	1	0.522
Kr-87	76 m	1	2.05
		2	2.57
		3	0.847
		4	0.347
Kr-88	2.8 h	1	2.4
		2	2.21
		3	0.19
		4	1.55
		5	0.85
		6	0.17
		7	0.02
Xe-131m	12 d	1	0.164
Xe-133m	2.3 m	1	0.233
Xe-133	5.27 d	1	0.081
Xe-135m	16 m	1	0.53
Xe-135	9.2 h	1	0.604
		2	0.36
		3	0.244
Xe-138	14 m	1	0.42
Particulate Daughters[a]			

Isotope Name	Isotope Half-Life	Disintegration Gammas Emitted Number	Disintegration Gammas Emitted Energy
Rb-88	18 m	1	0.91
		2	1.28
		3	1.85
		4	2.18
		5	4.2
Cs-138	32.2 m	1	0.14
		2	0.19
		3	0.23
		4	0.41
		5	0.46
		6	0.55
		7	0.87
		8	1.01
		9	1.43
		10	2.21
		11	2.62
		12	3.34

[a] Significant particulate daughters only

3.2 External Dose (Beta β)

The range of β particles in air is only a few meters. Hence, for β calculations, a cloud of material, which expands to fairly large dimensions (say >20 meters or 60 feet) at downwind distances is frequently considered an "infinite" cloud. In such a cloud, the air dose rate is calculated assuming that the rate of energy release per unit volume in the cloud is equal to the rate of absorption in that volume (no buildup). The body is considered a small volume within the flux in the cloud, and therefore, causes no perturbation in the flux.

β flux incident on the human body comes from one direction only, so that the air dose rate at the surface of the body is only one half of that in the air. In addition, the cloud is not infinite since the ground represents a boundary to the cloud, such that at the ground the cloud is a hemisphere of "infinite" radius but approaches the "infinite" cloud at some height above ground equal to the range of the β in air. Thus, the dose rate varies across the body (vertically) and so an average value of 0.64 for the actual dose rate compared to the "infinite" cloud calculation is used from Taylor[7]. Thus the β dose is given by:

$$(D)_\beta = 0.15(X)\overline{E} \quad \text{(rad)} \qquad \text{(D-10)}$$

3.3 Internal Dose from Inhalation

Internal dose from inhalation may be related directly to ground-level air concentration. The air concentration at ground level is as given in Equation (D-2) for any specific meteorological condition. The dose due to inhalation of the cloud is calculated by first, determining the quantity inhaled and then, multiplying by the conversion factor of dose per unit amount inhaled. The Quantity inhaled (Q_i) is calculated from

$$Q_i = 230(X) \qquad (\mu Ci), \qquad\qquad (D\text{-}11)$$

where 230 is taken as the standard average breathing rate from ICRP[8] in cc/sec.

The dose conversion factor (k) for a unit amount inhaled is calculated from ICRP[8]. In ICRP the permissible body burden (q), which is equivalent to a permissible dose rate (weekly, quarterly, yearly dose rate) for each isotope is given. Considering the effective half-life of the isotope in the critical organ (or other organ) permits calculation of the lifetime dose to the organ. Since the permissible body burden (q) refers to total quantity in the body, some factor to account for the fraction of total burden, which is in an organ or interest must be applied. This factor is given as (f_2) by ICRP. Additionally, to convert quantity breathed to quantity deposited in the organ of interest, an additional factor (f_a) from ICRP is used. Thus, the dose from inhalation (D_i) is calculated from

$$D_i = 230(X)qf_2f_a\frac{t_{1/2}}{0.693} \quad \text{(Rem)}, \qquad\qquad (D\text{-}12)$$

where:

q = Quantity (µCi) in <u>total body</u> equivalent to a dose rate of Y Rem/week (from ICRP);

$\frac{t_{1/2}}{0.693}$ = Mean life of isotope in organ; and

$qf_2f_a\frac{t_{1/2}}{0.693}$ = k Rem/µCi inhaled.

Values for the factor k are given in Tables D-3, D-4, and D-5 for the halogen, volatile solid, and nonvolatile solid mixtures. In the case of the halogens and nonvolatile solids, if they are assumed to be soluble, the thyroid and bone are the critical organs, respectively. If the volatile solids are assumed insoluble then the lung is the critical organ.

3.4 Fallout Dose (Gamma Dose)

The fallout dose (D_f) is almost entirely due to the halogens because of their larger assumed release fraction and the larger deposition velocity assigned to them. Fallout dose is calculated by determining the deposition (Ci/m^2), and multiplying by the dose rate conversion factor (R rad/h per Ci/m^2), and integrating over the decay during the time of dose received:

$$D_f = (X)V_d R\left(\frac{1-e^{-\lambda t'}}{\lambda}\right) \quad \text{(rad).} \qquad \text{(D-13)}$$

where:

$(X)V_d$	=	The deposition ($curies/m^2$);
R	=	Dose rate conversion factor;
λ	=	Decay constant of the isotope; and
t'	=	Dose period.

Values of the dose rate conversion factor (R) are given[9] in Figure D-2 for the various gamma energies. Since these values are for an infinite plane source and the cloud size and deposition pattern is not always infinite, a correction factor must be applied in some cases. The correction factor[9] is given in Figure D-3.

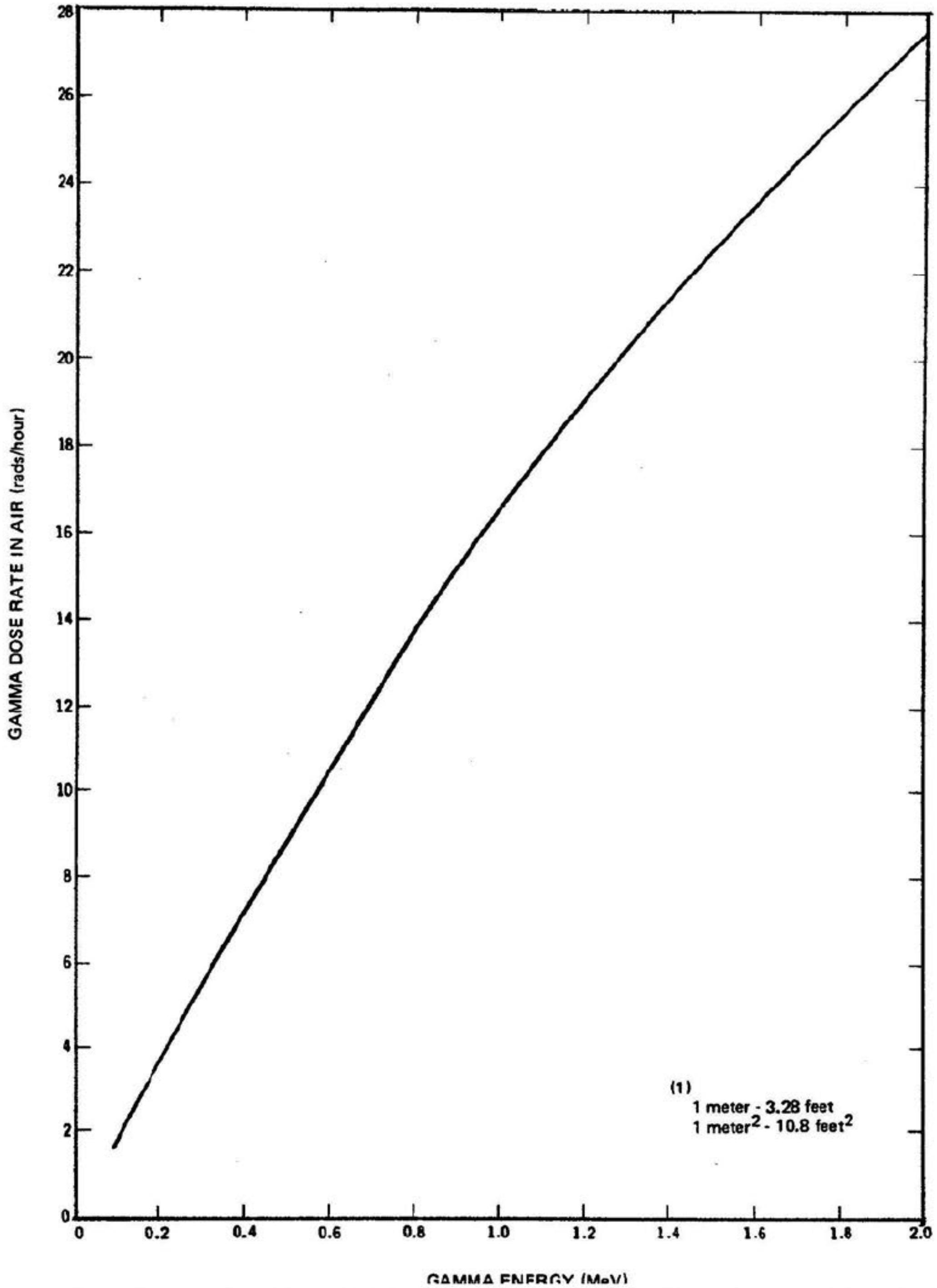

Figure D-2. Gamma Dose Rate at One Meter Height Above Smooth Infinite Plane Source of One Curie Per Square Meter.

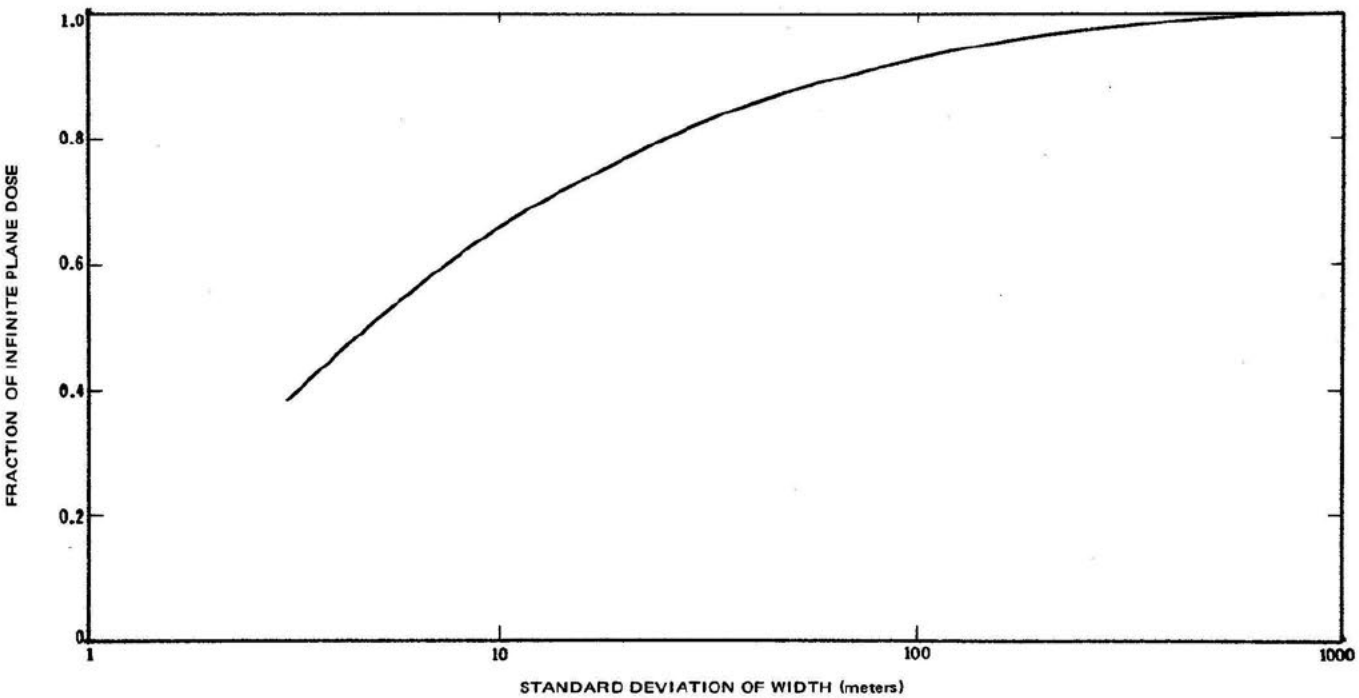

Figure D-3. Ratio of Gamma Dose from Finite Pattern to Infinite Plane Dose.

TABLE D-3
RADIOBIOLOGICAL FACTORS -- HALOGEN RADIOISOTOPES

Isotope Name	Isotope Half-Life[a]	Eff Half-Life[b]	$\overline{E}_{\gamma}$ (MeV)	$\overline{E}_{\beta}$ (MeV)	$\overline{E}_{Eff}$ (MeV)[c]	k (Rem/μCi)[d]
*I-131	8.05 d	7.0 d	0.39	0.191	0.23	1.6
*I-132	2.3 h	2.3 h	1.992	0.434	0.65	$4.5x10^{-2}$
*I-133	21 h	21 h	0.444	0.45	0.14	$4.0x10^{-1}$
*I-134	53 m	53 m	1.27	0.6	--	$2.6x10^{-2}$
*I-135	6.7 h	6.7 h	1.54	0.308	0.066	$1.3x10^{-1}$

a Radioactive half-life
b Effective half-life in the thyroid from ICRP
c Effective energy in the thyroid from ICRP
*d Dose per μCi inhaled based on IAEA recommended values in IAEA Safety Series No. 7.

TABLE D-4
RADIOBIOLOGICAL FACTORS -- VOLATILE SOLID RADIOISOTOPES

Isotope Name	Isotope Half-Life[a]	Eff Half-Life[b]	$\overline{E}_{\gamma}$ (MeV)	$\overline{E}_{\beta}$ (MeV)	$\overline{E}_{Eff}$ (MeV)[c]	k (Rem/μCi)[d]
*Mo-99	66 h	66 h	0.24	0.376	0.45	$2.6x10^{-2}$
*Te-127m	105 d	105 d	0.0885	0	0.083	$1.7x10^{-1}$
*Te-127	9.3 h	9.3 h	--	0.23	0.24	$4.6x10^{-3}$
*Te-131	25 m	25 m	0.475	0.577	0.73	--
*Te-132	78 h	78 h	0.231	0.073	0.13	$6.4x10^{-2}$
*Cs-134	2.1 y	120 d	1.41	0.52	0.074	$5.6x10^{-1}$
*Cs-137	30 y	138 d	0	0.192	0.192	$4.6x10^{-1}$

[a] Radioactive half-life
[b] Effective half-life in the lung from ICRP
[c] Effective energy in the lung from ICRP
*[d] Dose per μCi inhaled based on IAEA recommended values in IAEA Safety Series No. 7.

TABLE D-5
RADIOBIOLOGICAL FACTORS -- NONVOLATILE SOLID RADIOISOTOPES

Isotope Name	Isotope Half-Life[a]	Eff Half-Life[b]	$\overline{E}_{\gamma}$ (MeV)	$\overline{E}_{\beta}$ (MeV)	$\overline{E}_{Eff}$ (MeV)[c]	k (Rem/μCi)[d]
*Sr-89	50.4 d	50.4 d	0	0.487	0.49	$4x10^{-1}$
*Sr-90	28 y	17.53 y	0	0.2	1.1	36
*Sr-91	9.7 h	9.7 h	0.845	0.523	3.3	$5.0x10^{-3}$
*Y-90	64.2 h	64.2 h	--	0.73	4.4	$2.6x10^{-2}$
*Y-91	59 d	59 d	0.551	0	2.9	$3.3x10^{-1}$
*Zr-95	65 d	59.5 d	0.733	0.127	0.57	$5.5x10^{-2}$
*Nb-95m	90 h	59.5 d	0.235	0	3.8	--
*Nb-95	35 d	33.8 d	0.745	0.053	0.36	$1.2x10^{-2}$
*Ru-103	40 d	2.4 d	0.473	0.08	0.43	--
*Ru-106	1.0 y	15 d	--	0.013	0.013	--
*Rh-105	36 h	1.39 d	0.032	0.183	0.86	--
*Ba-140	12.8 d	10.7 d	0.237	0.268	1.5	$8x10^{-2}$
*La-140	40.2 m	1.68 d	2.11	0.495	2.7	$5.0x10^{-3}$
*Ce-141	32.5 d	31 d	0.097	0.163	0.17	$2.2x10^{-2}$
*Ce-143	33 h	1.33 d	0.344	0.355	2.2	$3.8x10^{-3}$
*Ce-144	285 d	243 d	0.043	0.087	1.3	1.1
*Pr-143	13.7 d	13.7 d	0	0.311	1.6	$2.0x10^{-2}$
*Nd-147	11.1 d	11.1 d	0.286	0.228	1.2	$1.8x10^{-2}$
*Pm-147	2.7 y	570 d	--	0.074	0.22	$2x10^{-1}$
*Pm-149	53 h	2.2 d	0.285	0.35	1.9	$3.3x10^{-1}$
*Pu-240	$6.7x10^{3}$ y	$1.95x10^{3}$ y	0.011	0	0.88	$7x10^{+3}$

[a] Radioactive half-life

[b] Effective half-life in the thyroid from ICRP
[c] Effective energy in the thyroid from ICRP
*[d] Dose per μCi inhaled based on IAEA recommended values in IAEA Safety Series No. 7.

4.0 APPLICATION OF METHODS

In utilizing the methods of calculation described here, several factors are of significance. These are discussed in the following paragraphs.

4.1 Height of Release

From Equation (D-2) it is evident that the dose is significantly affected by the height of the cloud above ground level. In case of stack releases this height is made up of the physical stack height plus cloud rise due to exit velocity and buoyancy. Many formulae are available to calculate the cloud rise. The method used here is the Holland formula[7] as modified by Moses[10].

$$\Delta H = c\frac{\left(1.5V_s d + 4x10^{-5} Q_h\right)}{\overline{\mu}_h}, \quad \text{(D-14)}$$

where:

ΔH	=	Cloud rise (m);
V_s	=	Exit velocity (m/sec);
Q_h	=	Heat emission of effluent (cal/sec);
$\overline{\mu}_h$	=	Wind speed at stack exit (m/sec);
c	=	Correction factor from Moses; and
d	=	Stack diameter (m);

In proposing the correction factor "c" in the plume rise formula, Moses used data from an experimental stack at Argonne with a diameter of about 1.5 feet and from a stack at Duisburg, Germany which has a diameter of 3.5 meters. His conclusions are that a value of 3 for the correction factor is proper for large stacks with appreciable buoyancy, whereas a factor of 2 is recommended for small stacks with modest buoyancy. In applying the Moses correction to individual situations, a linear interpolation is made from the actual stack diameter compared to those from which data were obtained (see Figure D.4).

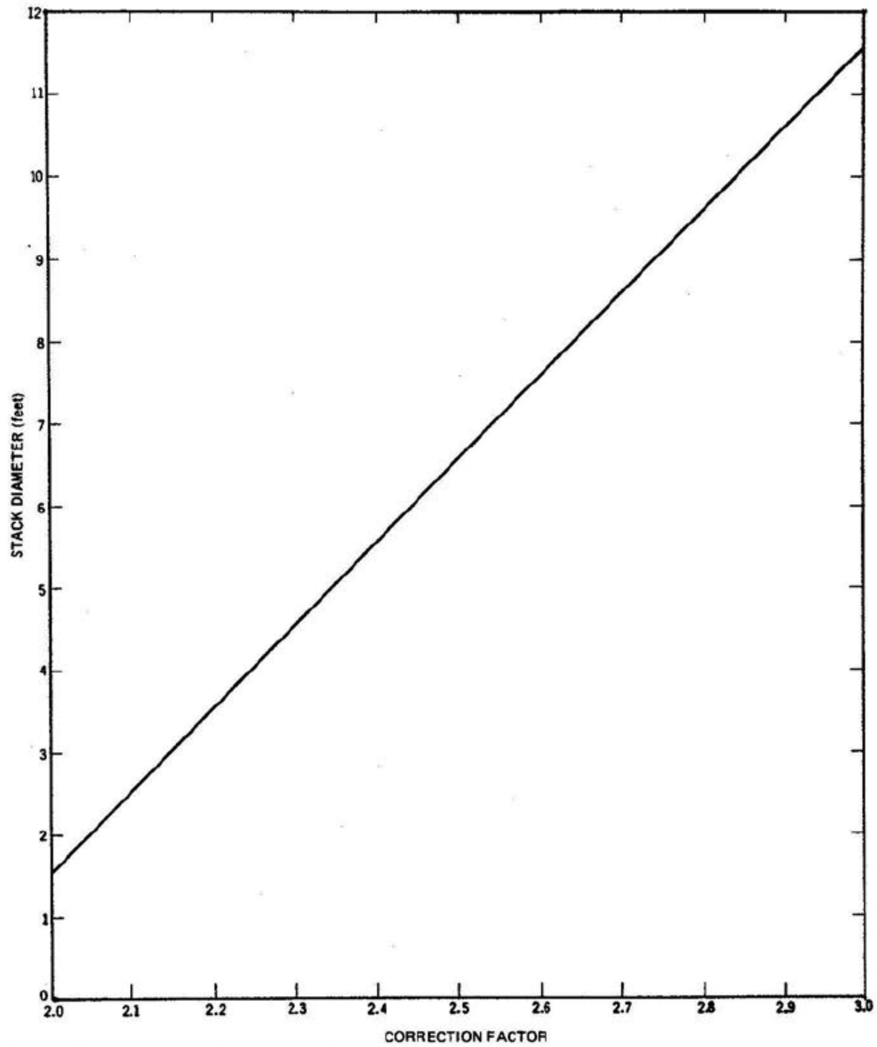

Figure D-4. Holland Plume Rise Formula Correction Factor

4.2 Prolonged Release

For calculations of air concentration in the prolonged-release case the application of two parameters is significant. These parameters are the duration of persistent wind direction during which transport in the same direction is likely, and the second is the wind fluctuation as measured by σ_θ during the persistent direction. This latter parameter is of particular interest since it is not generally available in standard meteorological data. It is suggested that since, theoretically, any duration of persistence is possible as is any value of $\sigma_\theta \bar{\mu}_h$, that a probabilistic approach be used in the choice of these parameters.

Wind direction persistence data have been summarized by the Weather Bureau for several locations. The data are partially shown in Table D-6 for ten locations including valley, desert, coastal, and lake-shore locations. These data do not differentiate between stability conditions or wind speed (see Table D-7 for typical wind speed frequencies). However, the distribution of various periods of persistent wind direction is indicated. From these data the amount of persistence applicable to an analysis can be chosen on the basis of the probability level deemed appropriate.

Subsequent to choosing a period of persistent wind direction, a representative value of $\sigma_{\theta}\overline{\mu}_{h}$ must be selected. A sample of the distribution of this parameter for three time periods is given in Figure D-5. These data are solely for daily periods of inversion observed during an entire year. Additionally, these data are the minimum values observed in each 24-hour day during the time increment indicated. It is considered that a similar analysis for non-inversion conditions (neutral or unstable) would not be markedly different from the one described. Therefore, use of these data would seem to give a reasonable indication of the over-all distribution of the parameter desired.

4.3 Cloud Depletion

In Equation (D-2) it will be observed that there is a term accounting for depletion of the cloud contents due to prior deposition on the ground. Within this equation is inclusion of the effect of vertical wind speed variations (wind shear). This is used primarily in calculations for elevated release of a cloud where a significant vertical shear may exist. The ratio of wind speed at any height compared to the ground level speed is calculated using a logarithmic profile as in Equation (D-15).

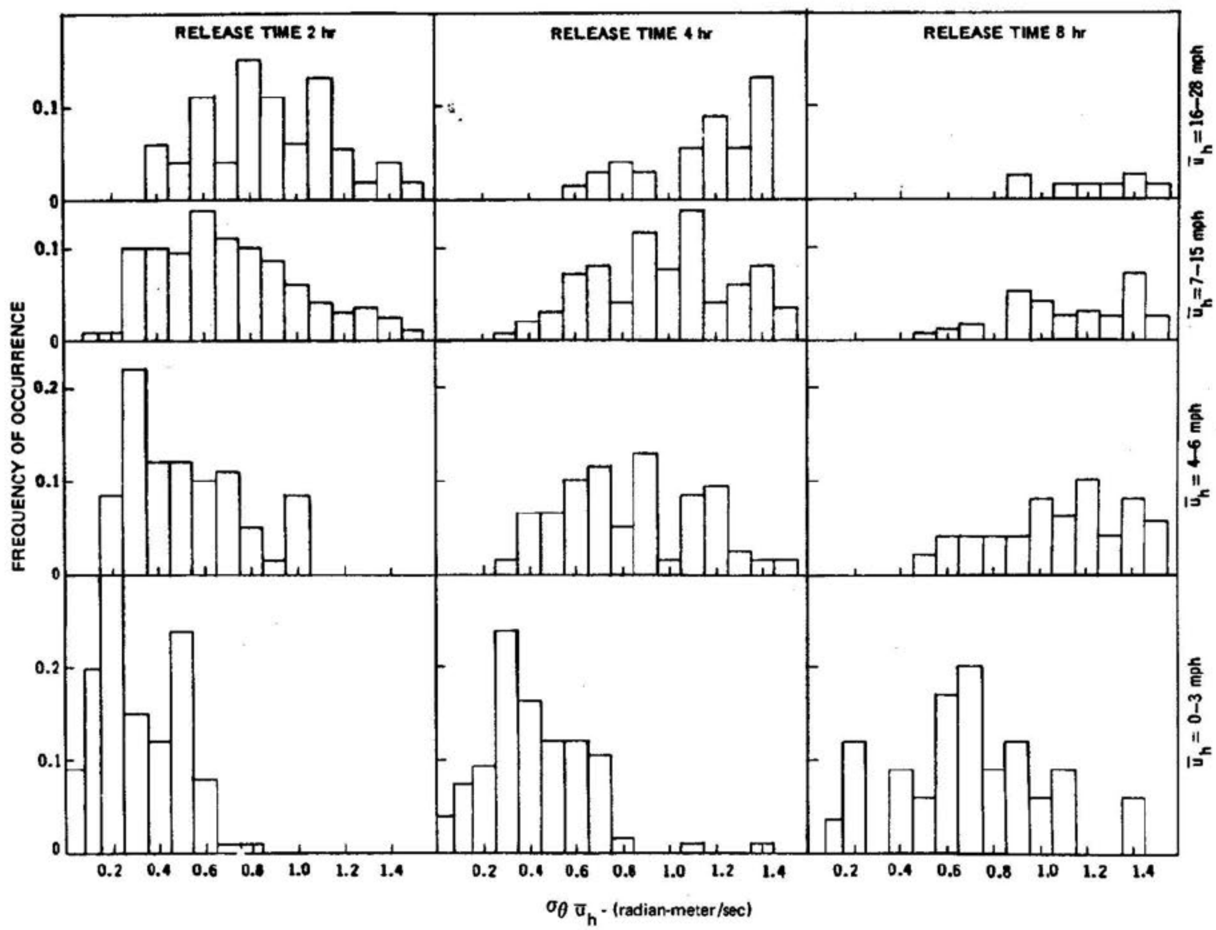

Figure D-5.

$$\overline{U}_0 = \overline{U}_h \frac{(1 - \ln h)}{R} \quad , \qquad \text{(D-15)}$$

where:

h	=	Height of cloud centerline (release height); and
R	=	Constant dependent on stability (see Table D-1).

4.4 Sample Calculation

A sample calculation is described for purposes of completing the discussion of the methods presented here.

Assumptions:

1. Quantities of materials released are:

   A. Noble gases - 1 curie $E\gamma$ = 0.65 MeV,

   λ = 1×10^{-4} sec^{-1}, and

   B. Halogens - 1 curie I-131.

2. Release period of 2 hrs.

3. Release height is 100 m (stack height).

4. Meteorological conditions are:

   A. Inversion (moderately stable);

   B. Wind speed at release height - 1 m/sec or 2 mph (about 12% chance of this for <u>any one</u> hour, from Table D-7);

   C. Wind direction is persistent during release (50% chance of this from Table D-6); and

   D. $\sigma_\theta \bar{\mu}_h$ = 0.1 radian-meters/sec (30% chance of this value or lower during 0-2 mph wind speed).

5. Radiation effects to be calculated:

   A. Dose point 1600 m (1 mile) downwind; and

   B. Passing cloud, lifetime thyroid and fallout doses to be estimated for a person standing at ground level under the cloud centerline during total time of cloud passage (2 hrs).

Calculations:

1. Using Equation (D-2) for the noble gases:

   (X) = 1.5×10^{-8} μCi-sec/cc at 1600 m,
   σ_y = 140 m, and
   σ_z = 25 m.

2. Integration of Equation (D-9)[11] gives a passing cloud dose of 1.0×10^{-6} rad.

3. Using Equation (D-2) for the halogens at 1600 m:

(X)	=	1.5×10^{-8} μCi-sec/cc,
σ_y	=	140 m, and
σ_z	=	25 m.

From Equation (D-11):

Q_i = $230 \times 1.5 \times 10^{-8} = 3.45 \times 10^{-6}$ μCi inhaled.

From Table D-3:

(k) for I-131 = 1.48 Rem/μCi inhaled

Therefore, the lifetime thyroid dose is:

D_i = $3.45 \times 10^{-6} \times 1.48 = 5.1 \times 10^{6}$ Rem.

From Equation (D-13):

$$D_f = (X)V_d R \frac{\left(1 - e^{-\lambda t}\right)}{\lambda},$$

where :

(X) = 1.5×10^{-8} μCi-sec/cc (or Ci-sec/m^3);

V_d = 3.4×10^{-3} m/sec (Table D-1);

R = 7.0 rad/h per Ci/m^2 (Figure D-2; and

$$\lambda = \frac{0.693}{t_{1/2}} = \frac{0.693}{8.05 \times 86{,}400} = 9.9 \times 10^{-7}\ \text{sec}^{-1}$$ (Table D-3).

Therefore;

$$D_f = 1.5 \times 10^{-8} \times 3.4 \times 10^{-3} \times \frac{7}{3600} \times \frac{\left(1 - e^{-7200 \times 9.9 \times 10^{-7}}\right)}{9.9 \times 10^{-7}}$$

= 7.1×10^{-7} rad.

TABLE D-6
WIND DIRECTION PERSISTENCE
(One Sector = 22 1/2 degrees)

Station	Direction[b]	Frequency of Duration in Hours[a] 50%	10%	1%	0.1%	Longest # Hours	Longest # Hours[c] In Any Direction	
Augusta, Georgia	W	2	3	8	13	18	W	18
Birmingham, Alabama	S	2	4	9	16	16	SSE	20
Chicago, Illinois	SSW	2	5	12	21	22	NNE	25
Little Rock, Arkansas	SSW	2	4	9	17	28	SSE	28
Phoenix, Arizona	E	2	3	6	9	12	E	12
Rochester, New York	WSW	2	6	13	23	28	WSW	28
Salt Lake City, Utah	SSE	2	4	7	13	15	S	17
San Diego, California	NW	2	6	12	16	17	WNW	33
Tampa, Florida	ENE	2	3	7	13	14	SSW	18
Yakima, Washington	W	2	5	8	14	17	WNW	19
Average	--	2	4	9	15	--	--	--

a The numbers should be read as follows: Augusta, Georgia (1) 50% of the hours are the beginning of a wind direction persistence period of at least 2 hours duration; 50% of less than 2 hours duration; (2) 10% of the hours are the beginning of a wind persistence period of at least 3 hours duration; 90% of less than 3 hours; (3) 1% of the hours are the beginning of a wind direction persistence period of at least 8 hours duration; 99% of less than 8 hours, etc. The data are standard Weather Bureau hourly observations (one observation per hour) so no time periods less than one hour are distinguishable, i.e., 100% of the hours are beginning of a wind direction persistence period of at least 1 hour. Persistence of direction is defined as within a sector of 22 1/2 degrees are centered on direction indicated.

b Direction examined is the one showing greatest frequency of persistent winds.

c Longest number of hours observed may not be same direction as direction showing most frequency of persistent winds.

TABLE D-7
WIND SPEED FREQUENCY[a]
(From U.S. Weather Bureau Data)

Site	Wind Speed (mph) 0-3[b]	4-7	8-12	13-18	19-24	25
Albany, New York	23	24	27	21	4	1
Chicago, Illinois	7	26	36	25	5	1
Jacksonville, Florida	10	33	35	18	3	1
Kansas City, Missouri	7	25	37	25	6	1
Los Angeles, California	28	33	27	11	1	1
Miami, Florida	14	30	34	20	2	1
New York, New York	6	15	30	31	12	5
Philadelphia, Pennsylvania	11	27	35	21	5	1
Springfield, Missouri	4	13	34	32	13	4
Tulsa, Oklahoma	9	24	34	26	7	1
Average	12	25	33	23	6	1

[a] Frequency of total time is represented, e.g., Albany, New York, 24% of the time the wind speed is 4 - 7 mph, etc.

[b] The data used are referred to as ground-level wind measurements with actual height of measurement varied from about 20 feet to 95 feet.

5.0 CONCLUSION

A method of estimating ground-level doses from a cloud of airborne radioactive materials has been described and a sample calculation is included for completeness. It has been assumed that the standard Gaussian diffusion equations describe the cloud dispersion. Situations where topographic or nearby manmade structures could have significant effects on the cloud were not considered. Special calculations should be used for such situations.
At locations where contemplated construction or operation of a facility includes a need to estimate environmental effects, the method described here may be used. Generally, the method lends itself to simple hand calculations. The exception is the passing-cloud dose calculation, which requires numerical integration. A digital computer program can perform such integrations and is recommended.

6.0 REFERENCES

1 Originally Appendix D, NEDO-10178, Safety Analysis Report, Midwest Recovery Plant, Morris, Illinois (Docket 50-268). Figure numbers, table numbers, and other identification within this appendix are those of the original document.

2 For radiation dose calculations, the time integrated $\frac{\mu Ci-\sec}{cc}$ air Concentration air concentration is of interest since dose rather than dose rate is calculated.

3 Simpson, C. L. Fuquay, J. J., and Hinds, W. T., "Forecasting Dispersion From a Source Near the Ground, "HW-SA-3192 (January 29, 1964).

4 Watson, E. C., and Gamertsfelder, C. C., "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria, "HW-SA-2809 (February 1963).

5 Fuquay, J. J., Simpson, C. L., and Hinds, W. T., "Prediction of Environmental Exposures from Sources Near the Ground Based on Hanford Experimental Data," Journal of Applied Meteorology, Volume 3, No. 6 (December 1964).

6 Glasstone, S., and Sesonski, A., "Nuclear Reactor Engineering, "D. Van Nostrand Co. (1963).

7 "Meteorology and Atomic Energy," AECU-3066.

8 "Report of Committee II (ICRP) on Permissable Dose for Internal Radiation" (1959).

9 "Meteorology and Atomic Energy," revised, to be published.

10 Moses, H., Strom, G. J., and Carson, J. E., "Effects of Meteorological and Engineering Factors on Stack Plume Rise," Nuclear Safety, Vol. 6, No. 1 (Fall, 1964).

11 A digital computer program was used for this calculation.

A.5 ATMOSPHERIC DIFFUSION CALCULATIONS

The atmospheric diffusion methods reported by Comply were used as a basis for these calculations.

Comply has four screening levels. In Level 1, the simplest level, only the quantity of radioactive material possessed during the monitoring period is entered. The calculations are based on generic parameters. Level 4 produces a more representative dose estimate by providing for more complete treatment of air dispersion by requiring site-specific information. GEMO demonstrates compliance at Level 4.

Using this method, the maximum off-site whole dose rate for GE-MO the past five years was calculated to be the following:

2019*	3.4 x 10-6 mRem/year
2018*	9.8 x 10-6 mRem/year
2017	3.1 x 10-7 mRem/year
2016	1.2 x 10-7 mRem/year
2015	1.8 x 10-7 mRem/year

* Changed method for obtaining Title 61 analysis. Using most conservative results, however, still extremely negligible.

A.6 FLOOD POTENTIAL - ELEVATION/DISCHARGE CURVE DES PLAINES AND KANKAKEE RIVERS

The following is a summary of an analysis of the Morris Operation (GEH-MO) site, and its vicinity, for susceptibility to severe flooding at flow rates of up to 600,000 cf/s. This study was originally performed as a result of a question asked by USAEC during evaluation of NEDO-10178, **Safety Analysis Report - Midwest Fuel Recovery Plant,** dated December 1970.

The Harza Engineering Company of Chicago, Illinois was engaged to develop preliminary water level-discharge rating curves for discharges up to 600,000 cfs as specified in USAEC questions, even though the maximum flood of record at the site is less than 100,000 cfs (See figure A.6-1). No studies were made to determine the discharge for the maximum probable flood at the site. However, as shown by the preliminary analysis, even at the discharge rating of 600,000 cfs, the maximum water level is still below the plant site elevation of 530 ft. (mean sea level). Thus, there will be no serious flood effects of safety significance at the GEH-MO.

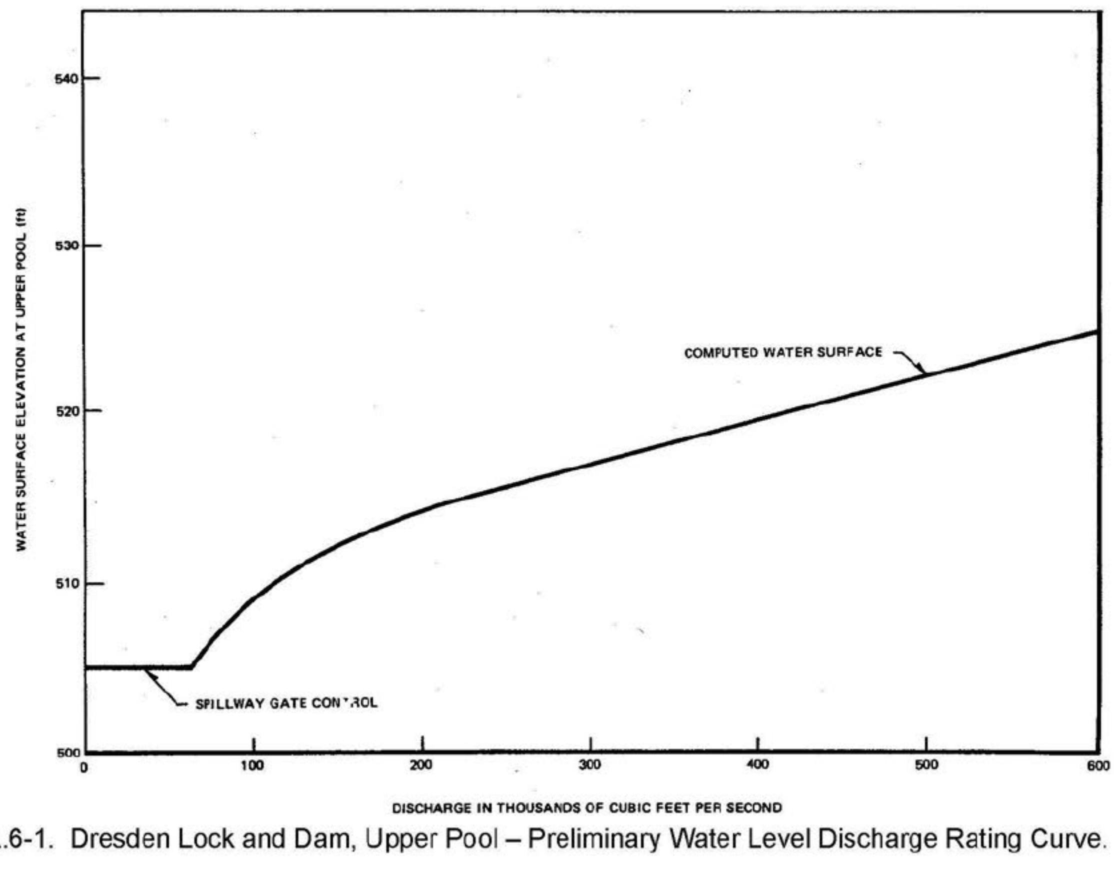

Figure A.6-1. Dresden Lock and Dam, Upper Pool – Preliminary Water Level Discharge Rating Curve.

The hydraulic analyses performed to determine the water levels for extreme and intermediate discharges were based on available topographic and hydraulic information. The analyses were limited to river and overbank cross sections in the vicinity of the plant site.

Method of Analysis. The direct step method was used for computing water surface profiles for selected discharges, floodway geometry and roughness coefficient. Computations were executed on an IBM 1130 computer using a Corps of Engineers program for computing water surface profiles. This program, used for 6 to 8 years, has been used in evaluating other sites for nuclear facilities.

Cross Sections. A total of 13 cross sections was selected in an 8-mile reach between the Morris Highway Bridge (route 47) and the Dresden Lock and dam Pool as shown on Figure A.6-2 attached. A section just upstream of the lock and dam passes through the plant site. At each cross section, channel and overbank geometries were determined from Illinois Water Charts prepared by the U.S. Army Corps of Engineers. Overbanks were described using USGS 7.5 ft. quadrangles which have 5 ft. contour intervals except for one map which has 10 ft. contour intervals. More refined definition of the overbank sections was not believed warranted for this preliminary study. Points in the cross sections were described at each major break in the side slope so that subareas computed by assuming trapezoidal sections would not differ from the true areas by a significant amount.

Roughness Coefficients. Roughness coefficients were established from photo interpretations, a reconnaissance of the area, and calibration runs of a recorded flood profile. The July 1957 flood profile for the study obtained from gage readings at Morris just below the Route 47 Bridge and below the Dresden Dam was reproduced by estimating "n" values and determining the backwater curves for the observed discharge. The "n" values were adjusted until a good reproduction of the flood profile was obtained. Roughness coefficients of 0.070 for overbank and 0.032 for the channel were determined from approximately 95,000 cfs discharge during the 1957 flood.

Starting Evaluation. For each selected discharge, critical depth was determined at the Morris Bridge section. Water surface profiles were then determined up to the Dresden Pool section starting from critical depth at the lower section. Start elevations were then determined by extrapolation from the slope of the upstream water surface. Water surface profiles were again computed using these starting elevations. Since the elevation change at the upstream section was not great after recomputing the profiles (1.5 feet maximum) it was concluded that a new starting elevation based on a new extrapolation would not materially affect the results.

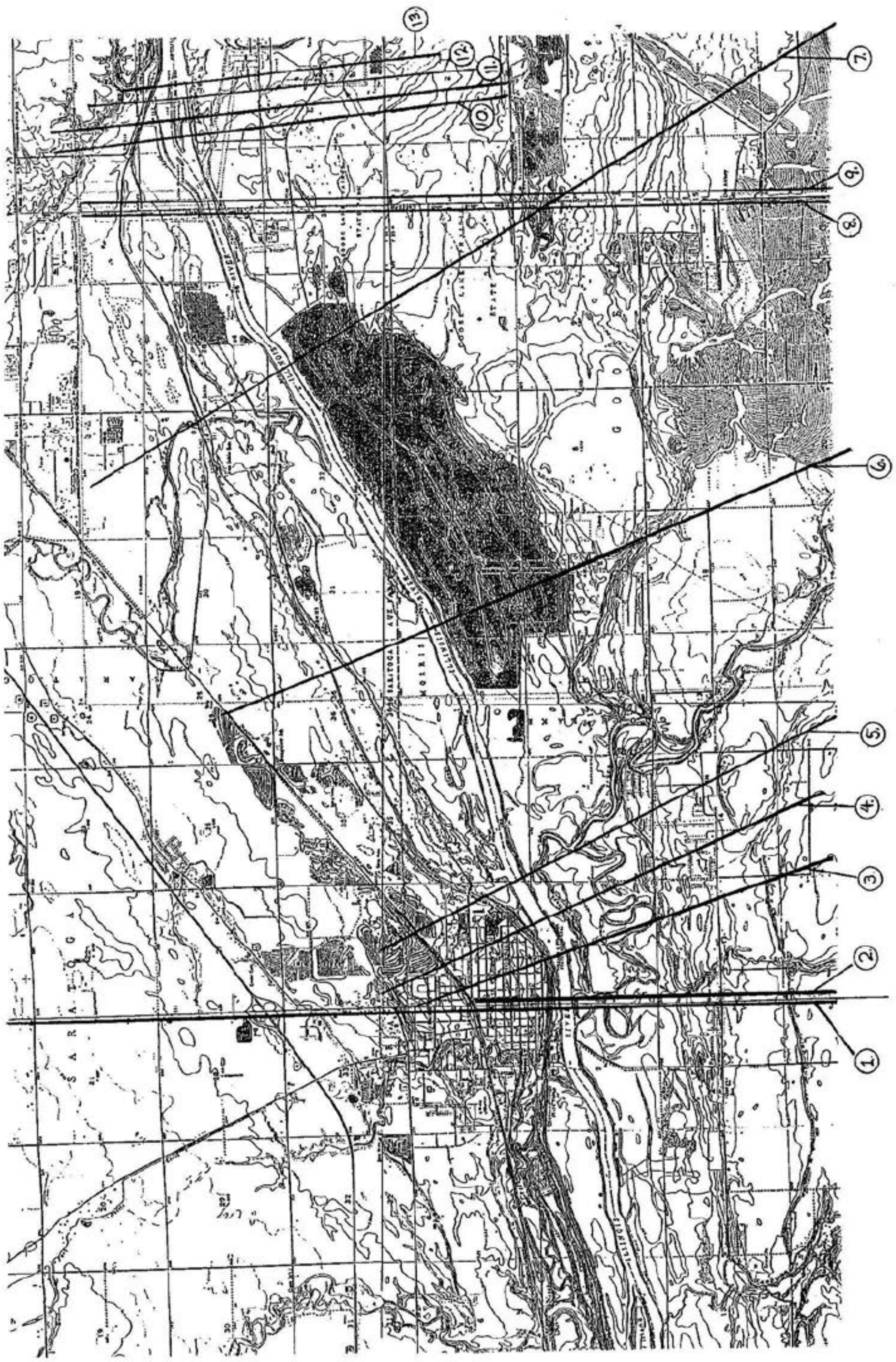

Figure A.6-2. GE-Morris Operation Site Study Reach.

Water Surface Profiles. Water surface profiles were determined for four discharges: 100,000 cfs, 200,000 cfs, 400,000 cfs and 600,000 cfs. Below about 100,000 cfs the water surface just above the dam is controlled by gate operations. Profiles for the four discharges are shown on Figure A.6-3. The profiles are shown for the two starting elevations.

Rating Curve. The water surface elevations computed at the Dresden Pool section for the four selected discharges were used to define the preliminary rating curve at the plant site. Elevations for other discharge were interpolated between the computed values.

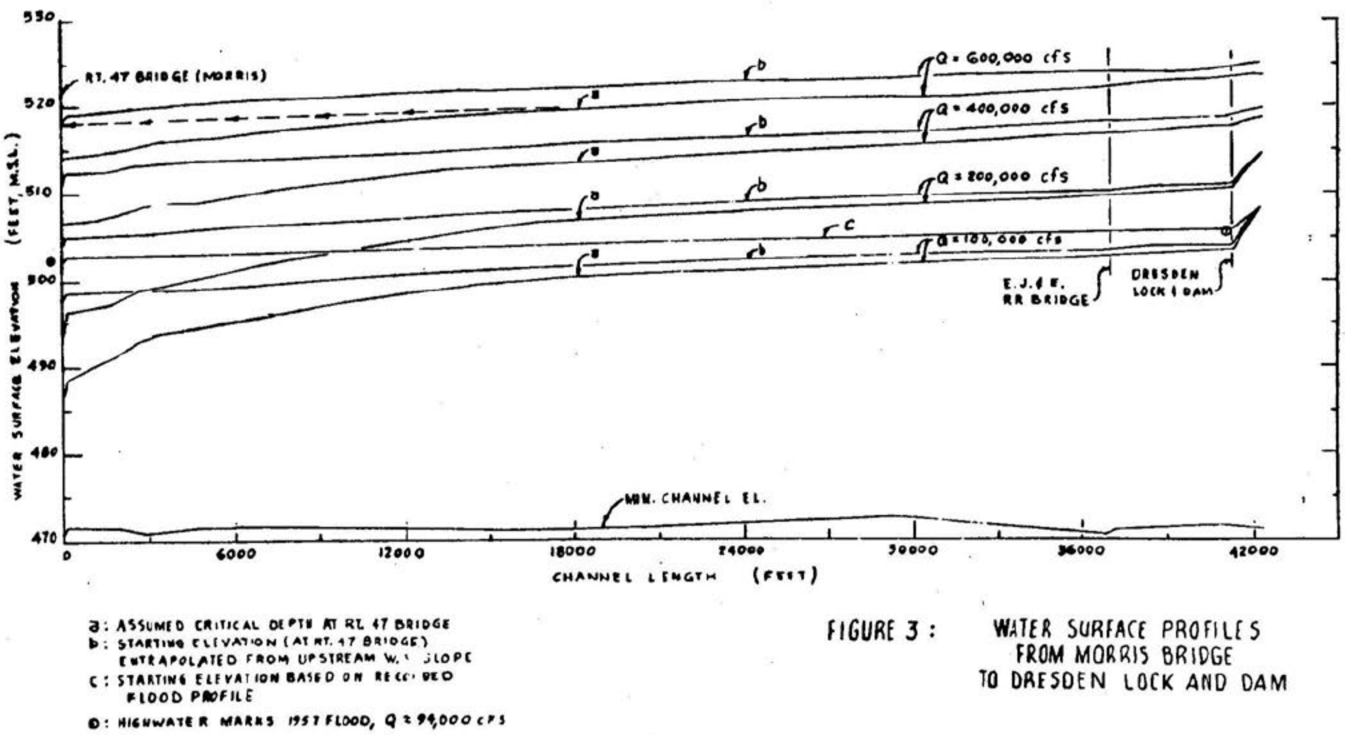

Figure A.6-3. Water Surface Profiles from Morris Beidge to Dresden Lock and Dam.

A.7 DECOMMISSIONING PLAN

A.7.1 INTRODUCTION

A.7.1.1 Purpose and Scope of Plan

This plan describes the method selected by GE for decommissioning of the GEH-MO site:

- The plan addresses GEH-MO decommissioning activities until the GEH-MO operating license is terminated.
- The plan applies to the entire GEH-MO site and is independent of subsequent utilization of the property.
- The plan considers what is currently technically feasible, assuming present regulations and conditions.
- The plan allows for revision or replacement of concepts as more data are obtained and improved technologies developed.

A.7.1.2 History of Operations

The GEH-MO facility was originally constructed to reprocess spent nuclear fuel and was named Midwest Fuel Recovery Plant (MFRP). The MFRP configuration included two water-filled storage basins - one for spent fuel storage prior to reprocessing and one for storage of high-level waste.

Startup testing operations pursuant to the then existing terms of SNM-1265 resulted in the contamination of certain process systems and canyon cells with unirradiated natural uranium and its daughter products. Startup testing was discontinued in late 1974 and the terms of SNM-1265 were changed to allow "storage only" of irradiated fuel.

Irradiated fuel was first received in early 1972 and receipts continued into 1989. Fuel storage capacity was increased twice as the need arose. First, the original waste storage basin was utilized by the addition of fuel storage racks in 1973. In 1975, removal of the original storage baskets and racks and installation of higher density baskets with a supporting grid system in both basins expanded capacity from approximately 100 tonnes to 750 tonnes.

The Low Activity Waste (LAW) Vault, Cladding Vault, Dry Chemical Vault (DCV), low-level waste evaporator system and the plant ventilation system, including the air tunnel, sand filter, exhaust blowers and stack are or were utilized in support of fuel storage operations. As a result, these systems contain varying levels of fission /and activation product contamination from fuel cladding leaks and reactor piping residue (crud) in addition to small quantities of unirradiated natural uranium and its daughter products from the startup testing operations.

A layaway program was initiated in February 1975 to place reprocessing equipment, instruments and certain facilities in protective status to minimize deterioration. Concurrent with fuel receipt and storage operations, procedures were developed and implemented to flush and purge vessels and piping to "mothball" mechanical and electrical equipment. As of December 1978, all reprocessing equipment is in layaway status at the site except for the uncontaminated fluorine production equipment, which was sold and has been removed.

In 1993, a decision was made to curtail further use of the three underground vaults and to commence emptying and disposal of their contents. As of October 1996, all three vaults are empty, dry and contain radioactivity only as contamination on the floors and walls.

The LAW Vault connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the LAW Vault. The DCV connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the DCV. The Cladding Vault is empty, dry, has been cleaned, and contains only low-level residual radioactive contamination on interior surfaces. CRA and CSF drains, which previously went to the Cladding Vault have been capped. Stack drain has been routed to the stack condensate system. This vault is being held available on a contingency basis.

A.7.2 PLAN ASSUMPTIONS AND BASES

A.7.2.1 Site Status

This decommissioning plan is based on the following assumptions:

- Off-site transfer of stored fuel will be completed by normal operating procedures rather than as a part of decommissioning efforts.
- The decision to terminate licensed operations at the site will be made in the course of normal (not emergency) business considerations.
- There is no plan for subsequent utilization of the site for nuclear activity requiring USNRC licensing.

A.7.2.2 Performance Objectives

The primary objective of the plan is to decontaminate the site to a point where continued USNRC licensing is no longer required. The following are supporting objectives:

- Reduce levels of residual contamination on exposed surfaces of site structures and components to permit unrestricted use or:

  a. Remove the contaminated surface from the site for authorized disposition.
  b. Apply surface covering (paint, etc.) only if contamination levels are as low as can be obtained by reasonable effort, or if such action is approved by regulatory authority.

- Remove piping, ducting and vessels for authorized salvage or disposal if their interior surfaces cannot be ensured of meeting unrestricted release limits.

- Dispose of scrap, rubble and other waste materials from site clean-up operations in accordance with applicable provisions of the Code of Federal Regulations, 10 CFR 72.130.

A.7.2.3 Other Considerations

Physical security requirements will be revised after the fuel has been shipped off-site. Access control and other protective measures will be maintained pursuant to regulatory requirements.

A.7.3 PLANNED TASKS

A.7.3.1 Radiation Survey

The first step in the decommissioning plan will be to prepare a comprehensive contamination survey of all site facilities, including the following:

- Main building - all areas
- LAW, Clad, and Dry Chemical Vaults
- Other buildings - utility service, CSF, warehouses, shop/warehouse and administration
- Grounds - walkways, asphalt driveways, gravel areas and ponds

The survey will determine the presence or absence of contamination and where present, the level of smearable and fixed contamination for comparison to unrestricted release limits. Samples of vault contents (if not empty) will be taken to determine bulk waste activity. The results of this survey will be analyzed to determine those structures, equipment, soil and bulk waste that are contaminated above unrestricted release limits and will establish the basis for preparing the final details of the decommissioning plan.

A.7.3.2 Supplementary Systems

Supplementary systems and equipment with temporary or mobile features may be utilized for special functions, such as aggressive surface decontamination, treatment of radioactive liquids, retrieval of bulk contaminated wastes and packaging of consolidated residues. The types, functions and amount of this equipment will be determined at the time of decommissioning.

A.7.3.3 Bulk Materials Removed

A.7.3.3.1 Waste Vault Contents

Removal of LAW Vault contents is complete (except for radioactive contamination) as of October 1996. The LAW Vault connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the LAW Vault.

The dry chemical vault (DCV) contained approximately 30,000 lb. of solid materials including alumina contaminated with unirradiated natural uranium. This material was retrieved from the vault and has been shipped to a disposal facility. The DCV connecting piping has been removed or capped, and the vault is laid away. There are no current plans for use of the DCV.

The Cladding Vault is empty, dry, has been cleaned, and contains only low-level residual radioactive contamination on interior surfaces. CRA and CSF drains, which previously went to the Cladding Vault have been capped. Stack drain has been routed to the stack condensate system. This vault is being held available on a contingency basis.

A.7.3.3.2 Contaminated Equipment

Preparation of empty fuel storage baskets and grids for removal from the basin may include vacuum cleaning and rinsing with water. After removal, cutting (with equipment such as a plasma torch) in a controlled area will be done as needed to facilitate fitting the components into containers for shipment to an off-site disposal facility. GEH-MO gained experience in basket and rack decontamination and disposal as part of a storage capacity expansion project undertaken in 1975. Underwater cutting using divers is an alternative that will be considered.

Also anticipated is removal of contaminated equipment is disposal of canyon vessels. Consideration will be given to selling equipment contaminated with natural uranium to licensed facilities or salvage operators. Otherwise, the equipment will be cut into appropriate sizes for off-site burial. Most of the approximately 40 major canyon vessels were designed to be remotely removable. Thus, the cutting operation for the vessels and equipment may be performed in place or in a convenient location such as on top of the mechanical cell covers, controlled ventilation and services are available in either case. Advanced planning will be utilized to minimize equipment cutting. Internal residual contamination of the canyon vessels is minimal due to the layaway flushing (described in Section A.7.1.2) that they received.

A.7.3.4. Residual Contamination Survey/Assessment

The contamination survey described in Section A.7.3.1 will be updated following the removal of bulk materials as appropriate. The survey update will determine the location, level and type of residual contamination. Subsequent assessments determine where additional decontamination is required.

Tests of proposed decontamination methods at this time will indicate modifications needed in order to meet the performance objectives set forth in Section A.7.2.2 above.

A.7.3.5 System Decontamination and Dismantling

A.7.3.5.1 Fuel Receipt and Storage Facilities

The fuel receipt and storage facilities include the cask receiving area (CRA), decontamination area (BDP), cask unloading pit, fuel storage basins 1 and 2, basin filter room, basin pump room (BPR), basin chiller heat exchangers and associated structures. The plan for these areas is to:

- Remove basin water, remove stainless liner and piping and survey concrete surfaces for contamination levels. Decontaminate concrete surfaces as required. Backfill basins and provide a cover over them.
- Remove contaminated equipment and piping from the BPR, filter room and chiller room.
- Decontaminate imbedded piping and fill with grout.
- Remove cranes and other equipment (the cask crane may be used for loading off-site shipments and removal may be deferred until later in the decommissioning work period).
- Decontaminate (or raze) the filter room and BPR structures.
- Decontaminate the concrete floor pads and other surfaces, or remove surfaces if necessary to achieve performance objectives.
- Decontaminate the CRA and BDP areas (these areas will be used for vehicle loading and other needs during most of the decommissioning period and this task will be scheduled later).
- Clean or package, as necessary, other contaminated structural components, walls, ceilings, etc.
- Package and ship contaminated waste to off-site disposal facilities.

A.7.3.5.2 Canyon

The plan for the canyon cells is to:

- Remove all fixed piping (other than imbedded) and instrument and electric cables.
- Decontaminate all surfaces. Remove stainless cell liners if the performance objectives (Section A.7.2.2) cannot be met with them in place.
- Decontaminate or package canyon cell covers and the canyon crane.
- Decontaminate imbedded piping and fill with grout.
- Leave the main building concrete structure including the canyon area in place after decontamination.
- Package and ship contaminated waste to off-site disposal facilities.

A.7.3.5.3 Other Main Building Areas

Several areas are not used for fuel storage operations. Other areas used during fuel storage operations may be minimally contaminated. The plan for these areas is to:

- Remove contaminated equipment.
- Remove and package other contaminated items such as instruments, piping ducts, services.
- Decontaminate area surfaces with techniques employed in the canyon cells.
- Package and ship contaminated materials to off-site disposal facilities.

A.7.3.5.4 Waste Storage Vaults

It is anticipated that minimal contamination (principally natural uranium) remains in the DCV. Assuming successful decontamination, the DCV will be backfilled with dirt and sealed, leaving the concrete walls and liners intact.

Radioactive contamination in the LAW and Cladding Vault consists almost exclusively of radiocobalt, radiocesium and radionickel. The plan for these vaults is to:

- Investigate the feasibility of further decontamination of inner walls.
- Backfill and seal the maintenance pit and off-gas cell openings, leaving the walls, inner tank and liners intact.
- If residual contamination levels prove unacceptable, the inner tank/liner shall be removed and shipped for burial or metal melt.

These structures will be decommissioned last permitting use of main building ventilation for the majority of the decommissioning work. The plan for these structures is to:

- Flush the floor of the air tunnel. Route the flush solution to the radwaste system.
- Either fill the air tunnel with concrete over its entire length or decontaminate to acceptable limits and fill. Seal the cell openings to the air tunnel.
- Remove the exhaust blowers and duct work located next to the sand filter.
- Remove the contaminated sand and gravel from the sand filter as required. Package it and ship it to an off-site disposal facility.
- Decontaminate and backfill the sand filter concrete structure and seal the filter openings.
- Decontaminate and backfill or package the horizontal duct between the sand filter and stack.
- Decontaminate and cap (ground level and top) or dispose of the 300 ft. stack.
- Package and ship contaminated materials to an off-site disposal facility.

A.7.3.5.6 Final Waste Removal

The remaining items to be considered are:

- Decontaminate potentially contaminated underground piping and fill with grout or dig up and package for disposal.
- Decontaminate special equipment used in decommissioning work and package for disposal.
- Package miscellaneous tools, which are no longer useful for disposal.

A.7.3.6 Final Survey

A comprehensive final survey similar to the initial one described in Section A.7.3.1 will be performed. The survey report will include:

- Description of scope and general procedures used in the survey.
- Description of remaining contamination.
- Results of survey for comparison with performance objectives.
- Surveillance recommendations and future use restrictions.

A.7.3.7 Inspection and Acceptance

A final survey report will be submitted to the USNRC.

It is anticipated that the USNRC will terminate Materials License No. SNM-2500 and release the facility for unrestricted use following their review and inspection.

A.7.4 PLAN ENVIRONMENTAL EFFECTS

A.7.4.1 Balancing of Effects

The decommissioning plan described in this document presents what is believed to be the most balanced approach to limiting environmental effects as they relate to potential risks to the public and site personnel. In summary, the approach involves evaluating each task of the plan at the time of implementation, and making the final decision for disposition based on a comparison of the alternatives below:

- Decontamination to unrestricted limits
- Removal to off-site disposal facilities
- Fixation and isolation

This approach ensures an optimization of effects.

A.7.4.2 Conclusions

Dispersal of significant radioactivity as a result of the implementation of this plan is highly unlikely. The main building ventilation system will be operated to provide normal filtration of particulate and aerosol matter. There are no radioactive liquid effluents from the site during normal license operations and there will be none during decommissioning activities. Radioactive wastes will be disposed of by transporting to licensed repositories in approved containers. Approved shipping practices shall be followed, thereby creating no significant impact on the environment.

After the performance objectives of the plan have been attained, the site will be available for unrestricted use with no impact on the environment.

A.7.5 RESOURCE REQUIREMENTS

A.7.5.1 Manpower Estimates

General Electric will carry out the specific tasks defined in Section A.7.3, utilizing Company personnel, contractor personnel, or a combination of both. Table A.7-1 depicts cost estimates for the various tasks using General Electric 2003 manpower rates for the on-site work. The removal of vault bulk materials was assumed to be carried out by subcontractors.

In estimating manpower requirements, it is anticipated that total implementation of the decommissioning plan will take 3 years. Some tasks will be performed in parallel but the general sequence of tasks is that described in Section A.7.1.

A.7.5.2 Shipping and Disposal Costs

Shipping and burial cost estimates include 1996 costs of shipping containers (nonreusable), transportation fees, and burial charges at a low-level waste disposal site. The cost estimate includes weights and volumes of materials based on past experience of GEH-MO. The transportation costs assume that the waste will be transported to specified out of state disposal facilities.

Disposal of "clean" materials is not included in the costs shown in Table A.7-1 since noncontaminated items are not addressed in this plan. (See Section A.7.2.2.)

A contingency of 25% of the decommissioning cost (Table A.7-1) was included in the total cost shown.

A.7.5.3 Certification of Financial Assurance

A payment surety bond was established to provide decommissioning financial assurance that was submitted to the NRC by letter dated March 27, 2018, Subject: GEH/GNF-A Financial Assurance of Decommissioning Funds – Surety Bonds. (ADAMS Accession Number ML18087A172). Supplemental Riders to increase the bond amounts were submitted to the NRC by letter dated March 19, 2019 Subject: GEH/GNF-A Financial Assurance of Decommissioning Funds – Surety Bond Riders (ADAMS Accession Number ML19080A060).

In addition, GEH has established a Master Standby Trust Agreement with the Bank of New York Mellon for the benefit of the NRC in the case of default or inability to direct decommissioning activities by GEH. An amended standby trust agreement was submitted to the NRC by letter dated April 5, 2018, Subject: GEH/GNF-A Financial Assurance of Decommissioning Funds – Standby Trust Agreement Amendments" (ADAMS Accession Number ML18096A036

A.8 AGING MANAGEMENT

This appendix provides a summarized description of the activities for managing the effects of aging at GEH-MO. The evaluations of time-limited aging analyses (TLAAs) for the renewal period are also presented.

An assessment of the GEH-MO inspection activities identified new and existing activities necessary to provide reasonable assurance that Systems, Structures, and Components (SSC) within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the renewal period. This section describes these aging management activities.

This section also discusses the evaluation results for each of the applicable SSCs specific time-limited aging analyses (TLAAs) performed for license renewal. The evaluations have demonstrated that the analyses remain valid for the renewal period; the analyses have been projected to the end of the renewal period; or that the effects of aging on the intended function(s) will be adequately managed for the renewal period.

GEH-MO is an away from reactor ISFSI storing spent fuel under 10CFR72 license until such time that the fuel may be shipped off-site for final disposition. The fuel storage basins at GE-MO are designed for below grade storage. Accordingly, the exterior materials can withstand the anticipated effects of "weathering" under normal conditions.

Structures, systems and components at GEH-MO that, while not performing a safety-related function, but do perform a function that demonstrates compliance with NRC regulations on environmental qualification, are identified in the CSAR, section 11, paragraph 11.3, as follows:

11.3 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY
No credible event, planned discharge, or design basis accident at GEH-MO is identified that would expose a member of the public to radiation in excess of limits specified in 10 CFR 72.104 or 10 CFR 72.106.

It is, therefore, the position of GEH-MO that the term "basic components" in the sense defined by 10 CFR 21.3(a)(2) and 10 CFR 21.3 (m) is not applicable to GEH-MO.

However, "structures systems and components important to safety" as promulgated in 10 CFR 72.122, "Overall Requirements" are identified below.

a. Fuel storage basin (FSB) - concrete walls, floors, and expansion gate are principal elements in protection of stored fuel, and in isolation of basin water from the environment.

b. Fuel storage basin - stainless steel liner forms a second element in fuel protection and basin water isolation, facilitating decontamination.
c. Fuel storage system, including baskets and supporting grids is a principal element in protection of stored fuel.
d. Steel expansion gate – Identified as Gate #4, along the south east corner of fuel basin II. The gate is constructed of reinforced concrete with a thickness of 8" and height and width dimensions of 29'-6" and 5'-0", respectively. The water side of the gate is lined with 16-gauge stainless steel to prevent the reinforced concrete from coming into contact with the water in the basin.
e. Unloading pit doorway guard - is designed to prevent a loaded fuel basket from being tipped so that fuel bundles could fall into the cask-unloading pit. The unloading pit doorway guard is an element in protection of fuel during movement of a loaded basket.
f. Filter cell structure (FCS) - the concrete cell part of the basin pump room area provides radiation shielding to reduce occupational exposure.
g. Fuel Storage Basin building – the steel structure that surrounds/protects the fuel Basins.
h. Fuel Basket Grapple – Used to remove the fuel baskets from their storage location in the fuel basin support grid.
i. Fuel Grapple – Used to remove the fuel bundles from the fuel baskets when they are in the unloading pit.
j. Fuel Basin Crane – Crane utilized to move the full fuel baskets to the unloading pit.
k. Fuel Handling Crane – Crane used to remove the fuel bundles from the fuel storage baskets and place into a cask.
l. Cask Crane – 125 Ton overhead crane used to lift a fully loaded cask from the unloading pit and place cask onto transport vehicle.
m. Spent Fuel Cladding – Fuel in GE-MO basins are clad with SS or zircalloy cladding.

However, since these systems do contain the stored fuel or provide support functions, they have been reviewed for aging management. These SSCs are organized in accordance with NUREG 1801 in Table 1 of this appendix.

STRUCTURES MONITORING AGING MANAGEMENT PROGRAM (AMP)

As identified in Table 1, SSCs involving concrete or structural steel, and necessitating periodic examination, are inspected and monitored according to this Structures Monitoring AMP. AMP elements are consistent with those in XI.S6 from NUREG 1801 Rev. 2 and are as follows:

Scope of Program – Inspection and monitoring of SSCs important to safety ensures there is no loss of function. This is facilitated with periodic examinations and select monitoring in accordance with this AMP. The SSCs identified during the AMR that are covered by this AMP are denoted as "Structures Monitoring" in Table 1.

Preventative Actions – Preventative actions delineated in NUREG-1339 are not applicable to the fuel basin building structure, as bolting used for construction does not form a pressure boundary, is not a reactor internal component, and does not involve high strength >150ksi bolts. Bolted connections used to construct the basin building structure are, however, inspected every 5 years by qualified personnel in accordance with SOP 16-17. Preventative actions for inaccessible portions of concrete and liner structures include maintenance of water chemistry within approved license specifications through continuous filtration and addition of ultra-pure water (typically 0.056 μmho/cm) as needed to maintain basin level (see Water Chemistry AMP).

Parameters Monitored or Inspected – For each structure / aging effect combination designated as "Structures Monitoring" in Table 1, the following parameters are inspected:

Concrete Structures: loss of material, cracking, increase in porosity and permeability, loss of foundation strength, and reduction in concrete anchor capacity due to local concrete degradation.

Steel Structures including Galvanized Steel: loss of material due to corrosion of any kind.

Stainless Steel Basin Liner: evidence of bulging or depressions and leakage rate via leak detection channels.

Structural Bolting: loose bolts, missing or loose nuts, and other conditions indicative of loss of preload. Loss of material due to general, pitting, and crevice corrosion.

Ground Water Quality: Ground water chemistry (pH, chlorides, and sulfates) are monitored periodically to assess its impact, if any, on below grade concrete structures.

Detection of Aging Effects – Aging is detected by periodic visual inspections for each structure / aging effect denoted in Table 1. Parameters are examined every 5 years by qualified inspectors in accordance SOP 16-17. This SOP incorporates relevant sections of ACI 349.3R and suggested parameters from the NUREG 1801, XI.S6 AMP. Additionally, ground water quality is periodically sampled to ensure a non-aggressive environment for inaccessible concrete structures[GB(Pc1)].

The purpose of the GEH-MO Inspection Activities is to:

1. Determine that no significant deterioration of the basin structure has occurred, such that it can still perform its intended function, and

2. Confirm that no significant degradation of the fuel storage components in the basin has occurred.