Environmental Impact Appraisal Related to the Renewal of Materials License SNM-1265 for the Receipt, Storage and Transfer of Spent Fuel

Morris Operation
General Electric Company
Docket No. 70-1308

Office of Nuclear Material Safety
and Safeguards

U.S. Nuclear Regulatory
Commission

ENVIRONMENTAL IMPACT APPRAISAL

related to the

RENEWAL OF MATERIALS LICENSE NO. SNM-1265

for the

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at

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FOREWORD

This environmental impact appraisal was prepared by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards (the staff), in accordance with the Commission's regulations at 10 CFR (Code of Federal Regulations), Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection." The regulations implement the requirements of the National Environmental Policy Act of 1969. The environmental impact appraisal includes:

- a description of the proposed action
- a summary of the probable impact of the proposed action on the environment
- the basis for the conclusion that no environmental impact statement need be prepared.

Single copies of this appraisal may be obtained by writing to the:

Director of the Division of Fuel Cycle
and Material Safety
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. A. Thomas Clark, Jr., is the NRC Environmental Project Manager on this project. Dr. Clark may be contacted at the above address or at (301) 427-4205.

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1.0 INTRODUCTION

1.1 PROPOSED LICENSE RENEWAL

By letter dated February 27, 1979, the General Electric Company (licensee or GE) applied pursuant to 10 CFR Parts 30, 40, and 70, for renewal of Materials License No. SNM-1265, Docket 70-1308, for a period of twenty (20) years. The license authorizes GE to receive, possess, store and transfer spent nuclear fuel at the General Electric Company Morris Operation (Morris Operation) near Morris, Illinois. This environmental impact appraisal discusses the environmental impacts of the proposed licensing action. The detailed safety aspects of the proposed amendment have been identified separately and will be addressed in the Nuclear Regulatory Commission Staff's Safety Evaluation Report.

1.2 BACKGROUND INFORMATION

General Electric Company's Morris Operation is located 11 km (7 miles) from the city of Morris in Grundy County, Illinois. Also located on the site is the Boiling Water Reactor Training Center (BWRTC). Although both are General Electric operations, the Morris Operation does not include BWRTC activities, and this licensing action does not apply to the BWRTC.

The Morris Operation was originally designed as a spent fuel reprocessing plant and named the Midwest Fuel Recovery Plant (MFRP). The Morris Operation facility has been licensed for the receipt and storage of spent nuclear fuel since December 1971, and storage of spent fuel there began in December 1972. The environmental considerations associated with spent fuel storage were among the environmental considerations associated with the operation of the MFRP.

The General Electric Company submitted an Environmental Report for the MFRP in June 1971 (GE Document No. NEDO-14504, including supplements and amendments). Subsequently, a Final Environmental Statement (FES) relating to the MFRP was issued by the U.S. Atomic Energy Commission (AEC), Fuels and Materials, Directorate of Licensing, in December 1972.

Detailed descriptions of the site location and surrounding environs, including population, geology, hydrology, topography, climatology and meteorology, seismology and background radiation are reported in the FES. It also describes the Morris reprocessing facility including operational activities and effluent monitoring plans for (1) the receipt and storage of spent fuel, and (2) subsequent reprocessing operations to recover residual useful products and to store the radioactive waste. The environmental impact of fuel recovery operations under normal and accident conditions was evaluated quantitatively and the relative environmental costs and benefits weighed.

On August 23, 1974, the then Atomic Energy Commission terminated the MFRP Construction Permit,[a] No. CPCSF-3, but reissued Materials License No. SNM-1265 for receipt and storage of spent nuclear fuel for a full term (5 years). At that time the facility became known as the Morris Operation. In December 1975, Materials License No. SNM-1265 was revised and reissued, permitting the receipt and storage of up to 750 MTU of spent fuel (actual capacity is about 700 MTU). Since storage of fuel at the Morris operation began in 1972, a total of 315 MT of fuel has been received and stored as of April 30, 1980.[1]

1.3 PREPARATION OF THE STAFF'S ENVIRONMENTAL IMPACT APPRAISAL

This environmental impact appraisal relates to the request of the General Electric Company for renewal of Materials License No. SNM-1265 for receipt, storage and transfer of spent nuclear fuel at the Morris Operation. This appraisal of the license renewal request utilizes portions of the data described in Final Environmental Statement related to the Operation of the Midwest Fuel Recovery Plant, (Docket No. 5 -268) dated December 1972,[2] and the Consolidated Safety Analysis Report for the Morris Operation, (Docket No. 70-1308) dated January 1979.[3] Other information in this appraisal was developed by independent review of Operating Experience, Irradiated Fuel Storage, Morris Operation,[4] and the licensee's Summary Environmental Report.[5] Although the staff examined data from all of these sources in

(a) When GE determined that the reprocessing facilities did not operate as anticipated, a decision was made to terminate the construction permit.

detail, only summaries of the most pertinent data are provided in this appraisal. References to the sources of detailed information are cited throughout this appraisal and are listed separately at the end.

As part of its review of the license renewal application, the staff also met with the licensee to discuss information provided and to seek new information that might be needed for an adequate assessment. This was done to ensure that the staff had a thorough knowledge of past operating experience. In addition, the staff sought information from other sources to assist in the evaluation, and visited and inspected the project site and surrounding vicinity. On the basis of all the foregoing and other such activities or inquiries as were deemed useful and appropriate, the staff made an independent assessment of the proposed action. Based on this assessment, which is discussed in the following sections of this document, the staff determined that the impacts associated with continued operation of the facility will not significantly affect the quality of the human environment. Therefore, in accordance with 10 CFR Part 51, the staff prepared this environmental impact appraisal of the proposed licensing action.

Copies of this appraisal are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Local Public Document Room for the Morris facility, which is located at the Morris Public Library, Morris, Illinois.

1.4 STATUS OF LICENSES

The licensee has provided a list of related licenses required from Federal and state agencies in connection with operation of the facility at Morris. These licenses are listed in Table 1.4.1.

1.5 RELATED FEDERAL PROGRAMS

Federal programs and policies related to spent fuel storage are briefly discussed in this section for informational purposes, since the Morris Operation may be under consideration as a Federal away-from-reactor (AFR) storage location for spent fuel.[6] However, no determination has been made as to

TABLE 1.4.1. Licenses Required from Federal and State Agencies

Action	License No.	Issuance Date	Expiration Date
Registration Radiation Installation State of Illinois Department of Public Health	none	8/6/71	none
State of Illinois Environmental Protection Agency Div. of Air Pollution Control	063-806-AAC	4/26/78	4/18/81
Illinois Environmental Protection Agency Water Pollution Control Permit, Evaporation Pond Permit	1979-EO-440	5/11/79	5/1/84
U.S. Nuclear Regulatory Commission	SNM-1265	12/3/75	8/31/79[a]
Materials License Revised and Reissued for Increased Capacity of Facility			
State of Illinois Dept. of Public Health Radioactive Material License Amendment No. 5	IL-00329-01	8/31/79	8/31/80
Illinois Environmental Protection Agency Water Pollution Control Permit, Land Disposal System Permit	1976-EB-408-1	9/17/76	none

(a) Pursuant to 10 CFR§70.33(b), the licensee made a timely filing for renewal of its license. Therefore, in accordance with 10 CFR§70.33(b), the license shall not expire until the application for renewal has been finally determined by the Commission.

whether the Morris Operation will be selected as a Federal AFR storage location and the NRC staff has no basis upon which to predict if and when such a determination would be made. Therefore, this environmental impact appraisal assumes that the Morris Operation will be owned and operated by the licensee for the term covered by the proposed licensing action.

1.5.1 NRC and Department of Energy Environmental Impact Statements on Spent Fuel Storage

In August 1979, a "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" was issued by the Office of Nuclear Material Safety and Safeguards of the NRC.[7] The Generic Environmental Impact Statement (GEIS) on spent fuel storage was prepared by the Nuclear Regulatory Commission staff in response to a directive from the Commissioners published in the Federal Register on September 16,1975 (40 FR 42801). The Commission directed the staff to analyze alternatives for the handling and storage of spent light water reactor (LWR) fuel, with particular emphasis on development of long-range policy. Accordingly, the scope of the GEIS examines alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

Since the Commission's directive was issued, there have been significant policy developments relating to spent fuel storage. In April 1977, President Carter announced that the U.S. would indefinitely defer domestic reprocessing of spent fuel. In light of this development, in December 1977 and May 1978, the NRC terminated its proceedings on the General Environmental Statement on Mixed Oxide Fuel (GESMO), and other matters related to the reprocessing and recycling of spent light water reactor fuel. This policy decision highlights the importance of the GEIS.

The storage of spent fuel addressed in the GEIS is considered to be an interim action, not final disposition of the fuel. The Commission has clearly distinguished between permanent disposal and interim storage.[8] Nonetheless, it has expressed its concern that storage of spent fuel not be used to justify retarding the development of a practicable method of permanent disposal.[9] This concern is shared by groups who have studied this situation.[10, 11] The Commission has initiated and is conducting a generic proceeding to review its basis for confidence that safe waste disposal will be available.[12]

The study documented in the GEIS covers considerations pertinent to the interim storage of spent fuel. The Commission announcement of September 16, 1975, outlining this study stipulated that the staff was to examine the period through the mid-1980's. In the absence of a national policy directed to final disposition of spent fuel, the staff extended the time period of this study to the year 2000. The GEIS concludes that storage of light water reactor (LWR) spent fuel at away-from-reactor (AFR) facilities is economically and environmentally acceptable, and that the storage of LWR spent fuel in water pools has an insignificant effect on the environment whether in at reactor (AR) sites or in AFR sites.(7) The GEIS also concludes that there is an increasing need for AFR spent fuel storage starting in the early to mid-1980's.(7)

In October 1977, the Department of Energy (DOE) announced a Spent Fuel Storage Policy for nuclear power reactors. Under this policy, as approved by the President, U.S. utilities will be given the opportunity to deliver spent fuel to U.S. Government custody in exchange for payment of a fee. Under this policy, spent fuel transferred to the U.S. Government would be delivered at the user's expense to a U.S. Government-approved storage site.(13)

If this policy is implemented, spent fuel storage could be accommodated in either large centralized Independent Spent Fuel Storage Installations (ISFSI) owned or operated by the U.S. Government or decentralized storage in small Government-approved, privately owned ISFSIs. The GE Morris Operation may be under consideration as a Federal AFR storage facility for spent fuel.(6)

Two bills have been introduced in the House of Representatives to implement this policy. One, H.R. 2586, was introduced on March 1, 1979, and the other, H.R. 2611, was introduced on March 5, 1979. Identical bills have been introduced in the Senate.

The DOE policy actions presume continued light water reactor power generation with discharge of spent fuel and Federal government responsibility for the storage and disposition of spent fuel

DOE used the NRC GEIS as a source in their draft generic environmental impact statement on their announced spent fuel policy, "Storage of U.S. Spent Power Reactor Fuel."(14) The DOE GEIS and its supplement(15) analyze the impacts associated with alternative actions resulting from implementation of the spent fuel policy announced in October 1977. Alternatives that are assessed in the statement include: (1) no Federal action and (2) implementation of the policy with (a) centralized storage, and (b) decentralized storage. The effects of these alternatives on transportation are also assessed. The supplement to the DOE GEIS assesses the additional alternative of interim storage of domestic spent fuel in expanded reactor basins at reactor sites. The DOE GEIS concludes that differences in the environmental impacts between the options are small and for all options the associated environmental effects are within existing national standards and guidelines.[14]

1.5.2 National Waste Terminal Storage Program

This program was established in February 1976 and represents the principal programmatic effort of DOE for disposal of commercial nuclear waste or spent fuel in a geologic formation(s). It interfaces with the disposition of spent fuel in DOE's spent fuel policy. The original emphasis of the NWTS program was disposal of wastes from commercial reprocessing facilities. After the President's announcement of a plan to defer commercial reprocessing, the emphasis was shifted to disposal of spent fuel that may be classified as waste and to retrievable storage of spent fuel that may later be reprocessed.[14]

2.0 NEED FOR CONTINUED OPERATION OF THE FACILITY

As previously stated, current U.S. policy has placed a ban on the reprocessing (and recycling) of LWR fuel for an indefinite period of time. In addition, the Commission has terminated the hearings on the Generic Environmental Impact Statement on the Use of Mixed Oxide Fuels in Light Water Cooled Reactors (GESMO). As a consequence, the reprocessing part of the fuel cycle has not been a successful commercial development and there has been a halt to further construction by private companies of facilities to store and reprocess spent fuel from reactors. This course of events, coupled with the Department of Energy's spent fuel policy, under which it would accept ultimate responsibility for storing spent nuclear fuel, has resulted in the situation many utilities face today, a lack of spent fuel storage space.(14)

The Morris Operation has been used to store spent fuel from a number of utilities. As a result, General Electric is storing 315 MTU in the form of spent fuel assemblies, and has contracted to receive an additional 33 MTU of spent fuel. If General Electric's license is not renewed, GE will be faced with the disposition of spent fuel presently stored on site. The alternatives available to GE for disposing of the spent fuel are discussed in Section 11.0. Because of the reasons set forth in that section, there are no other facilities available to take the fuel.

A renewal of Materials License No. SNM-1265 is necessary to provide a means for continued storage of the current spent fuel inventory at the Morris Operation. The facility is licensed to receive 750 MTU. As discussed previously, 315 MTU are in storage and an additional 33 MTU are under contract to be received. Therefore, there is a remaining licensed capacity of approximately 400 MTU. Although GE is not committed to future acceptance of spent fuel beyond the 350 MTU previously discussed, the facility is available to meet emergency needs for the storage of spent fuel. In the event of an imminent shutdown of a nuclear power plant because of a lack of spent fuel storage space, such storage space could be provided by the Morris operation.

While GE is not committed to assist in meeting emergency spent fuel storage needs of the nuclear industry, the recent acceptance of 8 BWR fuel assemblies from LaCrosse Boiling Water Reactor to facilitate expansion of onsite storage capacity indicates how the Morris Operation has been used to alleviate certain fuel storage problems. A second example would be the acceptance of spent fuel to facilitate emergency full-core offloads when a reactor has already lost its full-core reserve.

The need for AFR storage facilities, either Federal or private, has been confirmed by NRC's "Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel,"(7) DOE's "Draft Environmental Impact Statement on Storage of U.S. Spent Power Reactor Fuel,"(14) and by the General Accounting Office study "Federal Facilities for Storing Spent Nuclear Fuel -- Are They Needed."(32)

3.0 THE SITE AND REGION ENVIRONMENTAL DESCRIPTIONS

Descriptions of the Morris Operation Environment are given in this section. These environmental descriptions relate to: site location, land use, water use, demography, background radiology, meteorology and climatology.

3.1 SITE LOCATION

The Morris Operation is located in Grundy County, Illinois. The existing facilities occupy about 21 hectares (ha) (52 acres) at the north edge of 330 ha (815 acres) of land owned by the General Electric Company. The property occupies portions of Section 35, Township 34 North, Range 8 East; and Section 2, Township 33 North, Range 8 East. The nearest major city, Joliet, Illinois, is approximately 25 km (16 mi) northeast of the site.

Notable landmarks in the vicinity of the Morris Operation include towns, rivers, outdoor recreation areas and a highway. The town of Morris is located on the north bank of the Illinois River approximately 11 km (7 mi) west of the storage facility. Other nearby towns include Minooka, 10 km (6 mi) north; Channahon, 6 km (4 mi) northeast; Wilmington, 14 km (9 mi) southeast; and Coal City, 10 km (6 mi) south of the site. The Des Plaines River and Kankakee River join to form the Illinois River about 2 km (1 mi) northeast of the site. A large tract of public land used for outdoor recreation, the Des Plaines Wildlife Conservation area, lies between the Des Plaines and Kankakee Rivers about 3 km (2 mi) east of the site. The area is bordered on the north by Commonwealth Edison Dresden Nuclear Power Station (DNPS); on the west by Goose Lake Prairie State Park, a major portion of which has been designated a nature preserve (Goose Lake Prairie); on the south by A. P. Green Refractory Company; and on the east by the DNPS Cooling Lake 516 ha, (1275 acres) and 20 ha (50 acres) of recreation property. Interstate Highway 55, a major north-south arterial through Illinois, passes approximately 7 km (4 mi) east of the site and Interstate Highway 80, a major east-west highway, passes about 8 km (5 mi) to the north. The Santa Fe and Illinois Central Gulf Railroads pass approximately 6 km (4 mi) to the southeast. The Chicago Rock Island and Pacific Railroad passes about 7 km (4 mi) to the northwest.

3.2 LAND USE

The land use patterns in the area of the Morris Operation are discussed below. Land use at the site and in its vicinity [a 10-km (6 mi) radius], and land use in the region [80-km (50 mi) radius] are discussed separately.

3.2.1 The Site and Vicinity

Natural features and resources dictate to a large extent the land use pattern within a 10-km (6 mi) radius of the Morris Operation. This region of Illinois consists of flat to gently rolling topography that was originally covered with native tall-grass prairie. The prairie vegetation contributed to the formation of deep, fertile soils that are a valuable agricultural resource. It is not surprising, then, that agriculture is the greatest single land use within the 10-km (6 mi) radius of the Morris Operation.

Strip mining for coal is another nearby activity. It has modified a large block of land within 10 km (6 mi) of the site. A tract of abandoned strip-mined land approximately 10-km (6-mi) long and 2- to 3-km (1- to 2-mi) wide lies generally south of the site. Other smaller strip-mined areas are in the vicinity; the nearest one is a clay mine less than 0.5 km (0.3 mi) south of the site. Much of the abandoned strip-mined land is now used for residential development.

Two large public recreational properties owned by the Illinois State Department of Conservation and some private recreational land are situated within a few kilometers of the Morris Operation (Figure 3.2.1). Goose Lake Prairie State Park and Nature Preserve, which joins the west boundary of the site, is the last large native tall-grass prairie in Illinois.(16) It is also one of the largest tall-grass prairie preserves in North America and, thus, is of national significance.(17) Specific uses of the park lands and nature preserve include preservation for historical purposes, research, education and nonconsumptive recreation such as hiking, picnicking and birdwatching. Illinois Department of Conservation attendance reports show that 60,728 people visited Goose Lake Prairie in 1976, and 77,806 visited in 1977.(18) The Des Plaines Wildlife Conservation Area, several kilometers east of the site, is a sizable block of land managed for wildlife and is used primarily as a public

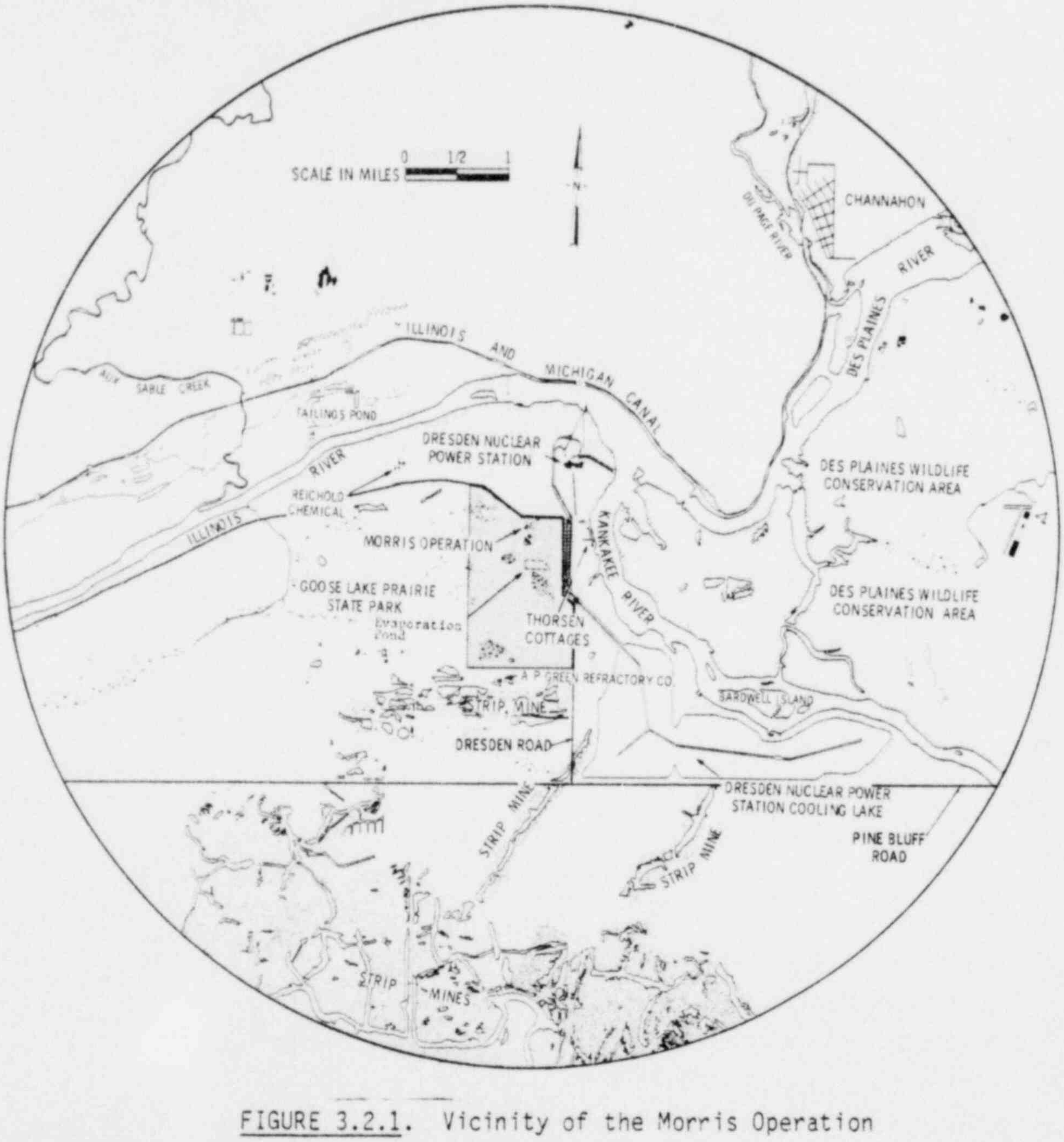

FIGURE 3.2.1. Vicinity of the Morris Operation

hunting area. Attendance figures for 1976 and 1977 were 92,043 and 173,689, respectively.[18] Other recreational property, situated near the Kankakee River just east of the site, consists of about 20 ha (50 acres) of privately owned land. It is divided into approximately 30 cottage sites (Thorsen cottages), which the owner leases.

Except for the state park, land within 4.0 km (2.5 mi) of the site is zoned for industrial use - medium to heavy manufacturing.(3) Thus, further nonindustrial development near the site is limited.

Much of the property within 10 km (6 mi) of the site and along the Illinois and Des Plaines Rivers is used by industry. Among the industrial users are the adjoining Dresden Nuclear Power Station with its associated cooling lake and the A. P. Green Refractory Company. To the west, just beyond Goose Lake Prairie State Park, is a fossil-fueled power plant. Other industrial sites located within 10 km (6 mi) are listed in Table 3.2.1.

The General Electric property (Figure 3.2.2) is largely fallow land but does contain 101 ha (250 acres) of row crops (soybeans and corn) and 50 ha (124 acres) of pasture used for grazing beef cattle. The principal plant structures are located at the north edge of the property, within a 6-ha (15 acres) fenced enclosure. Sanitary waste treatment facilities and a 6-ha

TABLE 3.2.1. Industrial Installations Within 10 km (6 mi) of the Morris Site

Installation	Function	Proximity
Reichold Chemical Plant	Chemical Plant	2.4 km (1.5 mi) NW
Amax	Aluminum mill products	5 km (3 mi) NW
Northern Illinois Gas Company	Natural gas manufacturer	5 km (3 mi) NW
Rexene Polymers Company	Chemical plant	6 km (4 mi) ENE
Mobil Oil Company	Oil refinery	6 km (4 mi) ENE
Collins Power Station	Electricity generation (fossil-fired)	6 km (4 mi) WSW
ARMAK Company	Manufacturer of fatty acid derivatives	6 km (4 mi) WNW
Northern Petro Chemical Company	Manufacturer of polyethylene and ethylene glycol	6 km (4 mi) NW
Joliet Arsenal	Munitions plant (inactive)	10 km (6 mi) ENE
Demert and Dougherty	Filling aerosol cans	10 km (6 mi) S

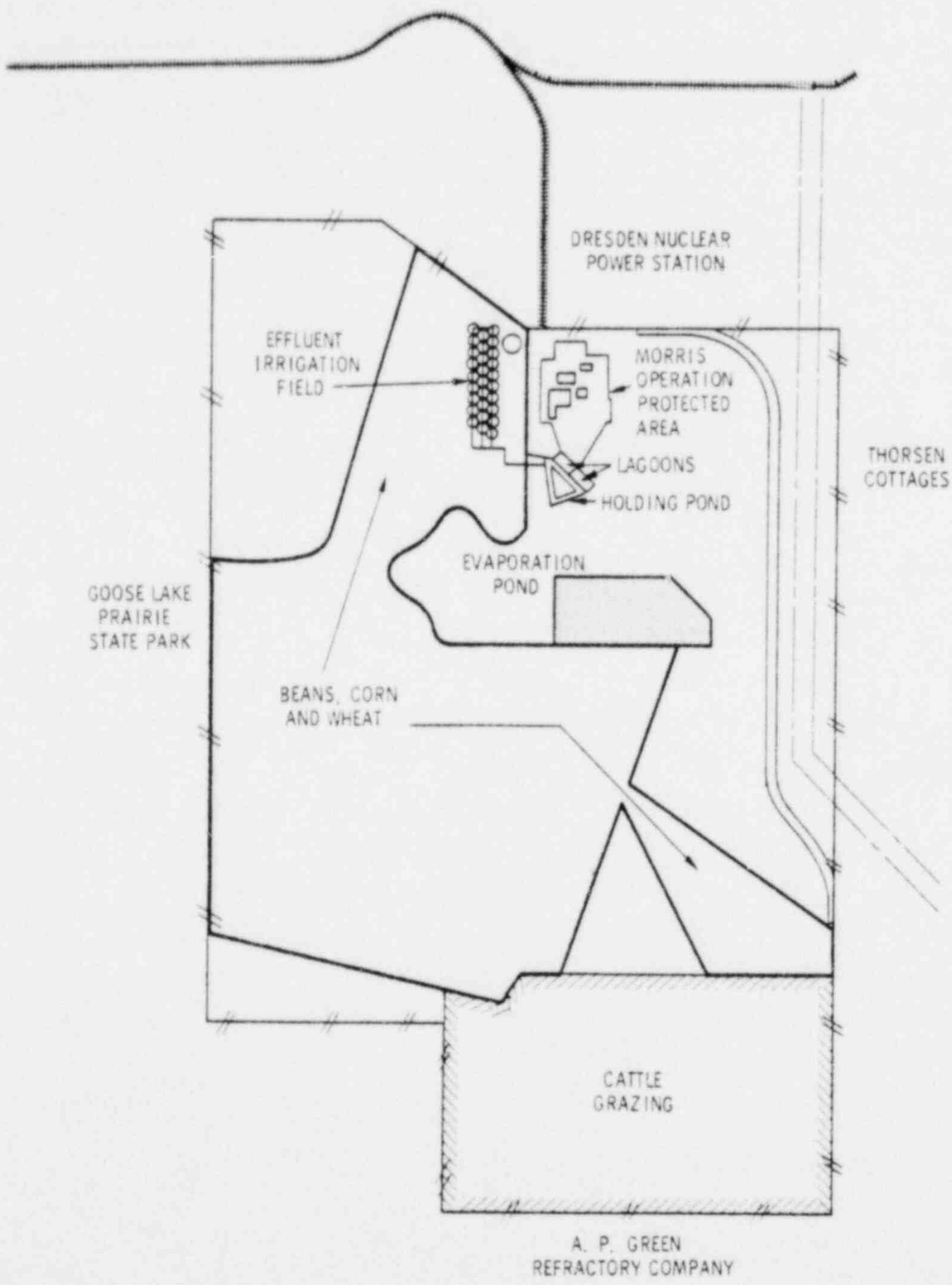

FIGURE 3.2.2. The Site

(15-acre) industrial waste evaporation pond are located immediately south of the facility enclosure. Also, an effluent irrigation system is located in the field west of the main facility.

The Des Plaines and Kankakee Rivers join to form the Illinois River and are known collectively as the Illinois Waterway. The Illinois and Des Plaines Rivers are used principally for navigation, sewage disposal and dilution, and as a water source for industrial usage. The Kankakee River is a smaller and shallower stream used primarily for recreational boating, fishing and domestic water supply.

3.2.2 The Region

The area within an 80-km (50-mi) radius of the Morris Operation is largely agricultural. Soils of the region are deep and fertile with a nearly level to gently rolling topography. In contrast, soils at the site are shallow with occasional rock outcroppings. This region is part of the Midwestern Corn Belt and its crops are used predominantly for animal feeds. Corn and soybeans are the main cash crops, but lesser amounts of cereal grains and forage crops are also grown. Cattle and hog feeding operations are common to the area. Somewhat less common are dairy operations.

Large urban and industrial centers in and around the Chicago area are northeast of the site some 60 to 80 km (35 to 50 mi) and beyond. Much of the industry associated with the towns and smaller communities away from the Chicago area is related to agriculture. Grain storage facilities situated along highways, railways and the Illinois Waterway form a prominent part of the rural landscape.

3.3 WATER USE

Water used at the site for potable, firefighting and cooling water is supplied from a 240-m (788-ft) well located within the exclusion area, southeast of the administration building. The closest deep wells that tap the same aquifer as the Morris Operation well are located at Dresden Nuclear Power Station, 900 m (3,000 ft) northeast of the site. There, two deep wells tap the lower Cambrian-Ordovician aquifer and pump about 1.1×10^2 to 1.5×10^2 ℓ/min

(30 to 40 gpm). Groundwater withdrawals in Grundy County were 2.2×10^7 ℓ/day (5.7×10^6 gpd) in 1970 and are predicted to be 3×10^7 ℓ/day (8×10^6 gpd) by 1980. Groundwater uses are expected to double by the year 2000. The groundwater requirement for the Morris Operation represents 0.06% of the county requirements for 1980.

3.4 DEMOGRAPHY

The projected 1980 population distributions within 16-km (10-mi) and 80-km (50-mi) radii of the site are shown in Figures 3.4.1 and 3.4.2, respectively. The total population estimated to reside within 80 km (50 mi) of the site by 1980 is 7,300,000.[19] About 84% of the total 80-km (50-mi) population resides in a northeasterly direction from the site. Less than 1% of the total population resides within 16 km (10 mi). The nearest population center exceeding 100,000 persons (Joliet and vicinity) is about 25 km (16 mi) northeast of the site. The nearest permanent resident is located about 0.8 km (0.5 mi) east of the site. The staff has attempted to project the estimated population for the vicinity and regional area for the anticipated duration of the license (20 years). This attempt has resulted in a range of estimates.

Information provided to the staff by the licensee,[3] the State of Illinois (Bureau of the Budget),[19] and Commonwealth Edison (DNPS)[20] as well as an NRC study[21] have resulted in estimates ranging from a low of 7,500,000 to a high of 13,000,000 people in the year 2000.

All of the estimates started with the same basis, the 1970 population census. The variation is attributed to the assumed growth rate for the area, which varies from 0.6% to 2.5% per year.

For the purposes of this appraisal, the staff has calculated the growth rate for the period of 1970 to 1980, and assumed it to remain constant until the year 2000. The method will result in a growth rate of 1.4% and a

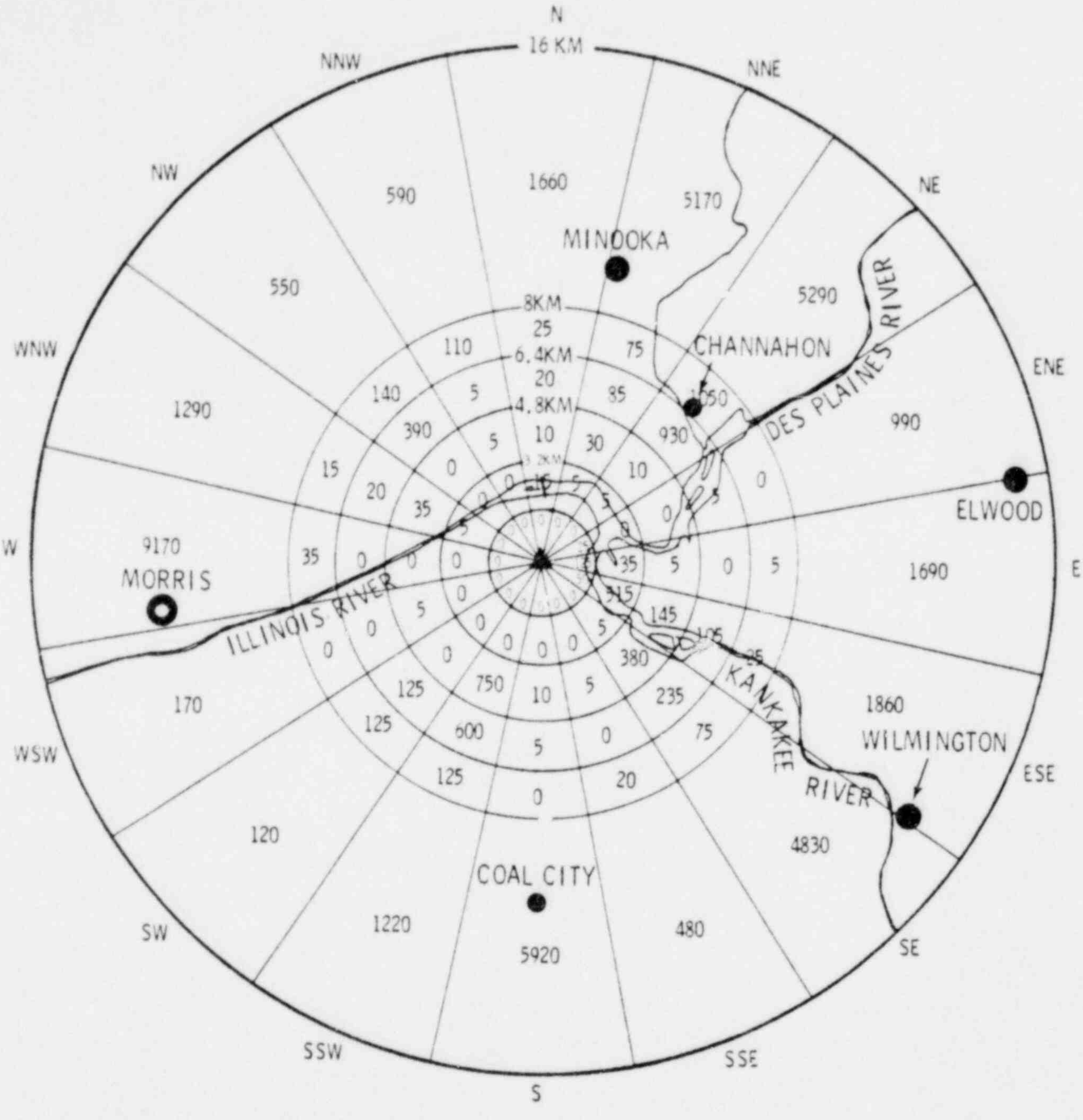

FIGURE 3.4.1. Population Within 16 km (10 mi) of the Morris Operation

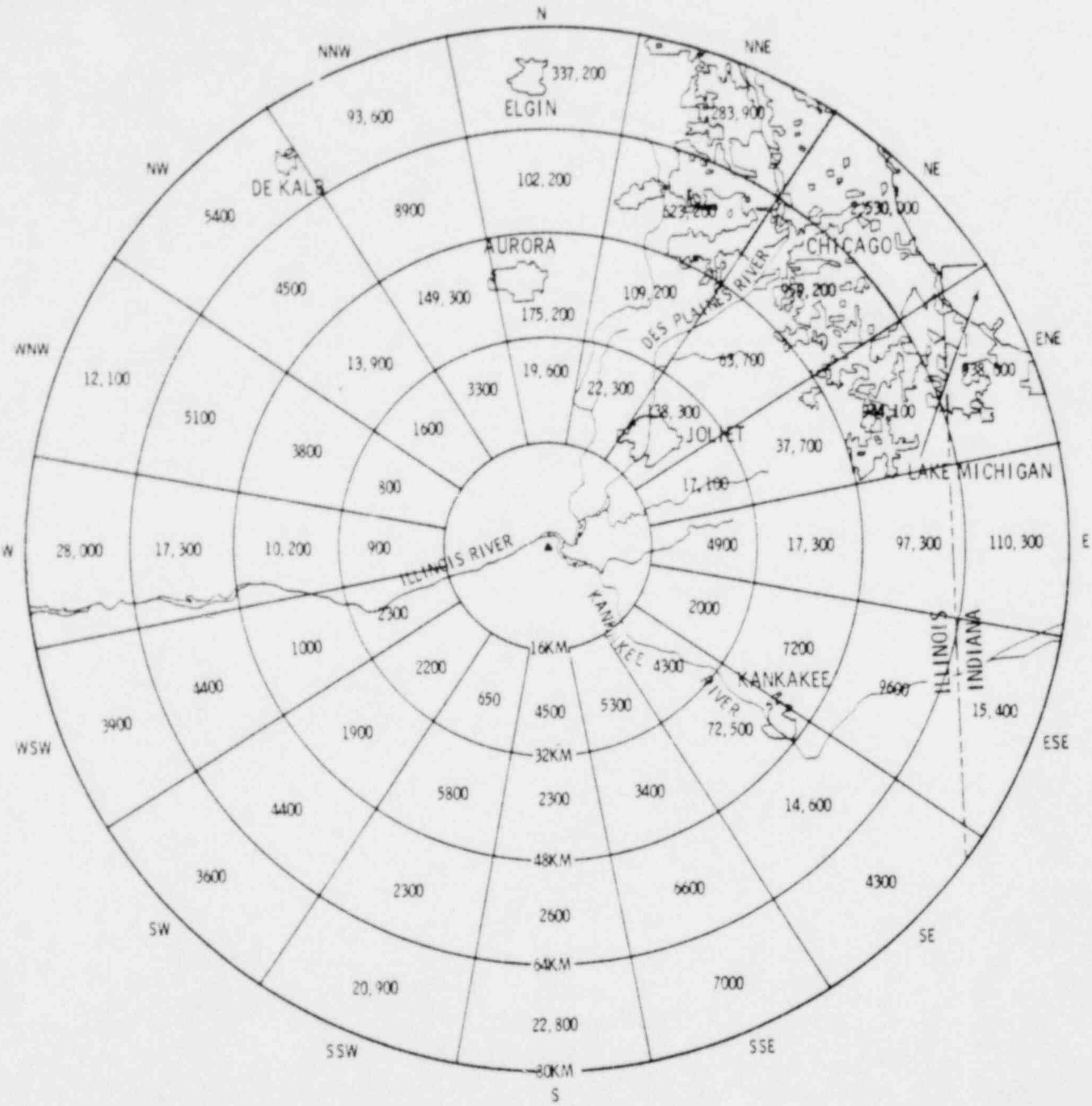

FIGURE 3.4.2. Population Within 80 km (50 mi) of the Morris Operation

respective population on the order of 10,000,000 people for the regional area [80-km (50-mi) radius] for the year 2000. This assumption is contrary to the State's position that the population of the regional area will actually decrease for a period of time.

3.5 BACKGROUND RADIATION OF THE REGION

The natural background radiation of the region, as presented in the Final Environmental Impact Statement for Dresden Nuclear Power Station Units 2 and 3 and Radiological Quality of the Environment in the United States, 1977 is 135 mrem/yr.[20,22] Of this total, 45 mrem/yr is attributed to cosmic radiation; 65 mrem/yr is attributed to external gamma radiation (primarily from ^{40}K and the decay products of the uranium and thorium series) and the remainder of the whole-body dose, 25 mrem/yr, is due to internal radiation (mostly ^{3}H, ^{14}C, ^{40}K, ^{224}Ra, and ^{226}Ra and their decay products).[20]

3.6 METEOROLOGY AND CLIMATOLOGY

The licensee's Consolidated Safety Analysis Report (CSAR)[3] and the Final Environmental Statement for the MFRP[2] describe the meteorology and climatology of the site. Onsite weather records are available from a 122-m (400-ft) instrumented meteorology tower, which has served the Dresden and Morris Operation since 1967. Long-term weather data are available from the Argonne National Laboratory, 43 km (27 mi) northeast of the Morris Site, and the Joliet Municipal Airport, 19 km (12 mi) northeast of the Morris Site.

The climate in the vicinity of the Morris plant is typically continental with cold winters and warm humid summers. Local topographic or other surface features (e.g., Lake Michigan) do not significantly affect the climate near the Morris Operation. The most severe weather conditions experienced in the area are tornados. Tornado frequency at the site is similar to that expected for all of Illinois, which in turn is typical of midwestern states. About five tornados per year are expected in the state. Three have been reported within 18 km (11 mi) of the Morris Operation since [illegible], but none have caused damage at the site. The region is classified by the Commission as being in Tornado Intensity Region I as defined in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

4.0 FACILITY DESCRIPTION

The Morris Operation was built as an integral part of the Midwest Fuel Recovery Plant (MFRP). The MFRP was designed to reprocess irradiated fuel from nuclear power reactors. Reprocessing was to include separating uranium, plutonium and neptunium from the fission product wastes generated in power production. This reprocessing was to be accomplished utilizing equipment installed in a reinforced concrete "canyon" containing several processing cells within the Main Process Building. Adjoining the west end of the canyon building is an underwater storage area designed for unloading irradiated fuel from shipping casks and storage of the irradiated fuel. The fuel transfer opening allowing for access to the canyon from the fuel storage area has been sealed. Major structures on the site are shown in Figure 4.0.1.

The Morris Operation Facility is licensed to store UO_2 fuels from nuclear power reactors moderated and cooled by light water. The compacted UO_2 pellets are clad in zirconium, zirconium alloy, or stainless steel rods arranged in bundle geometries to form a fuel assembly.

Shipment of irradiated fuel to the Morris Operation and of radioactive solid wastes from the site to licensed disposal sites can be made either by truck or by rail. Sections 7.5 and 8.2 of this document address the impacts of radioactive material movement.

The basin itself is a below-grade water-filled pool made of reinforced concrete, poured against bedrock and lined with stainless steel. The pool is divided into three interconnected parts, the cask unloading basin, and fuel storage basins 1 and 2. The cask unloading basin is 14.6 m (48 ft) deep. It has a shelf under 5.6 m (18.5 ft) of water where casks are placed while handling equipment is changed. The floors of the shelf and the unloading basin include provisions to dissipate impact loads, i.e., in the event of a cask drop. The shelf has an 18 cm (7 in.) thick crushable impact pad made of 2.5 and 3.8 cm (1 and 1 1/2 in.) steel sheets; and the deep pit has a 5 cm (2 in.) steel plate. The floors of the storage basins are under 8.7 m (28.5 ft) of water. The areas of these two storage basins are 107 m^2 (1,150 ft^2) and 172 m^2 (1,850 ft^2), respectively.

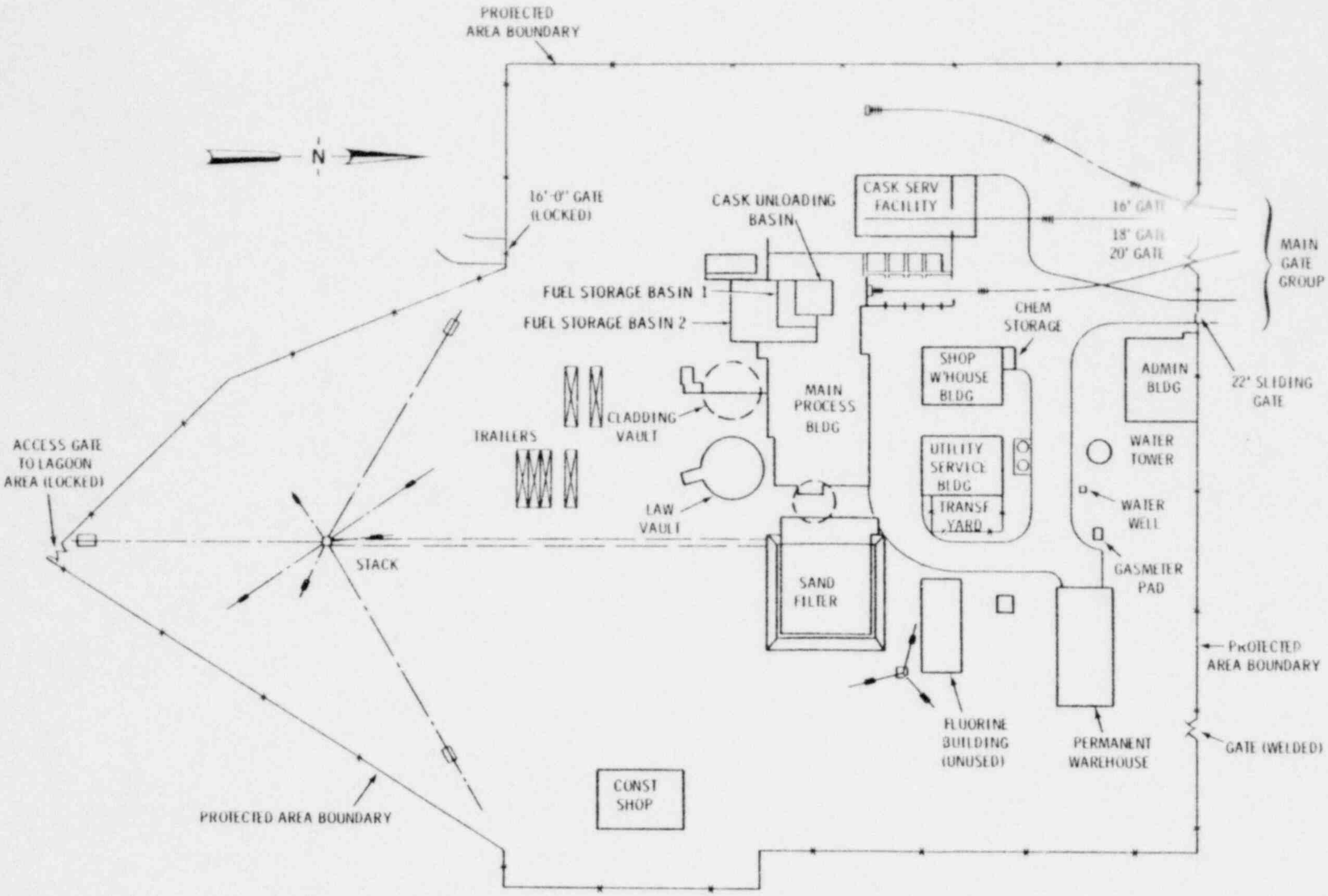

FIGURE 4.0.1. Major Structures of the Morris Operation

The passageway between the unloading pit [14.6 m (48 ft) deep] and the storage basins [8.7 m (28.5 ft) deep] is protected so that a fuel basket, if dropped, cannot tip upside down and empty its contents on the unloading pit floor. The unloading pit doorway guard which accomplishes this is a 1.2 m (4 ft) x 3 m (10 ft) rectangular stainless steel pipe frame hinged at the level of the storage basin f oor. The top ordinarily extends into the unloading pit about 23° so that the only way that a fuel basket may pass through it is vertically in the upright position.

A stainless steel grid of 69 cm (27 in.) squares occupies most of the floor area of each storage basin. The grids are about 0.3 m (1 ft) high and are braced against the side of the basins. A casting at each intersection in the grid provides latching and support for fuel baskets.

There are 151 of these latching squares in basin 1 and 264 in basin 2, making a total of 415. Each one is capable of holding a single basket. Three additional latching squares are installed in the unloading pit to hold baskets in place during loading.

There are two kinds of baskets; one holds up to four PWR fuel assemblies and the other up to nine BWR assemblies. The two kinds of baskets are interchangeable and may be moved to any vacant storage location.

Fuel handling equipment consists of the fuel handling crane and grapples, and the basin crane and basket hooks. The fuel handling crane and grapples are used to move fuel from a shipping cask to a storage basket, and the basin crane and basket hooks are to move the basket of fuel from the unloading pit to a storage location. Rigid tools are used to move fuel so that fuel cannot be lifted too close to the surface. The length of the tool is selected so that even with the crane fully up the minimum water depth is maintained. The grapples used with the fuel handling crane are 12.5 m (41 ft) and 15 m (50 ft) long for PWR and BWR fuel assemblies, respectively. Basket hooks used with the basin crane are 4.3 m (14 ft) long.

Two underground vaults are used to receive low-level waste at the Morris Operation. These are designated as the Low Activity Waste Vault (LAW) and the

Cladding Vault. They are located just to the south of the canyon portion of the Morris Operation. The LAW is the primary storage point for low-level waste. The waste is accumulated until the tank is sufficiently full to require evaporation. The Cladding Vault was originally intended to receive the zirconium, zircaloy, or stainless steel hulls remaining after fuel dissolution.

The LAW vault consists of a steel tank inside a steel-lined concrete vault. The open topped inner vault rests on a concrete pad poured on the bottom liner. The concrete of the steel-lined vault is poured against the solid rock underlying the site. Waste lines reach the vault through a tunnel to the off-gas cell of the canyon. Air enters the vault through a filter installed in a reinforced concrete access pit which also provides access to the piping between the vault and the evaporator.

The Cladding Vault is a stainless steel-lined concrete vault also poured against the rock. It is also connected to the mechanical cell in the canyon by a tunnel through which a cart operated to deliver fuel cladding hulls to the vault. It receives liquids directly from the cask receiving area, the cask service facility floor drain, and the stack drain. The LAW and Cladding Vaults are equipped to transfer liquids between them as necessary. When necessary, the accumulated liquid is evaporated and bottoms returned to the LAW vault.

The LAW vault is equipped with separate systems for collection of possible leakage from the tank and inleakage from the ground. Inleakage is transferred to the process sewer but could be sent to the vault if radioactivity were found to be present. Liquid collected in the tank leakage sump is transferred to the vault. Liquid from the cladding vault sump is returned to the cladding vault.

4.1 FACILITY WATER USE

Water is drawn from an onsite well and discharged to a 1.9×10^5 ℓ (50,000 gal.) elevated water sphere located near the well (Figure 3.0.1), at a rate of 3.8×10^2 ℓ/min (100 gpm). The maximum continuous pumping rate of the well is 9.5×10^2 ℓ/min (250 gpm). Backup water is supplied by a 117 m (383 ft) well, equipped with 1.1×10^2 ℓ/min (30 gpm) pump. This well is located 11 m (35 ft) south of the Boiling Water Reactor Training Center [approximately 46 m (150 ft) east of the primary well].

Underground piping distributes the water to various points of the utility service building for supplying the demineralizer system as well as potable, firefighting and nondemineralized water. Normally, an estimated 7.6×10^2 ℓ/day (200 gpd) of cooling water makeup is required; sanitary water needs are estimated at 1.1×10^4 ℓ/day (3,000 gpd).

Demineralized water is used for utility boiler makeup, for fuel storage basin supply and for cask flushing. This water is supplied from the cation-anion demineralizer located in the utility service building which is capable of treating 95 ℓ/min (25 gpm) continuously and 190 ℓ/min (50 gpm) instantaneously from the utility water supply system

4.2 STORAGE BASIN WATER TREATMENT SYSTEMS

The storage basin complex is supplied with demineralized water from the onsite well and treatment facilities. Water is withdrawn from the basin by skimmers and circulated through a filter system before returning to the basins. A basin water cooling system maintains basin water temperature within operating limits,[2] while the filtration system maintains water clarity, minimizes the concentration of radioactive materials in the water, and reduces the potential for corrosion.

4.2.1 Basin Water Cooling System

Basin water is cooled by circulation of the water through force-draft, expanded-surface, air coolers (fin-fan coolers) with a nominal capacity of 6.3×10^3 MJ/hr (6×10^6 Btu/hr). Two 2,800 ℓpm (750 gpm) pumps are provided to move the water. The pumps may be used singularly or together.

An additional bank of coolers with the same capacity remains in standby, and a third bank with a capacity of 4.2×10^3 MJ/hr (4×10^6 Btu/hr) is available, with some delay, if needed. However, experience has indicated that during most of the time since 1972, heat generated could be dissipated to the atmosphere without the aid of coolers. Additional details of the basin water cooling system are contained in the CSAR.[3]

4.2.2 Basin Water Filtration System

Water removed by skimmers is filtered through a Powdex filter system consisting of a single 11 m^2 (115 ft^2) DeLaval filter. The filter is precoated with Solka Floc, which is overlaid with diatomaceous earth, Powdex resin or Zeolon as desired. This system utilizes a 9.5×10^2 ℓ/min (250 gpm) pump. Sludge from the filter is discharged to the Low Activity Waste (LAW) vault. Additional details about the filtration system are provided in the CSAR.[3]

4.3 VENTILATION SYSTEM

The ventilation system filters exhaust air from the main Process Building through a filter bed consisting of layers of graded gravel and sand 4.5 m (15 ft) thick. Air is drawn through the sand filter by up to three exhaust blowers capable of providing 450 m^3/min (16,000 cfm) each. Filtered air is exhausted to the atmosphere via a 91 m (300 ft) stack. The ventilation system is designed so that air passes sequentially from areas of low contamination potential to areas of higher potential.

4.4 RADIOACTIVE WASTE MANAGEMENT SYSTEMS

Liquid wastes are collected either in the LAW vault or the Cladding vault depending on the source of the liquid. Except for liquid waste from the cask receiving area, stack drain and the Cask Service Facility, all liquid waste streams are routed to the LAW vault. Provisions can be made to pump liquid to and from the LAW and cladding vaults.

Excess liquids in the LAW vault are evaporated to reduce liquid volumes stored in the vault. Ultimately, equipment and facilities for solidification and packaging of the evaporator bottoms for offsite shipment and disposal will be provided as stated in the Decommissioning Plan for the Morris Operation.[3]

Wastes contained in the LAW and Cladding vaults are stable and chemically inert. The vaults are monitored under a program designed to detect any leakage of radioactive material from the vaults.[3] Solid radioactive wastes are compacted, if possible, packaged and shipped for disposal by a licensed disposal contractor.

4.5 NONRADIOACTIVE WASTE SYSTEMS

Nonradioactive waste systems are described in the following sections. The flow directions of Morris Operations effluents are shown schematically in Figure 4.5.1.

4.5.1 Waste-Containing Chemicals or Biocides

Chemical waste solutions, containing principally sodium nitrate, are from resin regenerations in the water demineralizer system. These solutions are discharged to an onsite evaporation pond. Currently, the evaporation pond is being used under Water Control Permit No. 1979-EO-440, issued on May 11, 1979, by the Illinois Environmental Protection Agency. Chemical wastes produced as blowdown water from the utility boiler (treated for removal of phosphates) and air compressor cooling tower blowdown are router to the sanitary lagoons and holding pond. Blowdown denotes draining, under pressure, of some water from the lower part of a boiler to remove whatever impurities may have accumulated.

4.5.2 Sanitary System Wastes

Sanitary wastes are comminuted and discharged to two onsite lagoons, each with a surface area of 1,340 m^2 (14,400 ft^2). The lagoons discharge to a holding pond. The lagoons can be connected either in series or in parallel, depending on need. Blowdown wastes, 1.6×10^4 ℓ/day (4300 gpd), are discharged to this system. The discharged waste water from the holding pond is piped to a spray field irrigation system on GE-owned land onsite. This overflow, containing approximately 1 ppm chlorine, will be less than 3×10^6 ℓ/yr (7.9×10^5 gal./yr). The sanitary sewage treatment system is operated under Permit No. 1976-EB-408-1 issued by the State of Illinois, Sanitary Water Board, as amended.

4.5.3 Other Wastes

Other nonradioactive wastes are gaseous exhausts and nonradioactive solid wastes. The gaseous exhaust from the facility contains combustion products - CO_2 and water vapor. The exhaust is produced from a natural gas-fired boiler with a rated capacity of 11,000 kg (25,000 lb) of steam per hour. These combustion products are discharged through the boiler stack

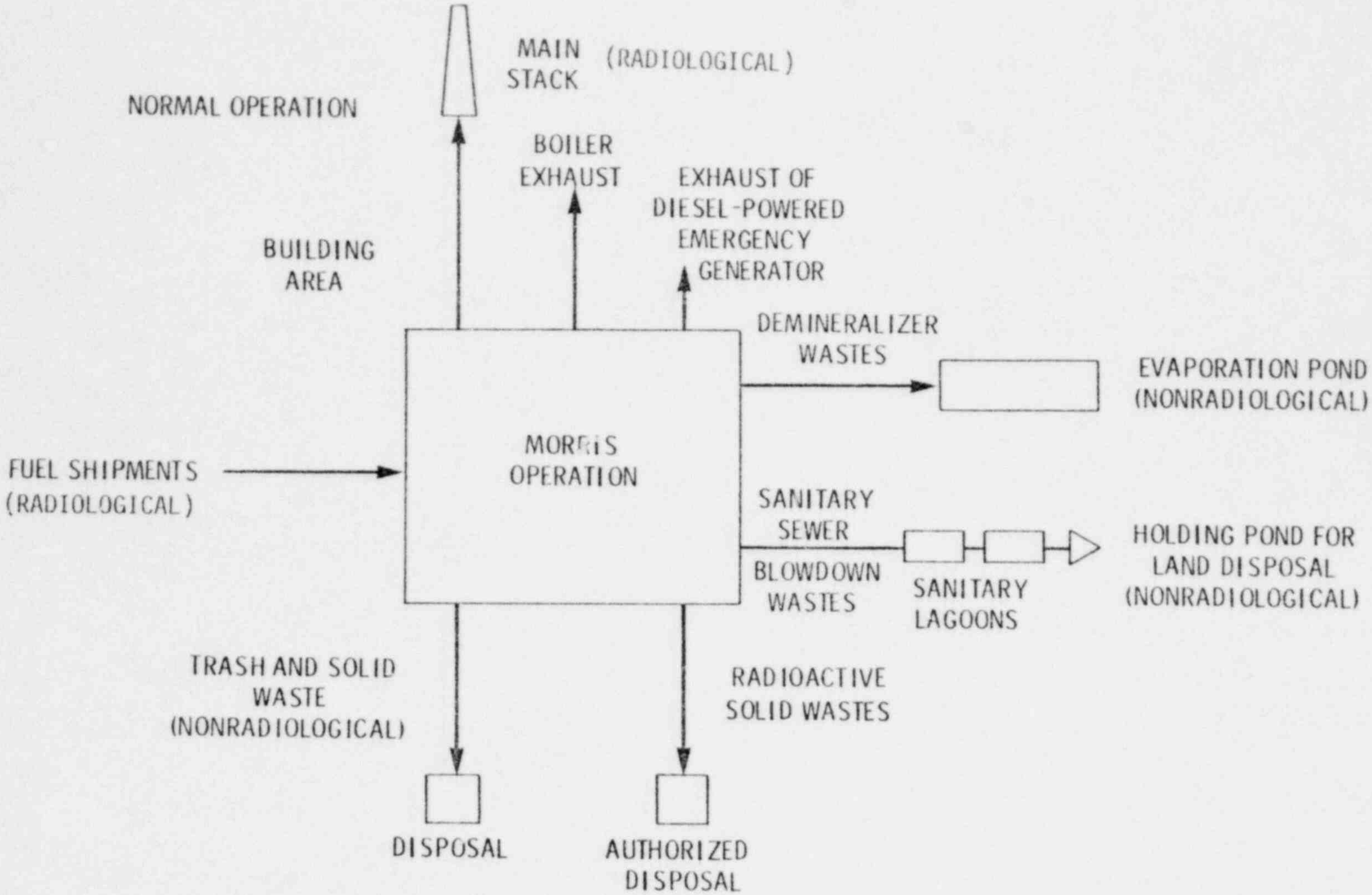

FIGURE 4.5.1. Effluents from the Morris Operation

at a height of 16 m (53 ft). Miscellaneous nonradioactive solid wastes such as trash, garbage and shop wastes are shipped offsite for treatment and disposal by a licensed contractor.

5.0 OPERATIONAL HISTORY

Since 1972, approximately 1200 fuel assemblies (443 PWR and 761 BWR) have been received at the Morris Operation, totaling 315 MTU or 42% of the licensed basin capacity. The fuel received was irradiated at the Haddam Neck, La Crosse, Point Beach, San Onofre and Dresden Nuclear Stations. Some additional assemblies are presently being received from San Onofre. Several operational aspects of the Morris Operation are discussed in this section. These include the:

- basin water cooling system
- basin water cleanup system
- ventilation system
- waste management
- occupational radiation exposure.

5.1 BASIN WATER COOLING SYSTEM

With 315 MTU of spent fuel in the basin, an average burnup of 15,500 MWd/MTU and an average cooling period of about 80 months, basin water temperatures have been maintained between 28°C (82°F) and 38°C (100°F). The calculated heat load from this fuel is approximately 1.4×10^3 MJ/hr (1.3×10^6 Btu/hr). The highest basin heat load reported by the applicant was 2.2×10^3 MJ/hr (2.1×10^6 Btu/hr) during the first half of 1977. Basin cooling is presently accomplished by one bank of coolers capable of dissipating 6.3×10^3 MJ/hr (6×10^6 Btu/hr). This is approximately one-third of the maximum cooling capacity. The heat load measured during August 1979 was 7.9×10^2 MJ (7.5×10^5 Btu/hr). Loss of basin cooling is discussed in Section 8.1.6.

5.2 BASIN WATER CLEANUP SYSTEM

In 1976 the basin water cleanup system was modified to include Zeolon-100. Zeolon-100 is an inorganic ion exchange resin capable of effectively removing cesium from the basin water. Before this modification, radioactive cesium was the predominant radioactive contaminant in the basin water. Present operating experience indicates the radioactive cesium and cobalt concentrations in the basin water average 3×10^{-4} and 1×10^{-4} μCi/ml, respectively, during periods of filter operations.

Any removal of the filters from operation results in a slow increase in basin water contamination levels. Based on gross beta activities measured during 1976-1978, the basin water contamination levels can be expected to increase by as much as a factor of 10 (about 3% of the concentration of 0.1 μCi/ml permitted by the license condition) during interruptions of normal filter operation lasting a few days. Operating experience indicates these levels are quickly lowered to the normal levels listed above when filters resume operations.

Operation and history of the basin water treatment system is discussed in detail in Section II of reference (1). In 1978, the filter was operable 98.5% of the time with the 1.5% (5.5 days total) downtime due to a minor design change and to 15 filter changes. Five of the filter changes were necessitated by minor power interruptions.

5.3 VENTILATION SYSTEM

Monitoring of airborne radioactive material concentrations in working areas and of releases to the environment via the main facility stack indicate no significant exposure of operating personnel nor any significant release to the environment. Comparisons of the applicant's gross beta and gross alpha measurements on particulate air filters indicate less than 1×10^{-10} μCi/cm^3 with average concentrations in working areas 0.001 of applicable maximum permissible concentrations during the operation of the facility. During the same period gross alpha and gross beta measurements of particulates released via the main facility stack were less than 1×10^{-18} μCi/cm^3, or more than a factor of 0.00001 of applicable standards for such releases. Releases of radioactive gases via the main facility stack have been less than the detection levels of the measurement systems. Calculated annual release rates of these gases are presented in Section 7.3.

5.4 WASTE MANAGEMENT

The facility is designed to release no liquid effluents offsite under normal conditions. Discussions of the operating experience with solid and liquid wastes are presented below.

5.4.1 Radioactive Waste Water

Radioactive wastes are held in two vaults: the LAW vault and the cladding vault. Liquids in these vaults are periodically evaporated to reduce the volume of liquid. As a result of the evaporation, small quantities of tritium and halogens are released to the environment via the main stack (see Table 7.3.1). The releases as shown in Table 7.3.1 are small in quantity and they are not considered significant.

Measurements of liquids in the annulii surrounding these vaults indicate that there has been no significant leakage from these vaults. Measured leakage rates have been a maximum of 2.4 ℓ/day from the low-activity waste vault and 0.2 ℓ/day from the cladding vault. Leakage into the annulii is collected in sumps and is routed to the low-activity waste vault. There are no indications of any leakage from the annulii to the surrounding soils and groundwater.

5.4.2 Solid Radioactive Waste

Through 1978, solid radioactive waste (disposable clothing, used laboratory equipment) were packaged and shipped to Sheffield, Illinois, for burial. These shipments totaled 41 consisting of a total of 765 m^3 (27,000 ft^3) of waste.

5.4.3 Industrial Waste Water

Industrial waste water is routed to an evaporation pond or through the sanitary sewer system, depending on the origin of the waste water. The evaporation pond was designed for containing liquids generated by the reprocessing facility. Without reprocessing of spent fuel, a comparatively low volume of water is routed to the pond. As a result the pond remains mostly dry.

During November 1977, a leak in the evaporation pond via an old septic tank drain field was found. This leak was plugged during December 1977. Measurement of nitrates (the predominant contaminant in the evaporation pond) at the Corps of Engineers Pump Station on the Kankakee River indicated no significant concentrations. The quantity of liquid that leaked from the evaporation pond via this abandoned drain field is not known. However, the quantity of waste water routed to the pond has been small compared to the capacity of the pond. This leak or any others that may exist in the clay liner of the evaporation pond have not resulted in any significant deterioration in the quality of receiving waters.

5.4.4 Sanitary Waste Water

There have been no releases of sanitary waste water offsite. Sanitary waste water is routed to a treatment lagoon and collected in a holding pond. Treated water can be used to irrigate an adjacent field.

5.5 OCCUPATIONAL RADIATION EXPOSURE

Sources of occupational exposure within the Morris Operation include the decontamination pad, basin pump room, basin coolers and the various fuel handling operations. The radionuclides contributing to this exposure are $^{58,60}Co$ and $^{134,137}Cs$. In accordance with 10 CFR Part 20, the Morris Operation requires that exposure of personnel to ionizing radiation be kept as low as reasonably achievable. The internal and external occupational exposures described below are within the limits of 10 CFR Part 20 and are considered to be as low as reasonably achievable.

5.5.1 External Exposure

The occupational dose at the Morris Operation was 33 to 46 man-rem/yr for the years 1976-1978, respectively. During the 2-yr period 1976-1977 a total of 211 MTU of spent fuel was received; the total occupational exposure was 79 man-rem or 0.37 man-rem/MTU of spent fuel received.

5.5.2 Internal Exposure

In 1978, 25 Morris Operations employees received detectable internal depositions of ^{137}Cs and ^{60}Co. The average body burden was approximately 1.3% of the allowable burden. The number of employees having detectable burdens and the magnitude of those burdens have remained essentially constant over the 1972-1978 period. Assuming the 25 employees maintain these average depositions for 1 year, an additional 1.6 man-rem of exposure would result. This dose would be in addition to the 33 man-rem from external sources. Since the dose received from internal depositions is a small fraction of the allowable burden, the consequences of these doses are considered to be minor.

6.0 ENVIRONMENTAL MEASUREMENTS AND MONITORING PROGRAM

Environmental monitoring programs usually consist of many diverse elements, owing in large part to the various reasons for conducting environmental surveillance. For example, environmental measurements may monitor the magnitude and extent of a readily quantifiable impact, or these measurements verify that the impact is indeed less than the calculated value. When the accuracy of predictive models is doubtful, or when the variance of environmental parameters is great, monitoring may be performed to add credibility to the calculational model. The toxicity, persistence, degree of bioaccumulation, and type of biological or chemical effect may indicate a need for more or less stringent controls on a given component of the effluent. Similarly, an innocuous but easily identifiable agent may be used as a marker for other components of an effluent so that only the innocuous agent has to be monitored. The detection of significant quantities of the innocuous agent may trigger remedial effluent control actions and/or intensified monitoring for a wide range of other agents.

The data from the monitoring programs are used for several purposes. One purpose is to determine the need for population studies of potentially affected biota. Another purpose is to assure that potential impacts are not being ignored.

Thermal, radiological, hydrological, meteorological and biological monitoring programs are discussed in the following sections.

6.1 THERMAL

The fuel basins do not discharge heat to surface waters of the Des Plaines or Kankakee Rivers or to any offsite surficial waters. All waste heat is discharged to the atmosphere.

The calculated heat load from the basin is discussed in Section 5.1. Rates of heat dissipation to the atmosphere are discussed in Section 7.4.1. As discussed in those sections, the quantity of heat dissipated (ranging from 0 to 0.07 Fahrenheit degrees) will not significantly change air temperatures in the local environment and will not be capable of measurement. Therefore, this source of thermal rejection has not required either monitoring or measurement programs. The meteorological monitoring program at DNPS provides records of ambient air temperatures, although these air temperature measurements are not mandated by the magnitude of the thermal effluent.

6.2 RADIOLOGICAL

The environmental monitoring program for the Morris Operation is conducted as a part of a larger monitoring program conducted by Commonwealth Edison Company for the Dresden Nuclear Power Station (DNPS). The DNPS program is designed to assure compliance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I. The intent of the regulations and guides is to protect the public and workers from radiation hazards arising out of activities under licenses issued by the NRC and to assure that any exposure to the public and workers that does exist is as low as reasonably achievable.

General Electric joined with Commonwealth Edison in 1967 to expand its program to accumulate baseline data for GE's proposed fuel reprocessing operation. This cooperative effort has continued to date. Evaluation of the data supplied by the monitoring program indicates an absence of detectable off-site contamination or any increase in off-site exposure resulting from spent fuel storgage at Morris. The present program is outlined in Table 6.2.1.

6.3 HYDROLOGICAL

Sanitary waste water and industrial process water, including the blowdown from nonradiological systems, are diverted through clay-lined lagoons to a holding pond. This water is chemically treated and may be used to irrigate the farmland owned by the General Electric Company adjacent to the Morris Operation.

Chemical and radiological wastes are not discharged to the holding pond. Rather, chemical wastes from resin regenerations are discharged to an evaporation pond. There are not discharges of liquid radiological waste.

Discharges to the evaporation pond may seep into the local ground water and eventually enter the Kankakee River. Nitrates are the primary effluent from Morris Operation in this pathway. If seepage occurs, it may collect in a

TABLE 6.2.1. GE Radiological Monitoring Program

Sample Medium	Collection Sites	Analysis	Frequency
1. Particulates in air	a) Near field: Collins Road, Bennitt farm, Pheasant trail	Filter gross β	Weekly
	b) Far field: Clay products, Prairie Park, Coal City, Goose Lake Village, Morris, Minooka, Channahon, Joliet	Filter exchange only	Weekly
2. Exposure by TLD (Thermoluminescent Dosimeter)	a) Near field: see 1.a) b) Far field: see 1.b)	γ radiation	Quarterly
3. Milk	Davidson, Dorin & Mather farms	Iodine-131	Weekly
4. Water	a) Site waste ponds (2 samples)	Gross α, β	Monthly
	b) Corps of Engineers pump station c) Illinois River at Rt. 47 d) Thorsen well e) LAW vault well f) West pond	Gross β, H^3	Quarterly

small stream bed. The stream is sampled at the Corps of Engineer's pumping station where it is raised to the Kankakee River. No contamination at the station has been reported.

Possible leakage from the spent fuel storage, cladding vault and LAW vault is collected in sumps and transferred to the LAW vault. A network of wells are used to monitor potential seepage to ground water from the storage basins, LAW vault or the cladding vault. These wells can be used to sample ground water at intersections between the potential flow of seepage from the Morris Operation and the nearest surface water.

6.4 METEOROLOGICAL

A meteorological program is required to obtain sufficient data to permit calculation of radiation doses from radiological releases. The present monitoring program which is conducted by DNPS and is used by GE satisfies U.S. NRC Regulatory Guide 1.26. A 120 m (400 ft) fully instrumental meteorological tower on the DNPS site provides the data used by the Morris Operation. Argonne National Laboratory [43 km (27 miles) northeast] provides additional and backup meteorological data.

6.5 BIOLOGICAL

Ecological studies of the changes in the flora and fauna population are mandated whenever a facility's operation is capable of inflicting significantly acute or chronic impacts upon its environment. The program at the Morris Operation was established when reprocessing was the planned facility use. When Morris was converted to a fuel storage installation, the requirements for biological monitoring were greatly reduced.

Since operation involves no disturbances to the land, the only terrestrial monitoring is for potential radiological impacts. The potential radiological impact on the terrestrial ecology is monitored through surveillance of air and water by the radiological monitoring program discussed in Section 6.2.

In the absence of point-source liquid discharges to the surface waters, no measurable impacts to the aquatic environment exist. No aquatic monitoring efforts are required.

7.0 ENVIRONMENTAL IMPACTS OF PROPOSED ACTION (CONTINUED FACILITY OPERATION FOR UP TO TWENTY YEARS)

The environmental impacts resulting from the proposed licensing action are discussed in this section. Impacts discussed include those from heat and radiological sources. In addition, there is discussion of land use, water use, nonradiological impacts (including socioeconomic) and transportation.

7.1 LAND USE

The proposed licensing action will not result in any additional land usage. The facilities necessary for the storage of 750 MTU now exist at the Morris Operation. No additional facilities would have to be constructed. The present land usage of the Morris Operation is described in Section 3.2.

7.2 WATER USE

Present water use is described in Sections 3.3 and 4.1. As previously stated, there are two uses for water at the site: 1) cooling the stored spent fuel, and 2) meeting sanitary needs. In the first instance only evaporative makeup is required, i.e., replacing that water evaporated from the pool surface. Since the pool water is maintained at a constant temperature by circulation through an air-cooled heat exhanger (cooler), the amount of fuel stored does not affect water use, i.e., as more fuel is stored, producing more heat input to the water, the coolers remove more of the heat to keep the temperature constant. The evaporative losses account for only about ten per cent of water used at the site.

The second use, that of meeting sanitary needs, accounts for the other ninety percent of water use. Since no additional people are required to operate the installation as additional fuel is received, this use also will not increase. Therefore, no change in water use is anticipated.

As described in Sections 3.3 and 4.1, the water use (3,200 gallons per day) constitutes a small percent (0.06%) of Grundy County groundwater withdrawals for 1980. This use would have no detectable effect on the groundwater supplies.

7.3 RADIOLOGICAL IMPACTS

The radiological impacts of normal operation of the Morris Storage Facility were analyzed to determine potential exposure pathways, dose commitments and impact on man and biota. Radiological impacts of normal operation were calculated based on the facility capacity of 750 MTU of spent fuel as stored for up to 20 years.

7.3.1 Exposure Pathways

The Morris Operation's facilities are designed so that no surface liquids are released offsite as a result of routine operation. The features of the Morris Operation, which assure no routine releases of liquids, include 1) evaporation of low-level radioactive wastes, 2) monitoring of the annulii surrounding the low-level waste vault, cladding vault and spent fuel storage basins to assure no undetected leaks have developed, 3) sump pumps which route any liquids found in the annulii around these vaults to the low-level waste vault, 4) an irrigation system for disposal of liquid sanitary wastes to land surfaces onsite and 5) use of a clay-lined evaporation pond for disposal of chemically contaminated wastes. Therefore, any release of radioactive material from fuel storage operations would be via the ventilation air exhaust system. The ventilation air is passed through a sand filter and exhausted to the atmosphere from a stack height of 91 m (300 ft). Potential sources for the material released are:

- effluent from the low-level waste evaporator
- incidental contamination in ventilation air from decontamination and storage areas
- vented gases from a shipping cask
- off gas from leaking fuel rods in storage, or during fuel handling.

The Morris Operation waste management system is designed to limit airborne releases to as low as reasonably achievable in accordance with 10 CFR Part 20.

The major portion of doses to the general public will be due to noble gases released as gaseous effluents to the atmosphere. The significant pathways for exposure to noble gases will include air submersion and inhalation.

Less significant dose commitments from inhalation of halogens and ingestion of potentially contaminated crops will be incurred. Deposition on ground and direct exposure from sources onsite would not contribute a significant portion of the total dose commitment.

7.3.2 Radioactive Materials Released to the Atmosphere

Most of the fuel received has been aged over one year after reactor discharge. The license authorizes receipt of fuel aged at least 90 days. For the purpose of calculations of releases of radioactivity in which fuel age is a factor, we have assumed an age at receipt of one year. This assumption is based on past experience regarding fuel received at Morris and our understanding of reactor operation wherein fuel is discharged on an approximately yearly basis and then stored for several years prior to offsite shipment for further storage.

In the relatively benign environment of water basin storage, the rod failure rate is expected to be so low that it can be ignored.[23] The Morris Operation will analyze potential leaking fuel assemblies on a case-by-case basis to assure criteria of the facility are adhered to.

The defect rate during fuel irradiation in commercial nuclear power plants has been found to be 0.0001/yr per assembly irradiated.[24] The defect rate during receiving and shipping operations is assumed to be 0.0002 per assembly received. This is thought to be a conservatively high defect-rate estimate.[23] A major share of these defects is assumed to develop during shipment, and the radioactive gases that escape are released during cask venting. Fuel rod failures are postulated as a result of microscopic cladding failures during irradiation, and some defects are postulated to develop during shipping and handling.

The predicted annual releases of radioactive material at the Morris Operation (750 MTU) presented in Table 7.3.1 were calculated based on assumptions and estimates contained in Appendix A.

7.3.3 Dose Commitments

The actual radiological impact on man associated with routine operation depends on the manner in which the radioactive waste management system is operated, the quantity of fuel received and the age of that fuel. The staff evaluated and examined the Operating Experience Report[4] and concluded that the installation meets the objectives and limits of 10 CFR Part 20. The staff's evaluation assumes the fuel received has cooled one year, the average cooling period of stored fuel is five years, and the fuel storage basins hold licensed capacities.

TABLE 7.3.1. Predicted Annual Releases of Radioactive Material to the Atmosphere from the Morris Operation, Ci

	Receiving of Fuel[a]		Storing of 750 MTU of Fuel[b]	
Radionuclide	Release, Ci	% of 10CFR20[c] Appendix B	Release, Ci	% of 10CFR20[c] Appendix B
^{3}H	3.0×10^{-1}	1.5×10^{-7}	2.1	1.1×10^{-3}
^{54}Mn	9.2×10^{-6}	9.1×10^{-10}	---	---
^{58}Co	2.1×10^{-4}	1.1×10^{-8}	---	---
^{60}Co	3.1×10^{-4}	1.0×10^{-7}	6.9×10^{-5}	2.3×10^{-8}
^{85}Kr	1.6×10^{2}	5.3×10^{-5}	3.8×10^{1}	1.2×10^{-5}
^{90}Sr	3.7×10^{-5}	1.2×10^{-7}	1.3×10^{-5}	4.3×10^{-8}
^{95}Nb	1.5×10^{-5}	5.0×10^{-10}	---	---
^{95}Zr	3.1×10^{-5}	3.1×10^{-9}	---	---
^{106}Ru	1.7×10^{-4}	8.5×10^{-8}	4.1×10^{-6}	2.0×10^{-9}
^{129}I	2.0×10^{-6}	1.0×10^{-8}	1.9×10^{-6}	9.5×10^{-9}
^{134}Cs	4.2×10^{-3}	1.0×10^{-6}	5.3×10^{-4}	1.3×10^{-7}
^{137}Cs	2.0×10^{-3}	4.0×10^{-7}	8.8×10^{-4}	1.7×10^{-7}
^{144}Ce	2.4×10^{-4}	1.2×10^{-7}	2.6×10^{-6}	1.3×10^{-9}

(a) 200 MTU/yr of spent fuel cooled 1 year.
(b) Fuel cooled 5 years.
(c) Air concentration at maximum individual location, percent of 10CFR20 Appendix B Table II.

7.3.3.1 Offsite Exposure

Annual doses to the regional population and the closest occupants from routine operation are presented in Table 7.3.2. These doses cannot be measured and are estimated by means of calculations. The total-body dose calculated for the nearest occupants is 4.6×10^{-5} mrem or less than 0.0001% of the dose received from naturally occurring sources. Total-body dose to the regional population is calculated to be 1.4×10^{-2} man-rem. This can be compared with the dose to the regional population from naturally occurring sources, which is 9.6×10^{5} man-rem. Fifty-year dose commitments to the nearest resident and to the regional population are presented in Table 7.3.3.

7.3.3.2 Occupational Radiation Exposure

An analysis of potentials for exposure to operating personnel, based on information contained in the Operating Experience Report,[4] the present quantities of spent fuel stored, capacity of the facility, predicted fuel receipt rates and design criteria indicate that the occupational exposures should be approximately equivalent to past operating experience. This equivalence is based on the age of fuel to be received, the predicted fuel receipt rate and the age and quantity of fuel now in storage. Occupational radiation exposure as a result of past operations is presented in Section 5.5.

TABLE 7.3.2. Annual Doses to the Nearest Resident and the Regional Population from Routine Operation[a]

	Skin	Total Body	Thyroid	Lung	Bone
Nearest Resident,[b] mrem	2.9×10^{-4}	4.6×10^{-5}	2.8×10^{-5}	2.5×10^{-5}	1.8×10^{-5}
Regional Population,(c) man-rem	9.0×10^{-2}	1.4×10^{-2}	9.0×10^{-3}	7.8×10^{-3}	5.5×10^{-2}

(a) Facility storage capacity is 750 MT.
(b) Nearest resident resides 800 m (2600 ft) east of the Morris Operation stack.
(c) Regional population in 1980 is estimated to be 7.3×10^{6} persons.

TABLE 7.3.3 Fifty-Year Dose from One Years Release to the Nearest Resident and the Regional Population from Routine Operation[(a)]

	Skin	Total Body	Thyroid	Lung	Bone
Nearest Resident,(b) mrem	2.9×10^{-4}	6.8×10^{-5}	3.6×10^{-5}	2.9×10^{-5}	4.7×10^{-5}
Regional Population,(c) man-rem	9.0×10^{-2}	2.2×10^{-2}	1.1×10^{-2}	9.0×10^{-3}	1.5×10^{-2}

(a) Facility storage capacity is 750 MT.
(b) Nearest resident resides 800 m (2600 ft) east of the Morris Operation stack.
(c) Regional population in 1980 is estimated to be 7.3×10^6 persons.

7.3.3.3 Evaluation of Radiological Impacts

The calculated total body dose of 4.6×10^{-5} mrem/yr to the nearest resident is small when compared with the individual background dose of 135 mrem/yr or the dose limit of 500 mrem/yr specified in 10 CFR Part 20. The staff has also concluded that the total dose commitment of 1.4×10^{-2} man-rem/yr to the population within 80 km of the plant is not significant when compared with the natural radiation background dose commitments of 2.35×10^6 man-rem/yr. As a result, the staff has concluded that there has been no measurable radiological impact to man from normal operation of the plant, nor is any expected in the future.

Environmental standards for the uranium fuel cycle, as stated in 40 CFR Part 190, require that operations shall be conducted in such a manner as to provide reasonable assurance that the annual dose equivalent does not exceed 25 mrem to the total body, 75 mrem to the thyroid and 25 mrem to any other organ of any member of the public as a result of planned discharges of radioactive materials. The fifty-year skin, total body and thyroid doses to the nearest resident as a result of routine releases from the Morris Operation are 2.9×10^{-4} mrem, 6.8×10^{-5} mrem and 3.6×10^{-5} mrem, respectively.

The staff reviewed the FES for Dresden Nuclear Power Station as well as excerpts from the Commonwealth Edison Offsite Dose Calculation Manual to determine compliance with 40 CFR Part 190.[20] Based on this review it was concluded that Dresden Nuclear Power Station meets the requirements of 40 CFR Part 190. The calculated dose commitments from the Morris Operation are a small fraction of the dose commitments for Dresden (< 0.00001). It is therefore concluded that the Morris Operation meets the intent of 40 CFR Part 190.

7.4 NONRADIOLOGICAL IMPACTS

Nonradiological impacts include heat dissipated to the atmosphere and waste system and socioeconomic impacts.

7.4.1 Cooling System Impacts

Basin cooling is provided by finned-tube, forced-draft heat exchangers, which dissipate the excess heat to the atmosphere.

The cooling capacity of the system is 1.3×10^4 MJ/hr (1.2×10^7 Btu/hr) of which 6.3×10^3 MJ/hr (6×10^6 Btu/hr) is presently used. The system can be expanded to 1.7×10^4 MJ/hr (1.6×10^7 Btu/hr). The maximum heat load for storage of 750 MTU in basins 1 and 2 is 6.9×10^3 MJ/hr (6.5×10^6 Btu/hr) one-third of potential maximum cooling capacity.[3] The measured heat load as of August 1979, was 7.9×10^2 MJ/hr (7.5×10^5 Btu/hr).

There is a dissipation of 6.9×10^3 MJ/hr (6.5×10^6 Btu/hr) of heat to the atmosphere which is equivalent to the heat load generated by 180 homes [3.6×10^1 MJ/hr (3.4×10^4 Btu/hr) each]. This heat load dissipation does not measurably affect ground-level fogging and icing generated by Dresden Nuclear Power Station (DNPS) cooling lakes. Direct dissipation of heat to the atmosphere by cooling fins eliminates the problem of drift associated with wet systems. Based on the conservative assumptions, (1) discharge air at the maximum pool water temperature, and (2) stagnant meteorological conditions, we estimate that the heated air discharged from the coolers could raise the local air temperature surrounding the nearest ground vegetation, by about 0.07 Fahrenheit degrees (winter conditions). It is more likely that the temperature change effect would be less than 0.01 Fahrenheit degrees. This effect is not observable and would not be expected to affect any terrestrial ecosystem.

7.4.2 Waste System Impacts

Nonradioactive wastes consist of:

- exhaust combustion gases from the auxiliary diesel generator (operated 1 hr/wk)
- exhaust combustion gas from natural gas utility boiler
- liquid wastes from regenerating the water demineralizer units
- sanitary waste
- condensate blowdown from natural gas utility boiler and the air compressor cooling tower.

The licensee possesses a permit for operation of the natural gas utility boiler (see Table 1.4.1). The State Environmental Protection Agency does not set air pollution emission standards for existing boilers of this type and size.

Depending upon the nature of the liquid waste, it is routed to an evaporation pond or sanitary lagoons. No liquid wastes are discharged from the surface of the site.

Wastes from regenerating the water demineralizer units are routed to the evaporation pond. The pond's capacity is much greater than that required for spent fuel storage at the facility.

Sanitary wastes are routed to sanitary lagoons. From the lagoons liquid wastes are routed to a holding pond. Effluent from the holding pond can be chlorinated and used to irrigate the land in the facility tract under Illinois State Permit No. 1976-EB-408-1.

The lack of any off-site release of liquid effluents as a result of normal operation precludes any off-site impacts.

7.4.3 Socioeconomic Impacts

The principal socioeconomic impact associated with continued operation of the Morris plant results from employment of the labor force of 50 to 60 people. Any socioeconomic impacts resulting from the employment of a labor force of that size drawn from the labor pools of Grundy and Will Counties will be virtually incapable of measurement. Given the population of these counties, 50 to 60 family units will have little effect on demand for public services (school, fire and police protection) or on the general level of retail sales made in the area.

Another socioeconomic impact associated with the continued operation of the plant (Materials License No. SNM-1265) would be the taxes paid by the plant owner and its employees to local government jurisdictions.

7.4.4 Evaluation of Nonradiological Impacts

There will be no impacts to the surrounding community as a result of nonradiological effluents from the Morris Operation. Heat dissipated from the spent fuel stored is small (see Section 7.4.1) and will not have an effect on the surrounding environment. Combustion gases from the diesel generator and utility boiler will not be measurable offsite. The absence of any offsite release of liquid effluents precludes any impacts from this source.

Socioeconomic impacts on the surrounding community resulting from the renewal of Materials License No. SNM-1265 will be unchanged from current impacts of operation. After several years of operation, the impacts of operation of any plant become an integral part of the community and become a part of the norm. As discussed above, continued operation of the Morris plant results in no significant socioeconomic impacts to the surrounding community.

7.5 TRANSPORTATION OF RADIOACTIVE MATERIALS

All spent fuel transported to or from the Morris Operation will be shipped in heavily shielded casks by truck or rail. For truck shipments, the Nuclear Fuel Services NFS-4 cask is primarily used. This cask weighs 22.7 MT (25 tons) and will accommodate one pressurized water reactor (PWR) or two boiling water reactor (BWR) fuel assemblies. For rail shipments, General Electric's IF-300 cask will be used. This cask weighs 61.7 MT (68 tons) and will accommodate seven PWR or 18 BWR fuel assemblies.

Safety in transport of irradiated nuclear fuel is primarily based on the design of shipping casks. Casks used for shipment of spent fuel are certified by the Nuclear Regulatory Commission (NRC) to meet 10 CFR Part 71 requirements for Type B packages under both normal transportation conditions and transportation accident test conditions.

Spent fuel shipping casks are designed to withstand severe transportation accidents without significant loss of contents or increase in external radiation levels. The spent fuel contents of the casks are protected from damaging effects of impact, puncture, and fire by the massive structure and shielding of the casks. Auxiliary sacrificial structures around the casks on the vehicles afford additional protection.

These performance standards on packaging and the implementation of a quality assurance program assure that the packaging for radioactive materials is designed and constructed so that, under both normal and accident conditions, the radioactive material is unlikely to be released from the packaging.

7.5.1 Radiological Impacts of Transportion

Regulations of the Department of Transportation (DOT) for transport of spent fuel are given in 49 CFR Parts 170-189. These regulations set the criteria for radiation levels, surface temperature, surface contamination levels, shipping paper information, labeling, placarding, shipper certification, accident response, and general packaging. Spent fuel casks are transported in exclusive-use vehicles. Allowable radiation limits for closed, exclusive-use vehicles are:

- 1,000 mr/hr at 1 m (3 ft) from the external surface of the cask
- 200 mr/hr at any point on the external surface of the vehicle
- 10 mr/hr at 2 m (6 ft) from the edge of the vehicle
- 2 mr/hr in any normally occupied position in the vehicle.

The transportation of spent nuclear fuel to and from the Morris Operation is regulated by NRC requirements in 10 CFR Part 71 and DOT requirements in 49 CFR Parts 170-189.

The limitations on the radiation levels on the outside of packages of radioactive materials are provided to protect the employees, transport workers and the public from external radiation in the transport of radioactive material under normal conditions.

The analysis of radiological impacts from transportation of spent fuel to Morris is based on the quantities of spent fuel received by the Morris Operation through 1978, the origin of the fuel and the transport time from the point of origin to Morris. This data indicated an average travel distance of 2400 km (1500 mi), a transport time of 43 hours and a maximum of 400 truck shipments received per year. These data were applied to models contained in NUREG-0170 to obtain estimates of the dose to drivers and to the population along the route.[25]

The analysis performed assumes all transshipments are by truck. While the Morris Operation has received shipments by rail, these were all from Dresden Nuclear Power Station which is only one mile by railroad. These transshipments were not included in the analysis. The assumption that all transshipments are by truck results in conservative estimates of dose commitments.[25] Any increase in the fraction of transshipments by rail would result in lower estimates for dose commitments to crews and to the public.[25]

7.5.1.1 Radiological Impact on Drivers

On each 2400 km (1500 mile) trip two drivers would spend 43 hours in the truck cab. Based on the regulatory limit of 2 mrem/hr in normally occupied positions in the vehicle, the dose to the crew of drivers would be 0.17 rem per trip. Four hundred trips per year would result in a total of 68.8 manrem per year. The maximum annual dose to any one driver has been estimated to be 0.87 rem per year.[25]

7.5.1.2 Radiological Impact on the Public

The public exposed to low doses of radiation as a result of routine shipments include those exposed while traveling the same roadway as the transshipment, those living along the route and those exposed during stops along the route. The exposures from each of these situations was calculated based on data presented in NUREG-0170,[25] and past operating experience at the Morris Operation assuming a maximum of 400 MTU shipped at a rate of 200 MTU/year.

The population dose to persons traveling on the same highway as the transshipment is calculated to be 10.5 man-rem/yr, while the accumulated dose to persons living along the route is 3.1 man-rem/yr. The dose to persons during stops along the route is calculated to be 7.6 man-rem/yr. Thus the total dose commitment to the 20 million persons potentially exposed is 21.2 man-rem/yr or approximately 0.0008% of the annual dose due to naturally occurring sources.

The impacts on the public from the transshipment of low-level waste would not be any greater than those from transshipment of spent fuel. This analysis is based on the number of shipments per year required, the distance shipped and the DOT regulations governing such shipments.

7.5.2 Nonradiological Impacts of Transportation

The quantity of heat released to the environment from the transport of spent fuel is small. A one-year-old spent fuel assembly generates approximately 17 MJ of excess heat per hour (1.6×10^4 Btu/hr),[23] with all the heat generated in transit dissipated to the atmosphere. If the average speed of the truck transporting a spent fuel assembly was 55 km/hr (35 mph), approximately 0.5 MJ/mi (460 Btu/mi) of excess heat would be released. This release would amount to an addition of about 8% to the environmental heat load, produced by the estimated 6.3 MJ/mi (6,000 Btu/mi) of waste heat from the truck engine. The truck estimate is based on a 100-horsepower engine.[26] The heat that would be released to the environment as a result of transport of fuel assemblies is a small fraction of the heat generated by other traffic and is not considered to be significant.

8.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

A spectrum of accidents are discussed in this section. While it is unlikely that any of these accidents would occur, they are evaluated to determine the magnitude of any adverse effects. Facility accidents that are postulated to result in release of radioactive material to the biosphere include 1) a tornado-generated missile, 2) a fuel basket drop, and 3) a fuel assembly drop. Effects of accidents for which no release of radioactive material is postulated (criticality, cask drop and loss of basin cooling) are also evaluated in Section 8.1. Section 8.2 evaluates the effects of postulated transportation accidents. These accidents include 1) undetected leakage of coolant, 2) loss of neutron water shield, 3) cask over-pressurization and, 4) extra severe collision or overturn accident.

8.1 POSTULATED FACILITY ACCIDENTS

Three facility accidents are postulated that could result in the release of radioactive material to the biosphere. These include a tornado-generated missile, a fuel basket drop, and a fuel assembly drop. Source terms presented in Table 8.1.1 assume a fuel burnup of 44,000 MWd/MTU uranium and a fuel cooling period of 1 year. Fuel assembly inventory was calculated using the ORIGEN computer code.[27]

A comparison of the staff's calculated doses from these accidents with limits set forth in 10 CFR Part 20, 500 mrem total body, indicates that the dose from the tornado-generated missile that yields the highest release will be 1% of the limit. Doses to the closest resident and the regional population are presented in Tables 8.1.2 and 8.1.3.

Analysis of three additional accidents is included for which no release of radioactive material is postulated. These include criticality, cask drop and loss of basin cooling. The lack of any release is a result of the engineered safeguards included in the design and construction of the facility and the inherent characteristics of fuel assemblies, which include the fuel cladding and the chemical form of the fuel.

TABLE 8.1.1. Source Terms for Postulated Facility Accidents,[a] Ci

	Tornado-Generated Missile	Fuel Basket Drop	Fuel Assembly Drop
^{3}H	1.3×10^{1}	1.3×10^{1}	3.3
^{85}Kr	7.2×10^{3}	7.2×10^{3}	1.8×10^{3}
^{129}I	1.9×10^{-5}	1.9×10^{-5}	4.7×10^{-6}

(a) Fuel cooling time 1 year.

TABLE 8.1.2. Annual Dose to the Nearest Resident,[a] Mrem

	Skin	Total Body	Thyroid	Lung	Bone
Tornado-Generated Missile	4.1×10^{2}	5.7	3.7	5.8	1.8×10^{-3}
Fuel Basket Drop	5.4	7.5×10^{-2}	5.0×10^{-2}	7.6×10^{-2}	2.4×10^{-5}
Fuel Assembly Drop	1.3	1.9×10^{-2}	1.3×10^{-2}	1.9×10^{-2}	6.0×10^{-6}

(a) Located 800 m east of the Morris Operation stack.

TABLE 8.1.3. Annual Dose to the Regional Population,[a] Man-rem

	Skin	Total Body	Thyroid	Lung	Bone
Tornado-Generated Missile	2.6×10^{2}	3.5	2.3	3.6	1.0×10^{-2}
Fuel Basket Drop	1.2×10^{2}	1.8	1.2	1.8	5.2×10^{-4}
Fuel Assembly Drop	2.9×10^{1}	4.3×10^{-1}	3.0×10^{-1}	4.3×10^{-1}	1.3×10^{-4}

(a) 7.3×10^{6} persons.

8.1.1 Tornado-Generated Missile

This accident assumes a telephone pole 35 cm (13.5 in.) in diameter and 10.7 m (35 ft) long is propelled by tornadic winds. The telephone pole enters the basin and hits four PWR fuel assemblies with sufficient force to rupture all fuel pins. No credit has been taken for the protective factor of the overlying water that would retard the missile. The gap activity (noble gases and halogens) is released to the basin water. All noble gases and 0.002 of the halogens escape from the basin water and are vented to the atmosphere through the damaged facility structure. Total-body dose to the nearest resident would be 5.7 mrem or 0.042 of the dose from naturally occurring sources.

8.1.2 Fuel Basket Drop

This accident assumes that a fuel basket containing four pressurized water reactor (PWR) fuel assemblies is dropped while being transferred from the cask loadout basin to the storage basin. The gap activity (noble gases and halogens) in all of the fuel pins is released into the basin water. All noble gases and 0.002 of the halogens escape from the basin water and are vented to the atmosphere via the facility stack. Total-body dose to the nearest resident would be 7.5×10^{-2} mrem or 0.00056 of the annual dose from naturally occurring sources.

8.1.3 Fuel Assembly Drop

This accident assumes that a PWR fuel assembly is dropped during transfer from the shipping cask to a fuel basket. The gap activity (noble gases and halogens) in all of the fuel pins is released into the basin water. All noble gases and 0.002 of the halogens escape from the basin water and are released to the atmosphere via the facility stack. Total body dose to the nearest resident would be 1.9×10^{-2} mrem or 0.00014 of the annual dose from naturally occurring sources.

8.1.4 Criticality

The pool design coupled with fuel configuration and low fuel enrichments make the probability of a criticality accident extremely remote. Even so, as proven by operation of research reactors for many years, such a criticality accident would not generate sufficient energy to disperse any radioactive materials to the atmosphere.[7] Similarly, a criticality accident would not result in any exposure of personnel to any significant radiation if shielded by more than 3.7 m (12 ft) of water.[7] The cask unloading basin is 14.6-m (48 ft) deep and the spent fuel storage basins are 8.5-m (28 ft) deep. Under normal conditions there are 4.3 m (14 ft) of water covering the spent fuel in storage.

8.1.5 Cask Drop Accident

Rupture of the stainless steel basin liner is postulated as a result of a cask drop or tipping accident. A rupture of the stainless steel basin liner would result in the rapid flow of basin water into the channels between the stainless steel liner and the surrounding concrete. In 1972, the Morris Operation experienced a tipping accident which resulted in the tearing of the stainless steel basin liner and flow of water into the channels.

The accident in 1972 demonstrated that leakage of basin water into the channels behind the stainless steel basin liner would be limited to about 2,500 gallons. In that case, the leakage rate was rapid until the water in the collection channels was level with the pool water; then all water flow from the basin ceased. The water in the collection channels was transferred to the Low Activity Waste Vault and the stainless steel basin liner was repaired. This accident effectively demonstrated the confinement character of the reinforced concrete and the bed rock in which it is embedded. For this reason, the NRC staff expects no significant leakage of basin water through the surrounding rock as a result of a cask drop accident rupturing the stainless steel basin liner.

If the structrual concrete contained stress cracks of sufficient size to allow free flow of water, the hydraulic head caused by perched ground water would result in a flow of these ground waters into the channels between the concrete and the stainless steel basin liner. Continual monitoring of the sump which would collect water from the channels has confirmed that there has been no intrusion water leaking into the channels.

The cask itself is designed to survive, leak-tight, a drop from the maximum lift height in the unloading area. While a cask drop may cause limited structural deformation of the foundation and a tear in the liner, the integrity of the cask would not be jeopardized.

8.1.6 Loss of Basin Cooling

Based on the total storage capacity of the basin facility and the projected fuel receipt rate, the heat load will be on the order of 6.6×10^3 MJ/hr (6.5×10^6 Btu/hr).(3) Loss of basin cooling would cause the temperature of the basin water to rise at a rate of 1.1°C (2°F) per hour. Based on a basin operating temperature of 35°C (95°F) and disregarding heat losses by evaporation and to the concrete basin, boiling of the basin water could occur in 2 to 3 days.

An analysis of loss of basin cooling by the licensee, which considered heat losses to the concrete basin and by evaporation, concluded that basin water could reach a steady-state condition at a maximum temperature of 85°C (185°F) in approximately 4 days.(3) This analysis also concluded that 9 days would be required to evaporate enough water to expose the tops of the stored fuel assemblies if no makeup water were supplied.

The slow heat increase calculated from either analysis affords adequate time to initiate repairs and assure adequate makeup water is available. Makeup water is supplied from the two onsite wells but could be obtained from the nearby river if necessary. The slow heatup rates and adequate supplies of makeup water assure the integrity of the fuel stored in the basin will be maintained while repairs are made.

8.2 POSTULATED TRANSPORTATION ACCIDENTS

Transportation accidents occur in a range of frequencies and severities. Most accidents occur at low vehicle speeds. As presented in Table 8.2.1, the severity of accidents is greater at higher speeds, but the frequency decreases as the severity increases. Transportation accidents usually involve some combination of impact, puncture, fire, or submersion in water.

The accident risk projected for 1985 is 0.0166 latent cancer deaths per year for all types of radioactive shipments.[25] Accident risk from spent fuel shipments makes up 2.5% of the total risk from all types of radioactive shipments in the U.S.[25] Proposed shipments to the Morris Operation in turn represent only 10% of the projected spent fuel shipments in the U.S. in 1985. Therefore, the risk projected for spent fuel shipments to the Morris Operation is 0.25% of the latent cancer deaths per year predicted for all types of radioactive materials shipments in the nation.

In spite of the low annual risk of specific accidents, the occurrence in a very high-density population zone of a highest severity accident (severe enough to rupture the package and eject respirable radioactive material into the environment) is estimated to result in zero early fatalities and one latent cancer fatality for a spent fuel rail shipment and large decontamination costs.[25] Although such accidents may be possible, their probability of occurrence is very small.

TABLE 8.2.1. Classification of Accident Severity[26]

Accident Severity Category	Vehicle Speed at Impact, mph	Fire Duration, hr	Probability per Vehicle mile (truck)
Minor	0-30	0-1/2	1.3×10^{-6}
	30-50	0	
Moderate	0-30	1/2-1	3.0×10^{-7}
	30-70	1/2	
Severe	0-50	1	8.0×10^{-9}
	30-70	1/2-1	
	70	0-1/2	
Extra Severe	50-70	1	8.0 x 10-13
	70	1/2-1	
Extreme	70	1	2.0×10^{-14}

Four transportation accident scenarios are briefly discussed below to facilitate more of a perspective of what type of accident would be included in various severity classes. An example of a minor accident would be an undetected leakage of coolant, while a loss of neutron water shield or cask overpressurization would represent moderate accident conditions. Finally, a drastic impact and fire would represent extreme accident conditions.

8.2.1 Undetected Leakage of Coolant

This accident assumes that 0.12% of the free gases are released to cask cooling water.(26) The cask coolant cavity is assumed to leak at the rate of 0.001 cm^3/sec, with a total of 90 cm^3 of coolant released.(26)

Of the activity released from the cask, 0.1% would be dispersed as an aerosol. The small quantities of activity released over the length of the transportation route would result in insignificant doses. Total-body dose commitment to the nearest exposed individual would be on the order of 1×10^{-4} mrem. This dose assumes that all the aerosol is released at one time and that the nearest exposed individual is 100 m downwind.

8.2.2 Loss of Neutron Water Shield

This accident assumes that a truck collision causes a rupture of the jacket containing borated water, which shields against neutron radiation. The resultant loss of neutron shield water would cause no release of radioactive material. Neutron doses emanating from the damaged cask would be 0.6 mrem/hr at 10 m and 2×10^{-2} mrem/hr at 50 m. Assuming a member of the public remains 10 m from the cask for 2 hours, the dose received would be equivalent to 1% of annual background due to naturally occurring sources.

8.2.3 Cask Overpressurization

This accident assumes that the truck transporting the loaded shipping cask is involved in an overturn accident and associated fire that lasts longer than 30 minutes. The cask cavity would overpressurize and the pressure relief valve would operate to relieve the pressure, resulting in the release of 0.1% of the cavity coolant. The cavity coolant would contain radioactivity, due to transportation of assemblies and assuming 0.12% release of free gases. The total-body dose to the nearest exposed individual is calculated to be 0.01 mrem. This dose commitment assumes the individual is exposed for two hours to the maximum concentration achievable during the accident and is 100 m downwind.

8.2.4 Extra Severe Collision or Overturn Accident

It is assumed that collision or overturn accident causes the spent fuel truck cask to be subjected to extra severe impact and fire lasting more than 1 hour. Even if this accident does occur, the probability of cask failure is extremely low. The design of the cask, as discussed in Section 7.5, is such that a massive rupture and subsequent release is precluded. However, for this analysis, it is assumed that the integrity of the cask is broken by a breach of the closure head seal by gasket failure or by the cask lid bolts being sheared off. Of the fuel rods, 10% would be perforated and 100% of the cavity coolant would be released.[26] The first year total body dose to any one individual standing 100 m (300 ft) from the accident would be 32 mrem or 27% of natural background. Based on health effect estimates,[33] it is calculated that this dose might increase the normal mortality risk of cancer from 1 in 5 to 1.000004 in 5. This calculation, coupled with the already low probability of the occurrence of the accident, establishes that the radiological risk from this type of accident is insignificant.

8.3 EFFECTS ON THE PUBLIC FROM POSTULATED ACCIDENTS

As calculated in the above sections, in the event of an accident a release of radioactive materials would be very small, and the radiation dose to any individual very small, the effects to be considered are long-delayed somatic and genetic effects. Based on the following evaluation, however, the staff has concluded that there will be no detectable radiological impacts from the postulated accidents. Even as a consequence of the extra severe accident involving greater doses, as discussed in the preceding section, the radiological risk from that type of accident would be insignificant. The effects that must be considered are cancers that may result from external whole-body exposures and exposure from radioactive materials deposited in lung, bone, thyroid, and other organs; and genetic effects, reflected in future generations, due to exposure of the germ cells.

Assessment of delayed effects of low doses of radiation is by necessity indirect. This is because their incidence is too low to be observed against the much higher background incidence of similar effects from other causes. Both the maximum individual doses and the upper bound population doses resulting from the proposed action are fractions of the doses individuals and the population receive from naturally occurring radiation. Even in controlled studies with experimental animals, one observes a low incidence of effect that cannot be distinguished from the level of effect in unexposed animals, at exposure levels far higher than those predicted to result in this assessment. Hence, one can only estimate a relationship between health effects and radiation dose, basing this estimate upon observations made at very much higher exposure levels, where effects have been observed in man, and carefully studied animal experiments.

Utilizing the above dose models, the staff has concluded that at the level of exposure under consideration the postulated health effects (based on the BEIR linear dose response model)[28] would be less in quantity and no different in kind from the postulated health effects resulting from natural background radiation.

9.0 SAFEGUARDS FOR SPENT FUEL

Irradiated (spent) fuel removed from light water cooled power reactors (LWRs) contains low enriched uranium, fission products, and plutonium and other transuranics. It is highly radioactive and requires heavy shielding for safe handling. Theft or diversion of spent power reactor fuel by subnational adversaries with the intent of utilizing the contained special nuclear material (SNM) for nuclear explosives is not considered credible.[7] Sabotage of spent fuel might be within the capability of potential adversaries, however, and therefore may constitute a possible hazard to local populations. Sabotage of spent fuel could be attempted either in transit or at a fixed site. Since these two situations are covered by different regulations, they are discussed separately below.

There have been no deliberate acts of spent fuel sabotage directed against a licensed activity which culminated in a direct or indirect danger to the public health and safety by exposure to radiation.[29] It is apparent, however, that there may be people who have the skills necessary to plan and execute an operation against the nuclear industry, and that conceivably such people could be gathered together and motivated to conduct such an operation. The possibility always exists that at some point in time a disgruntled employee or politically motivated group may attempt some act that would be classified as a threat to nuclear activities.

9.1 SPENT FUEL IN TRANSIT

Spent fuel in transit is considered to be neither an attractive nor a practical target for sabotage. Shipments of spent fuel are protected in accordance with interim physical protection requirements as described in 10 CFR§73.37.

Massive, durable containers (casks) weighing 25 to 100 tons are used for transport of the spent fuel assemblies. Criminal acts involving the intentional opening of these casks would require an appreciable amount of time, elaborate planning, shielding and handling facilities. Spent fuel cask covers would be very difficult to remove by hand because of their bulk and weight. In practice, overhead cranes are employed for the uprighting of the massive cask (which in itself is a difficult operation) and for the removal of the cover. This operation is performed remotely, usually underwater, because of the high radiation levels experienced upon opening the cask. In the absence of shielding, an individual who was successful in unauthorized removal of the cover would be immediately exposed to lethal radiation.

Although it appears that no sabotage threat to spent fuel shipments exists, the response of the cask and its spent fuel contents to sabotage has been studied for a wide range of sabotage scenarios.

For any of the sabotage scenarios considered, it has been found that successful sabotage would involve breaching the cask in a way that would discharge a portion of the radioactive contents into the environment.

Deliberate acts directed at breaching the cask through mechanical means or through the use of projectiles most probably would not be successful owing to cask design and the great difficulties associated with mechanical disassembly.

Sabotage through the use of high explosives could likely produce cask penetration. However, the effort required would be extensive. Various sabotage scenarios involving the use of high explosives were considered in a recent NRC supported study.[30] The study has been issued in draft form and is currently under review by the NRC staff. The study concludes that the only realistic way to attack a spent fuel shipment in order to cause dispersal is with high explosives. The amounts of explosives considered range upward into several hundred pounds and even tons. The explosives configurations discussed include airblast, breaching charges, shaped charges, and platter charges. The details of the response of a cask and its contents to explosive sabotage are not well understood at this time and are under study as explained in the next section. There is, however, general agreement among the study authors and the NRC staff reviewers that skillful use of large quantities of high explosives would be required to achieve a release of the radioactive contents.

Although it is unlikely that a sabotage threat exists, and although it would require extensive effort to sabotage the cask so as to cause dispersal of radioactive materials, the consequences of such a scenario have been calculated and evaluated in NUREG-0170[25] and NUREG-0575.[7] However, presently available information is not conclusive. The NRC Staff has in progress a program designed to provide confirmatory data on the response of spent fuel and spent fuel casks to explosive attack. This program, however, is not expected to yield useful results until later this year.

The Commission has issued interim regulations (in 10 CFR Part 73) to strengthen the protection of licensed spent fuel shipments. These requirements may be modified in the future based upon this confirmatory research program.

These interim measures are designed to provide additional assurance that response forces can be summoned in a timely manner, if needed, and to further lower the level of risk. Permanent measures for the protection of spent fuel will be adopted to the extent that research shows they are needed.

After considering the absence of any information confirming an identifiable threat, the difficulty of breaching a spent fuel cask and fragmenting the spent fuel, the magnitude of the estimated consequences of successful sabotage,* and the applicable protection measures, the NRC Staff has concluded that the proposed shipments do not constitute a serious risk to the public health and safety.

9.2 FIXED SITE SAFEGUARDS

To the extent that acts of sabotage initiate sequences of events much like those initiated by accidents, the measures designed into the spent fuel facility at the Morris site for mitigation of consequences of such accidents also provide a degree of protection against potential releases resulting from sabotage. However, the possibility exists that potential saboteurs may be capable of overcoming the inherent protection and engineered safety features at the facility in an attempt to create a radiological hazard. Accordingly, analyses of the potential environmental effects of certain sabotage events involving aged spent fuel such as that to be stored at this station were

*See page 5-9 of reference 7.

developed and are presented in NUREG-0575, Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Reactor Fuel.[7]

Although there is no information available confirming the existence of any identifiable threat to commit acts of sabotage against a spent fuel storage facility,[29,31] protection against such acts and their possible consequences is dictated by prudence. For this reason, NRC regulations include requirements for the physical protection of spent fuel against sabotage.

The Commission's requirements for protective measures for spent fuel at fixed site facilities are contained in 10 CFR Part 73, especially Section 73.50. Principal features include requirements for a security plan, protective forces including armed guards, physical and procedural access controls, detection aids, communications systems and liaison with local law enforcement agencies.

The licensee has submitted to the Commission a physical security plan responding to these requirements. This plan has been reviewed by the staff and has been found to be satisfactory. The implementation of this plan has subsequently been inspected and also found acceptable. The commitments made in the plan and any additional license conditions will provide the level of protection required by 10 CFR Part 73 for the licensed activities at the facility.

Therefore, in consideration of

a) the absence of any information confirming an identifiable threat to the proposed storage activity,

b) the features of the spent fuel storage pool design that provide inherent protection against potential releases,

c) the protection features required by the regulations which provide deterrence and a capability for summoning response forces in a timely manner, and

d) the limited potential for radiological consequences as reflected in the staff's analysis of certain sabotage events, it has been determined that the sabotage-related risks to the public health and safety related to the storage of spent fuel, as may be authorized by renewal of the subject license, are acceptably small.

10.0 DECOMMISSIONING

At the end of the period specified in the operating license, the operator of a nuclear facility must renew the license or must dismantle the facility and dispose of its components. Before the operating license expires, if technical, economic, or other factors are unfavorable to continued operation of the plant, the operator may apply for license termination and dismantling authority. If the operator elects to apply for a license to operate, he must show that he possesses or has reasonable assurance of obtaining the funds necessary to cover the estimated costs of permanently shutting the facility down and maintaining it in a safe condition. General Electric Company's current general revenues and retained earnings are sufficiently large that the staff has reasonable assurance that the funds necessary to decommission the Morris Operation will be available at the time such an action is taken.

On December 22, 1978, General Electric submitted a decommissioning plan for its Morris Operation to the Commission for review. The decommissioning plan has also been incorporated as an appendix to the CSAR. The method was selected by General Electric for decommissioning the site at some future date. Specifically, the plan covers the events from the decision to terminate licensed operations until the time the Commission rules the license is no longer required.

Specifically, the plan addresses the history of operations at the site, describing the type of materials handled and the layaway program initiated in 1975 for unused portions of the fuel reprocessing facilities onsite. Also described are those facilities used in support of spent fuel and the type of contamination to be expected in these areas.

The plan further describes the objectives of the decommissioning effort and the tasks designed to meet these objectives. The primary objective is to decontaminate the site to a point where continued NRC licensing is no longer required, subsequently permitting unrestricted public use. The deposition of various bulk materials is addressed including waste vault contents and contaminated equipment. The plan as it is presently envisioned involves removal of contaminated equipment and materials, decontamination of surfaces, then backfilling and covering all large below-grade structures. The main building, including the canyon area, will be left in place after decontamination is complete. The plan assumes that decontamination efforts are acceptable to the regulatory authority.

The task of decommissioning the Morris Operation is estimated to take 3 years at a cost of $6 million (1978 dollars).

11.0 ALTERNATIVES

The following alternatives were evaluated:

- renewal of Materials License No. SNM-1265 (750 MTU);
- limited renewal of Materials License No. SNM-1265;
- termination of operations at Morris
- construction of an independent spent fuel storage installation at another site owned by GE;
- federal government acceptance of spent fuel;
- reprocessing;
- storage at another away-from-reactor storage location.

11.1 RENEWAL OF MATERIALS LICENSE NO. SNM-1265 (750 MTU)

The renewal of Materials License No. SNM-1265, which is the proposed action by the licensee, would allow continued operation of Morris. The associated impacts of continued operation are considered throughout this document.

11.2 LIMITED RENEWAL OF MATERIALS LICENSE NO. SNM-1265

Limited renewal of Materials License No. SNM-1265 would involve licensing the Morris Operation to store the existing inventory of approximately 315 MTU of spent fuel. The facility would continue to operate, but would not be licensed to receive any additional fuel with the exception of that already contracted. As a result, the impacts associated with transshipment of spent fuel would be eliminated, while the impacts of spent fuel handling onsite would be reduced. However, as illustrated in Sections 5.2, 7.1 and 7.2, the impacts from routine operations and transportation including the projected accident risks are small. Any reduction of impacts, both real and postulated, must be compared to the benefit derived from licensing the Morris Operation to receive and store 750 MTU of spent fuel.

Variations of the limited renewal concept include licensing for storage of the existing inventory plus:

- fuel storage space previously contracted to San Onofre Nuclear Station ($\sim$40 MTU);
- capacity to curtail reactor shutdowns caused by lack of storage space;
- capacity to curtail reactor shutdowns caused by lack of storage space with fuel returned to the generating reactor once the storage problem there is solved.

The first of these variations is essentially the same as licensing for existing inventory except transportation of 40 MTU of spent fuel would be required. San Onofre fuel is presently (at the time of publication) being shipped.

The second of these variations could result in the same impacts as granting a full license. As less reactor storage space becomes available, more reactors could face emergency shortfalls of storage space thereby requiring the use of storage space at Morris. This could eventually lead to a total use of the available storage space.

The third variation could result in significant transportation of spent fuel to and from Morris as reactor storage basins become full. The impact of this variation could possibly become greater than the impacts from renewal of the full license. The increase would be a direct result of the double handling and transportation of spent fuel.

Limited license renewal is a viable alternative that may or may not have smaller impacts than that of renewal of the existing license. The actual impacts encountered would be dictated in part by the restrictions placed on the license. Comparing impacts that may be saved with those of continued operation and the associated benefits, there are not any advantages to this alternative.

11.3 TERMINATION OF OPERATIONS AT MORRIS

Termination of operations could result from two causes, the first being a General Electric Company decision to cease operation and the second being an NRC denial of license renewal. Either of these has the same effect, i.e., termination of all future operations and a need to dispose of fuel presently stored at the Morris facility.

The most pressing problem to be overcome when initiating this alternative (the disposition of fuel now in storage) could be resolved by construction of a new independent spent fuel storage installation (ISFSI) at another site, sending the fuel to a Federally owned away-from-reactor storage location or returning the fuel to the originating reactor facility. The first two of these solutions are discussed in Section 11.4 and 11.5, respectively. As discussed in those sections, construction of an ISFSI would result in significant time delays and significant costs in terms of dollars expended and environmental impacts of construction and operation. Due to the environmental impacts and cost considerations, this is not the preferred alternative. Also, at this time the Federal Government has no facilities to accept the spent nuclear fuel now stored at Morris. The third solution, returning the fuel to originating reactors, is not possible due to fuel storage shortages at some of the originating reactors, contract and warranty obligations. In addition, all of the three solutions discussed would require the transshipment of 315 MTU of spent fuel stored at Morris to an alternate site.

As a future consideration, the use of this alternative removes one spent fuel storage installation now available to selected utilities, causing them to rely more heavily on transshipment and expansion of at-reactor storage capacities or to cease reactor operation. It should be noted that utilities are trying to take advantage of all options available to them. The options of transshipment, expansion and reactor cessation may all have an impact on the amount of fuel to be stored in the future; however, these alternatives are available to the utilities, not to General Electric.

For the reasons discussed above, the termination of Materials License No. SNM-1265 is not the preferred alternative.

11.4 CONSTRUCTION OF AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION AT ANOTHER SITE OWNED BY GE

The construction of an independent spent fuel storage installation (ISFSI) at another site would require approximately 60 months. During this time the Morris Operation would need a limited license for storage of the existing inventory.

Costs of such an ISFSI have been estimated to vary from $44,000 to $78,000 per MTU. To construct a facility capable of storing the 350 MTU of spent fuel now at Morris or under contract for storage at Morris would be $15,400,000 to $27,300,000 with the highest figure most accurate for a new site.

While this alternative is viable, there is no reason for the licensee to construct a new facility as a replacement for the Morris Operation. The environmental impacts of this alternative would be the accumulative impacts of construction, operation, and the need to transship the present inventory to the new site. The impacts of continued operation of a new facility would be comparable to those of continued operation of the Morris facility.

This alternative would not result in reducing any impacts, but instead would increase both environmental and economic costs. The alternative of constructing an independent spent fuel storage installation on another site is not preferred to that of continued operation of the Morris operation.

11.5 FEDERAL GOVERNMENT ACCEPTANCE OF SPENT FUEL

The Federal Government has proposed to accept and take title to spent nuclear fuel from utilities. This is an extension of President Carter's April 7, 1977, policy statement concerning commercial reprocessing of spent nuclear fuel. On October 18, 1977, the Department of Energy accepted ultimate responsibility for storing spent nuclear fuel.

While DOE has accepted the ultimate responsibility for storing spent fuel, no immediate relief from storage shortage problems has been forthcoming. Pending the evaluation of the various concerns and alternatives, utilities are faced with imminent fuel storage problems and no approved plans for ultimate deposition of spent nuclear fuel. Construction of factilities to store spent nuclear fuel cannot await the outcome of studies aimed at reducing the risk of proliferation or the implementation of geologic disposal.

The Department of Energy has announced plans possibly to use spent fuel storage capacities at reprocessing facilities as interim away-from-reactor storage installation. Approval of funding for such an undertaking is now before the Congress. Although legislation to provide for ultimate disposition of spent fuel by the Federal government is presently pending Congressional action, there have been no actions taken that would eliminate General Electric's responsibility of maintaining the Morris Operation. Use of such facilities is discussed in the following section. General Electric will not, however, be required to make use of government sponsored away-from-reactor storage. Such actions will be voluntary.

11.6 REPROCESSING

On April 7, 1977, President Carter issued a policy statement concerning commercial reprocessing of spent nuclear fuel. On October 18, 1977, the Department of Energy accepted ultimate responsibility for storing spent nuclear fuel. On December 23, 1977, the Commission made the decision to defer hearings on the Generic Environmental Impact Statement on the Use of Mixed Oxide Fuels in Light Water Cooled Reactors.[7] These actions stopped progress indefinitely toward reprocessing of spent nuclear fuel. While these events occurred after the decisions to cease operation of the Nuclear Fuel Services Facility (NFS) and Midwest Fuel Recovery Plant, they have affected licensing actions at the Allied General Nuclear Services, Barnwell Facility, and the process toward development of reprocessing facilities in general.

For a time the Nuclear Fuel Services (NFS) plant at West Valley, New York, actually operated. However, after a shutdown for extensive alterations and expansion, the conclusion was reached that these changes were commercially impractical and the facility was not reopened for reprocessing. The General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois, never operated as a reprocessing plant and is now licensed for spent fuel storage only and is the subject of the proposed action. A proposed plant, the Allied-General Nuclear Service (AGNS) plant in Barnwell, South Carolina (the subject of hearings before the Commission), and the Exxon plant proposed for construction in Tennessee (which was docketed for license review) have not been approved.

Because of current U.S. policy which has placed a ban on the reprocessing (and recycling) of LWR fuel for an indefinite period of time and the decision by the Nuclear Regulatory Commission to terminate proceedings on pending or future plutonium recycle-related license applications, the alternative of reprocessing is not viable.

11.7 STORAGE AT EXISTING AWAY-FROM-REACTOR STORAGE INSTALLATIONS

Away-from-reactor (AFR) storage installations now in existence were constructed as a part of reprocessing facilities. As of this writing, two facilities (other than GE) exist which have the capability to receive spent nuclear fuel for storage. These are the Nuclear Fuel Services (NFS) plant at West Valley, New York, and the Barnwell Fuel Receiving and Storage Station (FRSS) located near Barnwell, South Carolina.

The NFS plant has a spent fuel storage capacity of 260 MTU with 165 MTU presently stored in the facility. The NFS plant was shut down in 1972 for alterations and expansions. On September 22, 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for commercial storage.

Allied General's Barnwell plant has storage basins with a capacity to store 400 MTU of spent fuel. When construction of AGNS Barnwell Fuel Receiving and Storage Station (FRSS) was completed, Allied General Nuclear Services (AGNS) applied for a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell. Review of this licensing action has been stopped as a result of postponement of the Fuel Receiving and Storage Station hearings. Recent statements by Allied General indicate that this capacity will not be made available for interim storage.(14)

However, all of these facilities are under consideration by the Department of Energy as potential federal away-from-reactor spent fuel storage installation.(14) While storage space is available at existing fuel reprocessing facilities, the space is not available to GE or the industry.

11.8 SUMMARY

Based on the need for additional spent nuclear fuel storage capacity, cost consideration and the lack of environmental impacts, the staff recommends renewal of the Morris Operation license. The reasons for this recommendation are:

- Depending on the variation selected, limited license renewal can
  - limit the usefulness of the Morris Operation by not allowing use of up to 440 MTU of storage space;
  - result in ultimate impacts equal to or possibly greater than those for full licensing of the facility. (This is caused by the ultimate filling of the available space due to "emergency" needs for storage space and/or the continued shipping of fuel to and from reactors needing temporary relief from storage congestion for reracking operations.)
- Termination of operation is an available option. However, the utilization of this alternative relies heavily on other options that are either not available or due to environmental and economic cost are not considered reasonable.
- Construction of an independent storage installation at another site would result in greater environmental impacts and place a large financial burden on the licensee to replace an existing facility.
- While the Federal Government has accepted ultimate responsibility for storing spent fuel, relief from storage shortage problems has been slow. As of this writing DOE has requested authorization and funding to use existing storage capacities at reprocessing facilities as federally sponsored, away-from-reactor storage locations, although no Congressional action has been taken.
- With the Commission's decision to terminate the generic study on plutonium recycle use of mixed oxide fuel (GESMO) in December 1977 (42 FR 65334) in deference to the President's nonproliferation policy, commercial reprocessing has been indefinitely deferred in the United States.

- Other away-from-reactor storage sites in the United States are either not accepting spent fuel or are not licensed to receive spent fuel. Use of these facilities is, therefore, not available at this time.

12.0 SUMMARY AND CONCLUSIONS

12.1 SUMMARY OF ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

The environmental impacts associated with renewal of Materials License No. SNM-1265 are described and evaluated in Section 7.0. As discussed in that Section, the impacts from the Morris Operation are due to small releases from radioactive materials, nonradioactive effluents and heat to the atmosphere. The lack of liquid releases offsite precludes any impact as a result of normal operations.

Occupational exposures will be within the limits of 10 CFR Part 20 and are considered to be as low as reasonably achievable. See Section 5.5 and Section 7.3.3.2. Based on the calculated doses to the regional population and past operating experience, the staff has concluded that there will be no detectable radiological impact to the regional population from normal operation of the plant. See Section 7.3. Additionally, there are no impacts to the surrounding community as a result of nonradiological effluents from the plant. See Section 7.4. Heat dissipated from spent fuel stored is small and will not affect the surrounding environment. See Section 7.4. Combustion gases from diesel generator and utility boiler operation will not be detectable offsite. See Section 7.4.

The impacts associated with land and water use are insignificant as described in Section 7.0. The impacts from transportation of spent fuel are small. See Section 7.5. The maximum dose to individuals and the population along the routes used would be 0.0008% of the annual dose due to naturally occurring sources. See Section 7.5.1.2 No impact on the public is expected as a result of the dose calculated. See Section 7.5.1.2. There will be no significant socio-economic impacts as a result of this proposed action. See Section 7.4.3 and Section 7.4.4.

12.2 COST-BENEFIT BALANCE

The benefit from the proposed action, renewal of Materials License No. SNM-1265, is the continued use of the Morris Operation to store the present inventory of 315 MTU of fuel while allowing the maintenance of a reserve storage capacity of about 350 MTU to meet future needs within the nuclear industry.

The cost incurred by renewing the license is a small, undetectable dose to the regional population living near the plant and up to 40 man-rem/yr to the work force. See Section 7.0. These doses maintained over 20 years would not result in any health effects.[28] In addition, a total of approximately 180 man-rem would be incurred from transportation of spent fuel necessary to fill the remaining capacity at the Morris Operation.

The benefits derived from renewing the license for Morris outweigh the cost involved. This is especially true when the costs of not renewing the license are analyzed. These costs include the dose due to transportation of the 315 MTU of spent fuel now at Morris to a new location, about 140 man-rem, and the fact that doses to the work force at the new storage location will be similar to those incurred for routine storage operations at Morris.

12.3 CONCLUSION

On the basis of this Environmental Impact Appraisal, the staff concludes that the proposed licensing action will not significantly affect the quality of the human environment and that there will be no significant environmental impact from the proposed action. Therefore, the staff has found than an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Further, for the reasons set forth in Section 12.2, the staff concludes that the benefits associated with the proposed licensing action outweigh the costs. Accordingly, the staff believes that the action called for is renewal of the license.

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APPENDIX A

RADIOLOGICAL MODELS AND ASSUMPTIONS

INTRODUCTION

This Appendix describes the models and assumptions used to make estimates of the potential nearest resident doses from normal operations and from postulated accidents at General Electric Morris Operation. A separate reference list for the Appendix appears on page A-21.

A.1 MODELS USED FOR DOSE ESTIMATES TO AN, NORMAL PLANT OPERATION

The fundamental relation for calculation of radiation dose to man is given as follows for any radionuclide:

$$R_{ipr} = C_{ip} U_p D_{ipr} \qquad (1)$$

where

R_{ipr} = the dose rate to organ r from nuclide i via pathway p

C_{ip} = the concentration of nuclide i in the media of pathway p

U_p = usage: the exposure rate or intake rate associated with pathway p

D_{ipr} = a dose factor: a number specific to a given nuclide i, pathway p and organ r, which can be use to calculate radiation dose rate from exposure rate to a given concentration of a radionuclide or the intake rate of that radionuclide.

The three terms comprising Equation 1 are discussed in the following subsections.

A.1.1 CONCENTRATIONS IN ENVIRONMENTAL MEDIA, C_{ip}

Concentrations in air, soil or food are calculated as an integral part of computer programs developed for dose calculations.[1]

A.1.2 USAGES, U_p

Hours of exposure to external sources of radiation and intake rates of ingested food are supplied for each calculation. Since the principal contributors for external air submersion dose are noble gases, the assumption is made that the air concentrations of radionuclides will be essentially the same indoors as outdoors.

A.1.3 DOSE FACTORS, D_{ipr}

Equations for calculating internal dose factors are derived from those given by the International Commission on Radiological Protection (ICRP) for body burden and maximum permissible concentrations (MCP), and have previously been published.(2,3,4) Effective decay energies for the radionuclides are calculated from the ICRP model, which assumes all of the radionuclide is in the center of a spherical organ with an appropriate effective radius. Where data are lacking, metabolic parameters for the Standard Man are used. These dose factors have units of mrem/yr per pCi/yr taken into the body, either via ingestion or inhalation.

The dose factors for external exposure to air are derived on the assumption that the contaminated medium is large enough to be considered an "infinite volume" relative to the range of the emitted radiations. Under that assumption the energy emitted per gram of media is equivalent to the energy absorbed per gram of media. Conversion from MeV per disintegration per gram to rem is made and corrected for the difference in energy absorption between air or water and tissue, for the quality factor of the radiation under consideration, and for physical geometry of each specific exposure situation.

The dose from submersion in air is an external dose either to the skin or to both the skin and total body, depending on the penetrating power of the radiation emitted by the airborne radionuclides. Only beta and gamma radiation, which could penetrate 7 mg/cm^2 of tissue, is considered in calculating skin dose. Gamma radiation dose at a 5-cm depth in tissue is used for calculating external dose to the total body (and for internal organs). These dose factors have units of mrem/hr per pci/m^3 air and mrem/hr per pCi/ℓ water.

Material deposited from the air onto the ground represents a fairly large, nearly uniform, thin sheet of contamination. The factors for converting surface contamination in pCi/m^2 to gamma dose at 1 m above a uniformly contaminated plane have been described.(1,4,5) Dose factors for exposure to soil have units of mrem/hr per pCi/m^2 surface.

A.2 PATHWAY EQUATIONS

Individual equations tailored to each specific exposure pathway are derived from Equation 1. The principal difference among pathways is the manner in which the radionuclide concentrations are calculated. This section develops the set of equations required for the atmospheric pathway model.[1]

A.2.1 FARM PRODUCTS

The model presented for estimating the transfer of radionuclides (except for ^3H and ^{14}C) from air to plants through both leaves and soil to food products was derived by Soldat.[2] Soldat developed the model for a study of the potential doses to people from a nuclear power complex in the year 2000.

Deposition on Food Products

The source of the radionuclide contamination of the foods is by deposition of airborne radionuclides.

Deposition Directly from Air

$$d_1 = 86{,}400\ \chi_i\ V_{di}\ \text{(air deposition)} \qquad (4)$$

86,400 = dimensional conversion factor (sec/d)

V_{di} = deposition "velocity" of radionuclide i (m/sec)

χ_1 = annual average air concentration (pCi/m^3) of radionuclide i.

Concentration in Vegetation

The concentration of radioactive material in vegetation resulting from deposition onto the plant foliage and uptake from the soil of prior depositions on the ground is given in Equation 5

$$C_{iv} = d_i \left[\frac{r\ T_v(1 - e^{-\lambda_{Ei} t_e})}{Y_v\ \lambda_{Ei}} + \frac{B_{iv}(1 - e^{-\lambda_i t_b})}{P\ \lambda i} \right] e^{-\lambda_i t_h} \qquad (5)$$

where:

C^{iv} = concentration of radionuclide i in edible portion of plant v (pCi/kg)

r = fraction of deposition retained on plant (dimensionless), taken to be 0.25

T_v = factor for the translocation of externally deposited radionuclides to edible parts of plants (dimensionless). (For simplicity this factor is taken to be independent of radionuclide and set to 1 for leafy vegetables and fresh forage, and 0.2 for all other produce including grain. Reference 3 lists values of this parameter that vary with nuclide.)

λ_i = radiological decay constant for radionuclide i (d^{-1})

λ_{Ei} = effective removal constant of radionuclide i from plant (d^{-1}) $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_w = weathering removal constant = 0.693/14 (d-1)

t_e = time of above-ground exposure of crop to contamination during growing season (d)

Y_v = plant yield (kg(wet weight)/m^2)

B_{iv} = concentration factor for plant uptake of nuclide i from soil (pCi/kg(wet weight) per pCi/kg (dry soil)

t_b = time for buildup of radionuclide in soil (d), taken to be 30 years if the source of the radionuclide is an operating nuclear facility

Values for various plant concentration factors and animal product transfer coefficients for the element considered are given in Table A-1. Plant concentration factors were taken originally from UCRL-50163, pt, IV[6] and supplemented with radionuclide data as explained in HERMES.[2] Coefficients of transfer from feed to animal products for a limited number of radionuclides were available in the literature.[7] For those for which data were lacking, comparisons were made with the behavior of chemically similar elements in man and animals. In some instances, identified with an asterisk in Table A-1, the value used was set to 9.9×10^{-4}.

TABLE A-1. Plant Concentration Factors and Animal Product Transfer Coefficients

Element	Plant/Soil, (Dimensionless)	Egg/Feed (day/kc)	Milk/ Grass (day/k)	Beef/ Feed (day/kc)	Pork/ Feed (day/kc)	Poultry/ Feed (day/kc)
	-----b_{iv}-------	---------------------------S_{ia}---------------------				
Be	4.7E-04	2.0E-02	2.0E-06	8.0E-04	1.0E-02	4.0E-01
N	7.5E+00	9.9E-04*	1.1E-02	9.9E-04*	9.9E-04*	9.9E-04*
F	2.0E-02	9.9E-04*	7.0E-03	2.0E-02	2.0E-02	9.9E-04*
Na	5.0E-02	2.0E-01	4.0E-02	5.0E-02	1.0E-01	1.0E-02
P	5.0E+01	1.0E+1	1.2E-02	5.0E-02	5.4E-01	1.9E-01
Ca	4.0E-02	1.0E+00	8.0E-03	3.3E-03	3.3E-03	3.3E-03
Sc	1.1E-03	9.9E-04*	2.5E-04	6.0E-03	1.0E-02	4.0E-03
Cr	2.5E-04	9.9E-04*	1.1E-03	9.9E-04*	9.9E-04*	9.9E-04*
Mn	3.0E-02	1.0E-01	1.0E-04	5.0E-03	2.0E-02	1.1E-01
Fe	4.0E-04	1.0E-01	6.0E-04	2.0E-02	5.0E-03	1.0E-03
Co	9.4E-03	1.0E-01	5.0E-04	1.0E-03	5.0E-03	1.0E-03
Ni	1.9E-02	1.0E-01	3.4E-03	1.0E-02	1.5E-02	2.0E-03
Cu	1.3E-01	2.0E-01	7.0E-03	1.0E-02	1.5E-02	2.0E=03
Zn	4.0E-01	4.0E-03	6.0E-03	5.0E-02	1.4E-01	2.0E-03
Se	1.3E+00	2.1E+00	2.3E-02	1.0E+00	4.5E+01	3.7E-01
Br	7.6E-01	1.6E+00	2.5E-02	2.0E+02	9.0E-02	4.0E-03
Rb	1.3E-01	3.0E-00	1.0E-02	1.5E-01	2.0E-01	2.0E+00
Sr	2.0E-01	4.0E-01	1.5E-03	3.0E-04	7.3E-03	9.0E-04
Y	2.5E-03	5.0E-04	5.0E-06	5.0E-03	5.0E-03	5.0E-04
Zr	1.7E-04	1.2E-03	2.5E-06	5.03-04	1.0E-03	1.0E.04
Nb	9.4E-03	1.2E-03	1.2E-03	5.0E-04	1.0E-03	1.0E-04
Mo	1.3E-01	4.0E-01	4.0E-03	1.0E-02	2.0E-02	2.0E-02
Tc	2.5E-01	9.9E-04*	1.2E-02	9.9E-04*	9.9E-04*	9.9E-04*
Ru	1.0E-02	4.0E-03	5.0E-02	1.0E-03	5.0E-03	3.0E-04
Rh	1.3E+01	4.0E-03	5.0E-03	1.0E-03	4.0E-03	3.0E-04
Pd	5.0E-00	4.0E-03	5.0E-03	1.0E-03	5.0E-03	3.0E-04
Ag	1.5E-01	9.9E-04*	2.5E-04	9.9E-04*	9.9E-04*	9.90E-04
Cd	3.0E-01	9.9E-04*	6.2E-05	1.6E-02	1.6E-02	1.6E-02

* No data available, assumed to be 9.9E-04

Tritium and Carbon-14 Model

The concentration of tritium or ^{14}C in environmental media (soil, plants and animal products) is assumed to have the same specific activity (pCi of nuclide per kg of soluble element) as the contaminating medium (air). The fractional content of hydrogen or carbon in a plant or animal product is then used to compute the concentration of tritium or ^{14}C in the food product under consideration. Hydrogen content in both the water and the nonwater (dry) portion of the food product is used to calculate the tritium concentration. It is assumed that plants obtain all of their carbon from airborne carbon dioxide and that animals obtain all of their carbon through ingestion of plants.

When ^{14}C is present only in the water used for irrigation, it is difficult to model the transfer of this nuclide to vegetation, because plants acquire most of their carbon from the air. At this time we have not yet determined the transfer of carbon from the water to the air or soil. We have therefore conservatively assumed that plants obtain all their carbon from the irrigation water. Such an assumption could lead to plant concentrations that are high by about an order of magnitude or more. To date, no operating nuclear facilities have been identified that specify releases of ^{14}C in their liquid effluents. Table A-2 lists the parameters used in the computer program for tritium and ^{14}C. These values may be altered based on site-specific data.

TABLE A-2. Calculation of Fractions of Hydrogen and Carbon in Environmental Media, Vegetation, and Animal Products

Food or Fodder	Water	Carbon, Dry	Hydrogen, Dry	Carbon,[a] Wet	Hydrogen,[b] Wet
	f_w	f_c	f_h	F_{cv}, F_{ca}	F_{hv}, F_{ha}
Fresh Fruits, Vegetables and Grass	0.80	0.45	0.062	0.090	0.10
Grain and Stored Animal Feed	0.12	0.45	0.062	0.40	0.068
Eggs	0.75	0.60	0.092	0.15	0.11
Milk	0.88	0.58	0.083	0.070	0.11
Beef	0.60	0.60	0.094	0.24	0.10
Pork	0.50	0.66	0.10	0.33	0.11
Poultry	0.70	0.67	0.087	0.20	0.10

Absolute Humidity: 0.008 $/m^3$

Concentration of carbon in water: 2.0×10^{-5} kg/ [c]

Concentration of carbon in air: 1.6×10^{-4} kg/m^3[d]

(a) F_{cv} or $F_{ca} = f_c\ (1 - f_w)$.
(b) F_{hv} or $F_{ha} = f_w/9 + f_h\ (1\ f_w)$.
(c) Assues a typical bicarbonate concentration of 100 mg/ .
(d) Assumes a typical atmospheric CO_w concentration of 320 ppm_v.

Concentration of Tritium in Vegetation

The concentration of tritium in vegetation is:

$$C_{1v} = (C_{1w})\ (9)\ (F_{hv})^{(a)} \tag{7}$$

where

C_{1w} = concentration of tritium in the environmental water (pCi/ℓ)

= pCi $^3H/m^3$ air: absolute humidity, ℓ/m^3 (for airborne release)

1/9 = fraction of the mass of water which is hydrogen

F_{hv} = fraction of hydrogen in total vegetation (see Table A-2).[8]

The concentration of tritium in the animal product is:

$$C_{1a} = \frac{C_{1F}\ Q_F + C_{1aw}\ Q_{aw}}{F_{hF}\ Q_F + Q_{aw}/9}\ F_{ha} \tag{8}$$

where

C_{1F} = concentration of tritium in feed or forage (pCi/kg) calculated by Equation 7 above, where now $C_{1F} = C_{1v}$

F_{hF} = fraction of hydrogen in animal feed, where now $F_{hf} = F_{hv}$ (grain)

F_{ha} = fraction of hydrogen in animal product

C_{1aw} = concentration tritium in animal drinking water (set to 0 unless there is a release to water).

Similarly, the concentration of ^{14}C in vegetation is:

$$C_{3v} = C_{3w}\ F_{cv}^{\ (b)} \tag{9}$$

(a) The subscript 1 refers to tritium, which is the first nuclide in the isotope listing; similarly the subscript 3 in Equation 6 refers to ^{14}C.

(b) The subscript 1 refers to tritium, which is the first nuclide in the isotope listing; similarly, the subscript 3 in Equation 9 refers to ^{14}C.

where

C_{3w} = concentration of ^{14}C in the environmental media divided by carbon concentration in those media (pCi C/kg carbon)

= pCi $^{14}C/\ell$ divided by carbon concentration in irrigation water (kg/ℓ) for water release

F_{cv} = fraction of carbon in total vegetation.

The concentration of ^{14}C in the animal product is:

$$C_{3a} = \frac{C_{3F}\, Q_F + C_{3aw}\, Q_{aw}}{F_{cF}\, Q_F + F_{cw}\, Q_{aw}}\, F_{ca} \tag{10}$$

For an air release $C_{3aw} = 0$, and since F_{cw} is very small compared to F_{cf}, Equation 10 reduces to:

$$C_{3a} = C_{3F}\, \frac{F_{ca}}{F_{cf}} \tag{11}$$

Dose Calculations for Man

The dose, R_{vr}, in mrem to a person consuming vegetation is:

$$R_{vr} = \sum_{i=1}^{n} C_{iv}\, U_v\, D_{ir} \tag{12}$$

Similarly, the dose from consuming a particular animal product is:

$$R_{ar} = \sum_{i=1}^{n} C_{ia}\, U_a\, D_{ir} \tag{13}$$

where

U_v, U_a = annual consumption of contaminated vegetable or animal products in kg

D_{ir} = a factor that converts intake in pCi of nuclide i to dose in mrem to organ r.

The exposure mode is assumed to be a 1 yr chronic ingestion at a uniform rate. The dose factors employed have been derived from the ingestion and inhalation models given in ICRP Publication 2.[3]

A.2.2 AIR SUBMERSION

The formulas used to calculate doses from air submersion are given below:

$$R_{pr}(x,\Theta,d) = U_p \sum_{i=1}^{n} \overline{X}_i \; D_{ipr} \qquad (14)$$

where

$R_{pr}(x,\Theta,d)$ = the external dose rate from n nuclides via pathway p to organ r of a person located a point x meters from the source in a direction d averaged over a sector width of Θ radians, in mrem/hr

U_p = 8766 hr/yr for air submersion

D_{ipr} = dose factor for nuclide i, in mrem/hr per pCi/m^3 based on a half infinite cloud geomtry and corrected for the fractional penetration of beta and gamma radiations to the appropriate tissue depth (7×10^{-3} cm for skin, 5 cm for total body)

$\overline{X}_i$ = annual average concentration (pCi/m^3) of isotope i at point (x,Θ,d).

Equation 14 yields the yearly external dose to a person located at point (x,Θ,d). The population dose in man-rem/yr is determined by multiplying the dose from Equation 14 by the population located within the sector of the annulus of concern. Values of the dose at point (x,Θ,d) are assumed to be applicable to all individuals located in that sector.[9]

A.2.3 INHALATION

The equation used to calculate air inhalation doses is given by

$$R_{ipr}(x,\Theta,d) = \sum_{i=1}^{n} 3.169 \times 10^{4} D_{ipr} X_i U_p R_D \qquad (15)$$

$R_{ipr}(x,\Theta,d)$ = internal dose rate from n nuclides i via pathway p or organ r of a person located at a point x meters from the source in a direction d, averaged over a sector width of Θ radians, in mrem/hr

3.169×10^{-4} = dimensional conversion constant, in pCi/sec per Ci/yr

D_{ipr} = dose factor for organ r from inhalation of nuclide i, in mrem/yr per pCi/m^2

U_p = occupancy factor in fraction of a year

R_D = cloud depletion factor for iodines.

More information on the models used for calculating radiation doses maybe found in References 10 and 11.

A.3 ASSUMPTIONS USED IN ESTIMATING DOSES FROM NORMAL OPERATION

The assumptions used in estimating doses from routine releases via gaseous pathways follow in Section A.3.1. Assumptions for estimating doses from crops and animal fodder subject to deposition of radioactive materials released are present in Table A.3.

A.3.1 DOSES FROM THE GASEOUS PATHWAY

In estimating doses from the gaseous pathway:

- For external beta dose, 2π geometry was used.
- For external gamma dose, 2π geometry was used.
- The 1980 population distribution was used, and the regional population figure used was 9.167×10^6 persons.
- The X/Q for the regional population is 1.3×10^{-9} sec/m^3.
- The distance to the nearest resident is 800 m from the release point.
- The X/Q at 800 m is 3.1×10^{-8} sec/m^3.

For Spent Fuel Receiving

- Fuel rod defect rate = 2×10^{-4} per unit received
- Each defect releases 100% of the gaseous fission products present in the fuel rod gap and plenum
- The fraction of gaseous fission products present in the gap and plenum are:

  0.03 for ^3H

  0.3 for ^{85}Kr

  0.1 for halogens
- 100% of ^3H and ^{85}Kr released from rod is released to the atmosphere
- 1% of the halogens are released to the atmosphere during waste evaporation

TABLE A-3. Assumptions for Estimating Doses from Crops and Animal Fodder Subject to Deposition of Radioactive Materials

Food Types	Holdup, day	Consumption,[a] kg/yr or /yr	Atmospheric Dilution s/m^3	Yield, kg/m^2	Growing Period, day
Produce					
Leafy Vegetables	1	30	1.3×10^{-9}	1.5	70
Beans, Peas, Asparagus	1	30	1.3×10^{-9}	0.4	70
Potatoes	10	110	1.3×10^{-9}	5	100
Other Root Vegetables	1	72	1.3×10^{-9}	5	70
Berries	1	30	1.3×10^{-9}	2.7	60
Melons (water)	1	40	1.3×10^{-9}	1.4	100
Orchard Fruit	1	265	1.3×10^{-9}	2.1	90
Wheat[b]	10	80	1.3×10^{-9}	0.72	70
Other Grain (sweet corn)	1	8.3	1.3×10^{-9}	1.4	100
Eggs	2	30	1.3×10^{-9}	0.66	130
Milk	2	274	1.3×10^{-9}	1.3	30
Meat					
Beef	15	40	1.3×10^{-9}	2.0	130
Pork	15	40	1.3×10^{-9}	0.69	130
Poultry	2	18	1.3×10^{-9}	0.66	130

(a) Consumptions are for maximum individual. Average population member is assumed to eat one-half of those quantities.

(b) No irrigation of wheat.

- Release fractions for nongaseous fission products and activation products are:

Cs	7×10^{-11}
Other fission products	2×10^{-11}
Activation product	2×10^{-10}

- Age of fuel - 1 year

For Spent Fuel Storage

- Number of defective fuel rods = 5×10^{-4} of the total number of rods
- Rate of release from defective rods to the pool is 1% of that initially in the bulk fuel for

  Halogens
  ^{3}H
  ^{85}Kr

- 100% of the ^{3}H and ^{85}Kr released escapes the pool water
- 1% of the halogens are ultimately released during waste evaporation
- Release fractions for nongaseous fission products and for activation products are:

Cs	9×10^{-12}
Other fission products	2×10^{-13}
Activation product	2×10^{-11}

- Age of fuel - 5 years

A.4 MODELS USED FOR DOSE ESTIMATES TO MAN; POSTULATED FACILITY ACCIDENTS

Models used to estimate doses from postulated accident releases include those listed in Section A.1. Essentially all of the incurred dose from postulated accident releases was via inhalation. Air submersion and inhalation pathways were the only ones evaluated in accident cases.

A.5 ASSUMPTIONS USED IN ESTIMATING DOSES FROM POSTULATED FACILITY ACCIDENTS

The assumptions used to estimate doses from facility accidents are:

- The distance to the nearest resident is 800 m from the release point
- The X/Q at 800 m is: 1.7×10^{-5} sec/m^3 for elevated releases
  1.3×10^{-3} sec/m^3 for ground level releases
- The regional population is 9.167×10^6 persons uniformly distributed within an 80 km radius
- The X/Q for the regional population is:
  1.2×10^{-1} person - sec/m^3 for elevated releases
  1.0 person - sec/m^3 for ground level releases
- The pool decontamination factor for noble gases is 1, and for halogens it is 500. Mixed fission products are contained in the pool water; no release is postulated
- The sand filter efficiency is 99%
- The fraction of ^{85}Kr in the fuel pen gap is 0.3; the fraction of halogens is 0.1; and the fraction of tritium is 0.03

A.6 SOURCE TERMS

Source terms were calculated using the ORIGEN[12] Computer Code assuming:

- 44,000 MWd/Mt
- fuel enrichment of 3.24%
- 15.2 kg ^{235}U per assembly
- 453.4 kg ^{238}U per assembly
- 930 full power days.

Source terms for routine releases are presented in Section 5.1 and presented below in Table A.4. The source terms generated for accident scenarios are presented in Table A.5.

TABLE A-4. Source Terms for Routine Releases to Atmosphere

Nuclide	Ci/yr
^{3}H	2.4
^{85}Kr	198
^{90}Sr	6.0×10^{-5}
^{95}Zr	1.5×10^{-5}
^{95}Nb	3.1×10^{-5}
^{106}Ru	1.7×10^{-4}
^{129}I	3.9×10^{-6}
^{134}Cs	4.7×10^{-3}
^{137}Cs	2.9×10^{-3}
^{144}Ce	2.4×10^{-4}

TABLE A-5. Source Terms[(a)] for Postulated Facility Accidents, Ci

	Tornado-Generated Missile	Fuel Basket Drop	Fuel Assembly Drop
^{3}H	1.3×10^{1}	1.3×10^{1}	3.3
^{85}Kr	7.2×10^{3}	7.2×10^{3}	1.8×10^{3}
^{129}I	1.9×10^{-5}	1.9×10^{-5}	4.7×10^{-6}

(a) Fuel cooling time 1 year

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APPENDIX B

ACRONYMS

APPENDIX B

ACRONYMS

AFR	Away From Reactor
Btu	British thermal unit
BWR	boiling water reactor
BWRTC	Boiling Water Reactor Training Center
$^{\circ}$C	degree Celsius
cfm	cubic feet per minute
Ci	Curie
cm	centimeter
CFR	Code of Federal Regulations
CSAR	Consolidated Safety Analysis Report
DOE	Department of Energy
DOT	Department of Transportation
DNPS	Dresden Nuclear Power Station
$^{\circ}$F	degree Fahrenheit
FCR	full core reserve
FR	Federal Register
ft	feet, foot
GAO	General Accounting Office
GESMO	Generic Environmental Impact Statement on the Use of Mixed Oxide Fuels in Light Water Cooled Reactors
gpd	gallon per day
gpm	gallon per minute
ha	hectare
ISFSI	Independent Spent Fuel Storage Installation
kg	kilogram
km	kilometer
ℓ	liter
LAW	low activity waste
pm	liter per minute
LWR	light water reactor
m	meter
MFRP	Midwest Fuel Recovery Plant
mi	mile
MJ	megajoule
MO	Morris Operation
mrem	millirem
MTHM	metric tonne heavy metal
MTU	metric tonne uranium
MWd	megawatt days
NEPA	National Environmental Policy Act
NFS	Nuclear Fuel Services
NRC	Nuclear Regulatory Commission
PWR	pressurized water reactor
SNM	special nuclear material
TLD	thermoluminscent dosimeter
Ci	microcurie

<u>LISTING OF CHEMICAL ELEMENTS</u>

C	carbon
Ce	cerium
Co	colbalt
CO_2	carbon dioxide
Cs	cesium
^3H	tritium
I	iodine
K	potassium
Kr	krypton
Mn	manganese
Nb	niobium
Ra	radium
Ru	ruthenium
Sr	strontium
UO_2	uranium dioxide
Zr	zirconium