GE Nuclear Energy

NEDO-21326

ORIGINAL WHEN STAMPED IN RED

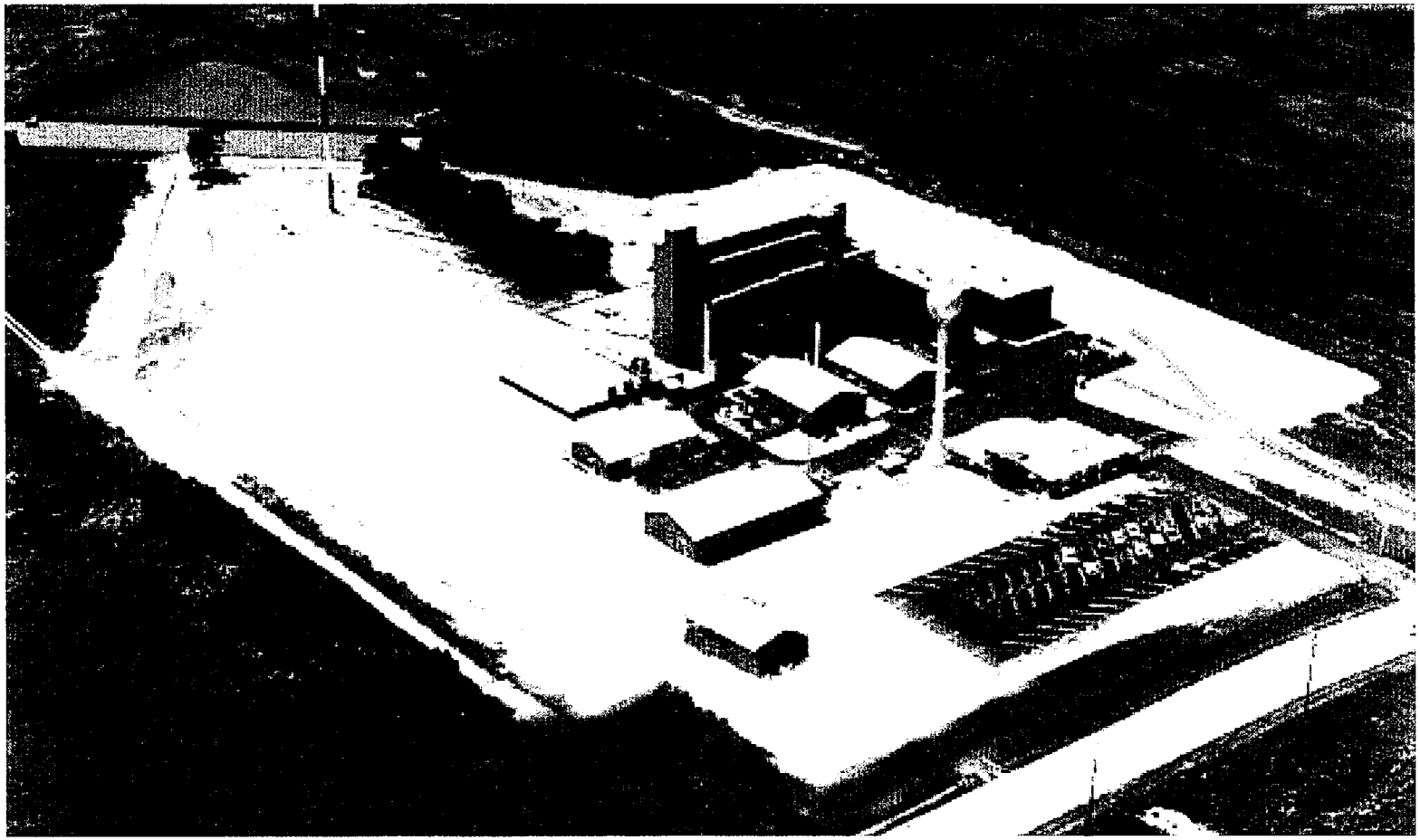
Consolidated

Safety Analysis Report

for
Morris
Operation
Morris, Illinois

GENERAL ELECTRIC'S MORRIS OPERATION (GE-MO)

Located near Morris, Illinois, this facility is licensed to receive, store and transfer irradiated nuclear fuel from Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs).



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Revision Summary

Rev. & Amendment	Date	Summary
NEDO-21326C	1/79	Reissue and update filed with license renewal application. Incorporates all "A" series revisions of original dated 1/72.
NEDO-21326C1	11/79	Incorporates demographic data through the year 2000, radiological monitoring update, and expanded table of contents.
NEDO-21326C2	- - -	Withdrawn.
NEDO-21326C3	1/81	Application amendment for IO.CFR/72.
NEDO-21326D	7/83	Editorial changes and clarifications.
NEDO-21326D1	5/84	Facility Changes Reported in 1984.
NEDO-21326D2	5/85	Organizational Changes
NEDO-21326D3	9/88	Editorial and organization changes and square tube basket revision.
NEDO-21326D4	3/90	Annual updated changes.
NEDO-21326D5	7/94	Organization changes, facility changes, Emergency Plan revision, update
NEDO-21326D6	10/95	Organization changes, facility changes, security plan changes.
NEDO-21326D7	10/96	Organization changes, facility changes, Decommissioning cost update.
NEDO-21326D8	4/98	Organization changes, facility changes.
NEDO-21326D9	5/00	Complete revision

Revision Coding Key: New or changed information is indicated by vertical bars in the right-hand margin opposite the new or changed information.

ATTACHMENT A

GUIDE TO SUBSECTIONS REVISED CONSOLIDATED SAFETY ANALYSIS REPORT FOR MORRIS OPERATION

NOTE: This revision incorporates all sections of the Consolidated Safety Analysis Report (CSAR), including all Appendices. All pages and sections have been changed to Revision D8. This revision is being done to ensure that all sections are in a consistent electronic format. Also, the format changes include different headers and footers, and software (word processor) changes. This includes the use of a different formula editor, so several formulas were re-entered. Also, an effort has been made to achieve a consistent use of abbreviations, acronyms after the first definition, and punctuation. Only changes other than these will be noted below.

Title	Previous Revision #	Information Changed Previous	New	Comment
Corporate Entities, Business, and Experience	D5	1.1.1	1.1.1	Editorial change
Plant Location	D5	1.1.2	1.1.2	added OCA, deleted evaporation pond (now N/A)
Existing Facilities	D5	1.1.3	1.1.3	deleted evaporation pond reference (now N/A), changed protected area to OCA
Site Characteristics	D5	1.2.1	1.2.1	changed protected area to OCA, added reference to crops
Regional and Site Meteorology	D5	1.2.1.1	1.2.1.1	changed cloudiness to cloud cover
Geology	D5	1.2.1.2	1.2.1.2	Editorial change
Hydrology	D6	1.2.1.3	1.2.1.3	Editorial change
Seismology	D6	1.2.1.4	1.2.1.4	Editorial change
Environs Summary	D6	1.2.1.5	1.2.1.5	deleted reference to controlled area
Tract Ownership	D5	1.2.1.6	1.2.1.6	Editorial change
Fuel Storage Areas	D7	1.2.2.1.1	1.2.2.1.1	name reference change
Other Structures	D7	1.2.2.2	1.2.2.2	updated status reference to DCV, Clad Vault, and LAW Vault
Other Structures	D7	1.2.2.2	1.2.2.2	updated title/acronym for building
Fuel Storage Operations	D5	1.3	1.3	deleted reference to photographs
Fuel Storage Operations	D5	1.3	1.3	deleted reference to shipping fuel away
Receiving and Cleaning the Cask	D5	1.3.1	1.3.1	clarified time of survey, and what is inspected
Survey and Inspection	D5	1.3.1.1	1.3.1.1	Editorial change
Survey and Inspection	D5	1.3.1.1	1.3.1.1	added note about cask owner responsibilities
Removal	D5	1.3.1.2	1.3.1.2	Editorial change, added acronym, deleted reference to figure
Flushing	D5	1.3.2.1	1.3.2.1	changed to reflect possible use of other casks
Preparation for Head Removal	D5	1.3.2.2	N/A	deleted, refers to specific IF-300 procedures

Title	Revision #	Previous	New	Comment
Placement in the Cask Unloading Basin	D5	1.3.2.3	1.3.2.2	edited to delete specific IF-300 procedure references
Unloading and Storing Spent Fuel	D5	1.3.3	1.3.3	edited to delete specific IF-300 procedure references, and figures
Preparing the Cask for Shipment	D5	1.3.4	1.3.4	edited to delete specific IF-300 procedure references, and figures
Support Systems	D5	1.4	1.4	removed reference to deleted steam system
Radwaste System	D6	1.4.1	1.4.1	deleted reference to Laboratory (removed)
Ventilation System	D6	1.4.2	1.4.2	editorial change
Basin Water Cleanup and Cooling Systems	D7	1.4.3	1.4.3	noted that room locked, and editorial changes
Cask Sampling, Cool Down, and Flush Systems	D7	1.4.4	1.4.4	edited to delete specific IF-300 procedure references, and figures
Leak Detection and Sump Systems	D7	1.4.5	1.4.5	editorial change
Energy Systems - Electrical	D7	1.4.7.a	1.4.7.a	editorial change
Energy Systems - Natural Gas	D7	1.4.7.b	1.4.7.b	reflect heating system changes
Steam System	D7	1.4.8	N/A	system deleted
Radiological and Other Monitoring	D7	1.5	1.5	updated reference for OCA, editorial changes
Site Description	D5	3.2.2	3.2.2	rephrased to delete obsolete term
Access Control	D5	3.2.2.3	3.2.2.3	editorial change
Boundaries for Establishing Effluent Release Limits	D5	3.2.2.4	3.2.2.4	updated reference for OCA
Population 0 and 5 Miles (Figures 3-4 and 3-5)	D5	3.2.3.1	3.2.3.1	added footnote reference, not previously indicated, but in reference list. Renumbered following footnotes.
Population Within 50 Miles (Figures 3-6 and 3-7)	D5	3.2.3.2	3.2.3.2	added footnote reference, not previously indicated, but in reference list. Renumbered following footnotes.
Transient Population	D5	3.2.3.3	3.2.3.3	Editorial changes
Meteorology	D5	3.4	3.4	Editorial changes
Tornadoes	D5	3.4.1.3	3.4.1.3	Editorial changes
Wind Data	D5	3.4.2.1	3.4.2.1	Editorial changes
Topography	D5	3.4.2.2	3.4.2.2	Editorial changes
Atmospheric Diffusion Characteristics	D5	3.4.4	3.4.4	Editorial changes
Surface Features and Drainage Patterns	D5	3.5.1	3.5.1	Editorial changes
Site Flood Potential	D5	3.5.2	3.5.2	Editorial changes
Site Characteristics	D5	3.6.2	3.6.2	Editorial changes
On Site Well	D5	3.6.2.1	3.6.2.1	Deleted reference to boiler, editorial changes
Regional and Tract Geology	D5	3.7.2	3.7.2	Editorial changes
Investigation of Faults	D5	3.7.3	3.7.3	Editorial changes
Seismic History	D5	3.7.4.2	3.7.4.2	Editorial changes
Meteorology	D5	3.9.1	3.9.1	Editorial changes
References	D5	3.10	3.10	deleted #6, not used
Material to be Stored	D5	4.1.1	4.1.1	Editorial changes

Title	Revision #	Previous	New	Comment
Storage Conditions	D5	4.1.2	4.1.2	Editorial changes
Structural and Mechanical Safety Criteria	D5	4.2	4.2	Editorial changes
Compliance	D5	4.2.1.2	4.2.1.2	Editorial changes
Compliance	D5	4.2.4.2	4.2.4.2	Editorial changes
Design Response Spectra Derivation	D5	4.2.4.2.1.b	4.2.4.2.1.b	Editorial changes
Rocking and Translational Response Summary	D5	4.2.4.2.2.b	4.2.4.2.2.b	Editorial changes
Seismic Analysis Methods	D5	4.2.4.2.4.a	4.2.4.2.4.a	Editorial changes
Compliance	D5	4.2.5.2	4.2.5.2	Editorial changes
Other Definitions	D5	4.2.5.2.1.e	4.2.5.2.1.e	Editorial changes
Load Combinations and Acceptance Criteria for Steel Structures	D5	4.2.5.2.3.c	4.2.5.2.3.c	Editorial changes
General	D5	4.3.1	4.3.1	Editorial changes
Criteria	D5	4.3.2.1	4.3.2.1	Changed reference from Control Room to CAS/SAS
Criteria	D5	4.3.3.1	4.3.3.1	Editorial changes
Generation	D6	4.3.3.2.a	4.3.3.2.a	Editorial comment
Criteria	D6	4.3.4.1.b	4.3.4.1.b	Editorial changes
Compliance	D6	4.3.4.2	4.3.4.2	Editorial changes, and changed reference from Control Room to CAS/SAS
Criteria	D6	4.3.5.1	4.3.5.1	Editorial changes
Compliance	D6	4.3.5.2.a	4.3.5.2.a	Editorial changes
Compliance	D6	4.3.5.2.b	4.3.5.2.b	Editorial changes
Compliance	D5	4.3.5.2.e	4.3.5.2.e	Editorial changes
Criteria	D5	4.3.6.1	4.3.6.1	updated reference for OCA
Compliance	D5	4.3.6.2	4.3.6.2	Editorial changes
Access Control (Controlled Areas)	D5	4.3.6.2.1	4.3.6.2.1	updated reference for Controlled Area vs. Restricted Area
Access Control (Controlled Areas)	D6	4.3.6.2.1	4.3.6.2.1	incorporated acronym for ALARA
General Electric Tract	D6	4.3.6.2.1.a	4.3.6.2.1.a	Editorial changes
OCA	D6	4.3.6.2.1.b	4.3.6.2.1.b	updated reference for OCA, and Editorial changes
Radiologically Controlled Area (RCA)	D6	4.3.6.2.1.c	4.3.6.2.1.c	updated reference for RCA vs. Operating Area
Radiologically Controlled Area (RCA)	D7	4.3.6.2.1.c	4.3.6.2.1.c	added statement allowing additional requirements, and reference to High Rad Area Controls
Radiologically Controlled Area (RCA)	D7	4.3.6.2.1.d	N/A	deleted, inserted statement above
Shielding	D7	4.3.6.2.2	4.3.6.2.2	substituted acronym ALARA for term
Radiation Alarm Systems	D7	4.3.6.2.3	4.3.6.2.3	editorial changes, added acronym
Radiation Alarm Systems	D7	4.3.6.2.3.b	4.3.6.2.3.b	editorial changes

Title	Revision #	Previous	New	Comment
Radiation Alarm Systems	D5	4.3.6.2.3.c.3	4.3.6.2.3 .c.3	combined with 4.3.6.2.3.c.5
Radiation Alarm Systems	D5	4.3.6.2.3.c.5	N/A	combined with 4.3.6.2.3.c.5
Radiation Alarm Systems	D5	4.3.6.2.3.c.6	4.3.6.2.3 .c.5	renumbered after deleting 4.3.6.2.3.c.5, and editorial changes
Radiation Alarm Systems	D5	4.3.6.2.3.c.7	4.3.6.2.3 .c.6	renumbered after deleting 4.3.6.2.3.c.5, editorial changes, and changed reference from control room to SAS
Compliance	D6	4.3.7.2	4.3.7.2	changed reference from control room to CAS/SAS
Codes, Guides, & Standards	D5	Table 4-3	Table 4-3	Deleted reference to API-650, not found in text
Introduction	D6, D7	5.1	5.1	Editorial changes, removed reference to shipment of fuel, updated reference to OCA, changed reference from exclusion areas to site boundary.
Restricted and Owner-Controlled Area	D7	5.2.1	5.2.1	changed reference from protected area (PPA) to OCA
Gates	D7	5.2.2	5.2.2	changed reference from protected area (PPA) to OCA
Principal Structure	D7	5.3	5.3	editorial changes, removed reference to fuel shipping
Fuel Storage Facility Layout	D5	5.3.2.f	N/A	removed reference to laboratory, renumbered following
Confinement Features	D5, D7	5.3.2.2	5.3.2.2	editorial changes, added LAW Vault status update, removed dimensional information about empty LAW Vault
Cask Handling and Fuel Storage Systems	D7	5.4	5.4	Inserted note on generic cask statements, specifics to be furnished by cask owner
Cask Receipt and Inspection Facilities	D5	5.4.1.1	5.4.1.1	editorial changes, removed reference to steam lines
Cask Handling Crane, and Handling Equipment	D5	5.4.1.2	5.4.1.2	revised to reflect only generic equipment, specifics to be furnished by cask owner, editorial changes, footnote 3 removed, rest renumbered
Damaged Cask Handling	D5	5.4.1.3	5.4.1.3	editorial changes
Cask Venting and Flushing Operations	D5	5.4.2.2	5.4.2.2	revised to reflect only generic equipment, specifics to be furnished by cask owner., editorial changes, cask-flush in-line monitors removed
Low-Level Solid Waste	D5	5.4.2.3	5.4.2.3	editorial changes, removed of figure reference
Description	D5	5.4.3.1	5.4.3.1	editorial changes
Doorway Guard	D5	5.4.3.3	5.4.3.3	editorial changes
Fuel Integrity in Storage	D5	5.4.4.1	5.4.4.1	editorial changes
Equipment Description	D5	5.4.4.2	5.4.4.2	editorial changes
Heat Transfer from Stored Fuel	D5	5.4.4.3	5.4.4.3	editorial changes
Storage Basin Description	D5	5.5.1	5.5.1	replaced control room with CAS/SAS, editorial changes
Concrete Structure	D5	5.5.1.2	5.5.1.2	editorial changes

Title	Revision #	Previous	New	Comment
Basin Liner	D5	5.5.1.3	5.5.1.3	editorial changes
Basin Liner Leakage Control	D5	5.5.1.4	5.5.1.4	editorial changes
Earthquake and Tornado Analyses	D5	5.5.1.5	5.5.1.5	added reference to SAS
Basin Water Clean-Up System	D5	5.5.2	5.5.2	editorial changes, added acronym ALARA vs. phrase
Radioactive Materials in Basin Water	D5	5.5.2.2	5.5.2.2	updated basin water activities, specified analysis whose activity has limits requiring action
Basin Water Filter System	D5	5.5.2.3	5.5.2.3	editorial changes
Typical Isotope Concentrations in Basin Water	D7	Table 5-1	Table 5-1	updated basin water activities, editorial changes
Equipment Description	D7	5.5.3.1	5.5.3.1	editorial changes
Air Tunnels	D5	5.5.4.1	5.5.4.1	editorial changes
Emergency Equipment Building (EEB)	D5	5.5.4.3	5.5.4.3	altered title of building
Compressor Room and Compressed Air System	D5	5.5.4.3.b	5.5.4.3.b	air compressors replaced, updated specifications, cooling towers and evaporative cooling system removed
Effluent Air Release	D5	5.5.4.4	5.5.4.4	editorial changes
Main (Process) Building Facilities	D5	5.5.5	5.5.5	editorial changes
Building Entrance Area	D5	5.5.5.1	5.5.5.1	deleted reference to monitoring room, editorial changes
Gallery Area	D5	5.5.5.2	5.5.5.2	edited door lock type, not all electrically controlled
Office Area	D6	5.5.5.3	5.5.5.3	added new term for control room, added alternate access route
Control Room, or Secondary Alarm Station (SAS)	D6	5.5.5.4	5.5.5.4	added alternate term for control room (SAS), editorial changes
Laboratory Area	D6, D5	5.5.5.5	N/A	deleted, removed reference to laboratory, renumbered rest
Central Alarm Station Monitoring of Fuel Storage Functions	D6	Table 5-2	Table 5-2	added new acronyms for alarm stations, removed reference to steam system
Off-Gas Cell	D5	5.5.5.6	5.5.5.5	editorial changes
Radwaste System Evaporator	D5	5.5.5.7	5.5.5.6	editorial changes
Ventilation Supply Room	D5	5.5.5.8	5.5.5.7	deleted references to the emergency power room, was removed
Basin Pump Room Addition	D5	5.5.5.9	5.5.5.8	editorial changes
Basin Pump Room Addition Building	D5	5.5.5.9.a	5.5.5.8.a	editorial changes
Basin Pump Room Addition Building	D5	5.5.5.9.b	5.5.5.8.b	editorial changes
Electro-Decontamination Room	D5	5.5.5.10	5.5.5.9	removed reference to IF-300 casks and parts, editorial changes
Waste Vaults	D7	5.6	5.6	editorial changes
Low Activity Waste (LAW) Vault	D7	5.6.a	5.6.a	edited to include statement of current status, tank OOS

Title	Revision #	Previous	New	Comment
Low Activity Waste (LAW) Vault	D7	5.6.a.4	N/A	deleted reference to laboratory
Cladding Vault	D7	5.6.b	5.6.b	edited to include statement of current status
Dry Chemical Vault	D7	5.6.c	5.6.c	edited to include statement of current status, tank OOS
Low Activity Waste Vault (LAW Vault)	D7	5.6.1	5.6.1	editorial changes
LAW Vault Construction	D7, D5	5.6.1.1	5.6.1.1	deleted text referring to construction details, editorial changes
Elevations	D5	5.6.1.2.a	5.6.1.2.a	deleted sections of construction details, tank OOS
Inner Tank	D5	5.6.1.2.b	N/A	deleted construction details, tank OOS
Construction	D5	5.6.2.2.b	5.6.2.2.b	editorial changes
Utility and Service Building	D5	5.7.1	5.7.1	removed reference to utility steam system, and edited to reflect revised layout and use.
Utility Section	D7	5.7.1.1	5.7.1.1	removed reference to utility steam system, and edited to reflect revised layout and use.
Service Section	D7	5.7.1.2	5.7.1.2	editorial changes
Outside Facilities	D7	5.7.1.3.a	5.7.1.3.a	editorial changes
Outside Facilities	D7	5.7.1.3.b	N/A	stack removed, boiler removed
Fire Protection	D5	5.7.2.3	5.7.2.3	altered control room reference to CAS/SAS
General Warehouse	D5	5.7.3	5.7.3	editorial changes
Administration Building	D5	5.7.4	5.7.4	editorial changes
Cask Service Facility	D5	5.7.5	5.7.5	deleted reference to LAW process equipment, edited rail spur end description.
North Warehouse	N/A	N/A	5.7.6	added reference to North Warehouse
Water Supply	D5	5.8.1	5.8.1	altered PPA reference to OCA, editorial changes, deleted references to utility steam, added reference to chlorination system installed.
Demineralized Water Supply	D5, D7	5.8.1.2	5.8.1.2	deleted reference to utility boiler, and laboratory, added CAS/SAS reference
Fire Fighting Water Supply	D7	5.8.1.3	5.8.1.3	editorial changes
Electrical Supply	D5	5.8.2	5.8.2	deleted reference to steam, editorial changes
Normal Electrical Power Source	D5	5.8.2.1	5.8.2.1	editorial changes
Operating Characteristics	D5	5.8.2.4	5.8.2.4	added CAS/SAS reference vs. control room
Site Natural Gas Supply	D5, D7	5.8.3	5.8.3	deleted utility boiler and laboratory references
Sanitary and Process Sewer System	D7	5.8.4.1	5.8.4.1	replaced PPA references with OCA, deleted reference to boiler, editorial changes
Rail Transportation Facilities	D5	5.8.5	5.8.5	replaced PPA references with OCA, deleted reference to IF-300, changed description of rail to CSF
Highway Access	D5	5.8.6	5.8.6	replaced PPA references with OCA

Title	Revision #	Previous	New	Comment
Grading	D5	5.8.7	5.8.7	replaced restricted area references with OCA, editorial changes
References	D5	5.10.3	N/A	deleted reference to IF-300 cask, renumbered rest of references.
Waste Management	D6	6.0	6.0	updated status of waste vaults, added method of waste management, editorial changes
Dry Chemical Vault	D6	6.1.1	6.1.1	updated status of waste vaults, editorial changes
Low Activity Waste (LAW) Vault	D6, D7	6.1.2	6.1.2	updated status of waste vaults, editorial changes
Cladding Vault	D7	6.1.3	6.1.3	updated status of waste vaults, editorial changes
Radwaste System	D7	6.2	6.2	Solidification not performed here
Non-Radioactive Waste	D7	6.4	6.4	Deleted references to utility boiler, and stack
Irradiated Fuel	D6	7.3.1	7.3.1	editorial changes
Storage Basin Water	D6, D5	7.3.2	7.3.2	incorporated acronym, editorial changes
Contaminants	D5	7.3.2.2	7.3.2.2	added current Cs-137 values, editorial comments
Airborne Radioactive Material Sources	D7	7.3.3	7.3.3	Footnote replaced
Facility Design Features	D5	7.4.1	7.4.1	editorial changes
Estimated Direct Radiation from Fuel in Storage	D5	7.4.2.1	7.4.2.1	updated data
Primary Safety Considerations	D5	7.4.3.1.c	7.4.3.1.c	incorporated acronyms
Principal Mechanisms for Ensuring Safety	D5	7.4.3.2	7.4.3.2	Editorial changes
Principal Mechanisms for Ensuring Safety	D5	7.4.3.2.a	7.4.3.2.a	editorial changes
Principal Mechanisms for Ensuring Safety	D5	7.4.3.2.b	7.4.3.2.b	editorial changes
Principal Mechanisms for Ensuring Safety	D5	7.4.3.2.c	7.4.3.2.c	incorporated acronym
Sampling Considerations	D5	7.4.4.3	7.4.4.3	editorial changes
Sampling Considerations	D5	7.4.4.3.a	7.4.4.3.a	editorial changes
Sampling Considerations	D7	7.4.4.3.b	7.4.4.3.b	editorial changes
Area Radiation Monitors (ARMs)	D7	7.4.5.2.a	7.4.5.2.a	editorial changes, added CAS/SAS reference, deleted reference to internal source (removed), incorporated acronym
Criticality Monitors	D7	7.4.5.2.c	7.4.5.2.c	editorial changes
Pipe Monitor	D7	7.4.5.2.d	N/A	deleted, as no longer installed. renumbered rest
Radiation Monitor Considerations	D7	7.4.5.3	7.4.5.3	incorporated acronym, deleted reference to internal source (removed)
Radiation Monitor Considerations	D7	7.4.5.3.a	7.4.5.3.a	editorial changes
Radiation Monitor Considerations	D7	7.4.5.3.a.1	7.4.5.3.a .1	editorial changes

Title	Revision #	Previous	New	Comment
Radiation Monitor Considerations	D7	7.4.5.3.b.1	7.4.5.3.b.1	editorial changes, ARMs made with local alarms
Radiation Monitor Considerations	D7	7.4.5.3.b.4	7.4.5.3.b.4	incorporated acronym, also added alarm location
Radiation Monitor Considerations	D7	7.4.5.3.c.2	7.4.5.3.c.2	deleted, internal sources removed, remainder of steps renumbered
Radiation Monitor Considerations	D5	7.4.5.3	7.4.5.3	incorporated acronym
Personnel Exposure Assessment	D5	7.5	7.5	editorial changes, added name of analysis with given specs
Health Physics Program	D5	7.6	7.6	incorporated acronym, editorial changes
Effluent and Environmental Monitoring Program	D5	7.7.1	7.7.1	removed reference to CECo. No longer performed, deleted footnote, renumbered
MO Radiological Monitoring Program	D5	Table 7-3	Table 7-3	editorial changes
Estimated Exposures	D5	7.7.2	7.7.2	updated, now use different software from NRC
Introduction	D5	8.1	8.1	editorial changes, deleted statement on shipping fuel
Direct Radiation	D5	8.1.1.1	8.1.1.1	editorial changes
Cask Drop into the Cask Unloading Basin	D7	8.4	8.4	editorial changes
Fission Gas Inventory in the Fuel	D5	8.5.a	8.5.a	footnote #2 not found in text, renumbered remainder of footnotes
Water Decontamination Factor	D5	8.5.b	8.5.b	editorial changes
Fuel Bundle Drop Accident	D7	8.5.1.a	8.5.1.a	editorial changes
Assumptions	D7	8.6.2.f	8.6.2.f	editorial changes
Dose Rate Calculations	D7	8.6.3	8.6.3	editorial changes
Cooling System Leak	D7	8.7	8.7	Fin-Fan coolers replaced with freon to water chillers
Reactivity Calculations	D7	8.8.2	8.8.2	editorial changes
Missile Impact	D7	8.8.3	8.8.3	editorial changes
Consequences of a Criticality Accident	D7	8.8.4	8.8.4	editorial changes
Assumptions	D5	8.8.4.1	8.8.4.1	reformatted formula
References	D7	8.9.2	N/A	footnote #2 not found in text, deleted and renumbered remaining
Introduction	D7	9.1	9.1	editorial changes
Organization Functions, Responsibilities, and Authorities	D7	9.2.1	9.2.1	editorial changes
Morris Operation	D5	9.2.2.1	9.2.2.1	editorial changes
Morris Operation Organization	D5	9.2.3	9.2.3	editorial changes
Manager - Morris Operation	D5	9.2.3.1	9.2.3.1	editorial changes
Morris Operation Organization chart	D7	Figure 9-2	Figure 9-2	updated to reflect staffing and responsibility changes
Operations, Engineering, and Projects Manager	D5	9.2.3.2	9.2.3.2	editorial changes

Title	Revision #	Previous	New	Comment
Regulatory Compliance Manager	D5	9.2.3.3	9.2.3.3	editorial changes
Safety Committee	D5	9.2.4	9.2.4	editorial changes, staff reductions have resulted in incorporation of functions in fewer personnel, reworded to match language of license
Operator Qualification, Training, and Certification	D5	9.3.1	9.3.1	deleted reference to loading casks (shipping fuel)
Environmental Health and Safety Plan	D5	9.4.1.3	9.4.1.3	deleted references to specific personnel by title for responsibilities (not required in this context in the CSAR)
Special Work Permits (SWPs)	D5	9.4.1.4	9.4.1.4	editorial changes, changed department name to HP/Safety
Regulated Work Permits (RWPs)	D5	9.4.1.5	9.4.1.5	editorial changes
Equipment Maintenance Programs	D5	9.4.1.6	9.4.1.6	editorial changes
Records and Reports	D5	9.4.2	9.4.2	editorial changes
Facility Modification	D5	9.4.3	9.4.3	editorial changes
Licensing Activity	D5	9.4.3.2	9.4.3.2	editorial changes
Project Implementation	D5	9.4.3.3	9.4.3.3	editorial changes
Responsibilities	D5	9.5.2	9.5.2	editorial changes
Contamination Accidents	D5	9.5.3.1.b	9.5.3.1.b	changed control room to CAS/SAS
History of Operations	D7	A.7.1.2	A.7.1.2	included current status of waste vaults
Performance Objectives	D5	A.7.2.2	A.7.2.2	editorial changes
Radiation Survey	D7	A.7.3.1	A.7.3.1	editorial changes
Waste Vault Contents	D7	A.7.3.3.1	A.7.3.3.1	updated Vault status
Contaminated Equipment	D5	A.7.3.3.2	A.7.3.3.2	editorial changes
Residual Contamination Survey/Assessment	D5	A.7.3.4	A.7.3.4	editorial changes
Final Waste Removal	D5	A.7.3.5.6	A.7.3.5.6	editorial changes
Balancing of Effects	D7	A.7.4.1	A.7.4.1	editorial changes
Cask Lifting Yoke Analysis	D7	A.8	A.8	Replaced with Aging Management
Introduction	D5	A.9.1	A.9.1	editorial corrections
Discussion of Calculations	D5	A.9.2	A.9.2	editorial corrections
Fuel Storage Basin Water Heat Transfer with Loss of Circulation	D5	A.9.3	A.9.3	editorial changes
Calculational Method	D5	A.9.3.1	A.9.3.1	editorial changes
Concrete Thermal Conductivity	D5	A.9.5	A.9.5	editorial changes
Fuel Basket System - Nuclear Design Analysis	D5	A.10.2	A.10.2	editorial changes
Criticality Prevention	D5	A.11.4.b	A.11.4.b	editorial changes
Fuel Basket System Design Analyses	D5	A.12	A.12	editorial changes
Cask Drop on the Shelf	D5	A.13.2	A.13.2	editorial changes, added note assuming review if other casks used

Title	Revision #	Previous	New	Comment
Floor Loading Analyses	D5	A.13.3.3	A.13.3.3	editorial changes
Potential Missiles	D5	A.15.2	A.15.2	editorial changes
Tornado Wind Velocities	D5	A.15.3	A.15.3	editorial changes
Buoyancy	D5	A.15.6.1.3	A.15.6.1 .3	editorial Changes
Impact Analysis	D5	A.15.6.1	A.15.6.1	editorial changes
Introduction	D5	B.22.1	B.22.1	editorial changes
H-3	D5	B.22.2	B.22.2	deleted reference to LAW Vault, also in equation #6
Accident Conditions (Ground Level Release)	D5	B.22.4	B.22.4	changed I-29 to I-129
High Activity System	D5	B.23.2	B.23.2	editorial changes, added reference to activity as a basis for deciding discharge path, refined description, revised dose rate on HIC based on last HIC moved. Added reference to CAS/SAS vs. control room.
Low Activity System	D5	B.23.3	B.23.3	deleted references to lab, editorial changes, revised volumes based on deletion of lab as source. Added reference to CAS/SAS vs. control room, revised dose rates per actual survey results.

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1.0 INTRODUCTION AND DESCRIPTION

1.1 INTRODUCTION

This document contains a consolidation of safety analysis information relating to receipt, storage and transfer of irradiated nuclear fuel in operations conducted by General Electric Company (GE or the Company) at Morris Operation (MO). Fuel receipt and shipping and cask handling are discussed here, but since 1989, the fuel basins at GE-MO are essentially full, and no further receipts of fuel are anticipated. Fuel shipments are not expected until the DOE repository is opened.

Almost all information in this document has been previously published or otherwise made a part of the public record regarding the Midwest Fuel Recovery Plant (MFRP) or GE-MO[1]. This document presents information regarding fuel storage operations, disregarding features of the facility not applicable to fuel storage. Not all information in this document describes important to safety structures, systems and components (SSC). Support SSC are also discussed as they apply to fuel storage. Section 8, "Accident Safety Analysis", and Section 11, "Quality Assurance", detail SSC important to safety.

The Company's facility is located near Morris, Illinois, adjacent to the Dresden Nuclear Power Station (DNPS). GE-MO is licensed for receipt, storage and transfer of nuclear fuel from boiling water reactors (BWRs) and pressurized water reactors (PWRs).

The GE-MO fuel storage facility includes two interconnected water-filled basins with cranes, water treatment system, and other facilities required to receive irradiated fuel and store it underwater for an indefinite period. Fuel storage equipment in the basins is designed to protect the integrity of fuel rods during seismic or meteorological events. Special procedures and isolation can be provided for storage of damaged or leaking fuel. Security measures are in effect to protect the facility against unauthorized access. Based on the service life of nonreplaceable components (concrete basin and basin liner), the normal service life of the facility would be more than 100 years, although it is intended for interim storage only.

In December 1975, GE received a license amendment to increase fuel storage capacity[2] from about 100 TeU to 750 TeU by installation of a fuel storage system of a new design and through appropriate changes in fuel handling and support systems. This modification, designed by GE as Morris Operation-Project I, converted the former high level waste storage basin to a fuel storage basin. The capacity expansion project was completed in 1976.

1.1.1 Corporate Entities, Business, and Experience

Facilities described in this report are owned and operated by General Electric Company, a corporation under the laws of the State of Connecticut, with its principal place of business at Fairfield, CT. The facility is operated through the Company's GE Nuclear Energy Division with headquarters in San Jose, California and operations in Morris, Illinois.

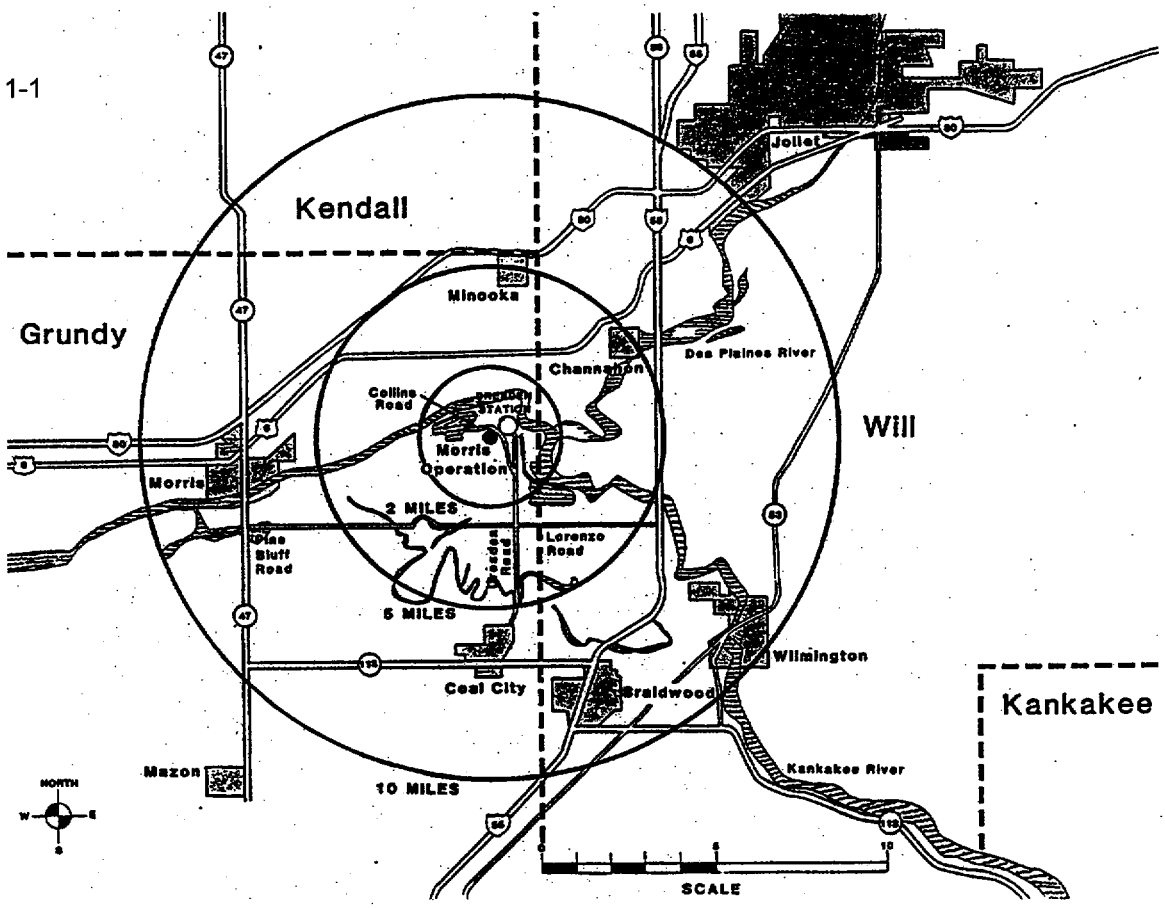
GE is a broadly diversified corporation involved in research, design, manufacturing, and marketing products and services in several fields including industrial products, technical systems and materials, consumer products, and power systems. The latter activity includes nuclear systems, equipment, fuel and services.

The Company's experience in nuclear activities includes research and development of prototype reactors for nuclear submarines, operation of the government's Hanford facilities for more than 17 years and development, design, manufacture, and erection of boiling water reactors currently operating at electric power stations in the United States and throughout the world. The staff of GE Nuclear Energy (GENE) includes hundreds of scientists, engineers, and technicians, representing one of the largest pools of nuclear knowledge and experience in the world.

1.1.2 Plant Location

GE-MO facilities are located on the northern end of a rectangular tract of about 886 acres owned by the Company in Gooselake Township, Grundy County, Illinois, near the confluence of the Kankakee and Des Plaines Rivers (Figure 1-1).

Figure 1-1



The tract (Figure 1-2) is about 15 air miles southwest of Joliet and about 50 miles southwest of the Chicago, Illinois - Gary, Indiana area. Morris, Illinois, the county seat of Grundy County is about 7 miles west of the tract. The Illinois Waterway and Kankakee River are separated from the tract to the north and east by lands owned by the Commonwealth Edison Company (CECo), the site of the Dresden Nuclear Power Station (DNPS) and related facilities, and a privately owned plot of about 50 acres. Gooselake Prairie State Park is to the west and a discontinued refractory mining operation borders the tract to the south. The GE-MO site consists of the developed area of the Company's tract, including the Owner-Controlled Area (OCA) and the protected area, and sanitary lagoons.

Figure 1-2

GE-MO Tract and Vacinity

The tract is shown in relation to Dresden Nuclear Power Station and other surrounding lands.

This Figure Withheld under 10 CFR 2.390

1.1.3 Existing Facilities

The existing facilities occupy about 52 acres at the north edge of the tract (Figure 1-3). The principal plant structures, including the ventilation stack, are within a 15 acre fenced area, while the sanitary waste treatment facilities are located immediately south of the OCA. The sanitary waste facilities are also fenced.

Figure 1-3
Principal Facilities – Site Area
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1.1.4 Fuel Type and Exposure

The design basis fuel to be stored is UO_2 fuel having had an initial enrichment of 5% U-235 or less, with stainless steel, zirconium or Zircaloy cladding, and in a "bundle of rods" geometry. Design basis fuel was assumed to be irradiated at specific power levels of up to 40 kW/kgU, with exposure to 44,000 MWd/TeU (reactor discharge batch average), and cooled for at least 1 year after reactor shutdown prior to receipt at GE-MO.

1.1.4.1 Fuel in Storage

Irradiated fuel from PWRs and BWRs has been received and stored at GE-MO since 1972. These activities have reaffirmed experience elsewhere that fuel can be handled and stored safely with no impact on the environment. There has been no significant fuel leakage (as determined by measurement of basin water activity), indicating the fuel is a stable, inert material

while in the storage basin environment. Effective control of water quality, radioactive material concentration in the water, cask contamination, and airborne radioactive material has been demonstrated.

1.2 GENERAL PLANT DESCRIPTION

The following descriptions are of those aspects of GE-MO facilities related to irradiated fuel storage or shipment.

1.2.1 Site Characteristics

The GE-MO site is in a developing industrial area. The terrain is typically "rolling prairie," with vestiges of long-abandoned coal strip mines. In general, the land in the area has been farmed for many years, but the GE-MO site is in an area of rocky outcroppings and thin top soil, unsuited to economical, large-scale farming of crops. Arable portions of the site outside of the OCA have been leased to local farmers and have been used for beef cattle grazing and raising crops. Both road and rail transportation services are available on the site (Figure 1-1). Rail access is via an extension of the DNPS siding from the Elgin, Joliet and Eastern Railway right-of-way to the west of the site. Road access is via county roads which connect with several state highways and provide routes to nearby communities and to interstate highways in the area. Water transportation access via the Illinois River is available through an agreement with CECo, but no docking facility is developed.

Investigations of site characteristics were made in support of the MFRP construction effort, and Morris Operation-Project I. These studies supplemented extensive information obtained in the course of DNPS development and operation. Factors significant to fuel storage activities are summarized below.

1.2.1.1 Regional and Site Meteorology

The climate of the Morris region of Illinois is typically continental, with cold winters and warm, humid summers. There are frequent short-term fluctuations in temperature, humidity, cloud cover, and wind speed and direction. Storm systems and weather fronts usually move eastward and northeastward through this area. The maximum recorded temperature for the area was 109 °F, with a minimum temperature of -22 °F, and an annual mean temperature of about 59 °F. There is a rather uniform distribution of wind direction, with the most frequent winds from the west and south at an average of 11 to 15 mph.

The most severe weather conditions experienced in the area are tornadoes. Over a 40 year period, there was an average of 4.8 tornadoes per year in Illinois, which is close to the average for all states east of the Rocky Mountains. While tornadoes have been reported near GE-MO since 1965, no damage to the site has occurred.

1.2.1.2 Geology

Exploration of the site's substructure, as well as actual excavation for facility construction confirmed the rock is sound at all depths with no evidence of active faults. All main building foundations and below-grade vault and basin structures are set in bedrock to ensure high structural integrity for these facilities.

1.2.1.3 Hydrology

Consideration has been given to subsurface water behavior in relation to operation of underground facilities, but because there is no liquid waste discharge, or storage of high activity liquid wastes at the fuel storage site, factors such as drainage patterns to water courses, soil ion-exchange capacity, etc., are not of major significance in ensuring the safety of fuel storage operation[3].

Potential flooding of the site is considered very unlikely. Site elevation at the plant location is 532.5 ft. compared with the maximum historical flood elevation of 506.4 ft. The normal pool elevation of the river as controlled by the Dresden Dam is 505 ft.

1.2.1.4 Seismology

Available references show the GE-MO site in Zone 1 (zone of minor damage) on the latest seismic probability map. In Richter's Seismic Regionalization map, the site is near the line of demarcation between an area assigned a probable maximum intensity of seven and one with a probable maximum intensity of eight of the Modified Mercalli (MM) scale. To ensure conformance with basin earthquake resistance criteria, design earthquake forces have been taken as those corresponding to a horizontal ground acceleration of 0.1G (MM7) and maximum earthquake forces at a horizontal ground acceleration of 0.2G (MM8).

1.2.1.5 Environs Summary

Distances from the plant stack to GE property boundaries are 2,265 ft. to the east, 6,512 ft. to the south and 3,100 ft. to the west. The tract boundary to the north is about 950 ft. from the stack; however, the DNPS site provides an effective boundary of about 5,950 ft. Studies of population and land usage in surrounding areas were made and reported in the course of DNPS development, during MFRP licensing, and during the GE-MO capacity expansion. Factors of specific interest are summarized below and discussed further in Section 3.

a. Industrial: On the DNPS site there are two operating nuclear power reactors situated about 0.7 miles northeast of the GE-MO stack. A large fossil-fired power plant is located about 4 miles west-southwest of the stack. A chemical plant is located about 1.5 miles from the stack to the northwest. Adjacent to the south boundary of the GE-MO tract there are discontinued clay mines 1.4 miles from the stack.

b. Residential: Residences nearest to the tract are on about 50 acres directly east of the facilities (about 0.5 mile from the stack) between GE's property and the Kankakee River. There are approximately 30 river front sites on which cottages have been built, largely for recreational purposes. There are other residences across the Kankakee river, the nearest about 0.7 mile from the stack.

   Total population within a 5 mile radius is estimated to be about 7,000 including summer visitors, increasing to about 9,000 by the year 2000. A population of about 49,000 reside within a 10 mile radius of the plant, and should increase to about 68,000 by the year 2000.

   Population in the 5 to 20 mile radius zone, which includes the cities of Aurora and Joliet, is about 350,000. This population should increase to about 450,000 by the year 2000. In general, population projections for the State of Illinois have been lowered in recent years. Current projections indicate a relatively slow growth rate as compared to the overall U.S. rate.

c. Recreational: In addition to fishing, hunting, and boating activities near the confluence of the Kankakee and Des Plaines Rivers 1 to 2 miles east of the plant, the Goose Lake Prairie State Park has been established adjacent to the GE-MO tract. This natural prairie preserve of about 1,800 acres is west of the tract, with the nearest point being about 0.6 mile from the stack.

1.2.1.6 Tract Ownership

The tract is wholly owned by GE. Since purchase of the original tract, which then totaled 1,380 acres, approximately 70 acres located at the southwest corner and approximately 50 acres in a 400 ft. wide strip along the south edge of the tract was sold to the A. P. Green Refractory Company, Illinois Products Division, which was used in connection with clay mining and clay products manufacturing activities. Clay mining and manufacturing was discontinued and the land sold to a private party. A parcel to the north and east was sold to the CECo for construction of canals to a cooling lake for DNPS reactors. Currently, GE property totals about 886 acres.

1.2.2 Facility Descriptions

Site facilities as they exist today are the result of using original buildings, where possible, and rearranging or adding new buildings, where necessary.

1.2.2.1 Main Building

The main building (also known as the process building) is a massive structure of reinforced concrete, about 204 ft. by 78 ft. in plan, and about 88 ft. above grade. The western end of the building houses most of the fuel storage facilities. This portion of the building is of steel frame and insulated metal siding construction, and is attached to the concrete main building.

1.2.2.1.1 Fuel Storage Areas

Fuel storage operation areas include (Figure 1-3):

a. Cask receiving area (3)
b. Decontamination area (6)
c. Cask unloading basin (7)
d. Fuel storage basins 1 and 2 (8)
e. Low level waste evaporator (15)
f. CAS/SAS (was Control Room)(14)
g. Basin water cleanup and cooling (11, 12)

1.2.2.2 Other Structures

Adjacent to the south wall of the main building are the underground Cladding and Low Activity Waste (LAW) vaults, which were originally part of the reprocessing plant waste system, and later part of the fuel storage system waste management facilities. The underground dry chemical vault (DCV), adjacent to the main building east wall, was used during reprocessing system testing. The Clad Vault is empty and is intended for contingency service only. The LAW Vault and the DCV are empty, connecting piping has been removed or capped, and the vaults are laid away. There are no current plans for use of the LAW Vault or DCV.

The sand filter building, a principal part of the plant ventilation system, is east of the main building. All air exhausted from the fuel storage areas and from supporting areas in the main building is passed through the sand filter, sampled, and vented to the atmosphere via the 300 ft. high stack (Item #20, Figure 1-3) located southeast of the main building. Attached to the sand filter building is the emergency equipment building (EEB) (16, Figure 1-3). Other prominent structures on the site include a utility and service building; a shop and warehouse building; the administration building; a water tower; and a cask service building.

Operation of the various facilities is described in Section 1.3. The basin areas are diagrammed in Figure 1-4.

1.2.2.3 Building Drawings

Drawings of the main building and the sand filter building are included in Appendix A.14. Elevations in these drawings are based on an arbitrarily selected reference point at 47.5 ft., which is grade elevation at the main building site. The site grade reference is 532.5 ft. above sea level, and the reference "zero" elevation is 485.0 ft. above sea level.

Figure 1-4

Schematic GE-MO Basin Facilities

This Figure Withheld under 10 CFR 2.390

1.3 FUEL STORAGE OPERATIONS

A description of fuel storage activities at GE-MO is provided in the text.

In addition to the rail casks, fuel can also be received in truck casks. They typically have capacity for one PWR fuel bundle, or two BWR fuel bundles, depending on internal basket configuration.

Fuel storage operations can be divided into four major phases: receiving and cleaning casks; preparing cask for unloading; unloading and storing fuel; and preparing casks for shipment.

1.3.1 Receiving and Cleaning the Cask

When a cask arrives and before it is admitted to GE-MO, it is first surveyed and inspected for physical condition and effects resulting from its transport to the site.

1.3.1.1 Survey and Inspection

The cask and vehicle are surveyed to determine external exposure rate and detect contamination by radioactive material. Procedures are in effect to ensure compliance with regulatory reporting requirements if contamination is found in reportable quantities.

The cask and vehicle are inspected for physical damage. If damage is found, the shipper or equipment owner is notified. Depending on the nature of the damage, repairs might be required before continuing the cask receiving process.

NOTE: It is the intent of this report to make generic statements only about fuel shipping, cask handling, loading, unloading, inspection, and receiving. It shall be considered the responsibility of the cask owner, since GE-MO does not own nuclear fuel shipping casks. Prior to receiving fuel shipping casks, the owner of the cask shall provide to GE-MO a certificate of compliance for the cask, and copies of all applicable handling, loading/unloading, and inspection procedures deemed necessary, for review and use. Alternatively, the cask owner may provide a representative to supervise and perform all necessary inspections. Any cask repair/rework is the responsibility of the cask owner. GE-MO may provide facility equipment and personnel support under the direction of the cask owner. QA program requirements shall be documented in the cask owner's QA program manual and are the responsibility of the cask owner.

1.3.1.2 Removal

Provision for cask removal from the vehicle is in the cask receiving area (CRA). A radio-controlled bridge crane with 125 ton capacity lifts the cask from the trailer, or rail car, and sets it upright on the decontamination area pad.

1.3.1.3 Cleaning

The cask is typically cleaned prior to placing it in the basin, removing road dirt and grime. This reduces the effort required to decontaminate the cask following removal from the basin and helps to maintain cleanliness of basin water.

1.3.2 Preparing the Cask for Unloading

1.3.2.1 Flushing

Provision has been made so that the interior of the cask may be flushed with basin water, which may then be sampled and analyzed for radioactive contamination as a means of detecting defective fuel. If defective fuel is suspected, special procedures may be required for opening and unloading the cask.

1.3.2.2 Placement in the Cask Unloading Basin

The cask is lifted from the decontamination pad and placed into the water over the cask setoff shelf. The cask is then lowered to the shelf and disengaged from the crane hook..

The crane typically shall engage an extension hook which enables the cask to be lowered to the bottom of the cask unloading basin without submerging the crane block or cables. When the cask is positioned in the cask unloading basin, the crane is disengaged from the cask.

1.3.3 Unloading and Storing Spent Fuel

Fuel is normally unloaded using the fuel handling crane - a crane of 5 ton capacity mounted on rails attached to columns below the cask crane rails. The unloading and storage basins are served by the basin crane - a manual control bridge crane of 7.5 ton capacity. As with other cranes, the basin crane is designed to prevent derailment under seismic conditions. The basin crane has a platform on the north side of the bridge that provides a work station with excellent viewing for the fuel handling crane operator. Additionally, an underwater closed-circuit TV system is available to support operations.

1.3.4 Preparing the Cask for Shipment

The loaded cask is lifted from the cask unloading basin to the cask shelf, and the extension hook returned to its stowed position. The cask is then lifted from the water.

The cask is rinsed with demineralized water to remove basin water from the cask surface. The cask is then moved to the decontamination pad, where the remaining water in the cask is drained. The cask is decontaminated, nuts (or bolts) are tightened and the cask is leak tested. Other procedures are completed, according to requirements for the individual cask prior to reloading the cask on its transport vehicle.

1.4 SUPPORT SYSTEMS

The principal support systems are:

a. Radwaste System

b. Ventilation System
c. Basin Water Cleanup and Cooling Systems
d. Cask Sampling and Flush Systems
e. Sump Monitoring and Pump-out Systems
f. Sewage Systems
g. Utility Systems, including air and water, electricity and gas
h. Radiation Monitoring Equipment

1.4.1 Radwaste System

The Radwaste System is split into two sub-systems identified as high and low activity. The purpose of this design is to separate highly radioactive basin filter sludge from other plant waste water such as laundry, sump waste and decon solutions. The Radwaste System for liquid waste is shown schematically in Figure 1-56a. Low activity liquid wastes consist primarily of laundry water, sump water, and decon solutions. This waste is processed through an electric evaporator.

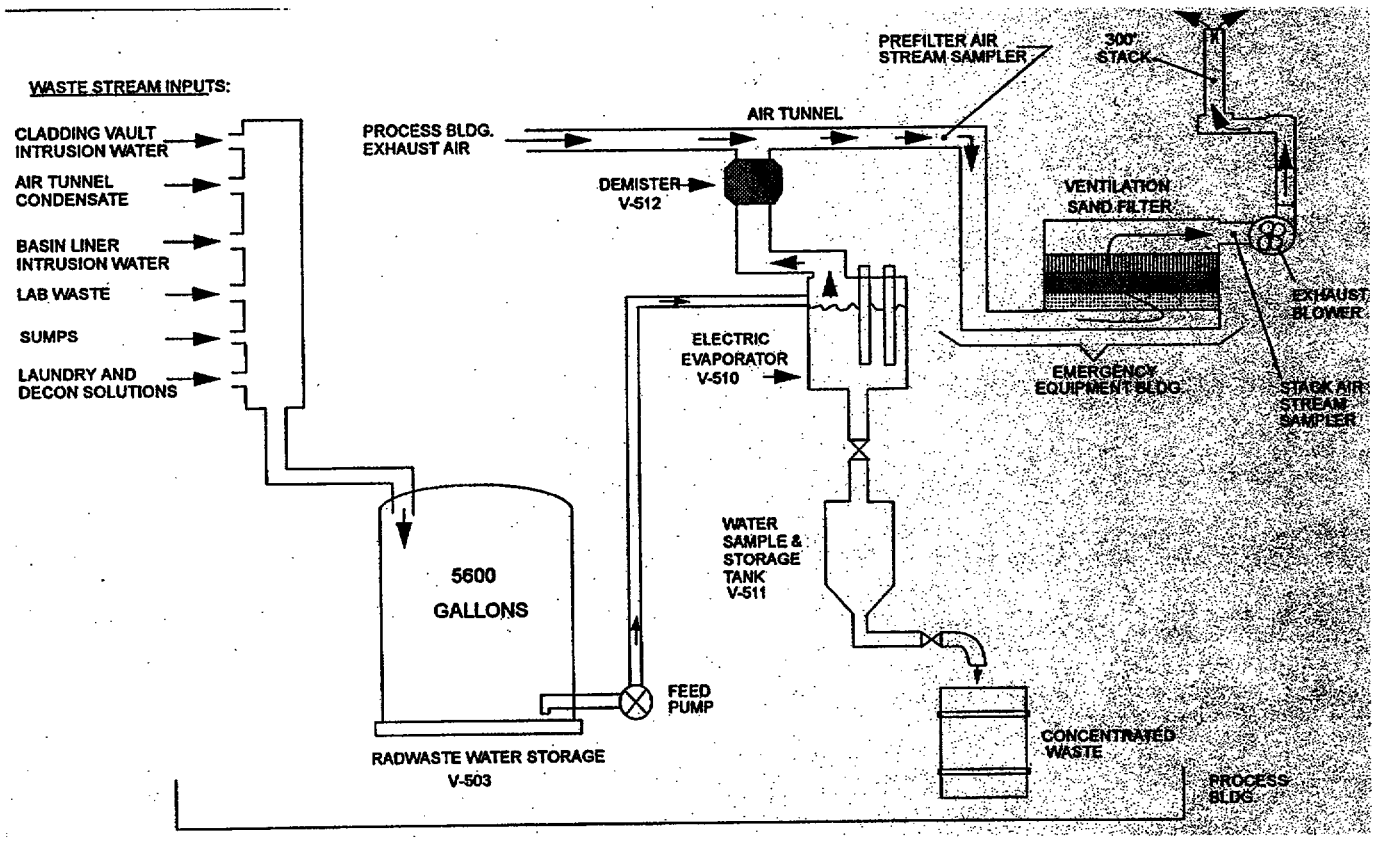

Figure 1-5a. RADWASTE SYSTEM: Low activity radwaste water streams are collected from various sources and piped to the Radwaste Water Storage Tank. Water from this tank is then pumped to an electric evaporator. Evaporator steam is demisted and exhausted via the ventilation system. Evaporator bottoms are put in barrels and shipped off site for processing.

The high activity part of the Radwaste System (Figure 1-5b) dewaters basin filter spent resins and returns the water to the basin. The dewatered filter resins and evaporator bottoms are disposed of as radwaste.

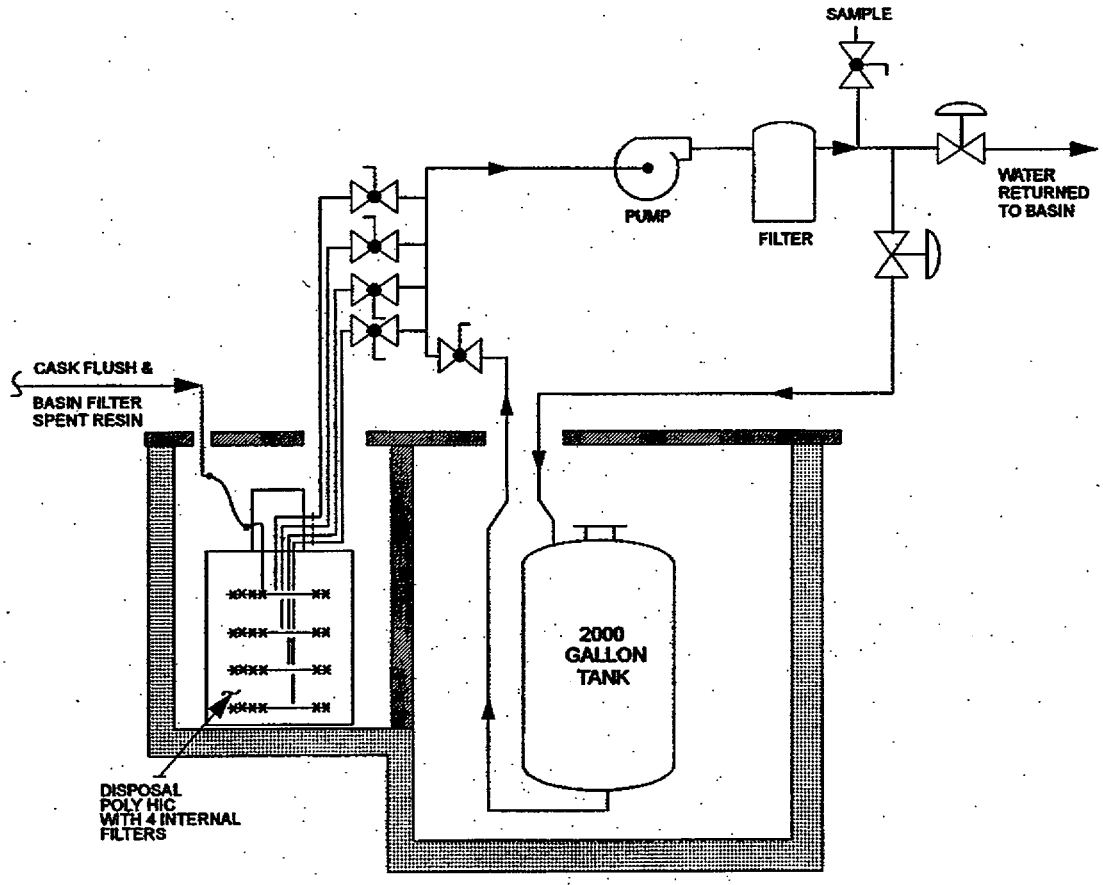

Figure 1-5b. BASIN FILTER SPENT RESIN SYSTEM: Spent resins from the Basin Filter and cask flush solutions are pumped to a shielded Poly High Integrity Container (HIC). Water is removed from the HIC, filtered and then returned to the Fuel Storage Basin. When filled, HICs are dried and shipped off site for burial.

In addition to the Radwaste System, the Cladding Vault is available to receive and hold contaminated water. This reinforced concrete vault is stainless-steel lined. The Cladding Vault is normally empty, but is maintained as a contingency if large volume water storage is required.

1.4.2 Ventilation System

A simplified diagram of the ventilation system is shown in Figure 1-17. Pressure differentials within and among connected areas ensure air flow from areas of low potential radioactive contamination (high air pressure) to areas of higher potential radioactive contamination (low air pressure).

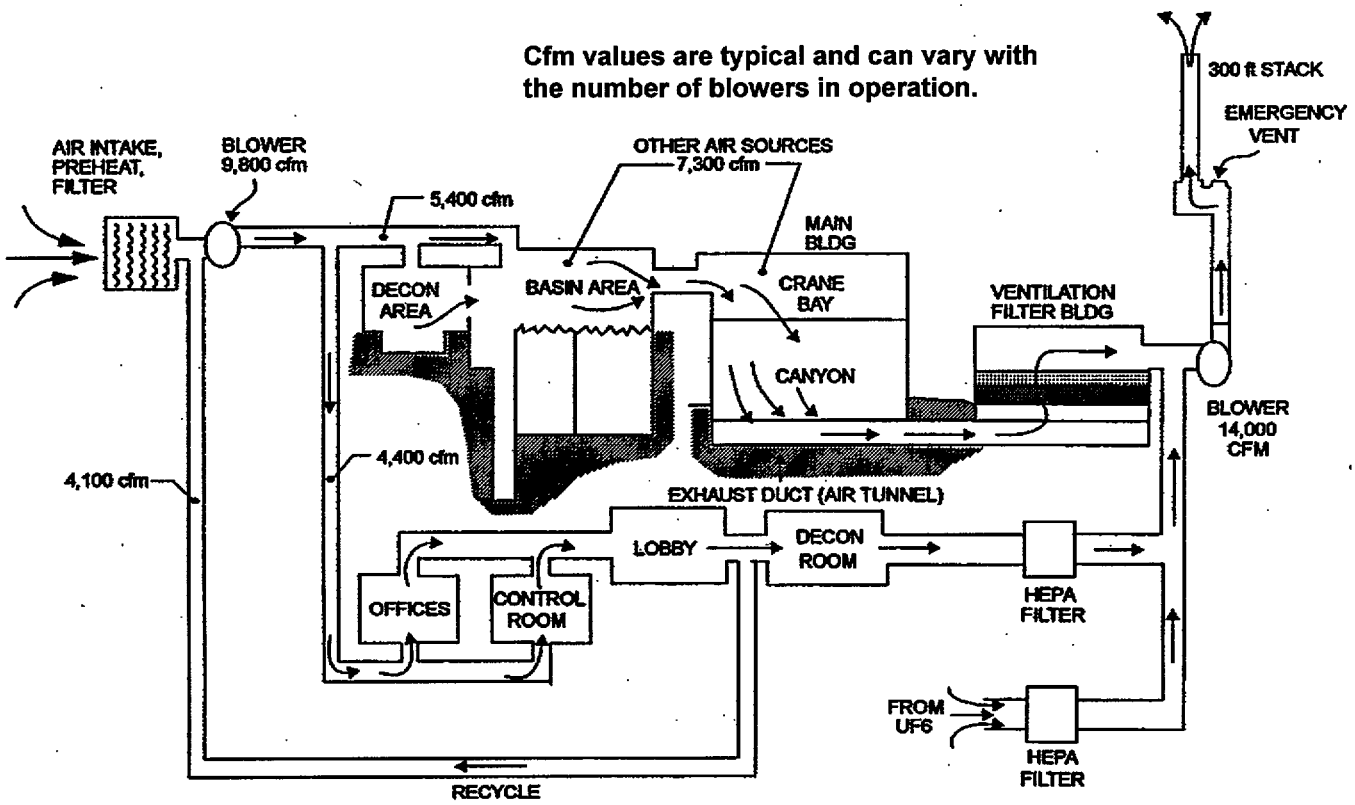

Figure 1-6. Outside air is combined with recycled air from the offices, control room and lobby and then split into two streams. One is a once through stream that passes through controlled areas to the air tunnel, through the sand filter and out the stack. The other stream ventilates the offices and is recycled with fresh incoming air. A small side-stream is diverted from this loop through a decontamination room and a filter to the stack.

Air to be passed through the sand filter flows to the air tunnel in the main building. The air tunnel provides means for draining liquids (such as condensate) to the off-gas cell sump where they are collected and pumped to the Radwaste System (Figure 1-5a and 1-5b).

1.4.3 Basin Water Cleanup and Cooling Systems

Simplified diagrams of the basin water cleanup and cooling systems are shown in Figures 1-7, 1-8, and 1-9. The filter unit is isolated in a shielded and locked room in the basin pump room. The pump room houses two 250 gpm pumps for the basin water chiller system, a 128 gpm pump for the heat pump cooling system, and a 250 gpm filter pump. Piping to the basin skimmers and water return piping is arranged to prohibit siphon action. Filter regeneration is accomplished remotely. Spent resins are pumped to the Radwaste System.

The water chiller system uses a water-to-freon chiller of stainless steel construction and rated at 1.2×10^6 Btu/hr. In addition, a separate heat pump system utilizes the waste heat from spent fuel to aid in heating personnel areas. It has a capacity of 480,000 Btu/hr.

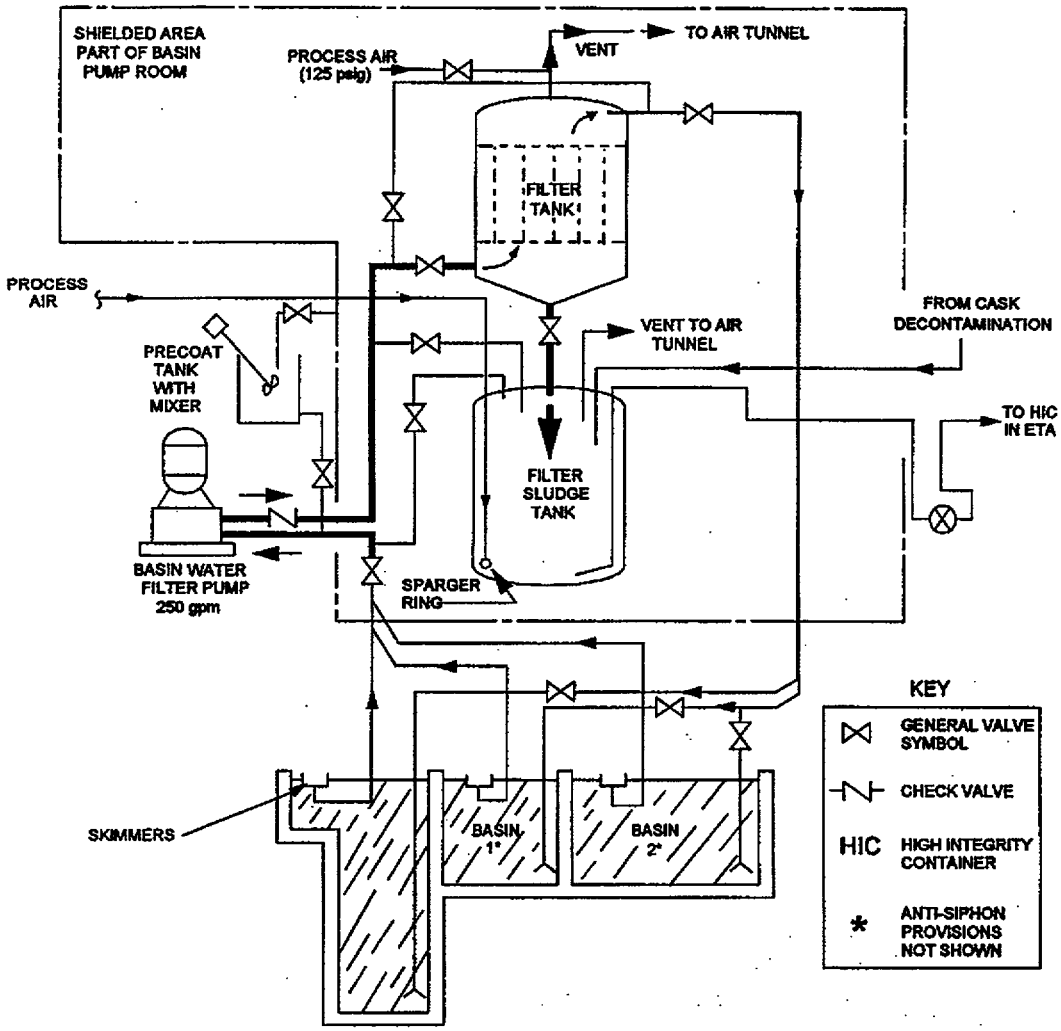

Figure 1-7. BASIN WATER CLEANUP SYSTEM: Water is continually drawn from basin skimmers at about 250 gpm, processed and returned to the basin. Filter sludge and cask decontamination water are collected in the sludge tank, then jetted to Radwaste Processing. Provisions are included for flushing tanks and precoating filters.

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Figure 1-8. BASIN WATER COOLING SYSTEM: Water is pumped from the basins to a three unit fin-fan cooler equipped with two fans.

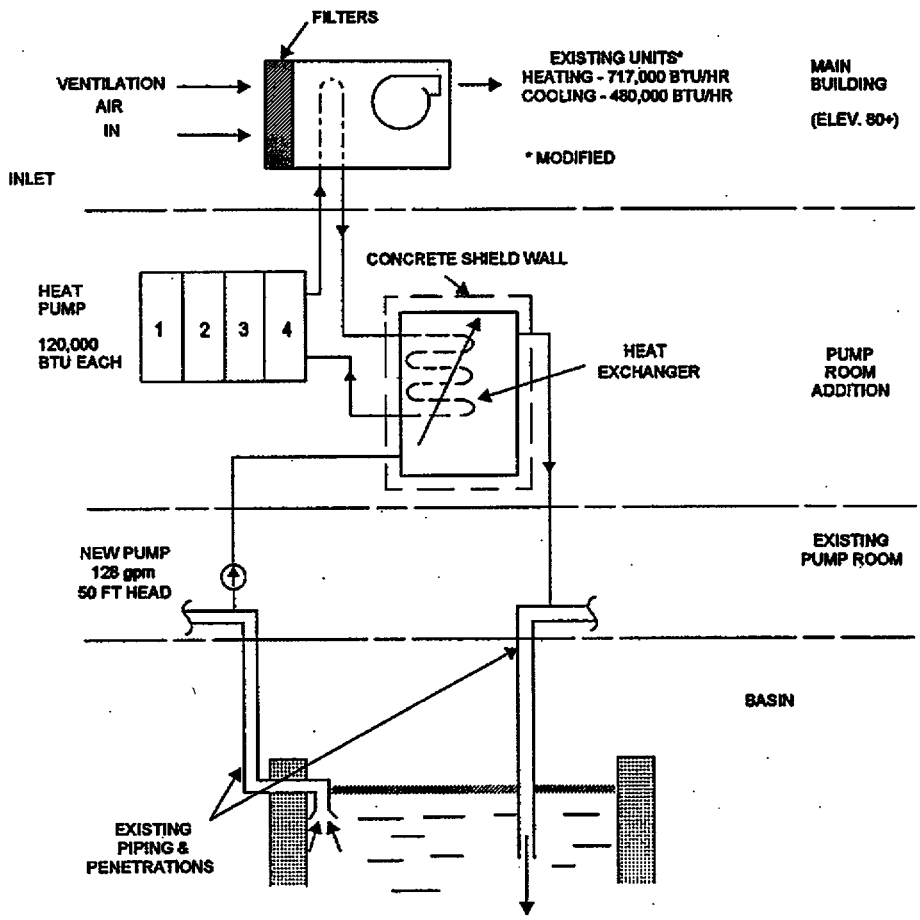

Figure 1-9. Basin Water Heat Pump Cooling System – Simplified Schematic

1.4.4 Cask Sampling, Cool-Down, and Flush Systems

The cask may be connected to the system by flexible lines. The cask may be vented to the Radwaste System and then sampled and flushed as required. Water from the basin is used as the cooling-flushing medium. Outflow from the cask may be piped to the Radwaste System or the Basin Filter inlet. Air from the process air supply is available to purge the cask. The purge air is discharged to the ventilation system.

1.4.5 Leak Detection and Sump systems

Basic to the leak detection system is a sump that accumulates leakage water as well as intrusion water (water entering from surrounding rock). A simplified schematic of the leak detection and empty-out system for the fuel storage basins is shown in Figure 1-21. The sump is emptied using a combination of an air lift and an air operated diaphragm pump. Provisions are included to sample sump water. All vaults are equipped with similar systems utilizing electric pumps in place of air lifts.

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Figure 1-10. LEAK DETECTION, EMPTY-OUT AND SAMPLING SYSTEM: Sumps are provided in several locations to collect leakage or other runoff. Water detection, empty-out and, in some cases, sampling and monitoring facilities are provided. This schematic shows fuel basin liner leak detection and empty-out system in simplified form.

1.4.6 Sewage systems

No sewage is discharged from the GE-MO site. Sanitary wastes are piped to the sanitary lagoons. A simplified schematic of the sanitary sewage systems is depicted in Figure 1-11.

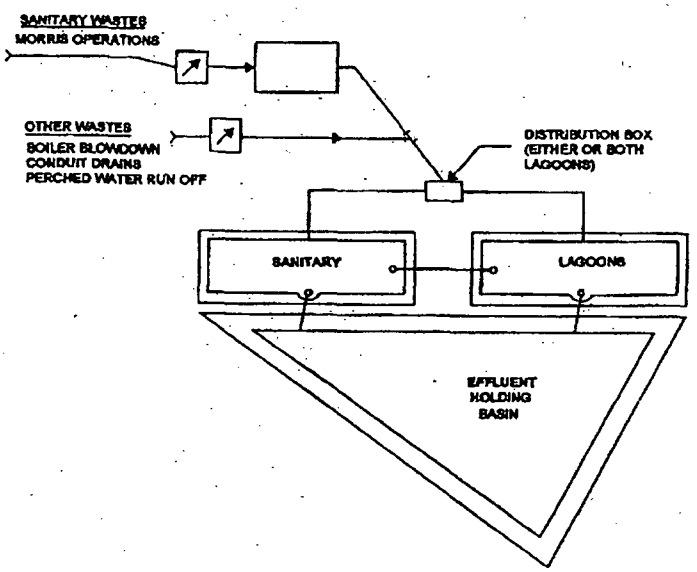

Figure 1-11. SEWAGE SYSTEMS: No liquid effluent is discharged off-site; only rain runoff is drained by open ditch, eventually discharging to the river. Holding basin retains lagoon effluent.

1.4.7 Energy Systems

There are two energy sources on site: the electrical system and the natural gas service.

a. Electrical: Electrical power is furnished by Commonwealth Edison Company (CECo) via two 34,000-volt lines. Distribution facilities are located in and near the utility service building (13, Figure 1-3). Principal loads at GE-MO are crane operation, ventilation system, control and instrumentation, and auxiliary systems and equipment.

   Although interruption of electrical power would not result in unsafe conditions, secondary power sources (originally intended as emergency sources for reprocessing activity) are provided to ensure continuing operation of electrical equipment during power outages.

b. Natural Gas: Natural gas is used as fuel for heating various areas on site. These include the cask receiving area, cask service facility, the Mock-Up Tower, the Maintenance shop, and the water tower. Because of the noncritical nature of gas usage, no alternative gas supplies are provided.

1.5 RADIOLOGICAL AND OTHER MONITORING

GE-MO monitors gaseous and liquid (ground water and surface water) effluent from the Morris Operation OCA boundary.

Within the GE-MO facility, sampling and laboratory analyses supplement the constant air and other monitoring devices to ensure a safe environment for employees and to detect trends or events.

1.6 EMERGENCY PROVISIONS

The GE-MO Emergency Plan (NEDO 31955) describes actions to be taken during emergency situations. Structures and systems at Morris supporting emergency action such as law enforcement, medical, fire, or other emergency services are identified. Assistance agreements exist with appropriate local agencies.

1.7 REFERENCES

1. License and docket information and a list of applicable documents are contained in Appendix A.1 and A.2.

2. Storage capacity expressed in terms of metric tons of uranium (TeU) as contained in LWR fuel rods.

3. See Chapter 8.

2.0 SUMMARY SAFETY ANALYSIS

In consideration of provisions in proposed regulator guides, and summaries contained in other chapters of this report, this chapter has been deleted.

3.0 SITE CHARACTERISTICS

3.1 INTRODUCTION

This section provides descriptions of geographical, demographic, meteorological, hydrological, seismological, and geological characteristics of the GE-MO site and vicinity. This information has been derived from various documents submitted during MFRP licensing activities[1] and site studies performed as part of actual and proposed capacity expansions. Applicable information from the history of experience in receipt, storage and transfer of irradiated nuclear fuel dating back to 1972 is also included.

3.2 GEOGRAPHY AND DEMOGRAPHY OF SITE

This section includes a description of site geography, population and land use considerations as applicable to the fuel storage facility.

3.2.1 Site Location

GE-MO facilities are located on a tract of about 886 acres owned by General Electric Company (GE or the Company) in Gooselake Township, Grundy County, Illinois, near the confluence of the Kankakee and Des Plaines Rivers. The tract is located 41°22'53" N latitude, 88°16'32" W longitude; about 15 air-miles southwest of Joliet and about 50 miles southwest of the Chicago, Illinois - Gary, Indiana area. Aurora is located about 25 miles north, and Kankakee is about 25 miles to the southeast. Morris, the county seat of Grundy County, is about 7 miles to the west. Interstate Highway 55 (I-55) is about 4 miles east, and Interstate Highway 80 (I-80) is about 5 miles to the north. Figures 1-1 through 1-3 depict the tract general location, and Figures 3-1 and 3-2 depict general plot arrangement and neighboring structures and activities.

3.2.2 Site Description

Figure 1-3 is a map of the site, showing the site, OCA, and other details, including transmission lines, gas lines, and other features. The GE-MO site is in a developing industrial area of typically "rolling prairie" terrain. In general, land in the area has been farmed for many years but the GE-MO buildings are in an area of rocky outcroppings and thin top soil, unsuited to economical, large-scale farming of crops.

3.2.2.1 GE-Morris Operation Boundary

GE-MO boundaries and surrounding lands and waters are shown in Figure 3-1. The tract's northern boundary is formed by Collins Road and the eastern boundary by Dresden Road. The Illinois and Kankakee Rivers are separated from the tract to the north and east by lands of Commonwealth Edison Company's (CECo), Dresden Nuclear Power Station (DNPS) and related facilities, and a privately owned plot of about 50 acres. To the south, the tract is bordered by discontinued clay mining operations now privately owned. Other lands bordering the GE tract include industrial areas to the northwest, and Goose Lake Prairie State Park

adjacent to the GE tract with the closest point about 0.6 miles west of the GE-MO stack. Both road and rail transportation services are available to the site.

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Figure 3-1. TOPOGRAPHIC MAP: GE Tract and Vicinity

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Figure 3-2. Contour Map – GE Morris Operation

3.2.2.2 Property Ownership

GE is the sole owner of the entire 886 acre tract, subject to easements which have been granted for power lines, and natural gas lines, as shown in Figure 1-2. The tract, as originally purchased, totaled about 1,380 acres and included that portion of Section 1, Township 33 North, Range 8 East that is south of the Kankakee River, all of Section 2, Township 33 North, Range 8 East and that portion of Section 35, Township 34 North, Range 8 East that was south of the DNPS site.

Since that time, about 70 acres located in the southwest corner of Section 1, Township 33 North, Range 8 East and about 50 acres in a 400 ft. wide strip along the south edge of Section 2, Township 33 North, Range 8 East were sold to A. P. Green Refractory Company, Illinois Products Division, for use in connection with clay mining and clay products manufacturing activities. Subsequently, the remainder of Section 1, Township 33 North, Range 8 East and a 525 ft. wide strip along the east edge of Section 35, Township 33 North, Range 8 East and extending into Section 2, Township 33 North, Range 8 East for a short distance have been sold to CECo for flume access to and from the DNPS cooling lake.

3.2.2.3 Access Control

Access to the GE-MO tract is controlled. GE-MO facilities occupy about 52 acres in the north portion of the tract, adjoining the DNPS site. Principal plant structures, including the ventilation stack, are located within an area of about 15 acres, fenced with chain-link-type fencing topped by multiple strands of barbed wire with an overall height of 8 ft. Access to the site is controlled by gates. The remainder of the tract is enclosed by an agricultural fence with posting advising unauthorized persons to keep out. In the conveyance of parcels previously described, provisions have been included to ensure their subsequent use and access will continue to be appropriately controlled. CECo similarly controls access to the DNPS site and security areas.

A lease agreement permits limited farming and beef cattle grazing on the tract outside the OCA.

3.2.2.4 Boundaries for Establishing Effluent Release Limits

The OCA boundary (the tract boundary shown in Figure 1-2) is the boundary for establishing dose equivalents as defined in 10 CFR 72.104 and 72.106.

No credible acts of nature, man-induced events or accidents have been identified that would result in biologically significant release of radioactive material or direct radiation dose in excess of limits of 10 CFR 72.106 outside the OCA boundary. Therefore, the Emergency Planning Zone (EPZ) for GE-MO coincides with the OCA boundary.

3.2.3 Population, Distribution and Trends

The data base for the following sections is founded on information developed by agencies of the States of Illinois and Indiana, as well as information developed by GE and CECo [2,3,4].

3.2.3.1 Population 0 and 5 Miles (Figures 3-3 and 3-4)

The population in the immediate vicinity of GE-MO is very low. Within a radius of 5 miles the population is about 14,700 including 5,256 (1993) in the village of Channahon, about 4 miles to the northeast. Included in this accounting are several residences at the Dresden Lock and Dam. The 1990 population figures within a 5 mile radius are based on local community estimates and are not intended to represent U.S. census data.

The population within 5 miles of the site is projected to increase to 30,000 by the year 2015[5].

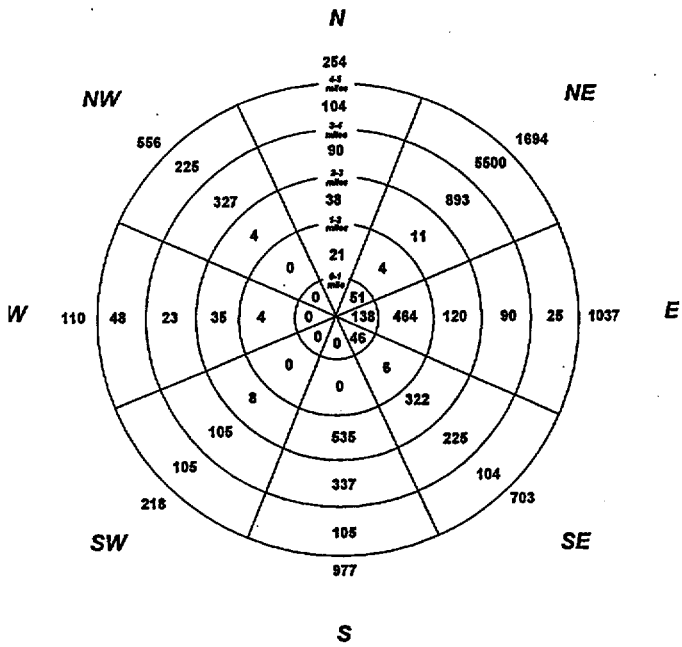


Figure 3-3. Estimated population within a five mile radius of GE-MO, 1990

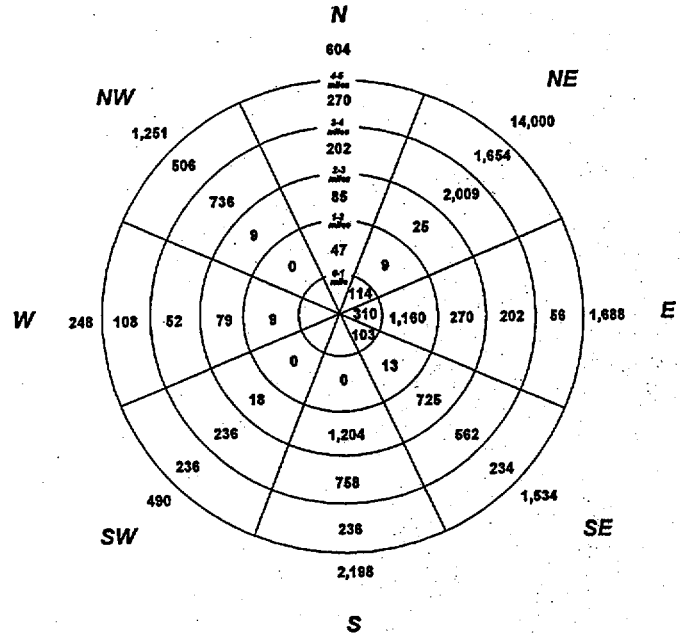


Figure 3-4. Estimated population in a five mile radius of GE-MO, 2015

3.2.3.2 Population Within 50 Miles (Figures 3-5 and 3-6)

The total population within the 50 mile radius was about 7,000,000 in 1990 and is projected to reach 8,000,000 by 2015 with about 91% of the total beyond the 30 mile radius[6,7].

Studies by CECo's Industrial Development Department indicate that since 1946, 82% of the new industries locating within the CECo's system are located within 25 miles of downtown Chicago. In 1965, 80% of the new industries also located according to this pattern. Current indications are that this industrial growth pattern is slowing but continuing within the 25 mile belt. Thus, the growth adjacent to the GE-MO-DNPS sites (which are outside of the 25 mile belt) should continue but at relatively low rates. Joliet and Aurora are the closest areas likely to experience significant population increases.

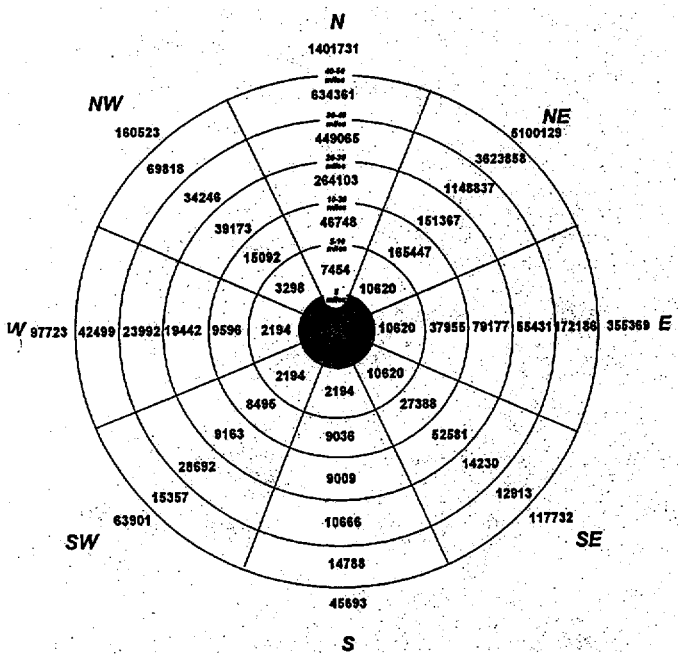


Figure 3-5. Estimated population within a 5-50 mile radius of GE-MO, 1990

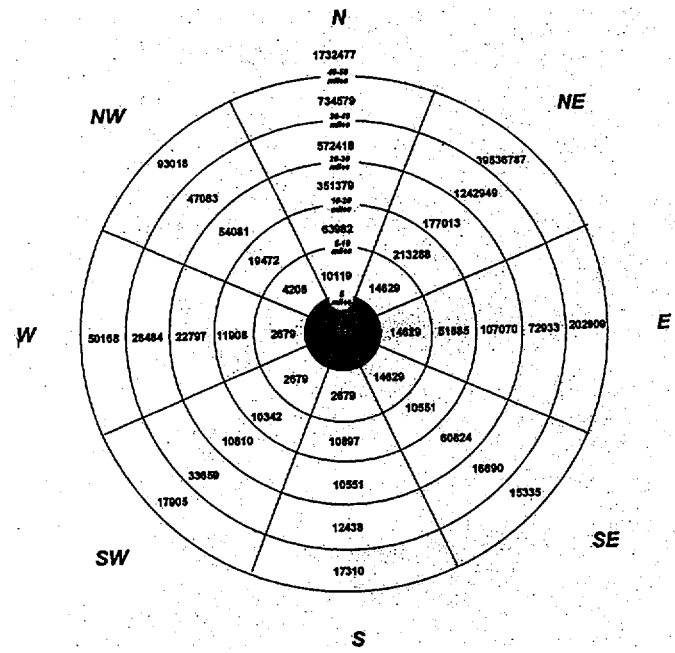


Figure 3-6. Estimated population in a 5-50 mile radius of GE-MO, 2015.

3.2.3.3 Transient Population

There are small seasonal variations in population in the area farm lands because of harvest personnel requirements. Unlike some farm areas, harvest activities are highly mechanized and relatively few additional workers are required.

Almost all manufacturing and other industrial activity is nonseasonal and draws upon a population base that resides in the same general area. For example, with the largest part of Chicago's industrial and residential areas within the 50 mile radius, daily movements of people within Chicago and environs result in a relatively insignificant statistical change from the viewpoint of considerations applicable to the GE-MO site.

As discussed elsewhere in this Section, recreational uses of lands and water in the area result in small seasonal changes in population in cottages, etc.

3.2.4 Users of Nearby Land and Waters

Immediate GE-MO neighbors (Figure 3-2) are the DNPS site on the north, abandoned clay pits and open farm land on the south and Goose Lake Prairie State Park to the west. To the east is the Dresden cooling lake and a privately owned property of about 50 acres, divided into about 30 cottage sites. Collins Station, a fossil-fired plant is to the west-southwest of GE-MO.

Present land use patterns in the area seem likely to continue for some time to come. The Northeastern Illinois Planning Commission does not expect a change in the pattern in the southwestern corner of adjacent Will County, either. (The county line is approximately 1.5 miles east of the GE tract.)

3.2.4.1 Industrial

In addition to CECo's holdings to the east, north, and northwest, another industrial area is located along Interstate Highway 55 (I-55). This highway runs north and south, about 4.5 miles directly east of the tract (Figure 1-1). Two miles east of I-55 is the inactive Joliet Army Ammunition Plant. A large Mobil Oil petroleum refinery is located where I-55 crosses the Des Plaines River. Industrial sites are also located on the north bank of the Illinois River.

3.2.4.2 Residential Use and Population Centers

Residential occupancy in the immediate vicinity of GE-MO is low. There is a cluster of about 30 cottages on the west shore of the Kankakee River, about 0.5 miles from the GE-MO stack. These are located between Dresden Road and the Kankakee River on a tract of about 50 acres adjacent to the GE-MO and DNPS sites. Residential development in the immediate vicinity of GE-MO would be limited to this tract which is now nearing saturation.

There is a similar group of cottages on the Kankakee River east bank greater than 1 mile from the GE-MO stack. Some homes in this area are permanent residences, although most have been developed for part-time recreational purposes. Surveys by CECo indicate that within 2.5 miles of the DNPS site there are a total of 129 permanent homes and 191 part-time recreational cottages along the Kankakee River. Other residences in the area include several at Dresden Dam about 1.2 miles to the north. There are no major residential centers developing south of the Kankakee and Illinois Rivers in the vicinity of the GE tract.

Cities and towns having populations greater than 1,000 located within 30 miles of GE-MO are listed in Table 3-1.

Other areas and sites involving intermittent and temporary congregations of persons within 5 miles of area are as follows (data as of May 1994):

a. Schools - Enrollment[8]

Minooka High School	895	Channahon School	1,200
Minooka Jr High & Grade School	1,031	Illinois Youth Center[9]	Closed

b. Churches - average attendance of largest service

Minooka Catholic	300	Minooka Methodist	200
Channahon Baptist	125	Channahon Methodist	120
Channahon Catholic	350	Goose Lake Baptist	140
Phelan Acres Bible	65		

c. There are no hospitals within the 5 mile area.

Table 3-1

CITIES GREATER THEN 1,000 POPULATION WITHIN 30 MILES OF GE-MORRIS OPERATION

Area	Name	Population (1990 Census)
0-5 Miles	Channahon	5,256
5-10 Miles	Braidwood	3,584
	Coal City	3,940
	Morris	10,270
	Wilmington	4,743
	Minooka	2,561
10-20 Miles	Crest Hill	11,000
	Gardner	1,237
	Joliet	76,836
	Lockport	9,401
	Manhattan	2,059
	Marseilles	4,811
	New Lenox	9,627
	Plainfield	4,557
	Rockdale	1,709
	Seneca	1,878
	Shorewood	6,264
20-30 Miles	Aurora	99,581
	Bolingbrook	44,000
	Bourbonnais	13,927
	Bradley	10,792
	Dwight	4,230
	Frankfort	7,180
	Kankakee	27,575
	Lemont	7,348
	Manteno	3,488
	Matteson	11,378
	Mokena	6,128
	Montgomery	4,267
	Naperville	85,351
	North Aurora	5,940
	Odell	1,030
	Orland Park	35,720
	Oswego	3,876
	Ottawa	17,451
	Peotone	2,974
	Plano	5,104
	Richton Park	10,528
	Romeoville	14,074
	Sandwich	5,567
	Somonauk	1,263
	Sugar Grove	2,005
	Tinley Park	37,121
	Woodridge	26,256
	Yorkville	3,925

3.2.4.3 Agricultural

There is no land suitable for large-scale farming operations within two miles of the GE tract. There are home gardens and some truck farms located near Plainfield and Joliet. Crops from truck farming in this area are generally for local consumption. Most farming operations raise corn, soy beans and grains. There is some farming and beef cattle grazing permitted on the GE-MO tract under a lease arrangement. The closest dairy herd is about seven miles south.

3.2.4.4 Recreational

Principal recreational activities in the area include swimming, boating, hunting and fishing. Most activities involve the Kankakee River and the "finger lakes" which have been left from earlier strip mining operations. Goose Lake Prairie State Park is located to the west of the tract. There is little sport activity, other than boating, on the Illinois and Des Plaines Rivers because of pollution of the Des Plaines River as it flows through the Chicago area.

3.2.4.5 Adjacent Waters

The only waters near the GE tract are the Kankakee and Illinois Rivers, DNPS cooling lake, Collins Station cooling lake, and small "finger Lakes".

CECo does not allow access to the Dresden cooling lake for recreational uses. A portion of the Collins Station cooling lake is managed by the Illinois Department of Conservation for fishing and waterfowl hunting. The Illinois Waterway, one of the major inland waterways, is adjacent to the DNPS site. An agreement between GE and CECo provides for access to the Illinois Waterway through the DNPS site so that facilities for boat docking and access roads to the waterway could be developed at some future time if required.

There are two small "finger lakes" about 2.5 miles south of the GE tract where homes have been built, while other lakes on which houses are being built are located about 3.5 miles southwest. Some houses are solely for recreational purposes.

3.3 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

None of the industrial, military, or transportation activities in the area present a credible hazard to the fuel storage facility nor to the transport of irradiated nuclear fuel. Fuel in storage is located well below ground level in a stainless steel-lined, reinforced concrete water basin, and held in stainless steel baskets latched in a supporting grid. Explosions or fires at "nearby" industrial facilities would be too far away to have any influence on fuel in storage. Even the explosion of a passing tank truck would not affect the safety of stored fuel. Likewise, the structural characteristics of fuel casks and the nature of nearby activities result in minimum hazard to transportation of spent fuel.

3.3.1 Nearby Nuclear Facilities

The location and identification of nuclear facilities within 50 miles of GE-MO site are shown in Table 3-2. The closest facilities are the DNPS Units 1, 2 and 3, located about 0.7 miles north of the GE-MO stack. The combined radiological impacts from GE-MO and DNPS are within requirements of 10 CFR 72.104 as indicated by calculations and environmental monitoring results. Calculated dose commitments from GE-MO are a small fraction of the dose commitments from DNPS, even considering design basis accidents evaluated in Section 8.

Table 3-2

NUCLEAR REACTORS WITHIN 50 MILES OF GE-MORRIS OPERATION

Type	Capacity (MWe)	On Line	Latitude	Longitude	Airline Miles to GE-MO	
BWR	200	1960	41°22'	88°14'	0.7	Dresden 1*
BWR	809	1970	41°22'	88°14'	0.7	Dresden 2
BWR	809	1971	41°22'	88°14'	0.7	Dresden 3
BWR	1,078	1983	41°21'	88°36'	20	LaSalle 1
BWR	1,078	1984	41°21'	88°36'	20	LaSalle 2
BWR	1,100	1986	41°16'	88°13'	10	Braidwood 1
BWR	1,100	1988	41°16'	88°13'	10	Braidwood 1

* Dresden 1 was shutdown in 1978

3.3.2 Industrial and Military

The GE tract is near several industrial sites along the Illinois River (Figures 1-1 and 1-2). Most development is north of the Illinois River over 1 mile from GE-MO. The development of the last few years is slowing as most suitable industrial sites are already occupied and Goose Lake Prairie State Park now occupies most of the remaining land south of the river.

In addition to DNPS immediately to the north, other industry in a 6 mile radius of GE-MO is listed in Table 3-3.

Table 3-3

INDUSTRIAL, TRANSPORTATION, AND MILITARY ACTIVITIES
(6-mile radius)

Installation	Function	Proximity
Reichold Chemical Plant	Chemical Plant	1.5 mi NW
Alumax	Aluminum Mill Products	3 mi NW
Northern Natural Gas Co.	Natural Gas Manufacturing	3 mi NW
Mobil Plastics	Chemical Plant	4 mi ENE
Mobil Oil Co.	Oil Refinery	4 mi ENE
Collins Power Station	Electricity generation (fossil-fired)	4 mi WSW
ARMAK Co.	Mfg of fatty Acid	4 mi WNW
Quantum	Mfg of polyethylene and ethylene glycol	4 mi NW
Joliet Arsenal	Munitions plant (Inactive)	6 mi E
Demert and Dougherty	Filling aerosol cans	6 mi S
Dow Chemicals	Styrofoam manufacturing	4 mi ENE
Durkee	Edible Oil processing	4 mi ENE

3.3.3 Transportation

One principal factor in the original selection of the GE-MO site was the ready availability of excellent rail and highway access to all parts of the United States and water transportation that could be developed if required in the future.

Highway access to the tract is via a paved county road, known as Dresden Road, extending south from the DNPS site parallel to the GE-MO tract and intersecting Pine Bluff Road (Figure 1-2). Pine Bluff Road (named Lorenzo Road in Will Country) runs in an east-west direction approximately 1 mile south of the GE tract boundary and provides access to I-55 approximately 4 miles east of the site, and Illinois 47 to the west. I-55 is a limited access highway between Chicago and St. Louis. Another limited-access highway, Interstate Highway 80, which traverses the State from east to west, is approximately 5 miles north of the GE lands and is accessible either from I-55 or from State Highway 47.

Railroad access to the tract is provided by a spur from the Elgin, Joliet and Eastern (EJ&E) Railway through the DNPS site. The EJ&E is a belt line which circles Chicago from near Wisconsin on the north to Indiana on the east, and connects with every major railroad serving Chicago. Through these connecting lines direct rail services to all parts of the United States are available.

There are no airports within 8 miles of the site and the closest major airports are Chicago O'Hare International Airport and Chicago Midway Airport, situated approximately 50 miles and 40 miles, respectively, to the north and northeast of the site. Commercial flights approach

Chicago airports from the southwest, so that most flights pass to the west of the GE-MO site. Data for aircraft flying the Visual Omni Range (VOR) - Joliet for the 37th busiest day (used for statistical purposes by the Federal Aeronautics Administration (FAA) to represent an above average day) in September 1979 are shown in Table 3-4.

Table 3-4

VOR - JOLIET FLIGHTS[a]
September 1979

Time Periods	Civilian Flights	Air Carriers
0800 - 1600 hrs.	124 (3000 - 9000 ft.)	111 10,000 ft. or above
1600 - 2200 hrs.	85 (same)	96 (same)
2200 - 0800 hrs.[b]	14	21

a. Track is about 3 miles west, 5 miles north of Minooka.
b Data for 2200-0800 hrs. is typical.

3.4 METEOROLOGY

The climate of Illinois is typically continental, with cold winters and warm humid summers. There are frequent short-period fluctuations in temperature, humidity, cloud cover, wind speed and direction. Winds are controlled primarily by storm systems and weather fronts which move eastward and northeastward through the area. Southeasterly and easterly winds usually bring mild and wet weather. The southerly winds are warm and showery while westerly winds are dry with moderate temperatures. Winds from the northwest and north are usually cool and dry. With the exception of tornadoes, there are no severe weather extremes in the area[10,11,12].

3.4.1 Regional Climatology

Topography of the area is not significant in affecting regional climatology except for some localized fog situations related to the rivers, strip-mine lakes, and the DNPS cooling lakes. The land is commonly referred to as rolling prairie and is without significant topographical features. Even Lake Michigan, the topographical feature of the area having the most meteorological significance, has only a general effect on the region's climate, and no specific effect on GE-MO.

3.4.1.1 Temperature and Precipitation

Temperature data for Morris, Illinois, is shown in Table 3-5. Annually, there are usually 28 days with temperatures above 90 °F occurring from May through October and 141 days with temperatures below 32 °F occurring from September through April. Average precipitation, including snowfall, and average snowfall data for Morris and Joliet, Illinois, are shown in Table 3-6[13]. The ANL record for June 1950 to June 1964 shows an annual average precipitation of

31.49 in. with a 24 hr. maximum of 6.24 in. A maximum annual snowfall of 100+ in. was recorded during the 1978-79 winter[14].

Table 3-5

LOCAL TEMPERATURE DATA (°F) FOR MORRIS, ILLINOIS

Month	Average	Low	High
January	25.8[a]	-22	68
February	27.5[b]	-22	67
March	37.3	-19	82
April	50.2	17	90
May	61.2	25	103
June	70.8	34	106
July	74.9	41	109
August	73.3	49	107
September	65.9	26	103
October	54.9	14	92
November	40.1	-9	82
December	28.7	-22	64

[a] Record period of 29 years [b] Record period of 28 years

Table 3-6

TOTAL PRECIPITATION[a] & TOTAL SNOWFALL (IN.) FOR MORRIS[b] & JOLIET, ILLINOIS

Month	MORRIS Precipitation (in.)	JOLIET Precipitation (in.)	JOLIET Snow (in.)
January	1.68	1.76	5.6
February	1.72	1.68	4.7
March	2.20	2.96	3.6
April	3.77	3.94	0.2
May	2.92	3.86	T[c]
June	4.65	4.31	T
July	4.48	3.93	T
August	3.36	3.50	0.0
September	2.00	3.01	0.0
October	2.27	2.33	T
November	1.74	1.91	2.2
December	1.81	1.98	6.5

[a] Amounts shown include equivalent inches of water for snowfall
[b] Snowfall data for Morris not available [c] T = Trace

3.4.1.2 Humidity and Fog

Average relative humidity in January is 85% at 8 a.m., 75% at noon and 80% at 8 p.m. (CST). Average relative humidity in July is 77% at 8 a.m., 55% at noon and 62% at 8 p.m. The 1% summer design wet bulb temperature is 78 °F[15].

Fog is more frequent in the region than at continental locations of similar latitude across North America. This is because of the influence of Lake Michigan, local rivers, and the DNPS cooling lake and related systems. The main physical processes causing radiation, advection, orographic and steam (ground) fog are evident in the region[16]. This natural fog occurs most frequently and persists the longest in winter. On the average, dense fog (visibility less than 0.4 km) occurs during less than 15% of the 300 to 450 hours of winter fog. Dense fog is recorded most frequently in the early morning. Winter fog occurs most frequently with temperatures between 14 °F and 40 °F and summer fogs with temperatures between 59 °F and 69 °F. Dense fog in winter occurs almost exclusively with surface saturation deficits of 0.5g per kilogram day air or less[17].

The closest meteorological station that has collected fog data is the Joliet Municipal Airport (about 12 miles NNE). Meteorological observations representing 99,165 hrs. (about 11 years) indicate that a total of 12,284 hrs. (12.4%) of fog with visibilities of 6 miles or less occurred at the airport. Dense fog having "zero" visibility (less than 330 feet) occurred 0.25% of the time, or about 23 hours per year. These critical cases occurred most often in winter, least in summer (most often in January and least in June) and most often in the early morning hours (0500-0900 CST). The "zero" visibility fogs had a median persistence of up to 3 consecutive hours. However, one occurrence lasted for 12 consecutive hours, with an estimated reoccurrence in 10-20 years[18].

3.4.1.3 Tornadoes

Information from the U.S. Weather Bureau indicates that over a period of approximately 40 years, there was an average of 4.8 tornadoes per year in Illinois, which is very close to the average for all states east of the Rocky Mountains. Of 192 tornadoes reported in the state, 52 were considered to have been "destructive" (i.e., damage of $50,000 or more and/or at least one death). The average area covered by reported tornadoes is about 8 square miles. Reported path widths range from 34 yards to 4 miles[19].

Several tornadoes have been reported near the DNPS site since 1965. On November 12, 1965, a tornado passed 4 miles west of the site while moving toward the east-northeast at approximately 70 mph. Several electrical transmission lines to the site were interrupted and, as a result, DNPS Unit 1 was shut down for about 24 hr. A second tornado, on May 24, 1966, passed near the site resulting in one transmission line being lost. However, the load was carried by other electrical transmission lines and DNPS Unit 1 operated normally. On July 17, 1972, a tornado passed northwest of the GE-MO site, and on April 3, 1974, a tornado touched down just north of Morris, Illinois. Neither caused damage in the GE-MO area.

3.4.2 Local Meteorology

Data and sources of data for site temperature, water vapor, precipitation and fog conditions are contained in Section 3.4.1.

3.4.2.1 Wind Data

Annual wind frequencies show a rather uniform distribution of wind direction (Figure 3-8). The most frequent wind directions are from the west and south sectors (based on 22.5 degree sectors). Average wind speed at the 300 ft. level is about 15 mph and at the 125 ft. level is about 11 mph. These observations are based on 1968 data taken from the DNPS meteorological tower. Maximum wind velocity reported in the area of the site is 109 mph, unofficially reported at Joliet on April 3, 1956, and on April 30, 1962, as the fastest gust during heavy thunderstorms and scattered tornadic activity. The fastest windspeed reported at various locations in the site area is 87 mph at Chicago and 75 mph at Peoria[20].

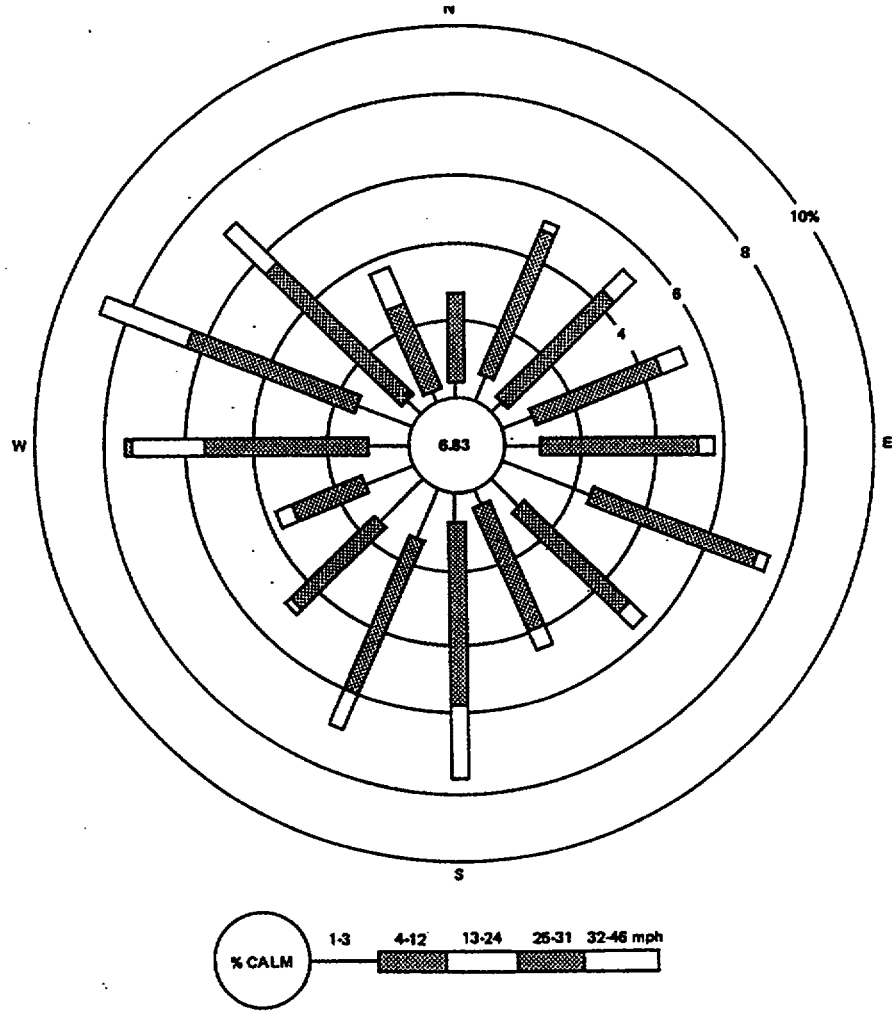


Figure 3-8. Annual Wind Rose at 35 foot Level at DNPS Site.

3.4.2.2 Topography

The only major topographic influence in the area is Lake Michigan which is 45 miles to the northeast and is considered to have an insignificant effect on site climatology. The only potentially significant topographical features around the site are the Dresden Heights, located on the north side of the Des Plaines River, about 1.5 miles northeast of the site ventilation stack. These bluffs rise to an elevation of 630 ft., compared to the elevation at the site of 530 ft. Since the stack extends 300 ft. above the grade, the perturbation in the flow of the plume over the bluffs located some 1.5 miles away is quite small.

These bluffs are the only significant topographical features near the GE-MO site or, in fact, in most of northeastern Illinois. The only other topographical disturbances in the area are spoil piles which remain from abandoned strip mines. These are located farther from the site and are not as high as the bluffs across the river. The highest topographical elevation in Illinois is Charles Mound, elevation 1,241 ft., located on the Illinois-Wisconsin border. The average elevation of the state is 600 ft.

3.4.2.3 Electrical Storms

Thunderstorm activity in the Chicago area for the years 1970 through 1975 is presented in Table 3-7 in terms of thunderstorm days per month. The incidence of thunderstorms over a 33 year period is about 39 per year[21].

Table 3-7

THUNDERSTORM ACTIVITY

	YEAR						
Month	70	71	72	73	74	75	33 Year Average
1	0	0	0	0	0	4	<0.5
2	0	1	0	1	0	1	<0.5
3	2	4	5	4	7	4	3
4	10	3	6	4	7	8	5
5	10	6	5	4	8	9	5
6	9	10	7	10	10	13	7
7	10	9	7	7	6	7	6
8	7	4	8	3	4	9	5
9	11	4	6	6	3	3	4
10	3	2	1	4	1	2	2
11	1	2	0	3	0	3	1
12	1	2	1	1	1	3	1
Total	64	46	46	47	47	66	39

3.4.3 On-Site Meteorological Measurement Program

In late 1967, a 400 ft., fully instrumented meteorological tower was placed in operation at the DNPS site. Actual data collected at levels from 35 ft. above ground to 400 ft. above ground has verified favorable atmospheric diffusion conditions exist at the site. Data obtained from the tower during the first year of operation was correlated hour for hour with atmospheric stability measurements taken at ANL and applied on a preliminary basis to calculations for Dresden reactors. Since ANL is not too distant (27 miles northeast), and located in similar terrain, the two locations are climatologically similar and joint use of data from the two sites is a valid technique.

Meteorological data used to model dispersion characteristics of gaseous emissions from GE-MO are based on data collected from 1971 through 1993 at the Dresden meteorological tower.

3.4.3.1 Diffusion Climatology

Hourly wind direction variability at the site shows that average direction range (angular change in direction) is 120 degrees in a 1 hr. period, for all wind speed conditions combined. During 0-3 mph wind speeds, the average range in direction is 100 degrees. Approximately 87% of the time when the wind speed is 0-3 mph (or 98.3% of all wind speeds) the wind direction range is 60 degrees or more which corresponds to a value of the diffusion parameter ($\sigma_\theta u_h$) of 20 degree-mph or 0.16 radian-meter per sec.

Environment surveys of the site and surrounding areas conducted by CECo, ANL, and the State of Illinois show that meteorological diffusion characteristics would cause a dispersion of small amounts of effluent emitted during normal operation to a degree such that these effluents have been undetectable off-site.

3.4.3.2 Wind Speed, Direction and Atmospheric Stability

At the 400 ft. meteorological tower on the adjacent DNPS site, wind speed, direction and persistence are measured at the 35 ft., 150 ft. and 300 ft. levels. In addition, temperature measurements are made at the same levels and dewpoint temperatures are recorded at these levels continuously. A weighing-bucket rain gage is used to measure precipitation. An example of winds at the site is shown in Figure 3-8[22], which is an annual wind rose for the 35 ft. level.

Dresden 1971 through 1974, 150 ft. wind data has been used to estimate dispersion rates and calculate radiation doses from GE-MO. Table 3-8 shows relative frequency of winds from a given direction by Pasquill stability classes. Variability of the 300 ft. wind direction is determined by computing standard deviation of the most recent 60 wind direction values (one value is reported each minute). The 300 ft. to 35 ft. differential temperature was used to determine the stability class. One year of wind data (1974) was used to prepare the table, with a data recovery rate of 85.0%.

Table 3-9 gives the frequency of each stability class and average wind speed at 150 ft. for that class, based on the 1974 data.

Table 3-8

JOINT FREQUENCY DISTRIBUTION OF PASQUILL

STABILITY CLASS AND WIND DIRECTION, DRESDEN

150-foot level

(percent of total observations)[a]

Class	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM	TOTAL	Number of Observations
A	0.08	0.01	0.03	0.01	0.04	0.04	0.04	0.05	0.19	0.15	0.24	0.21	0.27	0.09	0.05	0.07		1.58	118
B	0.46	0.20	0.11	0.24	0.50	0.52	0.46	0.60	0.71	0.75	1.38	0.83	0.75	0.48	0.35	0.60	0.08	9.03	673
C	0.56	0.35	0.16	0.27	0.68	0.82	0.64	0.87	0.98	1.48	0.98	0.78	0.60	0.60	0.43	0.38	0.04	10.63	792
D	2.70	2.63	2.54	2.74	3.41	3.05	3.29	4.44	6.27	6.01	4.31	3.76	4.87	5.25	4.20	2.74	0.09	62.30	4641
E	0.21	0.12	0.13	0.16	0.19	0.27	0.20	0.23	0.30	0.36	0.40	0.21	0.31	0.46	0.50	0.27	0.00	4.32	322
F	0.28	0.17	0.26	0.12	0.09	0.21	0.31	0.17	0.26	0.46	0.19	0.27	0.15	0.51	0.36	0.38	0.00	4.19	312
G	0.59	0.62	0.72	0.50	0.36	0.51	0.34	0.35	0.27	0.52	0.48	0.26	0.46	0.82	0.50	0.64	0.01	7.95	592
Total	4.88	4.10	3.95	4.04	5.27	5.42	5.28	6.71	8.98	9.73	7.98	6.32	7.41	8.21	6.39	5.08	0.22	100	7450

[a] 7450 valid observations

Source:
Joint wind speeds and frequency reported for the year 1974 at the Dresden Nuclear Power Station meteorological tower.

Table 3-9

STABILITY, FREQUENCY, AND WIND SPEED

Class	Frequency (%)	Wind Speed (mph)
A	1.58	7.7
B	9.03	8.8
C	10.63	9.8
D	62.30	12.8
E	4.32	12.6
F	4.19	13.6
G	7.95	13.4

	As Planned	As Operated
Stack Height	300 ft (91 m)	300 ft (91 m)
Discharge Volume	25,000 cfm	14,000 cfm

3.4.4 Atmospheric Diffusion Characteristics

A general discussion of techniques used in calculating atmospheric diffusion characteristics and the resulting off-site doses from normal operation of GE-MO is given in Appendix A-3. These same methods and characteristics have been applied to nearby Dresden reactors[23]. Application of these methods for GE-MO is described below and in Section 7.

Diffusion calculations are based on annual wind direction, frequency, and stability distribution around the stack. Exposures and concentrations are calculated for all areas off-site from the plant based on actual site meteorology, thus ensuring that points with the highest potential exposures and concentrations are identified. These calculations extend to distances of several miles from the site, providing a good profile of the distribution of the dose versus location and distance from the site.

The height of release of effluent is the physical stack height plus effluent rise due to momentum. No credit was taken for possible thermal buoyancy of the plume. The stack and ventilation system design characteristics used in the analysis are listed below.

Meteorological data used in calculating doses and concentrations from radioactive materials released via the stack are a combination of data gathered at the Dresden site and data taken at ANL. Wind speed and direction data taken at the Dresden site were used in the calculation. Atmospheric stability measurements taken at ANL were correlated hour for hour to determine joint wind frequency, stability and velocity distribution at the site.

Data obtained from the GE-MO/DNPS tower during the first year of operation was correlated and applied on a preliminary basis to calculations for the Dresden reactors. These meteorological data verified the validity of the earlier approach and indicated that application of site data to calculation of maximum effects from releases would reduce calculated effects. Since actual data gathered served to verify the approach which had been taken earlier, calculations were not repeated[24]. In summary, data collected from the meteorological tower at the Dresden site verifies predicted excellent atmospheric diffusion characteristics typical of the northern Illinois site.

3.4.4.1 Meteorological Diffusion Evaluation

Radiological effects of stack releases were evaluated at six points in the atmospheric diffusion spectrum, which should encompass conditions encountered at GE-MO. These are: (1) poor diffusion conditions caused by inversion (stable), at a wind speed of about 1 m/sec., typical of warm nights; (2) very stable and moderately stable conditions; (3) better diffusion conditions, typical of daytime, represented by neutral and unstable (lapse) diffusion, both at wind speeds of 1 m/sec. and 5 m/sec. Atmospheric diffusion methods reported by Watson and Gamertsfelder[25] and calculations for the site are described in Appendix A.

3.5 SURFACE HYDROLOGY

3.5.1 Surface Features and Drainage Patterns

GE-MO is located in the Illinois River Drainage basin, just south of the DNPS in eastern Grundy County, Illinois (Figure 1-1). The Kankakee River is 0.5 miles east of the site, flowing north until it meets the Des Plaines River 2 miles northeast of the site.

The two rivers join to form the Illinois River which flows west and south about 270 miles to the Mississippi. The GE-MO site is on a relatively high area about 30 ft. above normal pool level in the Kankakee River and between the flood plains of the two rivers.

The Illinois River and its tributaries are the primary surface water resources near the site. The Illinois and Des Plaines Rivers form part of the Illinois waterway which is a series of eight navigable pools (with the headwaters above a lock and dam) extending 327.2 miles from its confluence with the Mississippi River at Grafton, Illinois to the Chicago River outlet at Lake Michigan. The Illinois River is the stretch of the waterway from the confluence of the Kankakee and Des Plaines Rivers to the Mississippi River. The stretch of the Illinois River north of the site is part of the Dresden Island Pool of the waterway and includes the Dresden Island Lock and Dam which is almost due north of the site.

The Illinois River and tributaries drain an area of 32,081 square miles. The river is unique in the sense that during dry weather (low flow) its headwaters are essentially treated liquid wastes from about 5.5 million people and various industries in the metropolitan Chicago area mixed with water diverted from Lake Michigan.

Approximately 1.5 miles southeast of the GE-MO site, CECo has constructed a 1,275 acre cooling lake for DNPS. The intake/discharge flumes are located along the east boundary of the GE tract. The lake is confined by an encircling earth dam (or berm) with the top of the dam at an elevation of 527 ft. The elevation of the cooling lake is approximately 522 ft. No recreational use of this lake is planned.

A series of small marshes and ponds, primarily located in the Goose Lake Prairie Preserve, comprise the remaining surface water of the area. The ponds are approximately 1.5 miles southwest of the GE-MO boundary.

3.5.1.1 Stream Flows

Stream flows on the Illinois Waterway fluctuate significantly due to seasonal effects and water flow regulation by means of Lake Michigan diversion and the lock-and-dam system. For example, on September 20, 1971, flows in the Dresden Pool dropped to 2,400 cfs from about 17,000 cfs on the preceding day. Average flow rate over the period 1921 to 1945 measured at Marseilles (20 miles downstream of the Dresden Pool) was 12,050 cfs (5,400,000 gpm). A 7 day 10 year low flow of 3,300 cfs was determined from data collected from 1940 to 1965 at Marseilles. A maximum flow of 93,900 cfs occurred at Marseilles in April of 1957. The flow of the Illinois River at Marseilles is greater than 3,000 cfs 98% of the time. The average flow of the Illinois River (1920-1963) at Dresden Island Lock and Dam was approximately 10,900 cfs.

The normal pool elevation in the Illinois River, controlled at the Dresden Island Lock and Dam, is 505 ft., with a maximum historical flood elevation of 506.4 ft. (1957). The estimated maximum flood elevation is 520 ft.; the GE-MO site elevation is higher than 532 ft. Spillway capacity at the Dresden Island Lock and Dam is well in excess of the estimated maximum instantaneous flow of the Illinois River (1,000,000 cfs, based on the assumption that maximum flows for all contributory streams occur simultaneously). The site elevation is well above the valley storage upstream from the dam.

Compared to the Illinois River, the Kankakee River is a relatively small river, with an average flow rate of 3,810 cfs (1,710,000 gpm), a minimum of 204 cfs (91,600 gpm), and a maximum of 75,900 cfs (measured at Wilmington, Illinois).

3.5.2 Site Flood Potential

The highest flood of record in the region occurred in 1957 and involved flows of less than 100,000 cfs, and created far below the 532 ft. minimum elevation of the GE-MO site as referenced to mean sea level. A study has been performed to develop rating curves for discharges of up to 600,000 cfs where the water level would rise to less than 520 ft. or more than 10 ft. below the GE-MO site. This study is summarized in Appendix A.6.

There are no other credible flood situations affecting GE-MO.

3.5.3 Surface Water Quality

Agricultural activity, boat traffic, and dredging have increased the Illinois River silt load over the past years and keep it in a continuously turbid condition. Water quality data collected at Morris, Illinois, including temperature and dissolved oxygen values, are presented in Table 3-10.

The Kankakee is usually several degrees cooler than the Illinois (see Table 3-11) and is not disturbed by barge traffic or dredging, as is the Illinois. These are probably the major factors for the existence of a more diverse fish population in the Kankakee than in the Illinois. Water quality of the Kankakee is not spectacularly better than that of the Illinois, however, and in some aspects is even poorer (compare Table 3-10 and Table 3-11) based on data from the sampling station on the Kankakee I-55 bridge.

Table 3-10

CHARACTERISTICS OF THE ILLINOIS RIVER AT MORRIS, ILLINOIS[a]

PARAMETER	1957 - 1971 Range	Average	1990 - 1993 Range	Average
Water Temperature (°C)	1.1 - 29.4	15.6	0.7 - 26.9	13.3
Turbidity (mg/l)	16 - 330	67	0.3 - 150.0	24.6
Dissolved Oxygen (mg/l)	N/A	N/A	6.1 - 14.2	10.0
Alkalinity (mg/l)	96 - 208	174	104 - 206	160
Hardness (mg/l)	144 - 388	283	201 - 347	273
Total Suspended Solids (mg/l)	N/A	N/A	412 - 580	447.5
Chloride (mg/l)	23 - 162	58	42 - 110	67
Sulfate (mg/l)	11 - 125	48	51 - 125	75
Nitrite & Nitrate (mg/l) as NO_3	0 - 35	6	2.60 - 7.80	4.64
Ammonia (mg/l) as N	0 - 11	3.9	0.05 - 0.80	0.31
Total P (mg/l) as PO_4	0.1 - 37.0	3.8	0.22 - 0.57	0.35
pH	7.2 - 8.2	7.6	6.1 - 13.7	7.60
Fluoride (mg/l)	0.4 - 2.1	0.9	0.22 - 0.54	0.33
Dissolved Iron (μg/l)	0 - 500	100	23 - 5K	61
Specific Conductivity (μmhos)	410 - 1050	700	540 - 933	729
Fecal Coliform/100 ml	10 - 2000	977	60 - 4900	1094
Totals Dissolved Solids (mg/l)	250 - 670	448	332 - 927	448

[a] Compiled from Water Quality Network, 1971 and 1993, Illinois EPA

Table 3-11

CHARACTERISTICS OF THE KANKAKEE RIVER AT WILMINGTON, ILLINOIS[a]

PARAMETER	1957 - 1971 Range	Average	1990 - 1993 Range	Average
Water Temperature (°C)	0.6 - 30	13.9	0.7 - 26.0	13.1
Turbidity (mg/l)	1 - 400	58	2.5 - 210.0	29.0
Dissolved Oxygen (mg/l)	5.4 - 14.6	10.1	5.0 - 13.0	9.4
Alkalinity (mg/l)	116 - 220	178	104 - 228	184
Hardness (mg/l)	116 - 576	308	208 - 382	307
Total Suspended Solids (mg/l)	N/A	N/A	7 - 188	42.0
Chloride (mg/l)	9 - 56	21	17 - 33	24.3
Sulfate (mg/l)	20 - 152	78	35 - 123	82.3
Nitrite & Nitrate (mg/l) as NO_3	0 - 24	6	0.5 - 8.4	4.80
Ammonia (mg/l) as N	0 - 10.1	1.0	0.01 - 0.20	0.07
Total P (mg/l) as PO_4	0.0 - 10.0	1.1	0.04 - 0.39	0.12
pH	7.1 - 8.8	7.9	6.9 - 9.1	7.80
Fluoride (mg/l)	0.0 - 0.4	0.2	0.11 - 0.23	0.18
Dissolved Iron (μg/l)	0.0 - 12.0	1.1	5 - 5K	56.6
Specific Conductivity (μmhos)	N/A	N/A	432 - 773	615
Fecal Coliform/100 ml	10 - 800,000	31,848	10 - 2,750	136.6
Totals Dissolved Solids (mg/l)	170 - 530	362	N/A	N/A

[a] Compiled from Water Quality Network, 1971 and 1993, Illinois Environmental Protection Agency

3.6 SUBSURFACE HYDROLOGY

3.6.1 Regional and Area Characteristics

Groundwater in northeastern Illinois is drawn from four aquifer systems:

a. Sand and gravel deposits in the glacial drift;

b. Shallow dolomite formations mainly of the Silurian age;

c. Cambrian-Ordovician aquifers of which the Ironton-Galesville dolomite and the Galena-St. Peter sandstones are the most productive formations; and

d. The Mt. Simon aquifer consisting of the sandstone of the Mt. Simon and lower Eau Claire formations of the Cambrian age.

In the vicinity of GE-MO, glacial drift thickness ranges from none, with outcropping bedrock, to at most a few feet of drift. There is no evidence of the Silurian dolomite. As a result, groundwater in the vicinity of the site is drawn from the Cambrian-Ordovician aquifer which is used almost exclusively as the groundwater supply for municipal and industrial use in the area.

Glacial drift in the area is underlain by the Pennsylvanian-Spoon formation sandstone or the Ordovician-Fort Atkinson limestone, or both. Beneath these formations and directly over the Cambrian-Ordovician aquifers is a layer of Ordovician-Maquoketa shale approximately 65 ft. thick. The top of the Cambrian-Ordovician aquifers at the site is approximately 100 to 150 ft. beneath the surface and the piezometric surface of the Cambrian-Ordovician aquifers is about 100 ft. further down. The major source of near-surface groundwater in the area is from rainfall which seeps down through the alluvial overburden and upper strata of weathered and fractured rock to collect over relatively impermeable areas (clay seams, underlying shale).

3.6.1.1 Water Quality

Water from the glacial drift and Silurian dolomite aquifers ranges in hardness from 100 to 1,000 ppm, although the majority of samples analyzed for hardness ranged from 100 to 450 ppm. Temperatures range from 46 °F to 54 °F (Suter, et al., 1959). Hardness of water from the Cambrian-Ordovician aquifers ranges from 260 to 880 ppm. Both hardness and temperature increase eastward, and water quality noticeably deteriorates south of the Illinois River (Suter, et al., 1959). Mt. Simon waters are of poor quality in this region because of their brackish nature. This characteristic increases rapidly eastward across northeastern Illinois.

3.6.2 Site Characteristics

Geological structure under the GE-MO site is typical of the region, presenting no anomaly significant in hydrological considerations. In general, the upper 10 to 20 ft. of Fort Atkinson Limestone has high but variable permeabilities with permeabilities decreasing to less than 100 ft. per year near the base of the formation.

Water-level measurements from piezometers installed in the Fort Atkinson, Scales, and Galena formations indicate that the Scales Shale acts as an effective aquitard between the Fort Atkinson Limestone and the dolomite of the Galena group.

The historical record of groundwater variations within the Galena Dolomite (the upper unit in the Cambrian-Ordovician aquifer) shows a cone of depression has developed near Joliet and that the piezometric surface has dropped over 100 ft. from 1915 to 1958 to an elevation of about 400 ft. above mean sea level.

While the regional piezometric surface of the Galena at the present time is unknown, the number of wells which penetrate this aquifer has increased since 1958 and it is probable the surface has further dropped. During drilling of the water supple well on the GE-MO site in 1968, the static water level within the Galena Dolomite was at about 370 ft. while the static water level of the Cambrian-Ordovician aquifer as a unit was at about 395 ft. The piezometric level in the Fort Atkinson Limestone parallels the ground surface, is 3 to 5 ft. deep and reacts rapidly to

precipitation. The piezometric level in the Scales Shale is also near the ground surface, but reacts slowly to precipitation.

During LAW Vault construction, serious groundwater intrusion problems were encountered. The results of the investigation[26] indicated a complex groundwater system with several potential sources:

a. direct percolation from rainfall and runoff;

b. lateral seepage and flow from perched or confining zones in response to percolation from rainfall; and

c. lateral flow along joints, faults or fractured rocks.

3.6.2.1 On-Site Well

There is a single deep well on site into the Cambrian-Ordovician aquifer, and is equipped with a 100 gpm submersible vertical turbine pump. Principal use of water from this source is potable and sanitary water purposes with some water use (up to 1,400 gallons per day) for basin makeup. Well water could also be used for fire fighting. Characteristics of water from this well are contained in Table 3-12 & 3-13.

There is no release of liquids from GE-MO to potable ground water since site structures do not penetrate any principal aquifers. Even a major rupture of concrete basin walls could impact only on local on-site sample wells and would not penetrate to the Cambrian-Ordovician strata. (See Sec 8 and B.12, Ground Water Investigations by Dames & Moore, dated August 1977.)

3.6.3 Groundwater Investigation - 1977

As a part of a study of potential expansion of GE-MO facilities, a groundwater investigation was conducted in the spring and summer of 1977 by Dames and Moore[27]. The study included:

a. A review of previous site investigations
b. A review of literature
c. Evaluation of site boring data, groundwater level data, and pressure testing results
d. Evaluation of groundwater regime in the site area
e. Evaluation of groundwater movement and use at the site and in the region

Conclusions from this study (August 1977) are consistent with past studies, showing good availability of water for plant operations with negligible impact on aquifer performance. The more detailed analysis of permeabilities performed under this study further emphasize the suitability of the site for basin storage of irradiated fuel.

Table 3-12

WATER ANALYSIS - MORRIS OPERATION WELL

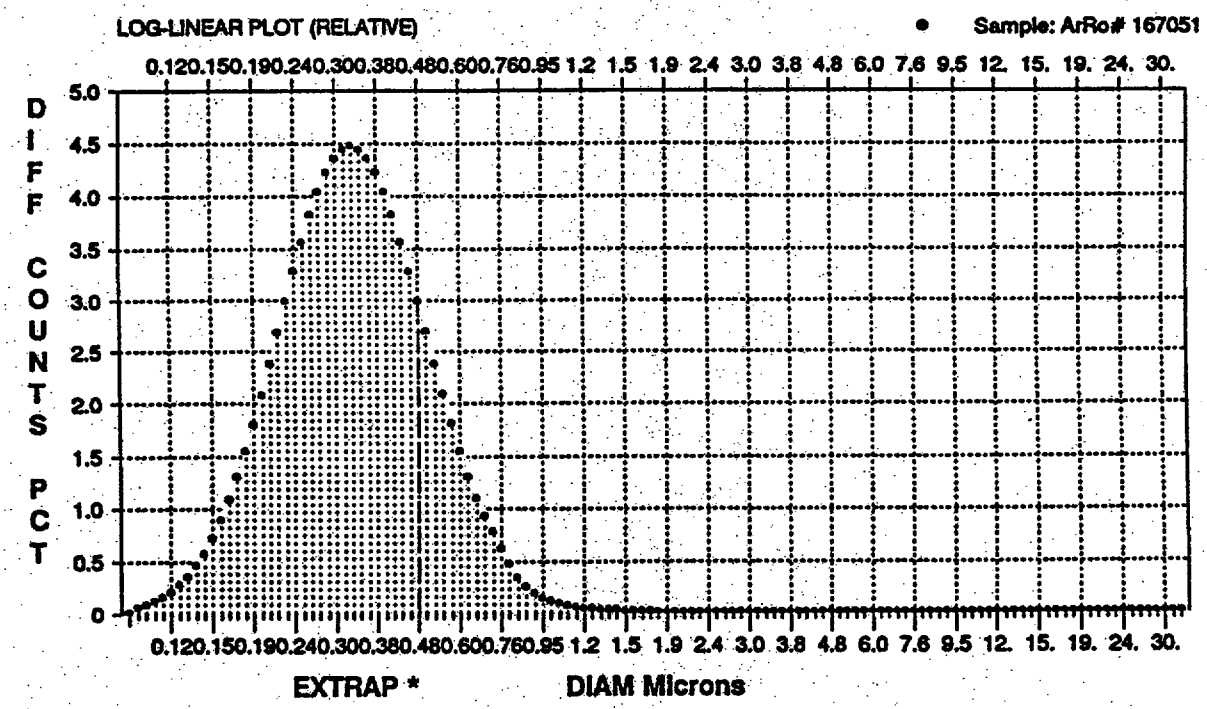
Material	Parts per Million
Chloride	100 ± 10
Nitrate	4.2
Iron	< 0.4
Silica (as Silicon)	5
Sulfate	225
Calcium	58
Magnesium	25
Sodium	159
Phosphate	None Detected
Manganese	< 0.1
Sulfide	None Detected[a]
Bicarbonate	295
Potassium	16
Tin	3
CO_2	11.6
pH	8.0
Conductivity	1.1×10^4 mhos/cm
Dissolved Solids	706
Total Suspended Solids	5
Turbidity	0.3[b]
Total Organic Carbon	2.8

[a] As much as 2.2 ppm H_2S (expressed as $CaCO_3$ equivalents) was present in 1968.

[b] NTU Units

Table 3-13

MICROSCOPIC PARTICLE SIZE DISTRIBUTION – MORRIS OPERATION WELL WATER



Geometric Mean Size:	0.333 um	– PERCENTILES –
Geom. Std Deviation:	1.517 um	0.100% Counts above 1.745 um
Geom. Skewness:	0.002	1.000% Counts above 0.907 um
Geom. Coeff Variation:	455.6	6.000% Counts above 0.633 um
		22.00% Counts above 0.456 um
Arithmetic Mean Size:	0.364 um	50.00% Counts above 0.331 um
Median Size:	0.324 um	78.00% Counts above 0.241 um
Mode Size:	0.330 um	94.00% Counts above 0.175 um
Kurtosis:	24.791	99.00% Counts above 0.130 um
Arith Std Deviation	0.176 um	99.90% Counts above 0.104 um

Source: Analysis by ARRO Laboratories, Inc., Joliet, Illinois.

3.7 GEOLOGY AND SEISMOLOGY

3.7.1 Geologic Studies

Geologic studies of the site have been performed by Dames & Moore. Studies were also performed by these consultants for DNPS and for the MFRP facilities. These studies are listed in Table 3-14. Reports of recent investigations, unique to fuel storage at GE-MO, are noted in Table 3-14 and are contained in the microfiche packet (Appendix B).

Table 3-14
MORRIS OPERATION SITE INVESTIGATIONS

M Report, Site Evaluation Study, Phase I - Part 1, Proposed Dresden Unit 2, Grundy County, Illinois, For General Electric Company Dated: April 13, 1965

M Report of Foundation Investigation, Proposed FRO Plant Project, Near Morris, Grundy County, Illinois, For General Electric Company Dated: December 13, 1967

Report, Subsurface Water Investigation, FRO Plant Project, Morris, Illinois, Fluor P.O. 4204-0-014, For General Electric Company Dated: February 25, 1970

Report of Drainage Well Pumping Tests, FRO Plant Project, Midwest Fuel Recovery Plant, Near Morris, Illinois, For General Electric Company Dated: January 11, 1971

M Report, Fault Investigation, Midwest Fuel Reprocessing Plant, Near Morris, Illinois, For General Electric Company Dated: October 1, 1974

M Report, Geological and Ground Water Investigation, Proposed Spent Fuel Storage Facility, Near Morris, Illinois, For General Electric Company Dated: September 3, 1975

M Letter Report, Evaluation of Foundation Recommendations, Project IV - Fuel Storage Capacity Expansion, Near Morris, Illinois, For General Electric Company Dated: May 12, 1977

M Report, Geophysical Investigations, Project IV - Fuel Storage Capacity Expansion, Near Morris, Illinois, For General Electric Company Dated: June 10. 1977

M Report, Ground Water Investigations, Project IV - Fuel Storage Capacity Expansion, Near Morris, Illinois, For General Electric Company Dated: June 17, 1977

Report, "Proposed Approach for Evaluate the Adequacy of Ground Water Monitoring System at Nuclear Spent Fuel Storage Plant - Morris, Illinois, Grundy County for General Electric Company" Dated: February 10, 1993

Report, "Groundwater Modeling and Specifications for Monitoring Wells at Morris, Illinois Operation for General Electric Company" Dated: August 18, 1993

Report, "Preliminary Estimates of Evaporation From Fuel Storage Basin at Morris, Illinois Facility for General Electric Company" Dated: September 29, 1993

Report, "Transport Modeling for Accidentally Released Water from Spent Fuel Storage Basin at Morris, Illinois Facility of General Electric Company" Dated: October 26, 1993

Report, "Groundwater Monitoring Well Network Summary and Installation Report – Morris, Illinois Facility for General Electric Company" Dated: January 28, 1994

Report, "Well No. DM-8, Groundwater Monitoring Well Network Installation Report, Morris, IL Facility, General Electric Company" Dated: January 4, 1995

M – Microfiche in Appendix B
Source : Dames & Moore, Consultants – Environmental and Earth Sciences, Park Ridge, Illinois

3.7.2 Regional and Tract Geology

The GE tract is situated in the Morris Basin, a relatively low area of slight topographic relief. Elevations range from 532 ft. on the site to about 500 ft. at the Illinois River bottom. The general appearance varies from flat to very gently rolling with slopes greater than 3% being rare. Surface topography is characterized by very shallow topsoils, with frequent outcroppings of bedrock. Dresden Heights is the dominant topographical feature and is located on the north side of the Des Plaines River about 1.5 miles northeast of the tract. Elevation of these bluffs is 630 ft. There are vestiges of abandoned strip mines in many parts of the area.

Regional structures in north and northeastern Illinois trend northwesterly and are characterized by asymmetrical folds with steep southwestern limbs and by vertical faults and joints that trend northwesterly. Fracture sets trending northeasterly also occur. Major regional geologic structures around the tract are shown in Figure 3-9.

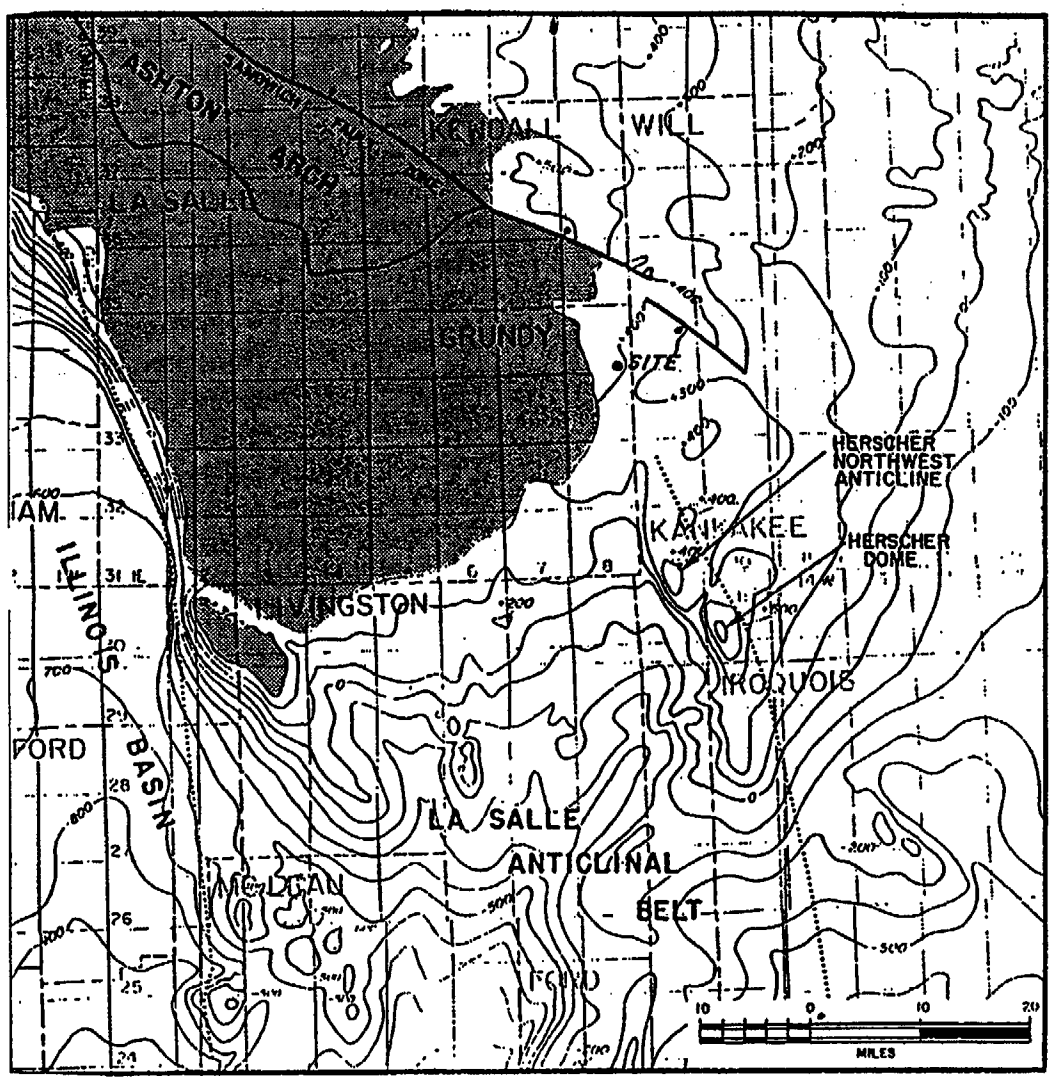
A major structural zone of the underlying Illinois Basin is the LaSalle Anticlinal Belt, a north-northwesterly trending band of en echelon folds. Within the northern two-thirds of the basin this folded zone separates the shallow eastern shelf of the basin from the larger and deeper western shelf. The rocks of the eastern shelf - the area of the GE tract - are nearly flat-lying. Initial deformation along the LaSalle Anticlinal Belt began in the northern end during the post-Mississippian, pre-Pennsylvanian period, and migrated southward with time[28].

Cambrian and Lower Ordovician rocks are exposed along the trend of the Ashton Arch, an anticline that merges with the northern portion of the LaSalle Anticlinal Belt. Uplift along the Ashton Arch was at least post-Silurian, probably occurring in the same period as along the LaSalle Anticlinal Belt[29].

The Ashton Arch is bounded to the north by the Sandwich Fault Zone, trending west-northwest across northern Illinois to within 6 miles of the Morris site. It is mapped on the surface and subsurface for nearly 90 miles. The fault zone is essentially vertical, with the northeastern block downthrown a maximum of 900 ft. by the main fault, with numerous associated short faults near the northwestern end. The throw decreases toward the southeastern end of the zone and a scissors effect causes the southwestern block along a subsidiary fault to be downthrown more than 100 ft.[30]. Movements along the Sandwich Fault Zone are dated as post-Silurian, pre-Pleistocene, but major movements along the fault may have occurred when the LaSalle Anticlinal Belt was uplifted in post-Mississippian, pre-Pennsylvanian time[31].

The attitude of folds and faults in the region indicate that compressive forces acted along northeast-southwest lines during deformation in the Paleozoic Era. Extension fractures from parallel to maximum compression and shear fractures are symmetrically inclined (angles less than 45 degrees) about the compressive force axis. Such fracturing has been mapped at the DNPS site by the Illinois State Geological Survey[32].

The locations of these faults and others between the LaSalle Anticlinal Belt and the Sandwich Fault provide strong evidence of direct relationship between faults mapped adjacent to the Morris site and regional structures[33].



EXPLANATION

STRUCTURE CONTOURS ON TOP OF THE GALENA GROUP CONTOUR INTERVAL 100 FEET; DATUM SEA LEVEL

MAQUOKETA ABSENT, TOP OF GALENA ERODED

FAULT, DOWNTHROWN SIDE INDICATED

APPROXIMATE BOUNDARY OF THE LA SALLE ANTICLINAL BELT

REFERENCE: MODIFIED FROM

BRISTOL, H.M. AND BUSCHBACH, T.C. 1973, ORDOVICIAN GALENA GROUP (TRENTON) OF ILLINOIS-STRUCTURE AND OIL FIELDS; ILLINOIS STATE GEOLOGICAL SURVEY, ILL. PET. 99, PLATE 1.

FROM: "REPORT OF GEOLOGIC INVESTIGATIONS - PROJECT IV - STORAGE CAPACITY EXPANSION NEAR MORRIS, ILLINOIS" AUGUST 1977; DAMES & MOORE, CHICAGO, ILLINOIS (APPENDIX B, 14)

Figure 3-9. Major Regional Geologic Structures

3.7.2.1 Site Geology

Stratigraphy was determined by test borings and trenching performed during several geologic studies[34] of the area, with the most recent study completed in August 1977. The spatial relationships found at the site are complex, but can be explained in terms of glacial erosion, deposition, and post-glacial erosion. The generalized stratigraphic column for GE-MO (Figure 3-10) consists of an upper layer Spoon Formation sandstone of varying thicknesses, underlain by Fort Atkinson Limestone about 46 ft. thick. Scales formation shale is beneath the limestone. The site is overlain with a thin topsoil. The Ordovician system has a thickness of about 1,000 ft., overlaying the Cambrian system. Brecciated rock is found in some cross sections, indicating ancient faulting.

Surface drainage is rather poor since the bedrock surface is undulating and entraps surface water. A perched water condition exists because of relatively impermeable limestone and shale underlying the site. This condition is encountered only a few feet below the surface (4 or 5 ft.). True groundwater occurs in the Cambrian-Ordovician aquifers at depths of about 120 ft. at GE-MO. Maximum frost penetration is about 4 ft. Clay is the known mineral deposit of value at the site, and this is limited to the shallow overburden.

3.7.3 Investigation of Faults

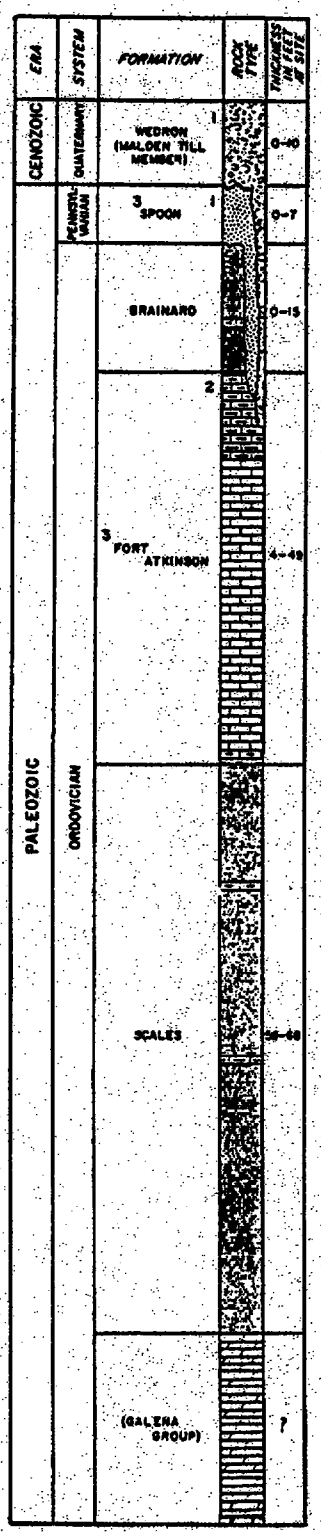
A northwest-trending fault passing southwest of the main building was originally identified by Dames & Moore from borings made for a foundation investigation in 1967. Another northwest-trending fault was inferred in 1971 during investigation of effectiveness of drainage wells but could not be otherwise confirmed.

The northwest-trending fault was studied by Dames & Moore in 1974, in more detail in 1975, and again in 1977.

The 1974 study identified the fault, showing it to have an offset of 35 to 40 ft. with the southwest side dropped in relation to the northeast side. It was concluded at the end of the 1974 study that the most probable time of faulting occurred between the late Ordovician and early Pennsylvanian periods. The 1975 study included a seismic refraction survey of the site and a site stratigraphic survey through use of test borings and trenching. Conclusions from the 1975 study placed the major movement of the fault contemporaneous or precontemporaneous with major development of the northern portion of the LaSalle Anticlinal Belt, which is generally accepted to be about 300,000 to 400,000 years ago.

3.7.3.1 1977 Fault Study

A geological investigation was conducted in the spring and summer of 1977 to determine structural and stratigraphic relationships of the northwest-trending fault zone and to substantiate age of faulting at the site.



EXPLANATION

- TILL
- SANDSTONE
- SHALE
- LIMESTONE
- ARGILLACEOUS AND SILTY LIMESTONE WITH THIN SHALE INTERBEDS
- DOLOMITE

1. NOTE

THE MALDEN TILL MEMBER OF THE WEDRON FORMATION OCCURS AS ISOLATED BODIES OVERLYING THE SPOON, BRAINARD, FORT ATKINSON, AND SCALES FORMATION. THE SPOON FORMATION ALSO OCCURS IN ISOLATED PATCHES AND WAS OBSERVED TO UNCONFORMABLY OVERLIE THE FORT ATKINSON FORMATION. HOWEVER, IN AREAS AROUND THE IMMEDIATE SITE THE SPOON MAY ALSO UNCONFORMABLY OVERLIE THE BRAINARD AND SCALES FORMATIONS.

2. NOTE

SOLUTION WIDENED JOINTS WITH ASSOCIATED CLAYEY SILT, SILTY CLAY, FINE GRAINED SANDSTONE AND SILTSTONE FILL MATERIAL ARE FOUND IN THE FORT ATKINSON FORMATION.

3. NOTE

THE SPOON FORMATION HAS PREVIOUSLY BEEN CALLED THE POTTSVILLE SANDSTONE. THE FORT ATKINSON LIMESTONE HAS PREVIOUSLY BEEN CALLED THE DIVINE LIMESTONE. THE POTTSVILLE SANDSTONE AND DIVINE LIMESTONE ARE NO LONGER RECONIZED AS ACCEPTABLE USAGE BY THE ILLINOIS STATE GEOLOGICAL SURVEY.

FROM: " REPORT OF GEOLOGIC INVESTIGATIONS - PROJECT IV - STORAGE CAPACITY EXPANSION NEAR MORRIS, ILLINOIS", AUGUST 1977 DAMES & MOORE, CHICAGO, ILLINOIS (APPENDIX B, 14)

Figure 3-10. Generalized Stratigraphic Column for the GE Morris Operation Site.

Field investigations included soil and rock core drilling, borehole water pressure testing, piezometer installation, geophysical surveys, trenching across the fault zone, and geological mapping of the trenches.

The investigation showed multiple northwest-trending faults are present in an en echelon pattern instead of a single fault as previously interpreted. Furthermore, it was interpreted that cross faults trend northeasterly and also occur in an en echelon pattern.

Relative movement of the northwest-trending fault zone is down-to-the southwest. Several faults exposed in trenches have downward displacement to the northeast, however. Most individual faults also are displaced down-to-the-southwest. The faults probably converge with depth creating step-like extensional blocks that have variable displacements relative to adjacent blocks as well as rotational displacements. The variability of displacements of fault blocks is characteristic of en echelon gravity faults produced by antithetic tensional forces. The excavations provided comprehensive information regarding detailed structural relationships of the fault zone including displacement of faults, orientation of faults and joints, and continuity of fault blocks. Faults mapped within the trenches correlated well with fracture zones measured in the angle borings (Figure 3-11; note shaded areas).

3.7.3.2 Conclusions - 1977 Study

Evidence of the Spoon Formation sandstone directly overlying a fault and fault block of Fort Atkinson Limestone conclusively dates the fault as having occurred no later than pre-early or early Desmoisian. Presence of clay-limestone rubble as a colluvial wedge-shaped deposit along the fault block supports a probable post-Chesterian age of faulting. Age of faulting (post-Chesterian/early-Desmoisian) at the site is supported further by the regional geologic history. Initial deformation along the LaSalle Anticlinal Belt and major movements of the Sandwich Fault occurrence during post-Mississippian/pre-Pennsylvanian time[35] is equivalent to the age of site deformation.

Continued uplift within the area occurred after Pennsylvanian time but this renewed activity was of less magnitude[36] and may be partially responsible for warping or increased inclination of bedding planes within the Spoon Formation during its unlithified, unconsolidated state. No displacement of offset is found within beds of the Spoon Formation at the site.

Criteria for faulting, as defined at 10 CFR 100, Appendix A, require that a fault has not moved in the last 35,000 years or has no history of recurrent movement in the last 500,000 years. The statigraphic evidence found throughout the site, both in this and previous investigations, indicates a pre-Spoon deposition age for faulting. Relationships observed in Trench CT-7 (Appendix B.14) provide substantiated evidence that faulting occurred in post-Chesterian to early Desmoisian time (approximately 280 million years before the present). Therefore, faulting at the site is not capable.

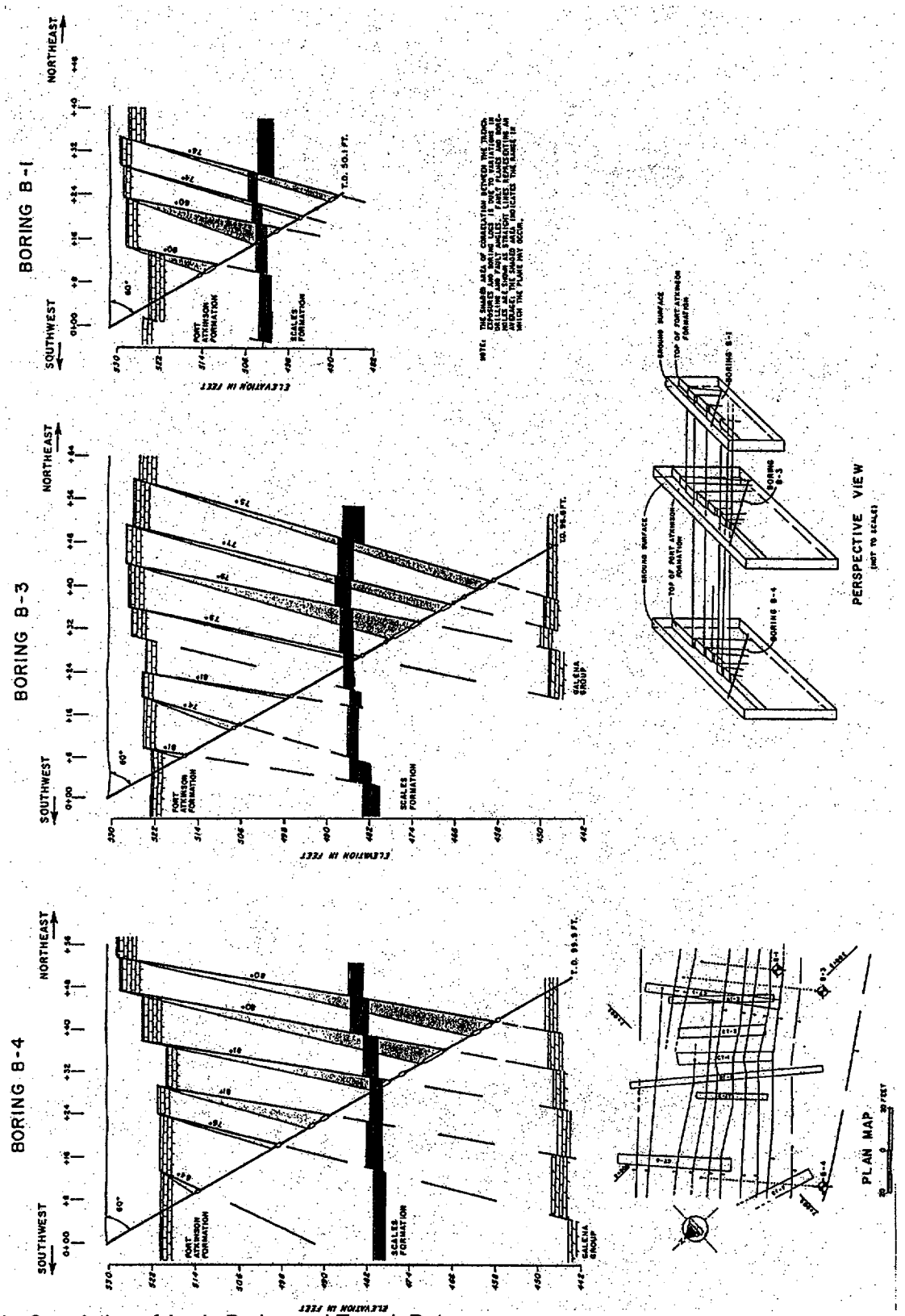


Figure 3-11. Correlation of Angle Boring and Trench Data.

3.7.4 Earthquake and Seismicity

Historical data shows seismic events in the vicinity of the site are relatively infrequent and are characterized by fairly low intensities and magnitudes.

3.7.4.1 Engineering Properties of Materials Underlying the Site

Static and dynamic properties of materials underlying the site have been summarized in a report of a foundation investigation[37]. In general, underlying materials have been found very suitable for heavy facility construction.

3.7.4.2 Seismic History

Several earthquakes of intensity MM V (Modified Mercalli (MM) scale) or higher have been listed as having epicenters in Illinois, including four of intensity MM VII. Only one significant earthquake has been centered within 50 miles of the site (intensity MM V or greater). It occurred on January 2, 1912, and was centered about 15 miles northwest of the site. It is described in "Earthquake History of the United States" (1973) as having an intensity of MM VI at Aurora, Freeport, Morris and Yorkville, and of V at Chicago. The shock was felt at Milwaukee and Madison, Wisconsin, and in Iowa, Indiana, and Fulton County, Kentucky. An intensity of MM VI was probably felt in the vicinity of the site as a result of this earthquake.

On September 15, 1972, an earthquake of epicentral intensity MM VI was centered about 55 miles northwest of the site. Press reports indicate the shock caused cracked plaster at Morris and Ottawa and a broken window at Rockton.

Only one earthquake of intensity MM VII has been centered within 100 miles of the site area. It occurred on May 26, 1909, about 88 miles NW of the site and according to "Earthquake History of the U.S.," it was felt from Missouri to Michigan and Minnesota to Indiana. A shock of intensity MM VII was noted over a considerable area from Bloomington, Illinois, to Platteville, Wisconsin[38].

The maximum intensity X-XII (MM) New Madrid, Missouri, earthquakes of 1811-1812 whose epicenters were approximately 350 miles to the south probably resulted in an intensity no greater than MM VI in the site area[39].

Another distant shock felt over a large area during historical times was the Charleston, South Carolina, earthquake of August 31, 1886. This shock may have been felt with about intensity MM III in the site area though it was reportedly not felt at Joliet and Kankakee.

The seismic risk map (Figure 3-12) of the conterminous United States was prepared by a group of research geophysicists headed by Dr. S. F. Algermissen of the United States Coast and Geodetic Survey and issued in January 1969. The site area lies well within zone 1 where minor earthquake damage can be expected. According to this map, zone 1 corresponds to intensities V and VI on the modified Mercalli (MM) scale.

MM VI seems to be the greatest intensity experienced historically in the site area. This was the result of the 1912 earthquake which was centered approximately 15 miles from the site, and may also have been the result of the 1811-1812 New Madrid, Missouri, earthquakes. MM VI, with its corresponding acceleration (according to Newmann's curve) of 0.01 G may be reasonably expected to occur again within the lifetime of the facility.

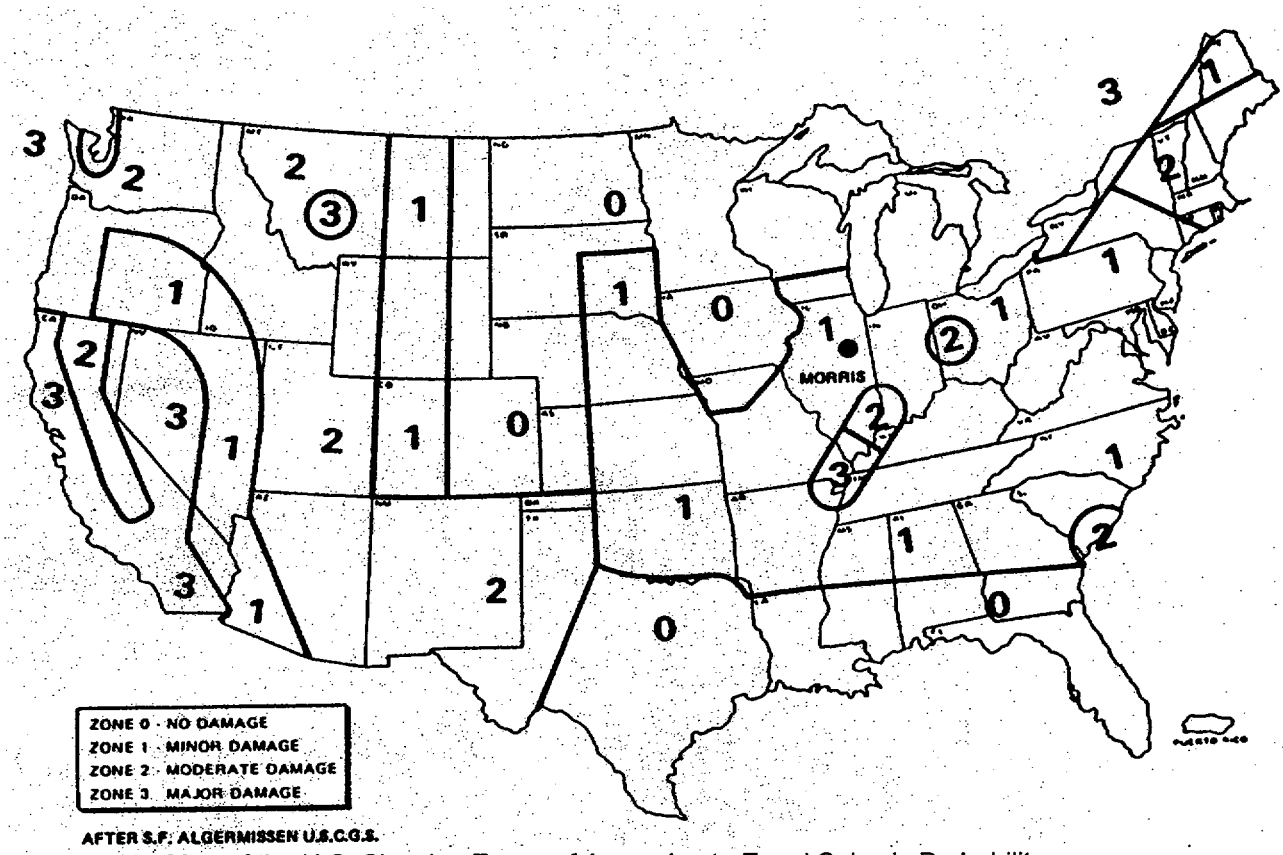


Figure 3-12. Map of the U.S. Showing Zones of Approximate Equal Seismic Probability.

3.7.5 Earthquake Design Basis

The design earthquake basis for the basin was a horizontal ground motion of 0.1 G. The basin structure and fuel storage system are designed to withstand the design basis earthquake without damage to structures or components essential to the integrity of stored fuel or fuel being moved in the normal process of storing or shipping fuel. The design earthquake is defined as a seismic event that has a reasonable probability of occurrence during the life of the facility, based on studies of seismic history and geology. A maximum earthquake with ground accelerations of 0.2 G is also considered in the seismic analyses. The design bases are discussed in Section 4.

3.8 TRANSPORTATION OF IRRADIATED FUEL

Irradiated fuel is received by truck or rail at GE-MO in casks certified to comply with applicable U.S. Nuclear Regulatory Commission (NRC) regulations[40]. Typical shipping casks are discussed in Section 1.3.

As of the end of 1989, 737 shipments of fuel had been completed, moving about 750 tonnes - heavy metal in 3,450 fuel bundles. Shipments to GE-MO have been completed without highway or rail accidents over about 744,300 miles.

Environmental impact of these transportation operations has been negligible, thus supporting conclusions of various studies and analyses[41,42].

Nonradiological and radiological impacts of transportation are analyzed in the literature[43]. Environmental impact assessments of GE-MO by the NRC staff have also found no significant environmental impact from spent fuel transport[44,45].

3.9 SUMMARY OF SITE CONDITIONS AFFECTING FACILITY OPERATING REQUIREMENTS

Irradiated fuel storage operations have been conducted at GE-MO since January 1972 when the first shipment of irradiated fuel was received under Materials License No. SNM-1265, Docket 70-1308, issued December 1971. Throughout this period of operating experience and during on-going environmental studies and monitoring programs, no condition has been found to detract from the desirability of this site as a fuel storage location. Factors significant in selection of design bases for GE-MO follow.

3.9.1 Meteorology

The climate at the site offers no severe extremes except tornadoes. Analysis of tornado activity, including official and unofficial records, indicates a frequency close to the average for all states east of the Rocky Mountains.

Site topography introduces little perturbation in diffusion calculations; only the 630 ft. elevation of Dresden Heights, about 1.5 miles north of the GE-MO stack is of concern in selecting stack design bases. Local fog conditions are involved in dispersion considerations. Diffusion climatology and characteristics have been firmly established and confirmed by the meteorological measurement program.

3.9.2 Hydrology

Site surface hydrology offers no characteristics significant to selection of design bases (except for usual consideration of natural drainage pathways, etc.). Subsurface hydrology shows excellent separation between upper strata and deeper aquifers that provide water for municipal and industrial use.

Intrusion of groundwater was of concern during construction. These flows indicate a complex near-surface groundwater system that becomes significant because of localized fracturing induced during construction.

3.9.3 Geology and Seismology

The site is located in a stable area which has experienced historically low seismic activity. The existing construction is founded on bedrock of Ordovician (Paleozoic) age. Design of the facility and its fuel storage equipment for horizontal ground motion of 0.10 G is considered conservative.

3.10 REFERENCES

1. See Appendix A.1 for document list.

2. State of Illinois, Bureau of the Budget, Illinois Population Projections (Revised 1977), Springfield, September 1977.

3. State of Indiana, State Board of Health, Indiana County Population-Projections, Indianapolis, 1978.

4. Northeastern Illinois Planning Commission, Regional Data Report, Chicago, June 1978.

5. The 5% growth in the 0 - 5 mile area was developed from the assumption that farmland will not experience growth (urbanization) except in a few selected areas. This growth was estimated and the overall area growth integrated. Most people working in local industries live in the Western Joliet and Morris areas; there has been little growth in smaller communities.

6. The USNRC staff reported an adjusted estimated 1980 population for the area within the 50 mile radius of about 9,169,337 (Environmental Impact Appraisal, Docket 70-1308m NR-FM-002).

7. During research for these data, differences were noted between (for example) the Northeastern Illinois Planning Commission data and Federal census figures. In general, however, the data appear mutually supportive, particularly at the county level.

8. Within 5 miles of the site the total school population is about 3,200.

9. Correctional institution (juvenile) at Channahon, 3 miles WNW. (Closed)

10. Climatography of the United States, No. 60-11, revised and reprinted June 1969.

11. H. E. Landsberg, "Climates of North America," World Survey of Climatology, Vol. 11, edited by Bryson, et al., Elsevier Scientific Publication Co. (1974)

12. S. S. Visher, Climatic Atlas of the United States, Harvard University Press, Cambridge (1966).

13. U.S. Department of Commerce, Climatography of the United States No. 86-9, "Decennial Census of United States Climate," for Illinois, Washington, D.C. (1964).

14. "Final Environmental Statement related to operation of the Midwest Fuel Recovery Plant by the General Electric Co.," Doc. 50-268, USAEC (December 1972).

15. Fluor Cooling Products Company, "Evaluated Weather Data for Cooling Equipment Design," Addendum No. 1, Winter and Summer Data, Santa Rose, CA (1964).

16. D. W. Phillips, et al., "The Climate of the Great Lakes Basin," Climatological Studies Number 20, Environment Canada, Toronto (1972).

17. J.L. Vogel, et al., "Fog Effects Resulting from Power Plant Cooling Lakes," Journal of Applied Meteorology. Vol. 14 (August 1975).

18. Final Environmental Statement related to the operation of Dresden Nuclear Power Station Units 2 and 3 by the Commonwealth Edison Co., Docket No. 50-237 and 50-249, AEC (November 1973).

19. Applicants Environmental Statement, Dresden Nuclear Power Station Unit 3, Commonwealth Edison Co., Docket No. 50-249 (July 1970).

20. Thom suggests an annual extreme-mile (fastest mile) wind speed of 82 mph for 30 ft. above ground and for a 100 yr. mean recurrence interval. Thom, H.C.S., "New Distributions of Extreme Winds in the United States, "Journal of the Structural Division,, Proc. ASCE, Vol. 94 No. St. 7 (1968) Applicants Environmental Report, Midwest Fuel Recovery Plant Morris, Illinois, June 1971.

21. Murray and Trettel, Inc. Consulting Meteorologist, Chicago, IL. Letter, Literski (M&T) to Eger (GE), September 23, 1976.

22. From Braidwood Station Environmental Report, Commonwealth Edison Co., Chicago, IL. Year of record: July 1971 - June 1972.

23. The application of these methods to the Dresden reactors and the description of the techniques used there can be found in Appendix A of the Final Safety Analysis Report for Dresden 2 and 3, Docket 50-237.

24. The description of the first year's data taken at the site can be found in Amendment No. 13, Question B-11, to the Dresden Unit No. 2 Final Safety Analysis Report, Docket 50-237.

25. E. C. Watson and C. C. Gamertsfelder, "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria," February 14, 1963, HW-SA-2809.

26. NEDO 10178-1, Water Intrusion Consideration, July 1971.

27. Dames & Moore report, "Ground-Water Investigations," (Appendix B.12).

28. Payne, 1940, page 7; and Eardley, 1962, page 45.

29. Willman and Templeton, 1951, page 123.

30. Bristol and Buschbach, 1973, Plate 1.

31. Willman and Templeton, 1952; also Bristol and Buschbach, 1971, Figure 3.

32. Ekblau, 1956; Dames & Moore, 1965.

33. Kempton, 1975.

34. See Table 3-14 for studies referenced in this section.

35. Payne, 1940; Willman and Templeton, 1951.

36. Willman and Templeton, 1951.

37. Dames & Moore, report dated December 1967 (Appendix B.2).

38. J. A. Udden prepared a report describing observations of this earthquake. He presents an isoseismal map for this earthquake and, according to his map, the site was in the area which experienced Rossi-Forel intensity VI (about V-VI on the modified Mercalli scale).

39. This intensity is based on an isoseismal map prepared by O. W. Nuttli and presented in the Bull. Seis. Soc. Am., Vol. 63, No. 1, 1973.

40. K. Eger, Operating Experience Report - Irradiated Fuel Storage at Morris Operation - January 1972 to December 1982, General Electric Company, (NEDO-20969B).

41. 10 CFR 51, Summary Table S-4, "Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water Cooled Nuclear Power Reactor," U.S. Nuclear Regulatory Commission, especially Note 4, "Although the environmental risk of radiological effects stemming from transportation accidents is currently incapable of being numerically quantified, the risk remains small regardless of whether it is being applied to a single reactor or a multireactor site."

42. Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, U.S. Atomic Energy Commission, December 1972 (WASH-1238); and U.S. Nuclear Regulatory Commission, April 1975 (Supplement 1, NUREG-75/038).

43. Final Environmental Statement of the Transportation of Radioactive Material by Air and Other Modes, U.S. Nuclear Regulatory Commission, December 1977 (NUREG-0170).

44. Environmental Impact Appraisal by the Division of Fuel Cycle and Material Safety Related to License Amendment for Materials License Amendment for Materials License No. SNM-1265 Morris Operation Facility - Grundy County, Illinois for General Electric Company - Docket No. 70-1308, Nuclear Regulatory Commission, December 1975 (NR-FM-002), especially Section 6.

45. Environmental Impact Appraisal related to the Renewal of Materials License No. SNM-1265 for the Receipt, Storage and Transfer of Spent Fuel at Morris Operation - General Electric Company - Docket No. 70-1308, U.S. Nuclear Regulatory Commission, June 1980, especially Sections 7.5 and 8.2.

4.0 DESIGN CRITERIA AND COMPLIANCE

4.1 INTRODUCTION

A general description of GE-MO and a summary of operational functions are contained in Section 1. Original design criteria for GE-MO facilities were developed and established as part of the design for a fuel reprocessing plant - the Midwest Fuel Recovery Plant (MFRP). Criteria herein are those applicable to the use of those facilities for spent fuel storage.

4.1.1 Material To Be Stored

GE-MO is licensed to store nuclear power station irradiated light water reactor fuel . Design bases are UO_2 fuel with initial enrichment of 5% U-235 or less, stainless steel, zirconium, or Zircaloy cladding and in a "bundle of rods" geometry. Design basis fuel may have been irradiated at specific power levels up to 40 kW/kgU, with exposure to 44,000 MWd/TeU (batch average), and must have cooled for at least 1 year after reactor shutdown before storage at GE-MO. The calculated fission product activity contents of fuel irradiated at 40 kW/kgU, exposed at 24,000 MWd/TeU and 44,000 MWd/TeU, and cooled 1 year are presented in Table 4-1.

Typical fuel received and stored has exposures of 33,000 MWd/TeU or less, with cooling periods much longer than 1 year. As of April 1, 1988, the average exposure of boiling water reactor (BWR) fuel in storage is about 17,000 MWd/TeU and that of pressurized water reactor (PWR) fuel about 25,500 MWd/TeU. As of the first of January 1999 the overall average cooling time is about 22 years.

Realistic exposures based on fuel in storage have been used in some analyses, as appropriate. Table 4-2 contains a list of analyses, fuel exposures and cooling times on which each is based.

Heat load calculations for basin water temperature and evaporation rates, basin water cooler design, and ventilation air cooling design are based on heat loads from fuel currently in storage and that expected to be stored.

4.1.2 Storage Conditions

Normal storage conditions at GE-MO impose much less stress on fuel than does the normal operational environment within a reactor. Maintaining integrity of fuel rods and monitoring release of off-gas provides protection against uncontrolled release of radioactive material from fuel in storage. Instrumentation and other equipment are provided to warn of unsafe conditions or the approach of unsafe conditions. However, the approach of unsafe conditions is relatively slow in all cases, so rapid response and prompt, automatic initiation of corrective action - as in a reprocessing plant or reactor in non-storage conditions - is not required.

Table 4-1
SPENT FUEL FISSION PRODUCT ACTIVITY
(Ci/TeU)
Specific Power = 40 kW/kgU
Cooling Time = 1 Year

CLASS	ISOTOPE	HALF LIFE	24,000 MWd/TeU	44,000 MWd/TeU
Noble Gases	Kr-85	10.701y	7,620	12,000
Halogens	I-129	1.57×10^7y	.021	.044
Tritium	H-3	12.346y	416	766
Transuranics	Am-241	432y	99	250
	Am-243	7370y	2.6	32
	Cm-242	162.76d	1,350	9,160
	Cm-244	18.099y	169	5,090
Total			1,621	14,532
All Remaining Fission Products	Rb-86	18.82d	.000693	-
	Sr-89	50.55d	9,410	7,140
	Sr-90	28.82y	64,700	103,000
	Y-90	64.06h	64,800	103,000
	Y-91	58.51d	20,800	16,500
	Zr-93	1.53×10^6y	2.3	3.9
	Zr-95	63.98d	41,500	38,300
	Nb-95m	86.6hd	527	487
	Nb-95	34.97d	87,800	81,800
	Tc-99	2.14×10^5y	10.8	18.9
	Ru-103	39.35d	2,680	3,280
	Rh-103m	56.116m	2,680	3,290
	Ru-106	366.5d	172,000	344,000
	Rh-106	29.8s	172,000	344,000
	Ag-110m	26.42d	12,300	51,700
	Ag-110	24s	160	672
	Cd-113m	14.6y	15	42.8
	Cd-115m	44.8d	3.5	4.9
	Sn-119m	250d	26.4	40.2
	Sn-123	129d	6113	801
	Sb-124	60.2d	3.2	8.1
	Sb-125	2.71y	4,840	10,100
	Te-125m	58d	1,180	2,470

Table 4-1 (Continued)
SPENT FUEL FISSION PRODUCT ACTIVITY

CLASS	ISOTOPE	HALF LIFE	24,000 MWd/TeU	44,000 MWd/TeU
All Remaining Fission Products	Sn-119m	250d	26.4	40.2
	Sn-123	129d	6113	801
	Sb-124	60.2d	3.2	8.1
	Sb-125	2.71y	4,840	10,100
	Te-125m	58d	1,180	2,470
	Te-127m	109d	1,320	1,870
	Te-127	9.35h	1,300	1,830
	Te-129m	33.52d	43.1	52.7
	Te-129	69.5m	27.4	33.5
	Cs-134	2.062y	88,900	283,000
	Cs-137	30.174y	77.900	142,000
	Ba-137m	2.5513m	73,700	134,000
	Ce-141	32.55d	800	772
	Ce-144	284.5d	530,000	594,000
	Pr-144	17.3m	530,000	594,000
	Pr-144m	7.2m	6,360	7,130
	Pm-147	2.62344y	104,000	91,400
	Pm-148m	41.29d	94.5	88.0
	Pm-148	5.37d	6.5	6.07
	Sm-151	87y	936	1,350
	Eu-152	13.2y	6.9	8.0
	Eu-154	8.5y	4,390	16,000
	Eu-155	4.96y	1,020	3,100
	Gd-153	241.6d	3.9	21.0
	Tb-160	72.1y	16.6	63.3
Total of All Remaining Fission Products			2.08×10^6	2.98×10^6

Table 4-2
ANALYSES, FUEL EXPOSURES, AND COOLING TIMES USED

Section	Type of Analysis	Exposure and Cooling Time Used MWd/TeU	Months
5.4.4.3	Storage Basket Heat Transfer	44,000	4
7.3.1	Radiation Sources	24,000	12
7.3.2	Fission Gases Released	24,000	12
7.4.2	Direct Radiation from Fuel	24,000	12
7.7.2	Maximum Off-site Exposures	24,000	12
8.6	Fuel Drop Accidents	44,000	12
8.7	Missile Impact Accidents	24,000	12

4.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

Systems, structures and equipment contributing to prevention of accidents (or to mitigation of consequences of accidents) which could affect public health and safety have been designed, fabricated, erected, operated, and maintained in compliance with established performance and quality standards. Under these standards, GE-MO will withstand, without loss of important protection capability, all credible operating and accident stresses, including forces that might be imposed by natural phenomena such as earthquakes, tornadoes, or flooding conditions.

Standards for ensuring systems, structures and equipment will adequately perform required safety functions for their intended service life with a low probability of failure have been based on temperatures, corrosion rates and other stress conditions derived from comprehensive analyses, including consideration of:

a. accessibility for in-service surveillance, monitoring and repair (or replacement);

b. potential for short-term exposure to abnormal operating or accident conditions;

c. consequences of component failure - no single component failure or multiple failures caused by a single initiating event shall result in significant radiation exposure to the public;

d. accessibility for emergency services, including ambulance attendants, fire and police services, and other emergency activity.

4.2.1 Wind and Tornado Loadings

4.2.1.1 Criteria

Final structures and components essential for safety shall be designed to withstand effects of short-term wind velocities of 300 mph with pressure differentials of up to 3 psi without damage

to fuel in storage to an extent endangering public health and safety. The site is located in USNRC Tornado Intensity Region I, as defined in Regulatory Guide 1.76.

4.2.1.2 Compliance

The fuel basin structure (enclosure) was analyzed with calculated wind loads applied as uniform static loads on vertical or horizontal projected areas of the walls and roof. Only dead load was considered as resisting uplift. Horizontal wind loads are distributed by the walls to the floor and roof systems, which transfer loads to the lateral load-carrying elements of the structures.

Plant structures and components were designed to withstand sustained wind velocities of 110 mph without loss of functions. At higher velocities, enclosure covering may fail or blow away.

These analyses included consideration of a drop in atmospheric pressure of 3 psi in 3 seconds. This condition would damage the basin enclosure, probably damage or even remove much of the roof and wall sheathing from the basin enclosure, but would cause no off-site radiological effect.

4.2.2 Tornado Missile Protection

4.2.2.1 Criteria

Plant structures and components essential for safety shall be designed to withstand effects of windborne missiles without damage to fuel in storage to an extent endangering public health and safety.

4.2.2.2 Compliance

Analyses in Appendix A.15 indicate the public health and safety would not be endangered as a result of tornado missiles impacting fuel storage structures or components.

4.2.3 Water Level (Flood) Design

4.2.3.1 Criteria

Structural integrity of fuel storage buildings and components shall not be endangered by flooding.

4.2.3.2 Compliance

Analysis has shown the maximum water level of a hypothetical flood greater than the maximum recorded flood at the site is below the site elevation (Appendix A.6).

4.2.4 Seismic Design

4.2.4.1 Criteria

Fuel storage structures and components essential to integrity of stored fuel, or fuel in the process of being transferred from shipping cask to the storage basin, shall be constructed to withstand a seismic event which, based on studies of area seismic history and geology, has a predicted recurrence of once per 1,000 years.

4.2.4.2 Compliance

The main building, including all portions of the structure now used for irradiated fuel storage, was originally constructed to seismic criteria based on a design earthquake and a maximum earthquake. The design earthquake was defined as a seismic event that has a reasonable probability of occurrence during the life of the facility, based on studies of historical seismically and structural geology. The design earthquake has a horizontal ground acceleration of 0.1 G.

The maximum earthquake is rated at twice the acceleration of the design earthquake, or 0.2-G. The design basis earthquake (DBE) can be sustained by these structures without exceeding allowable stresses. The maximum earthquake (ME) can be sustained without exceeding yield stress limits of the structure.

The 1940 El Centro, California earthquake has been thoroughly studied and well documented and provided most of the seismic data for time-history analyses available at the time of MFRP design. Illinois is not noted for earthquakes and no equally well studied seismic data base was available for Illinois.

Comparisons have been made between the El Centro earthquake spectrum and the spectrum in Regulatory Guide (RG) 1.60 for both Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) conditions. Results are shown in Figures 4-1 and 4-2. In generating spectra for the El Centro earthquake, damping values of 2% for DBE and 5% for ME were used. These damping values are consistent with those used in design of the basin structure. Sampling values for the RG 1.60 spectrum are 4% for OBE and 7% for SSE conditions, per RG 1.61. Differences between these spectra are insignificant.

A new fuel storage system was completed in 1976 to replace the original MFRP storage system. Since the new system is fabricated and installed as a separate entity in relation to the civil structures, it was designed to criteria in accordance with 10 CFR 100, Appendix A, and Regulatory Guide 1.60.

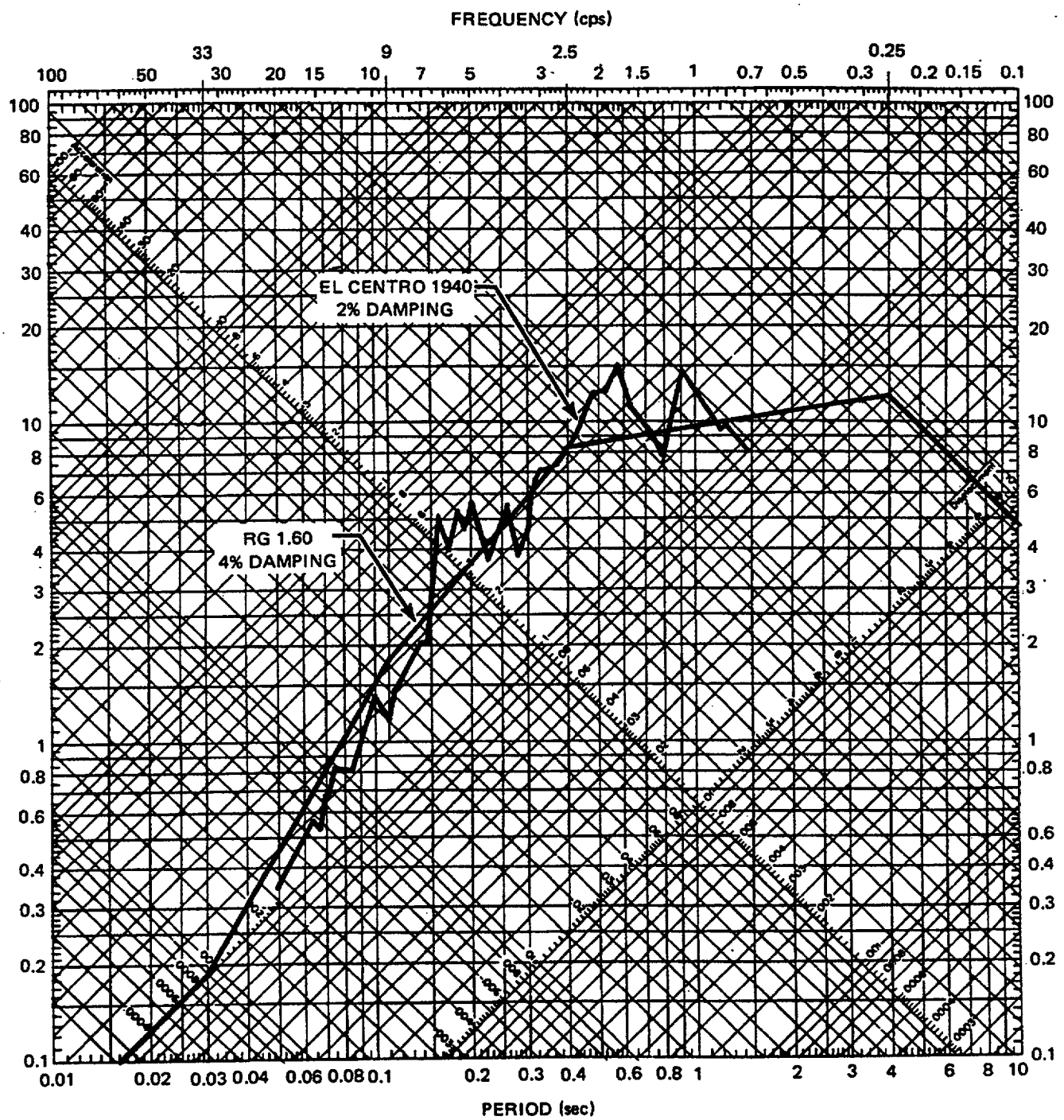


Figure 4-1. Spectra Comparison – 0,10G Ground Acceleration – RG 1.60 vs. El Centro 1940 N-S

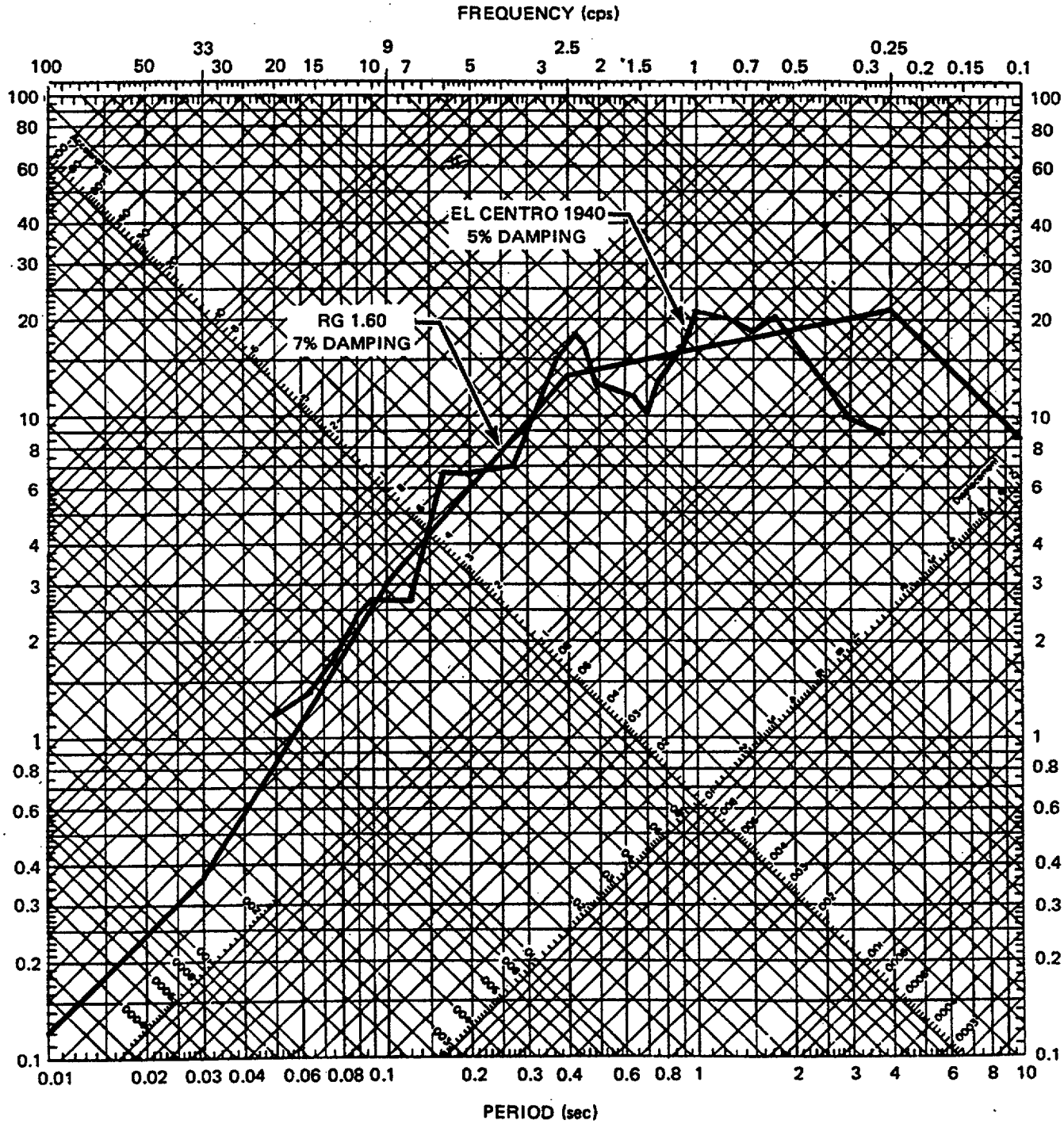


Figure 4-2. Spectra Comparison – 0.20G Ground Acceleration – RG 1.60 vs. El Centro 1940 N-S

4.2.4.2.1 Seismic Accelerations - Basins and Related Structures

a. Design Response Spectra

Structural (and equipment supported at grade) accelerations resulting from the DBE are defined by design response spectra. Design of fuel unloading and storage basins and underground vaults was based on north-south components of the 1940 El Centro earthquake normalized to 0.1G and 0.2G for the maximum earthquake case. The El Centro accelerogram is shown in Figure 4-3. The time used for the floor-level (main building) spectra was 6 seconds. Comparison of ground motion spectra for the 30 second period shows no measurable differences in the range provided.

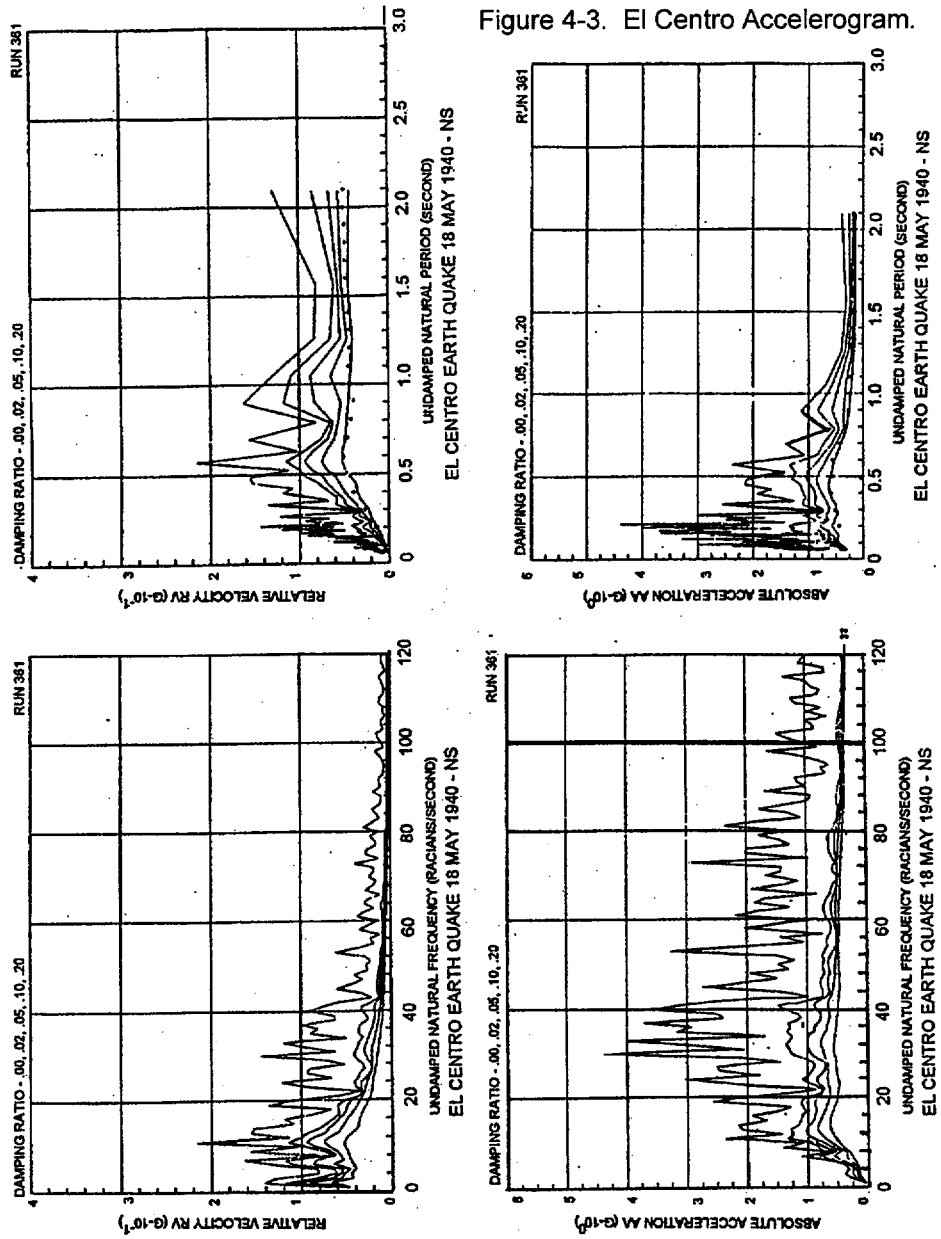


Figure 4-3. El Centro Accelerogram.

b. Design Response Spectra Derivation

Absolute acceleration response spectra for ground motion are shown in Figures 4-4, 4-5 and 4-6 for damping ratio values of 0.005, 0.010, and 0.020, respectively. These spectra result from a time-history analysis of the 1940 El Centro earthquake.

c. Damping values used for both design and maximum earthquake dynamic analyses of basin and vault structures, excluding basket and grid system, are:

ITEM	% CRITICAL DAMPING
Reinforced concrete structures	5.0
Steel frame structures	2.0
Welded assemblies	1.0
Bolted and riveted assemblies	2.0
Piping systems containing radioactive material	0.5
Underground vaults and basins containing radioactive material	0.5

d. Bases for Site-Dependent Analysis

A site-dependent analysis was not used. Section 3 describes the basis for specifying vibratory ground motion for design use.

e. Soil-supported Structures

Structures important for safety are founded on existing rock material exposed by excavation. The foundation support materials will withstand pressures imposed by appropriate loading combinations without failure (Appendix B.2).

4.2.4.2.2 Seismic System Analysis - Basins and Related Structures

Seismic system analyses applicable to basins, vaults, and related structures are discussed in the following paragraphs and Appendix B.4.

a. Seismic Analysis Methods

Hydrodynamic effects were a main consideration in analysis of vaults and tanks; specifically, cladding vault, fuel unloading basin, and fuel storage basins. Because the mathematically precise procedure for analysis is very complex, a simplified approach based on References 5 through 8 was used.

When a tank containing fluid of weight W is accelerated in a horizontal direction, a certain portion of the fluid behaves similarly to a solid mass in rigid contact with the wall. This mass exerts a maximum horizontal force directly proportional to the maximum acceleration of the tank bottom. Acceleration also causes another portion of the fluid to respond as

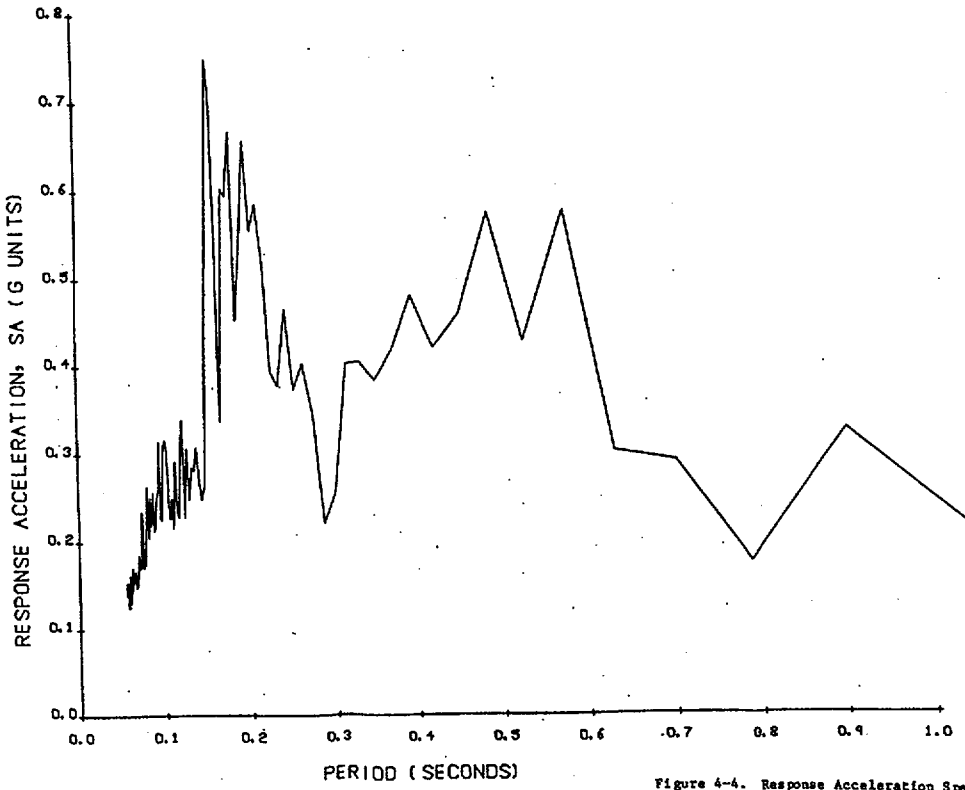


Figure 4-4. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio - 0.005

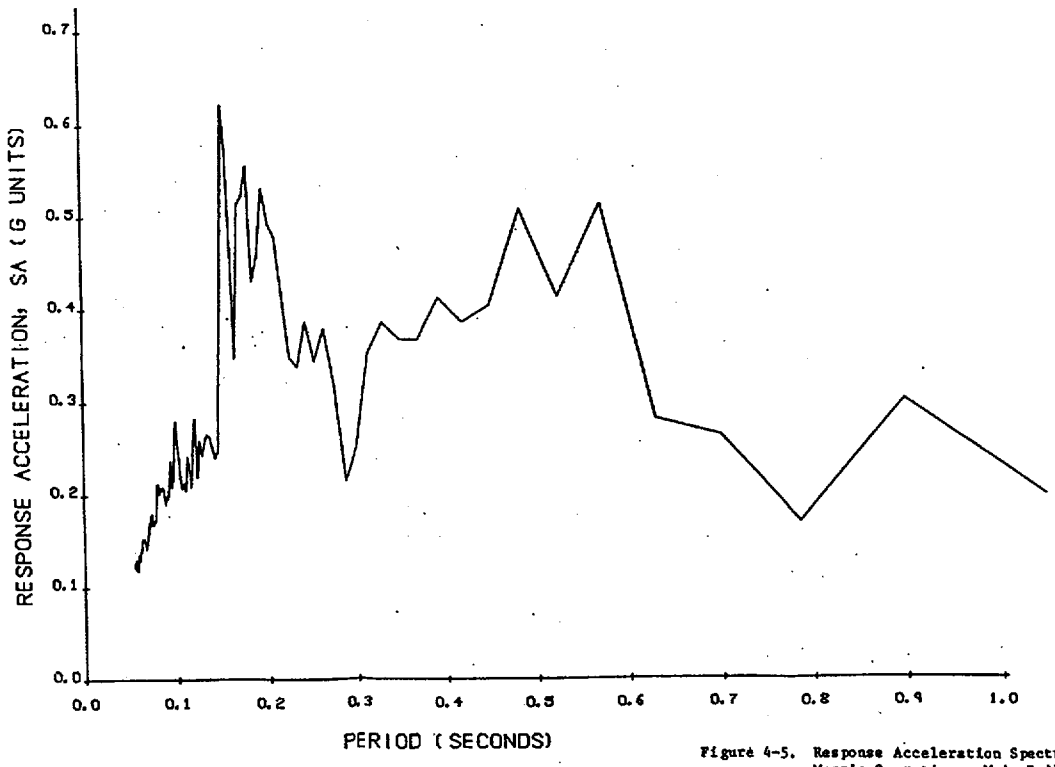


Figure 4-5. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio - 0.010

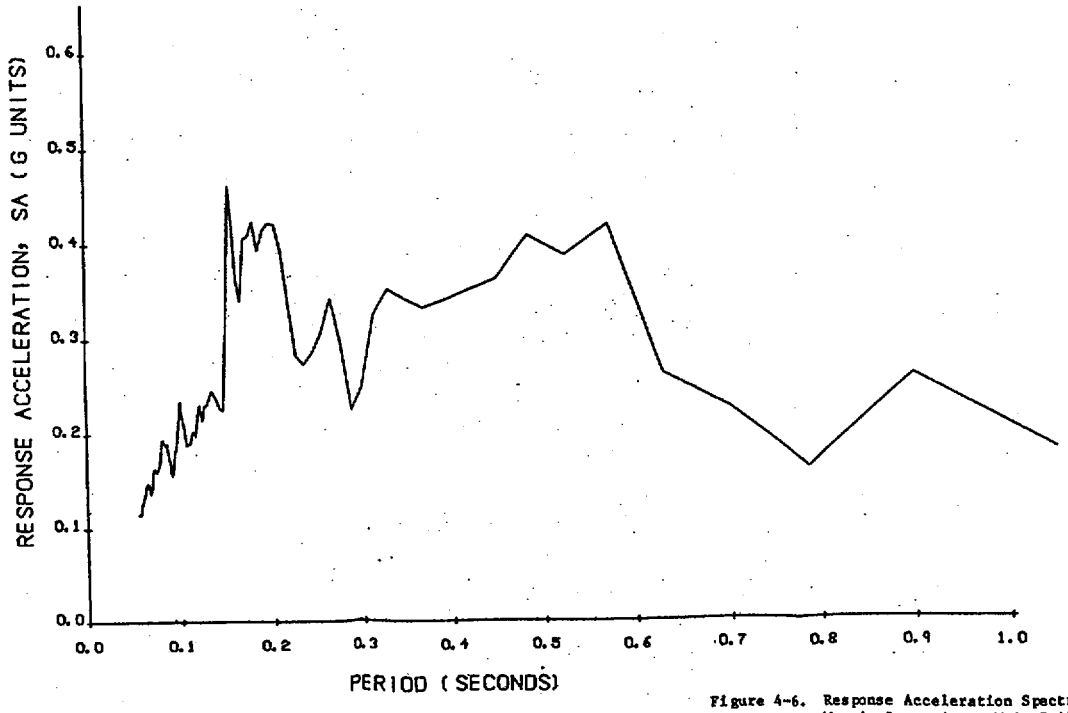
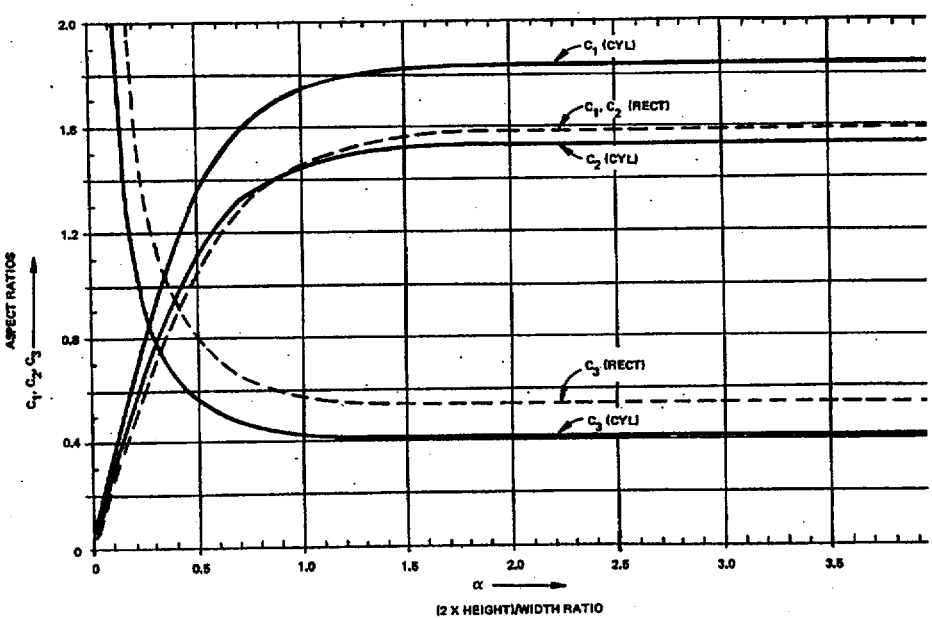


Figure 4-6. Response Acceleration Spectrum - Morris Operation - Main Building, Ground Motion, Damping Ratio = 0.02(

though it were a solid oscillating mass flexibly connected to the walls. The maximum amplitude of the mass relative to the walls determines both maximum vertical displacement of the water surface (slosh height) and horizontal force exerted on the walls.

Figure 4-7 provides dynamic constants (aspect ratios) used in determining period and magnitude of sloshing. In this figure, alpha is the ratio of twice the height to average width of the tank.



b. Rocking and Translational Response Summary

Because underground vaults and tanks are embedded in sound rock, lateral soil pressures on these elements are negligible. An evaluation of vaults and tanks (section a. above) was made on the basis of a shearing stress of 330 psi in the rock. Resulting deformations in the rock and concrete were used to calculate stresses. Special attention was given to points of stress concentration caused by cavities behind the concrete and to localized deformations at corners. Distortion was considered, caused by the discontinuity of rock at cavity sides and bottom of the cavities, and stresses in the vaults were calculated on the basis of resulting deformations.

Stresses were most severe at corners of thick walls of short span and where interior walls are formed into outer walls. Stresses in concrete walls were found to be less than allowable stresses in concrete or steel.

Periods of sloshing for vessels and tanks are given below.

Element	Period of Sloshing (Seconds)
Cladding Vault	3.7
Fuel Unloading Pit	2.2
Fuel Storage Basin I	3.5
Fuel Storage Basin II	3.9

Rocking and translational loads in the basket and grid system are transferred through the grid to walls of the fuel storage basin. An analysis was performed to determine if basin walls and liner can safely sustain maximum load combinations of the basket and grid system and water mass in the basin. The following stresses in the basin walls were found to be less than allowable stresses of concrete or steel:

(1) Bearing stresses at the base of the wall due to the support mechanism of the fuel storage system.

(2) Peripheral or punching shear at the base of the wall due to the support mechanism of the fuel storage system.

(3) Shear-friction of concrete in the wall; a crack is assumed to occur along the shear path. Relative displacement can be resisted by friction maintained by shear-friction reinforcement available across the potential crack.

(4) Stress due to skin-friction of the bearing plate (wedge) on the basin liner.

c. Methods Used to Couple Soil with Seismic System Structures

Cladding vault, cask unloading basin, and fuel storage basins are deeply embedded in rock. Consequently, they are assumed to be rigid and move with the rock.

d. Development of Floor Response Spectra

Floor response spectra are the same as those discussed in Section 4.2.4.2.1.

e. Differential Seismic Movement of Interconnected Components

Allowable stresses for extreme loads are 90% of yield strength. (In design of the fuel storage system, allowable stresses of 1.5 times AISC allowable stresses were used.)

f. Use of Constant Vertical Load Factors

No constant vertical load factors are used for structures, systems and components. The method of analysis used for both vertical and horizontal directions is the response spectrum method. Induced forces, moments and stresses due to motions in vertical and two horizontal directions are combined by the square root of the sum of the squares technique.

g. Seismic Restraint of Overhead Cranes

Overhead cranes that could potentially fall into the fuel unloading basin or fuel storage basins have seismic retainer attachments, or are designed otherwise to prevent dislodging during a seismic event.

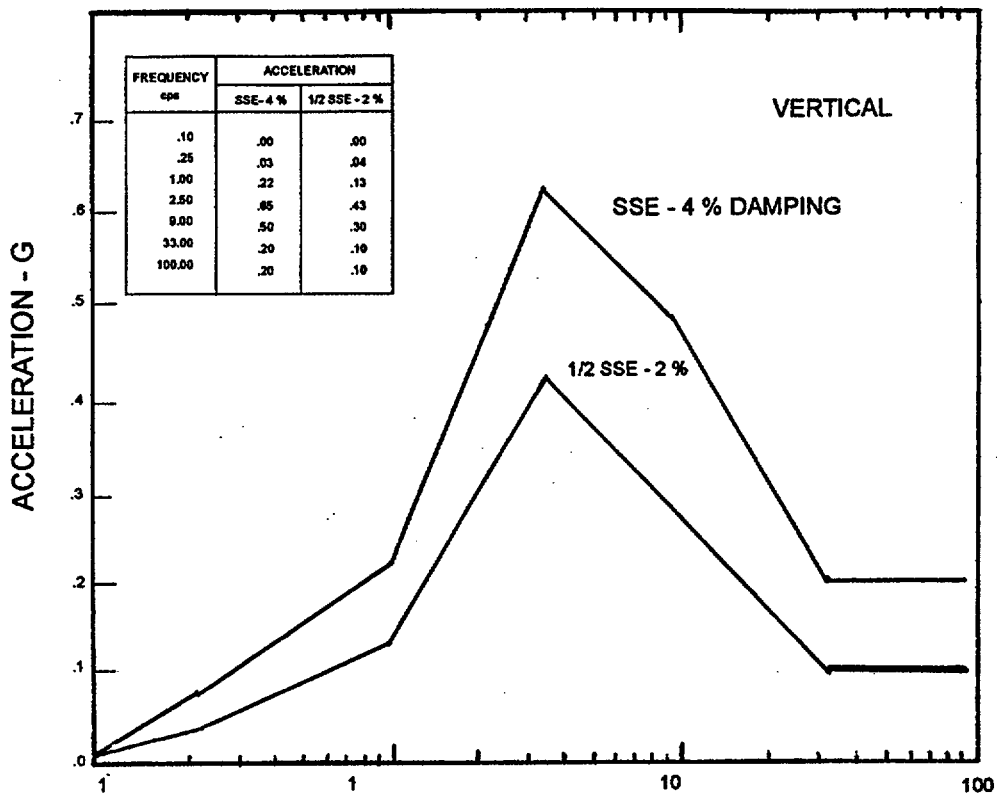
4.2.4.2.3 Seismic Acceleration and Response Spectra - Fuel Storage System

a. Response spectra for the fuel storage basket and grid system were derived as follows:

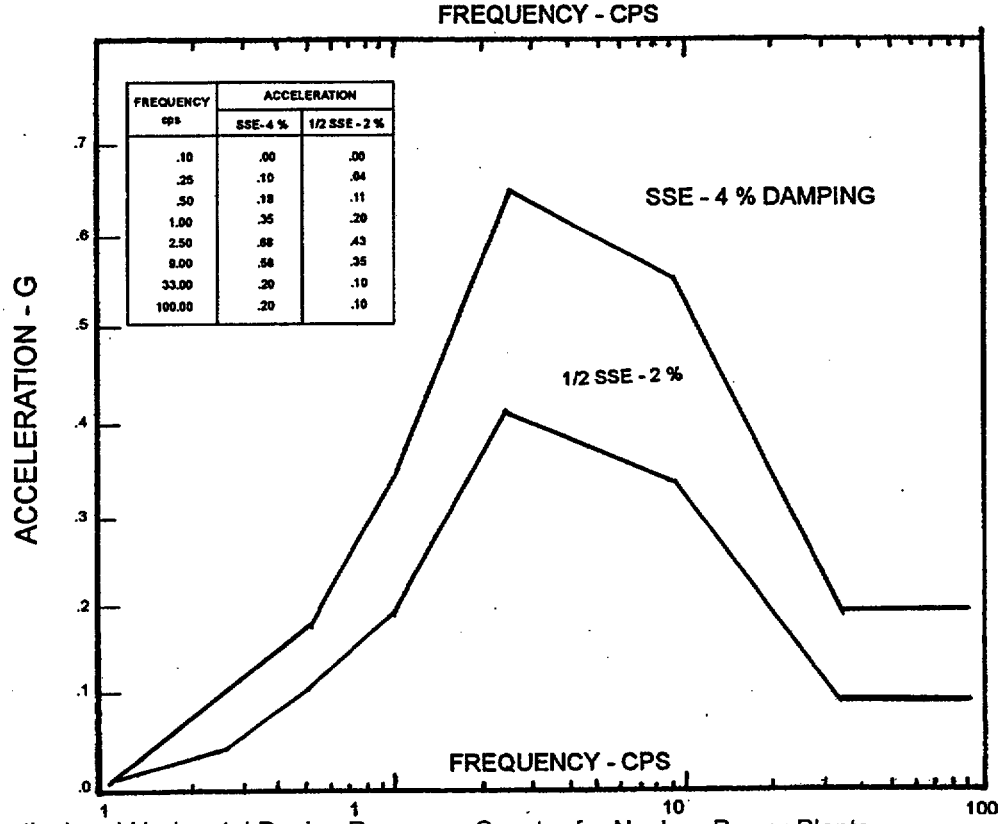
(1) Horizontal and vertical component design response spectra are scaled to a maximum horizontal ground acceleration of 0.20 G for SSE at 4% damping as specified in Regulatory Guides 1.60 and 1.61.

(2) Horizontal and vertical component design response spectra are scaled to a maximum horizontal ground acceleration of 0.10 G for 1/2 SSE at 2% damping as specified in Regulatory Guides 1.60 and 1.61.

A plot of these spectra is shown in Figure 4-8.



FREQUENCY cps	ACCELERATION SSE-4 %	ACCELERATION 1/2 SSE-2 %
.10	.00	.00
.25	.03	.04
1.00	.22	.13
2.50	.65	.43
9.00	.50	.30
33.00	.20	.10
100.00	.20	.10



FREQUENCY cps	ACCELERATION SSE-4 %	ACCELERATION 1/2 SSE-2 %
.10	.00	.00
.25	.10	.04
.50	.18	.11
1.00	.35	.20
2.50	.68	.43
9.00	.54	.35
33.00	.20	.10
100.00	.20	.10

Figure 4-8. Vertical and Horizontal Design Response Spectra for Nuclear Power Plants

b. Peak vertical acceleration of the response spectra for the basket and grid system occurs at a frequency of 3.5 cps. The fundamental frequency is 0.68 cps.

c. Damping values used for design and maximum earthquake dynamic analyses of the basket and grid design shall be (from Regulatory Guide 1.61) 2% (1/2 SSE) and 4% (SSE) for welded steel structures.

4.2.4.2.4 Seismic System Analysis - Fuel Storage System

a. Seismic Analysis Methods

In the seismic analysis a detailed mathematical model of the fuel storage baskets and support grid was subjected to horizontal and vertical design response spectra by the use of a computer system (SAP IV). The same mathematical model was used for both static and dynamic analyses.

The analysis used to obtain seismic response of the mathematical model is based on standard equations of motion for damped linear systems. Matrix equations were used to find the lowest natural frequencies, corresponding mode shapes of the system and response spectrum.

The SAP IV program calculates maximum responses in each of the lowest modes based on the spectra (accelerations) in the x, y and z directions. Total response for displacements and stress resultants is calculated as the square root of the sum of the squares of the modal maximum responses.

Seismic responses were obtained for N-S, E-W and vertical directions of storage baskets and grid. Degrees-of-freedom at the tops of the basket were "slaved" to six "master" baskets by partitioning 270 baskets into six groups. Seismic response of the lowest six modes was considered. Primary participation was derived from the first two modes.

Analyses used to obtain vertical dead load stresses and displacements were based on the same model as described above, except static loads were applied. The model was also subjected to two sets of static loads at 1.0G corresponding to N-S and E-W directions. A fourth load condition approximated a static equivalent analysis of the response spectra by applying horizontal loads at 0.6G and vertical loads at 0.2G.

b. Natural Frequencies and Response Loads

Frequencies and periods of vibration of basket modules and bottom holding grid are listed below for the first six modes.

MODE	FREQUENCY (HERTZ)	PERIOD (SEC)
1	5.99	0.167
2	7.59	0.132
3	8.71	0.115
4	9.97	0.100
5	10.05	0.100
6	10.32	0.097

c. Procedures Used to Lump Masses

Spent fuel storage baskets and grid were idealized as a finite element model consisting of over 1,300 nodal points and over 4,000 flexural beam-column elements. The grid was assumed to be on rollers on the basin floor and in axial contact with the wall at two adjacent sides of the basin. Basket modules were modeled as an equivalent cantilever beam connected to the grid by four artificial beam-type elements representing the holddown device. A small segment of mathematical model used in the analysis is shown in Appendix B.

Material and section properties for 12 basic elements were determined. In most elements these properties were extracted directly from the American Institute of Steel Construction (AISC) tables on steel sections. In other cases these properties were derived from combined shapes, built-up sections or castings. (See Appendix B.)

4.2.4.2.5 Seismic Subsystem Analysis

a. Determination of Number of Earthquake Cycles

Structures and equipment are designed on the basis of ground motion response spectra defined previously. Design of such structures and equipment is not controlled by fatigue because most stresses and strains occur only a small number of times. Full design strains from earthquakes and accidents occur too infrequently and with too few cycles to require a fatigue design basis for these structures.

b. Root Mean Square Basis

The total maximum value of any response quantity Q (shear, moment, deflection stress and acceleration) is based on the absolute sum, or on probability considerations, by the square root of the sum of the squares procedure according to the following equation:

$$Qmax = [(Q_1\ max)^2 + (Q_2\ max)^2 + (Q_3\ max)^2 + \ldots + (Q_n\ max)^2]^{1/2}$$

4.2.5 Combined Loads

4.2.5.1 Criteria

Stress levels for structures and equipment shall be limited to allowable stresses set forth in applicable codes, without allowance for short-term loading. Stresses arising from seismic motion in both vertical and horizontal directions shall be added to stresses arising from other applicable loadings. No significant concrete cracking shall occur as a result of design loading conditions. For maximum seismic ground motion or tornado wind conditions, combined stresses may approach but shall not exceed yield stresses.

4.2.5.2 Compliance

In general, concrete sections are designed so that failure would occur by yielding of the reinforcement rather than by crushing of the concrete. Where calculations indicated that a structure or component would be stressed beyond the yield point an analysis was made to determine its energy absorption capacity to ensure it exceeds the energy input from the initiating condition. In addition, such designs were reviewed to ensure any resulting deflections or distortions would not prevent performance of functions essential to continued confinement of radioactive materials and would not impair proper functioning of other structures and components from a safety point of view.

4.2.5.2.1 Loads - Definitions of Terms and Nomenclature

a. Normal Loads

Normal loads are those encountered during normal facility operation. They include:

D = Deadloads, or related internal moments and forces, including any permanent equipment loads.

L = Live loads, or related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.

T_o = Thermal effects and loads during normal operating conditions based on the most critical transient or steady state condition.

b. Severe Environmental Loads

Severe environmental loads are those that could be encountered infrequently during the life of the facility. Included in this category are:

E = loads resulting from the design earthquake

W = loads resulting from the specified design wind.

c. Extreme Environmental Load

Extreme environmental load is the load that is credible but highly improbable. It is:

W_t = loads resulting from design tornado, including wind velocity pressures, pressure differential and tornado-generated missiles, where applicable.

d. Abnormal Loads

Abnormal loads are those generated by a postulated accident, e.g., cask drop. They include:

T_a = Thermal loads resulting from an accident condition; specifically, this shall include design of fuel storage basins for thermal loads resulting from boiling basin water (212 °F), which could occur under certain conditions due to loss of basin cooling.

P_a = Pressure loadings resulting from an accident condition.

e. Other Definitions

u = section strength for concrete structures that is required to resist design loads and based on methods described by the American Concrete Institute in ACI 318.

s = section strength for structural steel based on elastic design methods, and allowable stresses against which calculated actual stresses are compared, are to be taken as 35/36 times allowable stresses defined by AISC Steel Construction Manual, Seventh Edition, Appendix A for 36,000 psi yield strength steel.

The yield strength for 304 stainless steel is used as 35,000 psi at 0.2% offset and a modulus of elasticity of 2.9×10^7. Allowable stresses for elements directly in the lifting load train are based on a safety factor of 5/1 on yield.

Y = section strength for structural steel required to resist design loads taken as 90% of yield strength. Allowable stresses of 1.5 times AISC allowable stresses are used, which are equal to or less than 90% of yield strength.

4.2.5.2.2 Load Combination and Acceptance Criteria for Concrete Structures

a. Load combinations used for normal operating conditions are:

(1) $u = 0.9D + 1.9E$

(2) $u = 0.75\ (1.4D + 1.7L + 1.7T_o)$

(3) $u = 0.75\ (1.4D + 1.7L + 1.9E + 1.7T_o)$

(4) $u = 0.9D + 0.75\ (1.9E + 1.7T_o)$

(5) $u = 1.4D + 1.7L + 1.9E$

b. Load combinations used for factored load conditions are:

(1) $u = D + L + T_o + W_t$

(2) $u = D + L + T_a$

4.2.5.2.3 Load Combinations and Acceptance Criteria for Steel Structures

a. Load combinations used for normal operating conditions are:

(1) $s = D + L$

(2) $s = D + L + 0.5E$

(3) $s = D + L + W$

(4) $1.5s = D + L + T_o$

(5) $1.5s = D + L + T_o + 0.5E$

(6) $1.5s = D + L + T_o + W$

b. Load combinations used for factored load conditions are:

(1) $Y = D + L + T_o + E$

(2) $Y = D + L + T_o + Wt$

(3) $Y = D + L + T_a$

(4) $Y + D + L + T_o + 1.5\ P_a$

c. Local yielding or buckling due to tornado winds and missile loadings is allowed unless this results in excessive release of radioactive materials to the environs.

4.2.6 Subsurface Hydrostatic Loadings

4.2.6.1 Criteria

Subsurface hydrostatic loading shall be considered in analysis of below-grade structures.

4.2.6.2 Compliance

Subsurface water is present at the interface between below-grade structures and surrounding rock, at least at the points of intersection with identified perched water zones. Lateral flow rates through rock are rather slow but are sufficient for hydraulic pressure head to accumulate outside below-grade structures. Magnitude of the pressure head varies with time and seasonal changes but only within the range of upper perched water zone level variations. This hydrostatic load is combined with other loads described in Section 4.2.5.

4.2.7 Basin Water Cooling

4.2.7.1 Criteria

Means shall be provided to maintain water temperature less than 200 °F (93.3 °C).

4.2.7.2 Compliance

Basin water is cooled by a system described in Section 5.5.3.

4.3 SAFETY PROTECTION SYSTEMS

4.3.1 General

There are no site-related factors sufficiently unusual to require protection systems or special design considerations beyond those normally required for a facility of this type. Operations take into account DNPS proximity to ensure cumulative effects of these operations do not constitute an unreasonable risk to public health and safety.

4.3.2 Protection by Multiple Confinement Barriers and Systems

The total confinement system consists of one or more individual confinement barriers and systems that successively minimize potential for release of radioactive material to the environment. These features also protect fuel in storage by protecting the fuel from damage and providing a favorable environment.

4.3.2.1 Criteria

Equipment and systems containing radioactive or potentially contaminated materials shall provide a continuous boundary against escape of such material and be designed to have a low probability of gross failure or significant uncontrolled leakage during the design lifetime.

Secondary confinement barriers such as vaults, ventilation system, etc., shall be designed and constructed to contain results of primary system failure, under conditions that may have initiated such failure, without loss of required integrity and to continue operation for the maximum anticipated period of stress.

Storage vaults and basins shall be designed and constructed for low probability of gross failure or uncontrolled leakage, with means provided to monitor leakage and preclude transport of radioactive materials to underlying aquifers. For lined structures containing radioactive or potentially contaminated liquids, leak detection and empty-out means shall be provided between liner and structure so that release of radioactive material to the environs can be avoided by pumping leakage back into storage, effecting repairs where leaks can be located and are accessible, or installing additional facilities in the event repair is not feasible. Water systems shall be designed to prevent accidental removal of water from basins by any means to less than a safe level. Basin water level shall be indicated and alarmed (low water alarm) in the CAS/SAS.

4.3.2.2 Compliance

All criteria described above have been satisfied; refer to Section 5.

4.3.3 Building Ventilation

4.3.3.1 Criteria

Radioactive material in building ventilation exhaust shall be reduced to levels **A**s **L**ow **A**s **R**easonably **A**chievable (ALARA) before being released to the environs. Special venting lines and enclosures shall be employed when necessary, such as during cask venting operations, to confine airborne radioactive particulate materials.

4.3.3.2 Compliance

Principal methods used to meet these criteria include:

a. Generation: Airborne radioactive material may originate from cask decontamination and venting operations; preparation of contaminated equipment for disposal; and from operation of low-activity liquid waste treatment systems. Other than these principal sources and minor H-3 and Kr-85 leakage from fuel in storage, no other significant source exists[1]. These activities (other than fuel storage) can be suspended on short notice whenever

higher than prescribed levels of radioactive materials are detected in the ventilation air exhaust stream. The waste evaporator system is designed to limit radioactive material in its effluent.

b. Confinement: The building ventilation system utilizes pressure differentials to maintain air flow paths to exhaust all ventilation air through the filter system and discharge stack. Special venting systems and special enclosures may be employed to confine airborne particulates from cask venting, decontamination activities, or similar sources to the filter - discharge stack system. The ventilation system is designed for all credible normal or anticipated off-normal conditions.

c. Release: Most of ventilation air is passed through a sand filter of demonstrated capability for removing particulate matter, and released through a 300 foot high discharge stack. Two streams are filtered through HEPA filters before release to the stack.

4.3.4 Protection by Equipment and Instrumentation

4.3.4.1 Criteria

Equipment and instrumentation shall be provided to monitor radiation and other parameters of operation, and to perform related control functions in accordance with the following:

a. Equipment and systems shall be set and adjusted to alarm and/or initiate action such that specified limits are not exceeded as a result of normal or abnormal occurrences.

b. Redundancy and independence shall be provided to a degree sufficient to ensure that no single failure of an instrument or equipment item can result in loss of control functions.

c. Equipment shall be designed to permit inspection, testing, and maintenance.

d. Monitoring of important systems and functions during normal operations and under anticipated off-normal or accident conditions is performed.

4.3.4.2 Compliance

Equipment is designed to permit inspection, maintenance, and periodic testing of functions to specified parameters. Temporary removal of single items of equipment from service has no safety significance.

Instrumentation is provided to ensure proper operation or notification of the failure of systems. Instrumentation is designed or specified to standards of known reliability.

Alarms that indicate a set point has been exceeded are annunciated in the CAS/SAS. Alarms with safety significance sound locally as well as in the CAS/SAS.

4.3.5 Nuclear Criticality Safety

4.3.5.1 Criteria

Every reasonable precaution is taken to ensure a criticality incident does not occur. Design controls are utilized and complemented by administrative control.

4.3.5.2 Compliance

The design of the spent fuel storage system includes the following controls to preclude a criticality incident:

a. Initial analyses were made in sufficient detail to demonstrate that criticality control concepts considered (e.g., control of geometry) were feasible under all credible conditions. Additional detailed nuclear criticality safety evaluations of the final design were made by qualified experts in the field to ensure final dimensions and other factors effecting safety margins were adequate to prevent a criticality incident. Additional detailed analyses required to confirm the final design are included in Appendices A.10, B.5 and B.15.

b. In the derivation of subcritical limits, the k_{eff} permitted for the most reactive credible conditions was specified as 0.95 at a 95 percent confidence level[2].

Operation of the spent fuel storage facility includes the following administrative controls to preclude a criticality incident:

a. Safety evaluation, review and approval of operating procedures related to design control parameters.

b. Verification of nuclear fuel parameters for fuel scheduled to be stored at GE-MO.

c. Verification of fuel identity for fuel received at GE-MO for storage.

d. Maintenance of fuel storage location records.

e. Specific fuel and cask handling procedures when these tasks are performed.

f. Personnel training.

Independent reviews and audits are utilized to determine adequacy of nuclear safety control provisions and effectiveness of implementing activities.

4.3.6 Radiological Protection

4.3.6.1 Criteria

Radiation and radioactive contamination conditions at GE-MO are controlled to provide protection of personnel health and safety at all times. Emphasis is placed on minimizing both individual exposures and total exposure (man-Rem) to **A**s **L**ow **A**s **R**easonably **A**chievable (**ALARA**).

During normal operations, including anticipated occurrences, the annual dose equivalent to any person located beyond the OCA boundary does not exceed 25 mRem to the whole body, 75 mRem to the thyroid and 25 mRem to any other organ as a result of either planned discharges or direct radiation from the facility.

Any person located at or beyond the nearest boundary of the OCA will not receive a dose greater than 5 Rem to the whole body or any organ from a design basis accident.

4.3.6.2 Compliance

Criteria are satisfied through the following design features and operational practices:

a. Confining radioactive materials to prescribed locations.

b. Clearly defining areas in which significant radiation or contamination levels exist.

c. Applying special provisions and appropriate procedures to assure personnel safety.

d. Applying rigorous surveillance, housekeeping, and clean-up practices.

e. Providing comprehensive personnel training in radiological safety.

Dosimeters are provided for ensuring accurate detection and assessment of personnel exposure to ionizing radiation in accordance with applicable procedures. Thermoluminescent dosimeters (TLDs) are positioned throughout the site to assess trends in background dose rates so that increases may be detected and corrective plans initiated.

4.3.6.2.1 Access Control (Controlled Areas)

Provisions have been established for controlling personnel access to areas in which radioactive material is present and are maintained to keep potential for contamination spread and exposure to radiation ALARA. This is accomplished by maintaining a series of access control barriers with increasingly restrictive occupancy constraints and access authorization requirements. These access controls were designed as follows:

a. General Electric Tract: Agricultural fencing with appropriate posting encloses the tract. Routine surveillance by operating and security personnel is utilized to ensure that unauthorized occupancy for significant periods of time is prevented.

b. OCA: An 8 ft. high chain link fence topped with barbed wire surrounds the OCA in which GE-MO storage facilities are located. Personnel and vehicle access gates are locked or guarded by security personnel at all times. Vehicles, materials and equipment are checked into and out of the area following procedures that require potentially contaminated or radioactive items to be monitored and cleared before entry or exit is authorized.

c. Radiologically Controlled Area (RCA): Personnel access to RCAs in which radioactive material is stored is controlled by limiting entrance such that occupancy authorization requirements can be strictly enforced. Access to various areas is controlled by structural compartmentalization and by authorization procedures commensurate with conditions existing in the particular areas. Access to all potentially contaminated areas is limited to specific routes in accordance with prescribed procedures and clothing and monitoring requirements which are varied according to conditions. Exit from RCAs, except under emergency conditions, is by the same controlled routes through necessary clothing change stations and monitoring facilities. Routine radiation surveys of the area are performed and TLDs are posted. Equipment requiring access (e.g., basin coolers) can be decontaminated to permit maintenance.

   Materials and equipment required for operation and maintenance will be checked into the areas and will be monitored before leaving the areas in accordance with prescribed control procedures. Access for transfer of such items is limited to specific points which are provided with means for precluding unauthorized usage.

   Additional requirements are utilized to limit access into areas of known or potential of high radiation levels or contamination levels. High Radiation Areas will be locked or guarded continuously.

4.3.6.2.2 Shielding

Radiation shielding is provided to control personnel exposure to ALARA levels.

4.3.6.2.3 Radiation Alarm Systems

Sampling and detection systems are provided that have sufficient sensitivity and scope of coverage to ensure any radiation or contamination condition of potential safety significance is accurately and promptly assessed.

Area radiation monitors (ARMs) meet the following requirements:

a. Monitors will detect gamma radiation within the range of 0.1 to 1,000 mRem/hr.

b. The high level alarm is audible locally.

c. The criticality accident alarm system meets the following requirements:

(1) The system has gamma-sensitive monitors that meet sensitivity requirements of 10 CFR 70.24(a)(1).

(2) The system produces a unique audible alarm.

(3) Two detectors are provided in the storage basin area, but are not underwater.

(4) The system is continuously functional.

(5) The high level alarm circuits for the system are arranged in parallel so that either alarm will energize all criticality alarms.

(6) The alarm circuit that energizes the criticality horns is designed to stay on until a manual reset in the SAS is employed to silence the horns (assuming radiation level is below trip point).

4.3.6.2.4 Effluent Monitoring

Sampling and monitoring systems and associated procedures are provided to measure radionuclides in ventilation effluent and in sample wells. Documentation and procedures for assessment of dose to the public from GE-MO effluents is contained in the GE-MO Off-site Dose Calculation Manual (ODCM).

4.3.7 Fire and Explosion Protection

4.3.7.1 Criterion

Structures, systems and components directly involved in storage of fuel shall be protected so that performance of their functions is not impaired when exposed to credible fire and explosion conditions.

4.3.7.2 Compliance

This criterion is met by using noncombustible and heat-resistant materials whenever practical throughout the facility, particularly in locations vital to functioning of confinement barriers and systems such as the basin areas and pump room. Fire detection, alarm, and suppression systems are installed in warehouse areas, and certain areas of the main building where deemed necessary. Fire extinguishers are strategically located throughout the facility. Fire training is furnished to all personnel. Fire alarms are audible in the CAS/SAS.

4.3.8 Fuel Handling and Storage

4.3.8.1 Criterion

Cask and fuel handling systems shall provide safe, reliable and efficient handling of casks and fuel.

4.3.8.2 Compliance

GE Morris Operation (GE-MO) is capable of receiving irradiated fuel bundles in shielded casks mounted on trucks or railroad cars. All major equipment such as cranes located above basin areas containing fuel are designed to ensure that components will not fall into the basin. The cask handling system has been designed to preclude a cask from being moved over fuel storage basins. Means are provided to preclude lifting a fuel bundle or a fuel storage basket to an elevation within a basin such that the shield provided by basin water is reduced to less than the prescribed depth.

Cask drop analyses have determined that energy absorption provisions in the fuel unloading basin are adequate.

Treatment of the storage basin water is adequate to minimize corrosion and prevent undue exposure of personnel.

4.3.9 Radioactive Waste Treatment

4.3.9.1 Criteria

Radioactive waste shall be stored in a manner that does not preclude retrieval and transfer off-site. Provisions shall be made for inspection and sampling of the material. No liquid radioactive waste shall be discharged from the site to the environs. Solid radioactive waste shall be disposed of in accordance with current regulations.

4.3.9.2 Compliance

Radioactive liquid waste is processed using the GE-MO or vendor radwaste system and is periodically concentrated by evaporation to reduce volume. Solid waste is disposed of via a licensed contractor.

4.3.10 Utility Systems

4.3.10.1 Criterion

Utility systems shall maintain the capability to perform safety related functions assuming a single failure.

4.3.10.2 Compliance

See Section 5.7.1.

4.4 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The objective of GE-MO is to prevent conditions that could result in undue risk to public health and safety by providing quality structures and reliable systems and components.

The degree of reliability that must be provided for various structures, components, and systems is determined primarily by consequences of failure of that unit. Failure of some structures, systems, or components could - if uncorrected - expose people to ionizing radiation (See Section 8). However, in a passive facility such as a fuel storage basin, repair or replacement of the failed structure, system or component can usually be accomplished long before consequences pose undue risk to public or employee health and safety. Failure of other structures, systems or components could result in an unacceptable loss of operating efficiency, but would pose no significant long or short-range risk to employees or the public.

Quality Assurance history and a list of safety related structures, systems and components are in Section 11. The quality assurance plan is NEDE-31559, "GE-MO Quality Assurance Plan".

4.4.1 Intensity of Natural Phenomena

Monitoring of natural remarkable events is provided by local, state, and federal agencies. These events are self evident and appropriate response is documented in the GE-MO Emergency Plan.

4.5 DECOMMISSIONING

4.5.1 Criterion

The GE-MO facility shall effect decontamination and decommissioning activities to an extent consistent with existing regulatory requirements.

4.5.2 Compliance

GE-MO design provides a stainless-steel-lined basin that includes cleaning, volume-reducing waste management facilities and a ventilation sand filter that will facilitate decontamination and decommissioning operations.

Codes, guides, and standards applicable to the GE-MO facility, as noted in this report, are listed in Table 4-3.

4.6 REFERENCES

1. K. J. Eger, Operating Experience - Irradiated Fuel Storage at Morris Operation, General Electric Company, January 1972 through December 1982 (NEDO-209969B).

2. See ANSI N18,2A-1975, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

Table 4-3
CODES, GUIDES, AND STANDARDS

Item	Section Where Referenced
Uniform Building Code	5.3.1
ASTM C150 (Cement)	5.5.1.2
ASTM A15 (Rebar)	5.5.1.2
ASTM 262 (Stainless Steel Liner)	5.5.1.3
Regulatory Guide 1.76	4.2.1.1
Regulatory Guide 1.60	4.2.4.2
Regulatory Guide 1.61	4.2.4.2
AISC Steel Construction Manual 7th Edition, Appendix	4.2.4.2.4[a]
ACI 318	4.2.5.2.1
ANSI-N18.2A 1975	4.3.5.2
ASTM A514 (Stainless Steel)	Appendix A.8
ASTM A285 (Stainless Steel)	Appendix A.13
ASTM A240 (Stainless Steel)	Appendix A.13
AWS-ASTM (welding rod)	Appendix A.13

[a] Other references, also.

5.0 FACILITY DESIGN AND DESCRIPTION

5.1 INTRODUCTION

This section contains descriptive information on buildings and other features of GE-MO used for receipt, or storage of irradiated fuel. Facilities associated with fuel reprocessing are discussed only as they relate to irradiated fuel storage activities.

This information has been consolidated from documents previously submitted and are part of the public record. The majority of descriptive material is based on the "Midwest Fuel Recovery Plant Final Safety Analysis Report" (MFRP FSAR) (NEDO-10178) with amendments and supplements and "The Safety Evaluation Report For Morris Operation Fuel Storage Expansion" (NEDO-20825).

Reproductions of maps and other illustrations in Sections 1 and 3 (especially Figures 1-1, 1-2, 3-1, and 3-2) provide geographical information about the GE-MO tract and show boundaries of property and general arrangement of buildings and other site features. (See Section 1 for use of terms "tract" and "site.") A more detailed layout and contour map of the site and environs is shown in Figure 5-1.

Radioactive material handling activities related to fuel storage are located within the Owner Controlled Area (OCA). There are no scheduled radioactive liquid effluent releases to the environs and no burial of radioactive or contaminated material on the tract. The only radioactive materials leaving the site are the gaseous effluents discharged through the ventilation stack or solid low-level radioactive wastes shipped for off-site burial. Off-site shipments are made in accordance with applicable United States Nuclear Regulatory Commission (NRC), United States Department of Transportation (USDOT), and other State and Federal regulations.

The entire GE tract (Figure 1-3) is enclosed by agricultural fencing with appropriate posting and forms the site boundary as defined in 10 CFR 20.1003 and described in Section 3.

5.2 CONTROLLED, RESTRICTED AND PROPERT PROTECTION

5.2.1 Restricted and Owner-Controlled Areas

Restricted areas, as defined in 10 CFR 20.1003, are within a 15 acre Owner Controlled Area (OCA) on the northern side of the tract (Figure 5-1), enclosed by a chain link fence topped with multiple strands of barbed wire for a total fence height of 8 ft. As depicted in Figure 5-1, facilities located within the OCA include the main building, adjacent ventilation sand filter and emergency equipment building (EEB), ventilation exhaust stack, cask service facility (CSF), utilities and service building, shop warehouse building, administration building, general warehouse, and water system well and elevated water tank. Liquid (nonradioactive) waste discharge lines are routed from the OCA to the sanitary treatment lagoons located south of the protected area. The sanitary lagoons are fenced to control access.

This Figure Withheld under 10 CFR 2.390

Figure 5-1. Site Plan showing principal facilities.

5.2.2 Gates

Entrance to the OCA is from the east-west county road (Collins Road), which bounds the tract on the north side. Entrances for personnel, road and rail traffic are at the northwest corner of the OCA. Entry is controlled from a guard station in the foyer of the administration building which includes the personnel entrance and is adjacent to the road and rail gates. An unmanned gate is located south of the OCA. The south gate provides access for construction equipment and is normally locked. A parking area for employees and visitors is provided north of the OCA.

5.3 PRINCIPAL STRUCTURE

The principal structure at GE-MO is the main or process building. This building was constructed to contain mechanical and chemical operations and processes for recovery of uranium and plutonium from spent nuclear fuel.

This Safety Analysis Report is concerned only with use of this structure for fuel receipt, and storage. Consequently, only those portions of the main building and other facilities associated with fuel storage and transportation activities are discussed in detail.

5.3.1 Main Building Design Basis

Design, materials and construction of the process building is in accordance with the Uniform Building Code and meets requirements of governing ordinances and authorities having jurisdiction (circa 1967). Facilities necessary for normal plant operation and confinement of radioactive materials were designed to resist earthquake and tornado conditions.

Section 4 describes significant criteria selected for design of the main building and other principal structures and describes principal means of satisfying these criteria.

5.3.2 Fuel Storage Facility Layout

Fuel storage facilities at GE-MO utilize the following portions of the process building:

a. cask receiving and decontamination areas;

b. fuel unloading pit;

c. fuel storage basins[1];

d. basin support systems (basin water cooling and filtration, etc.);

e. control room;

f. electro-decontamination room.

5.3.2.1 Process Building Plan and Sections

Appendix A.14 contains plan and section drawings of those portions of the process building associated with fuel storage. Drawings of other structures associated with fuel storage are included.

5.3.2.2 Confinement Features

The principal means of confinement of radioactive materials in a fuel storage facility is inherent in the fuel itself. Radioactive fuel pellets are contained within fuel rods; these stainless steel or zirconium alloy tubes are hermetically sealed when manufactured which prevents release of radioactive materials including gases that evolve from fuel during irradiation. Special vent hoods can be used for fuel bundles containing defective fuel rods to collect escaping gas, which would be filtered and then vented via the 300 ft. stack. Any such release would be a small fraction of 10 CFR 20 limits (Section 7). The fuel storage environment is benign relative to fuel cladding design conditions. Consequently, low temperatures and favorable water chemistry of the storage environment are not perceived to promote clad deterioration.

Irradiated nuclear fuel is received at GE-MO in shielded shipping casks which are designed, loaded, and transported in accordance with NRC and DOT regulatory requirements. Prior to shipment to GE-MO, fuel is inspected for defects; known defective fuel is not normally accepted for storage by GE-MO. Prior to unloading fuel for storage, cask flush water may be sampled to detect fuel damaged in transit. Fuel bundles are unloaded maintaining a minimum of 9 ft of water shielding for operating personnel. Cask unloading equipment and facilities are designed to minimize the effect of dropping or tipping over a cask.

Fuel bundles are stored in stainless steel basket assemblies designed to protect fuel from physical damage and to maintain fuel in a subcritical configuration. Baskets are locked into grids in the fuel basins to provide seismic restraint.

The basins are constructed below ground with stainless steel lined, reinforced concrete walls about 2 ft. thick poured in contact with the sides of a bedrock excavation. The south wall of the basin is about 4 ft. thick, because it was intended to stand independent of the surrounding rock to facilitate possible future expansion. Geophysical characteristics of the rock foundation would result in low permeability in the unlikely event of a major basin leak. A leak detection system and pump-out facilities are provided for the space between concrete walls and floor and the stainless steel liner.

A ventilation system is provided for the basins and other areas. It is designed so that air passes sequentially from areas of low contamination potential to areas of higher potential.

Basin water is circulated through a system that reduces radioactive contamination by ion exchange and filtration. A suction system is provided to vacuum basin floors and floating debris is removed by skimmer intakes. Radioactive materials collected by these systems are processed in the Radwaste System.

Irradiated fuel from light water reactors has been received and stored at GE-MO since 1972. These activities have reaffirmed irradiated fuel can be handled and stored safely with no impact on the environment[2]. There has been no detectable deterioration of fuel in storage (as determined by measurement of basin water activity) indicating the fuel is stable while in the storage basin environment.

5.4 CASK HANDLING AND FUEL STORAGE SYSTEMS

Following paragraphs describe cask handling and fuel storage systems by following, in general, the operational sequence of receiving and storing irradiated fuel. The functional sequence of fuel storage operations is described and illustrated in Section 1.

NOTE: It is the intent of this report to make generic statements only about nuclear fuel shipping, cask handling, loading, unloading, inspection, and receiving. It shall be considered the responsibility of the cask owner, since GE-MO does not own nuclear fuel shipping casks. Prior to receiving fuel shipping casks, the owner of the cask shall provide to GE-MO a certificate of compliance for the cask, and copies of all applicable handling, loading/unloading, and inspection procedures deemed necessary, for review and use. Alternatively, the cask owner may provide a representative to supervise and perform all necessary inspections. Any cask repair/rework is the responsibility of the cask owner. GE-MO may provide facility equipment and personnel support under the direction of the cask owner. QA program requirements shall be documented in the cask owner's QA program manual and are the responsibility of the cask owner.

5.4.1 Cask Receiving Area

Cask receipt and inspection operations take place outside of and in the Cask Receiving Area (CRA), a part of the process building.

5.4.1.1 Cask Receipt and Inspection Facilities

The rail spur and paved area (rails are flush with paved surface) lead into the CRA located at the northwest corner of the process building at the cask decontamination area (known as the Basin Decon Pad, or BDP) entrance.

The rail spur is provided with a car stop which is mounted on a large, reinforced concrete block designed to protect the building and storage basins from rail car accidents. The spur slopes away from the building and is equipped with a manual derail. The paved surface is designed to support trailers used for road transport of large shipping casks.

The CRA is about 70 ft. by 20 ft. and is paved and curbed. Although utility water is available in this area for cask washdown to remove road grime, this operation is normally performed on the BDP. The area is enclosed with insulated metal siding with a built-up roof of metal decking, 2 in. rigid insulation, and asphalt surfacing. The enclosure siding and roof will withstand sustained wind velocities of 110 mph but could fail under tornado conditions. There are four wind-driven roof ventilators.

5.4.1.2 Cask Handling Crane, and Handling Equipment

A two-motion, radio-controlled crane of 125 ton capacity is mounted on overhead rails which are parallel to and centered on the rail spur which serves the CRA. Lift height of the crane is approximately 34 ft. above grade. The horizontal travel area of the crane extends from the CRA over the BDP and finally over the cask unloading basin. The cask handling crane does not extend over any part of the storage basins.

The crane is equipped with rail keepers ("up-kick lug") to prevent the crane from derailing and falling into the CRA, BDP, or cask unloading basin.

Handling equipment will be used in conjunction with the cask crane to lift the cask from the transport vehicle and to move the loaded or empty cask.

5.4.1.3 Damaged Cask Handling

GE-MO does not own Type B nuclear fuel shipping casks and does not support "fabricator" QA program approval. Therefore, cask repair/rework is the responsibility of the cask owner. GE-MO provides facility equipment and personnel support under the direction of the cask owner. QA program requirements are documented in the cask owner's QA program manual.

5.4.2 Decontamination Area (BDP)

The decontamination area (BDP) (Figure 1-4) is used for incoming cask preparation and outgoing decontamination of fuel shipping casks. These operations include tightening or loosening cask head closures, incoming cask washdown, and sampling of cask coolant. The area is used for other activities involving decontamination.

5.4.2.1 Area Description

The BDP, about 27 ft. by 20 ft. in plan, is located inside the process building. The floor is a reinforced concrete pit, 3.5 ft. below grade, sloped to a sump located near the southwest corner of the pit. A stainless steel platform is centered on the north-south axis of the pit, welded to horizontal rails set in concrete. The platform is about 21 ft. by 8 ft. by 0.375 in. thick. The slightly raised platform allows for liquid runoff during cask wash and decontamination activities. The above-grade structure enclosing the decontamination facilities is of steel frame and

insulated siding construction adequately airtight to maintain ventilation control. The roof is approximately 50 ft. above grade and is of steel deck, rigid insulation and built-up roofing construction. The cask entry doors below the craneway are vertical dual doors about 30 ft. high and 11 ft. wide. A separate lift-type door is provided for the craneway.

Equipment, such as yokes, fixtures, and special tools required to receive and process casks, is moved into the BDP as required. Work platforms are provided for access to the upper parts of casks. A pump system is provided to flush casks internally. The BDP pit sump contents are pumped to the Radwaste System. Radiation shielding is provided for fixed lines carrying cask flush water.

5.4.2.2 Cask Venting and Flushing Operations

An incoming cask is surveyed for radioactive contamination and dose rates. Internal temperature and pressure of the cask may be measured to help determine internal conditions.

The cask may be connected to the cask sampling and flush system, and the cask may be vented to the Radwaste System. Flush water may be discharged either to the Radwaste System or the basin filter inlet. Casks may be filled with water from the basin and may be sampled prior to beginning flushing. Flushing can continue until at least one complete change of liquid content of the cask is completed. Fixed piping in the flush system is shielded..

Under normal conditions, concentrations of radioactive material in cask flushes are less than 1 μCi of radiocesium per milliliter. Concentration in excess of this limit requires further investigation for possibility of fuel damaged in shipment.

5.4.2.3 Low-Level Solid Waste

Solid radioactive waste generated at GE-MO is collected and periodically packaged for shipment to a commercial low-level contaminated waste disposal site. This waste consists primarily of disposable protective clothing, shoe covers, cleaning wipes, rags, rubber gloves, and similar materials used in various cask preparation and handling operations. While an on-site low-level waste compactor is available for volume reduction of solid LSA waste, incineration of combustible materials and re-melt of metals are preferred methods for treatment and disposal.

Contaminated resins are transferred to High Integrity Containers (HICs) and dewatered for subsequent burial at an approved site. Low-level waste packages are transported in shielded or unshielded trucks or semi-trailers dedicated to transfer of this type of waste[3].

5.4.2.4 Low-Level Liquid Waste

See Appendix Section B.23 for description.

5.4.3 Cask Unloading Pit

The cask unloading pit (Figure 1-5) is a two-level, water-filled basin adjacent to the BDP and connected to the fuel storage basins.

5.4.3.1 Description

The cask unloading pit is a reinforced concrete structure[4], poured against bedrock, with a stainless steel inner liner. General dimensions are shown in Figure 1-5. The cask unloading basin and other basin areas are filled with demineralized water to a reference level of 50 ft., or 2.5 ft. above grade, to provide cooling of stored fuel and radiation shielding during fuel unloading, transfer, and storage operations. The cask unloading pit is serviced by all three facility cranes: the cask handling crane, the fuel handling crane, and the basin crane.

Floors of the shelf and the cask unloading pit are provided with devices to dissipate impact loads from the maximum cask-drop accident.. The set off shelf is provided with a fabricated, stainless steel crushable pad and a 2 in. steel plate on top of the stainless steel liner. The floor of the cask unloading pit is covered with a 1.75 in. thick steel plate, under the 0.25 in. thick stainless steel liner.

5.4.3.2 Basket Positions

Three fuel basket positions are provided along the south wall of the cask unloading pit (fuel storage system components are described in Section 5.4.4). Empty baskets may be positioned in the basin before or after the cask is lowered to the floor. Using the fuel handling crane (Section 5.4.3.6), the crane operator engages a bundle with the fuel grapple, withdraws a bundle from the cask, and places the bundle in a predetermined position in a designated fuel basket. Basket designation and bundle position are determined by administrative procedures.

5.4.3.3 Doorway Guard

The only location throughout the facility where fuel basket contents could be discharged as a result of a postulated basket drop is at the cask unloading pit entrance to the fuel storage basin. During all other basket movements, the bottom of the basket is no more than 3 ft. above the basin floor (about 12 in. above the grid or about 27 in. above the floor). Under these conditions, a basket drop would not generate forces sufficient to eject fuel bundles from the baskets. Length of the basket assembly and height of the mounting grid prevent a base-up position with sufficient elevation to allow fuel ejection from the basket. (Also, see Section 8.6.2) However, if a basket were dropped in the doorway just inside Basin 1, the basket might tip toward the cask unloading pit and eject fuel bundles which could fall to the floor. Although consequences of this postulated accident do not present a serious safety hazard to either public or employees, a doorway guard is installed at the entrance to the fuel storage basin.

The doorway guard consists of a frame made of stainless steel pipe (Figure 5-5). It is supported in the doorway on the cask unloading pit side by hinges on the bottom attached to door brackets, and cables on the top. Each of the two cable assemblies includes a rod 0.25 in. diameter and 8.75 ft. long. Keepers are provided to ensure the cables stay on the pulleys. Underwater pulleys are attached to brackets on the cask unloading pit wall.

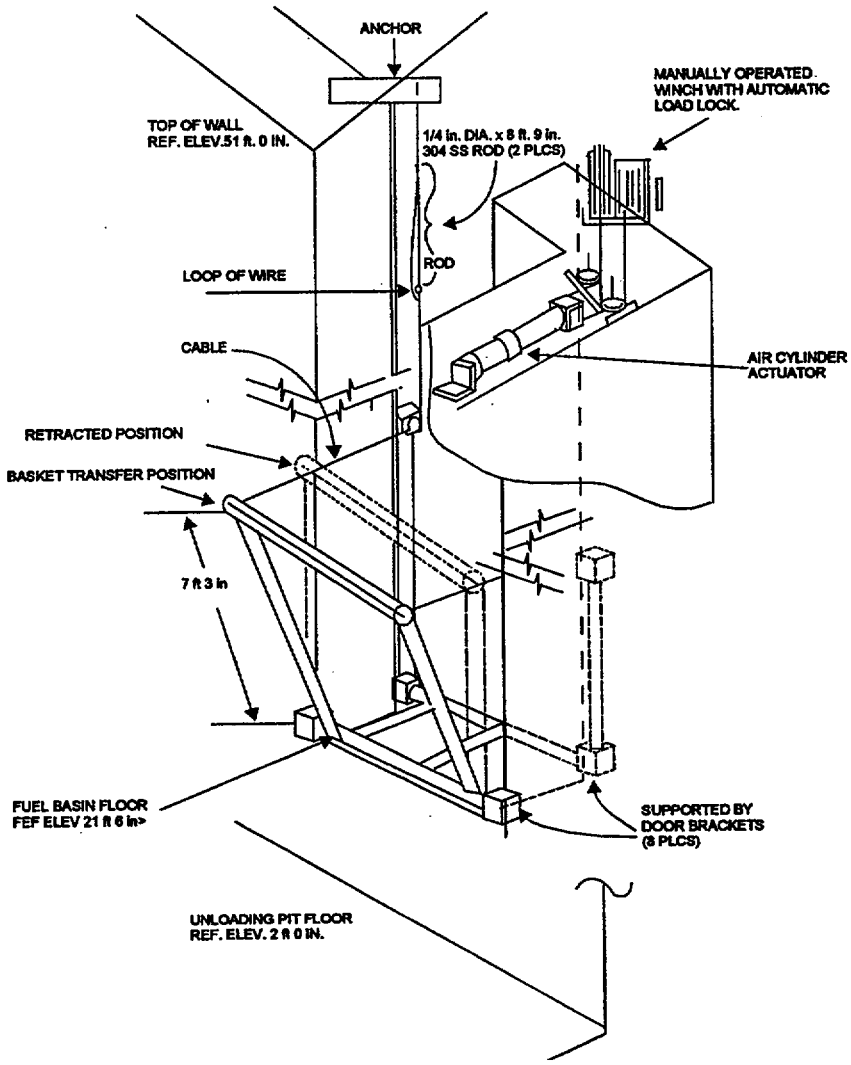


Figure 5-5. Unloading Pit Doorway Guard

Before fuel is loaded in a storage basket the guard is in the retracted or vertical position. The guard is lowered to the basket transfer (or angled) position prior to movement of a basket through the doorway. The basket lifting tool is lowered through the guard and attached to the basket located directly below the guard. The basket is then lifted through the guard and moved laterally into the fuel storage basin. Baskets must be moved to the eastward fuel basket position before being lifted to the doorway.

The doorway guard is designed to function as an energy absorbing device. Energy imparted to the guard by a basket falling against it is absorbed by stretching the two stainless steel rods (up to 40% elongation).

The fixed length of the basket lifting tool (grapple) prevents a basket from being lifted over the guard. A basket must be lifted through the guard and then moved laterally into the fuel storage basin. In this way, if a basket is dropped and it tilts toward the cask unloading pit, the guard will prevent it from tilting past a horizontal position.

5.4.3.4 Fuel Grapples

Fuel grapples are designed to remove fuel bundles from the cask and transfer fuel to storage baskets. The 5 ton capacity fuel handling crane is used to move the grapple to engage the fuel bundle. Grapples are fabricated to meet requirements of specific fuels. Typical BWR and PWR grapples are discussed in following paragraphs.

The BWR fuel grapple is constructed of two 20.5 ft. tubular sections joined lengthwise with a lifting bail and latching control mechanism at the top and a means of latching the fuel at the bottom. The grapple can be positively engaged through design features depicted in Figure 5-6. It can be disengaged only when the weight of the fuel bundle is not applied. The control mechanism for the grapple's hook is a manually operated handle connected to the latch by a cable running down the center of the grapple. An emergency release feature is incorporated into the design for use if the release cable fails.

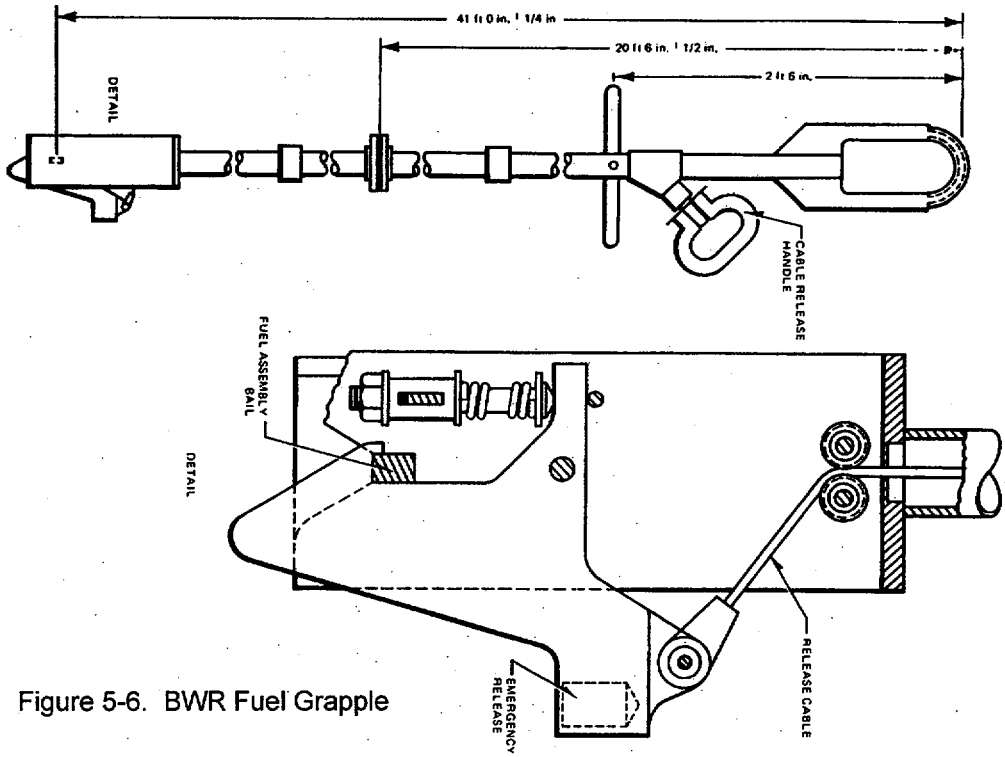


Figure 5-6. BWR Fuel Grapple

The PWR grapple (Figure 5-7) is 50 ft. long and constructed of stainless steel. At the top is a lifting bail and operating mechanism. At the bottom is the latching mechanism designed for the specific type of fuel bundles to be handled. The PWR fuel bundle has a "picture frame" upper plate. When the grapple is lowered onto the bundle, two guide pins on the grapple fit into holes in opposite corners of the picture frame, thus aligning the grapple. After the grapple is lowered to touch the upper plate of the fuel bundle, the four evenly-spaced grapple fingers are forced outward by manual rotation of the handle of the locking mechanism. This operation forces a cylinder down among the pivoted fingers, positively locking them in place. Once the bundle is locked in position on the grapple, it is ready for transfer to the storage basket.

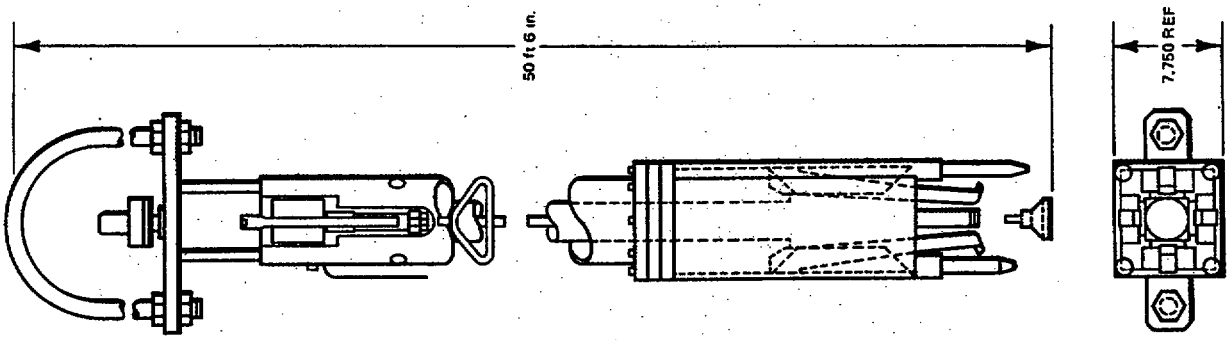


Figure 5-7. PWR Fuel Grapple

5.4.3.5 Basin Crane

The (fuel storage) basin crane is a manual control bridge crane of 7.5 ton capacity. Lift travel is limited by use of a long shank hook extension to prevent lifting of fuel baskets to within 9 ft. of the water surface. Travel limits of this crane extend from the cask unloading basin to the south end of the fuel storage basins. A platform on the south side of the crane bridge near water level facilitates operation of the basin crane. The fuel handling crane is operated from a platform on the north side of the basin crane. Bridge wheels and retainers are designed to maintain the basin crane in position under earthquake conditions. Derailment, if it occurs, would not result in either bridge or trolley falling into the basin. Repositioning on the rails can be accomplished manually with the use of hoists and jacks. Interruption of service of this crane has no safety connotation.

5.4.3.6 Fuel Handling Crane

The fuel handling crane (also referred to as basin auxiliary crane) is used to handle fuel bundles in the cask unloading basin. This crane has a 5 ton capacity with stepless speed control and is supported from rails attached to the underside of the cask crane support members. Provisions for meeting seismic conditions are similar to those for the basin crane including restraints (rail keepers) to prevent the crane from derailing and falling into the basin. The bridge is of the underslung monorail type, and the trolley is a rigid, one-piece weldment capable of withstanding vertical, lateral, or torsional strains. Bumpers for both bridge and trolley prevent overtravel. Interruption of service of this crane has no safety connotation.

5.4.4 Fuel Storage System

Fuel is transferred from a cask in the cask unloading pit to fuel baskets. Loaded baskets are moved into the storage basin by use of the basin crane (Section 5.4.3.5). Fuel baskets are latched into a supporting grid structure on the basin floor that provides seismic restraint.

The original intent for fuel storage at GE-MO was to provide short-term storage for fuel to be reprocessed. Thirty-two fuel baskets of relatively low storage density were provided to contain fuel bundles in the basin[5]. The unit storage densities[6] originally provided were approximately 0.2 TeU/ft^2 for BWR fuel and 0.1 TeU/ft^2 for PWR fuel in baskets and approximately 0.5 TeU/ft^2 for PWR fuel in storage racks. The present design provides more effective use of the total basin area for long-term storage by permitting unit storage densities of approximately 0.35 TeU/ft^2 for either BWR or PWR fuel. This modification was authorized by amendment to Materials License No. SNM-1265, dated December 1975.

5.4.4.1 Fuel Integrity In Storage

Regulations for safe storage of irradiated nuclear fuel require structural integrity to be maintained under severe accident conditions or catastrophic natural phenomena to prevent failure of fuel rods or a criticality excursion and to effectively control contamination levels in basin water. Integrity of fuel cladding is the primary barrier to release of radioactive material from fuel pellets.

Based on current experience and assessment of relevant literature, storage of spent nuclear fuel in storage basins for periods greater than 20 years is considered reasonable[7,8,9]. Fuel cladding is designed to withstand a far more severe environment in a reactor than that encountered in a storage basin.

Considerations include:

a. Zircaloy-clad fuel has been stored satisfactorily in basins since 1964 and stainless steel clad fuel has been stored since 1970. There are no indications of clad deterioration from the basin environment.

b. Low temperature and favorable water chemistry are not likely to promote cladding deterioration.

c. There are no obvious degradation mechanisms which operate on cladding under basin conditions at rates that would cause failures in the time frame of interim storage.

Literature[7,8,9] shows no significant effects of pool storage on fuel rods. Questions have been raised regarding long-term storage (20 to 100 years) because of possibilities of corrosive effects from inside the cladding and from effects at the external crud-cladding junction. However, tests

at Windscale on 9-year storage fuel do not show such attacks. It should be noted that the effect of small cladding defects in individual fuel rods is relatively minor due to chemical inertness of fuel pellets in water and cleanup capabilities of the filtration and ion exchange systems provided to control basin water contamination.

5.4.4.2 Equipment Description

The GE-MO fuel storage system utilizes uniformly-spaced baskets (26 in. square baskets on 27 in. centers). A schematic drawing of arrangement of PWR and BWR baskets is shown in Figure 5-8 and engineering drawings are located in Appendix A.14. Baskets for storage of BWR fuel bundles consist of either nine 8.5 in. stainless steel round tubes, or nine 6.25 in. stainless steel square tubes, while those for PWR fuel bundles consist of four 12 in. schedule 5S stainless steel pipes. The bottom of each basket is closed while holes in the basket wall permit convection flow through the basket. The closed-bottom area traps material that may fall from a fuel bundle, such as corrosion material on surfaces of the fuel bundle. The square tube BWR baskets have flow holes in the bottom and the wall to promote convective water flow through the basket.

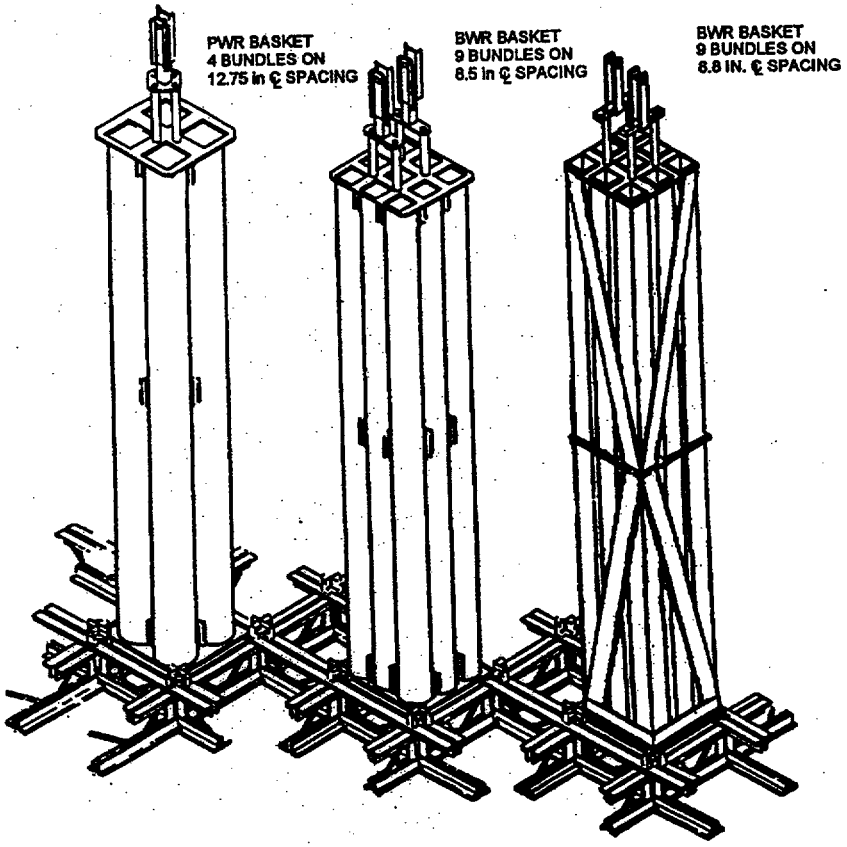


Figure 5-8. Morris Fuel Storage System. Constructed of stainless steel, the system provides a secure, flexible system for storage of LWR fuels. The three types of baskets mount interchangeably in the support grid.

Stainless steel baskets reduce neutron interaction between adjacent fuel bundles, permitting more efficient use of space. The resultant combination of separation and stainless steel neutron absorption ensures that the effective multiplication factor (k_{eff}) for an array of baskets will be < 0.95 at the 95% confidence level.

Pipes or tubes are attached firmly together and supported by a substructure, forming an independently movable basket. To lift the basket, special hooks are used to engage lifting rods that protrude above the basket. Outside substructure dimensions of PWR and BWR baskets are identical; therefore, each will fit interchangeably into a standard supporting structure.

Baskets are locked in position on a mounting grid of stainless steel members on the basin floor. A three-basket mount is installed in the cask unloading pit and a similar mount may be installed in the transfer corridor so baskets can be temporarily placed in the cask unloading pit or in the corridor in a manner equivalent in safety to that used in the main basin area. These mounts are called basket retainer frames, and are equivalent to the mounting grid used in the fuel basins.

Figures 1-13, 5-8 and 5-9 show views of the grid. Grids are installed in the basins in large modules (typically 4 by 14 basket units per module), which are limited to the size that can be moved and installed safely and conveniently in the fuel storage area. Grids are braced against the walls using wedges. An analysis of load effects on basin walls and liner indicates basin walls will withstand seismic and thermal loads transmitted by the support grid. As a result of the analysis, a solid film lubricant (Electrofilm)[10] was used on wedges to reduce the coefficient of friction between grid and wall to accommodate thermal and seismic movement.

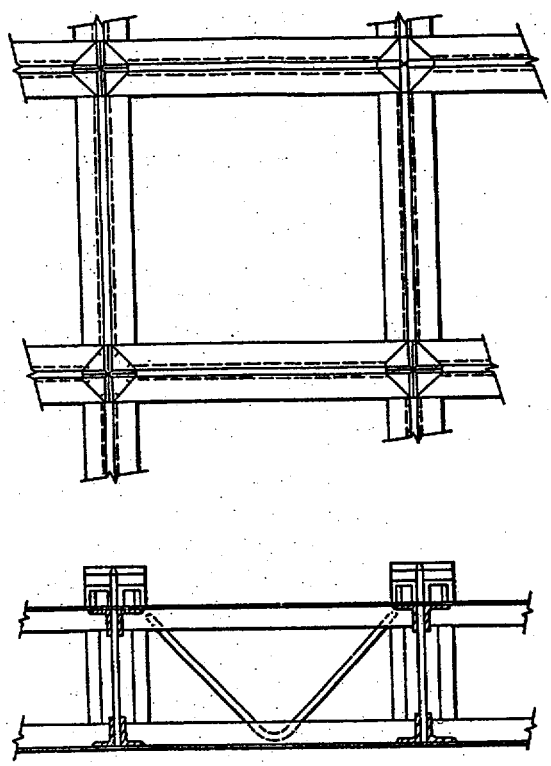


Figure 5-9. Typical Grid Assembly

Grids are fabricated from stainless steel structural material. Basket weight is supported by the stainless steel angle structure of the grid. At each intersection of cross members, a locking block is attached to the top of the grid structure, secure to the baskets in place.

Each basket has four cam-activated latches (Figure 5-10). Latches extend from each corner of the basket base and engage locking blocks on the grid when the latches are activated by linkage to the four lifting rods at the top of the basket assembly. When the weight of the basket (full or empty) is supported by the lifting rods, the cam-operated latch assemblies are retracted and will not engage the locking blocks. When the basket is set in place on the grid and lifting rods are released (tension removed), the weight of the lifting rod assemblies cause latch assemblies to engage locking blocks.

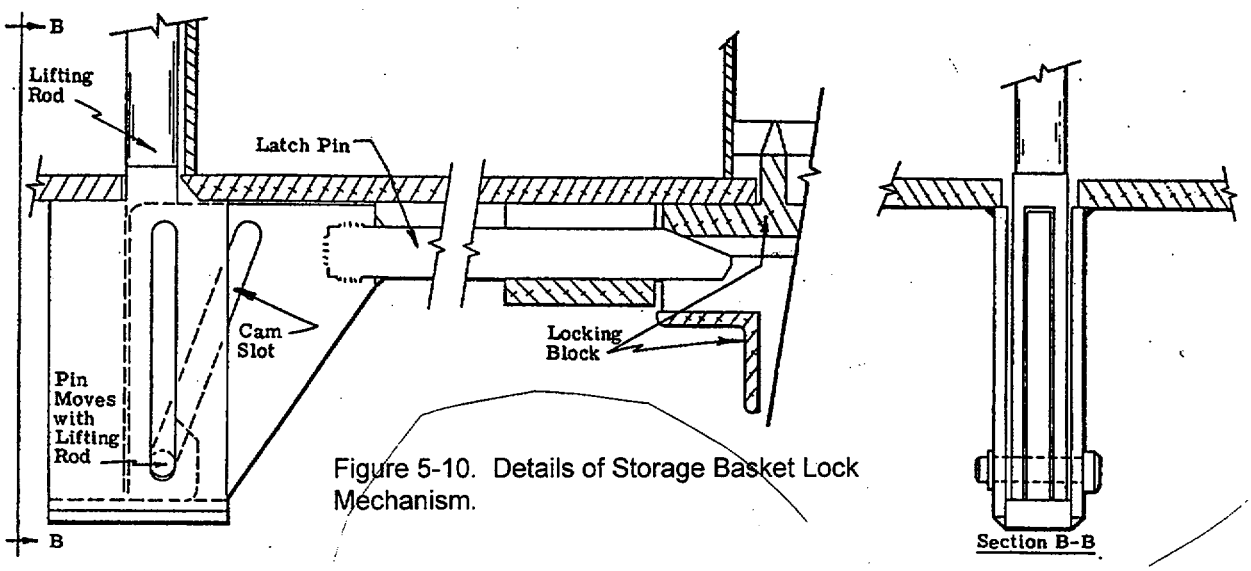


Figure 5-10. Details of Storage Basket Lock Mechanism.

Design criteria basis and safety analysis of the fuel basket system, including criticality analysis, are contained in Appendices A and B. The grid-basket system has been subjected to seismic testing to design criteria. Appendix A.14 includes engineering drawings of baskets and support grids. The fuel storage system has a minimum design life of 40 years but because of the nonaggressive service environment, a much longer useful life is indicated.

5.4.4.3 Heat Transfer from Stored Fuel

Heat transfer from stored fuel has been calculated for both BWR and PWR fuels and differential temperatures from fuel to basin water determined (Appendix A). Calculations included determining hole sizes in the basket wall that allow adequate water flow. Final basket assembly design is such that, even with some hole plugging (not considered a credible event), fuel temperatures remain satisfactory. Even with basin water cooling systems inoperative, maximum water temperature would be 183 °F[11]. See Appendix A.9.

5.4.4.4 Basket Lifting Tools

The BWR and PWR basket lifting tools (or basket yokes) are identical in function. However, the BWR yoke has two lifting hooks and the PWR yoke has one hook in order to match respective basket lifting bails.

Both lifting tools are constructed of stainless steel. Each tool is approximately 14 ft. long, a feature that precludes inadvertently lifting fuel closer than 9 ft. to the surface of basin water.

5.5 FUEL STORAGE BASINS AND SYSTEMS

This section describes fuel storage basins (Basins 1 and 2) and includes information about concrete and construction techniques employed when basins, main building, and other related facilities were built. Information regarding reinforced concrete construction is referenced in other sections of this report. General configuration and size of the water-filled fuel storage basins are shown in Figure 1-5.

5.5.1 Storage Basin Description

Basin 1 has an area of about 900 square ft.; Basin 2 has an area of about 1,500 square ft. There are a total of 414 fuel basket positions: 150 in Basin 1 and 264 in Basin 2.

Fuel storage basins and the cask unloading pit are constructed of reinforced concrete poured on bedrock with a welded stainless steel liner. Fuel storage basins are filled with demineralized water to a nominal depth of 28.5 ft. Water level may be lowered 2 ft. for maintenance or other purposes but at least 9 ft. of water is normally maintained above the top of stored fuel. If the water level falls more than 2 ft., pump suction inlets will be exposed. There is no means of accidentally draining the basin, nor can any basin water systems inadvertently drain the basin (i.e., the water systems are designed with nonreversible pumps, no drainage system, etc.). Basin water level is indicated in the CAS/SAS. The system includes an audible low-level alarm.

Cask handling, cask unloading, and fuel storage areas are constructed of concrete, steel, and other materials which are either nonflammable or fire-retardant No significant amount of flammable materials is used in these areas, and other potential fire dangers (bottled gases, etc.) are introduced only under stringent administrative control. No fire detection or automatic fire suppression systems are required in these areas or in the basin pump room and its extension. Fire extinguishers are strategically located and plant personnel are trained for fire surveillance. Further protection is provided by surveillance patrols.

Reinforced concrete in basin walls and floors is estimated to have a useful life of more than 100 years. The stainless steel liner can be expected to have a useful life of more than 100 years because of the non-aggressive service environment.

5.5.1.1 Foundation and Excavation

The basins are founded on shale bedrock (Figure 5-11). Samples of the shale have been tested at ultimate compressive strengths ranging from 6,000 to 11,000 psi. Appendix B contains a site survey and foundation report prepared for MFRP construction[12]. The excavation site was over excavated and back-filled to the south of Basin 2 to facilitate possible expansion of storage capacity at some later date. All loose and disturbed rock was removed prior to concrete construction. Backfill consisted of controlled and compacted granular soils. Concrete mud mats were poured to fill any area excavated more than 4 in. deeper than required (except for the south wall of Basin 2). The basin wall structure is designed to resist pressures from backfill and soil water where over excavations were made (south of fuel basin and vaults, Figure 5-11).

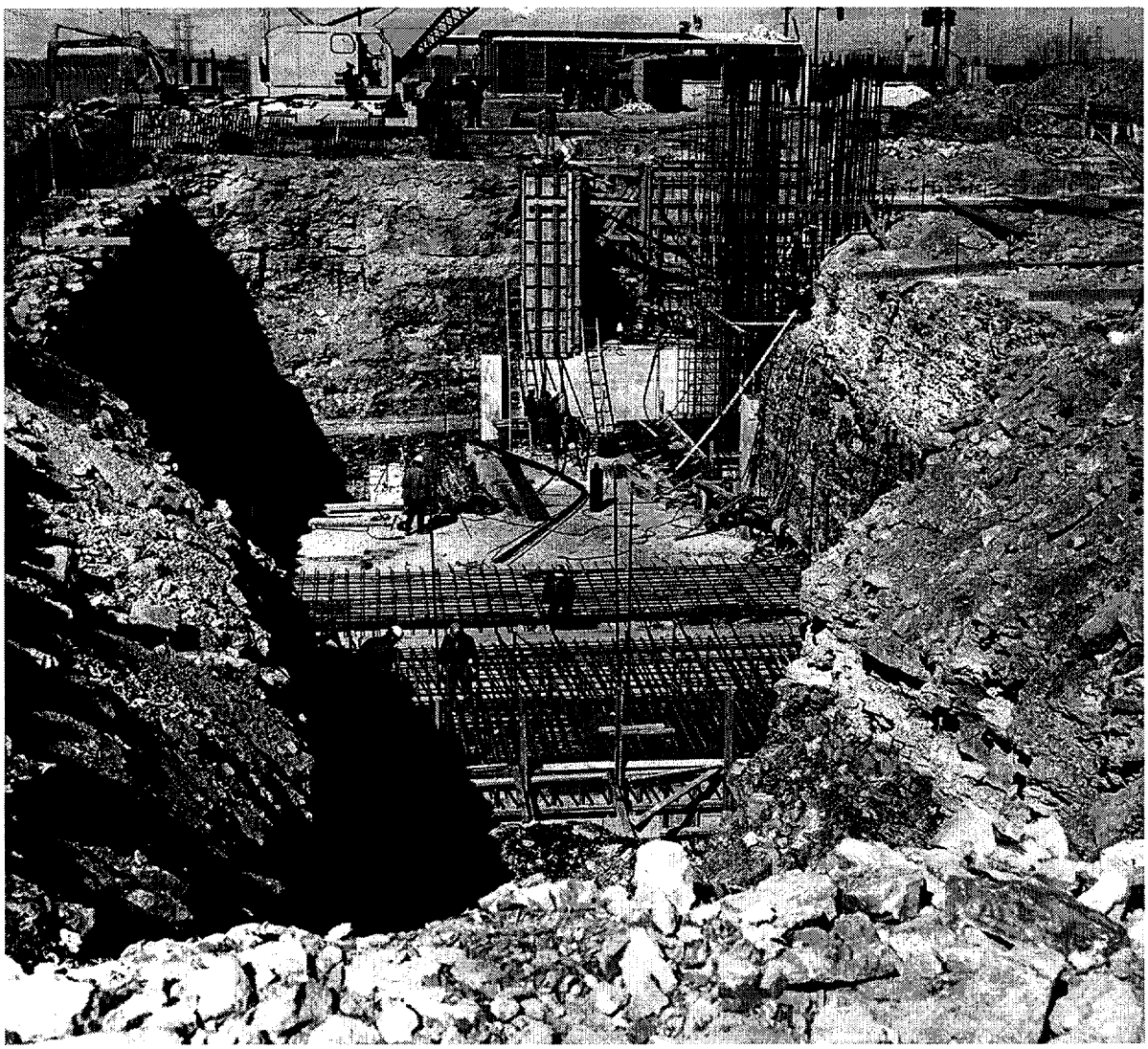


Figure 5-11. Excavation and Foundation Construction.

5.5.1.2 Concrete Structure

Storage basin floors were poured on bedrock and range in thickness from 30 to 54 inches. Basin walls extend 3.5 ft. above grade.

Materials used for basin concrete construction are typical of other concrete construction at GE-MO. Materials used for reinforced concrete structures were:

- Cement conforming to ASTM C150 type 2
- Washed sand
- Washed and graded aggregate
- Reinforcing steel per ASTM A15, intermediate grade

Concrete pours had slump tests and laboratory samples taken, usually at the truck discharge, but at times at the point of placement - particularly on canyon containment walls. Concrete samples were taken for every pour of 100 yards or less, whenever a pour composition changed and one per 100 yards for pours greater than 100 yards. A full-time concrete inspection program was in effect during construction.

Reinforcing steel used in the basins is intermediate strength with 40,000 psi minimum yield strength. Structural welds that carry loads from one element or reinforcing bar to another were not used. Where required, loads were transferred from bar to bar by conventional reinforcement bar laps secured in assemblies by steel tie wires. In special cases, U-bolts were used. The only welding permitted was tack-welding reinforcing steel to brace assemblies away from forms or to secure embedded items in position during the concrete pour. In most cases, assembly bracing or embed securing was done by use of additional reinforcing steel or structural steel tack welded to the reinforcing steel assembly. Embeds were either welded or clamped to this steel. Tack welds were made no larger than necessary to produce sound, crack-free welds.

5.5.1.3 Basin Liner

The unloading and storage basin complex is completely lined with 304L stainless steel sheets placed flush against concrete walls and floors and welded to a gridwork of stainless steel back-up members embedded in the concrete (Figure 5-13). The cask unloading pit floor liner is 0.25 in. thick and is placed over a 1.75 in. thick steel plate provided for distributing impact loads over the underlying concrete structure. Additional energy absorbing means, as may be required by cask drop accident considerations, will be installed for receipt of larger-sized casks.

The set off shelf liner, also 0.25 in. thick, is placed directly on the concrete structure with an energy absorbing assembly placed on top of the liner (seen in Figure 1-13).

For the remainder of the storage basin complex, the floor liner is 0.187 in. thick. Walls of the cask unloading pit, including shelf area are lined with 11 gauge sheet steel. For the fuel

storage basin walls, the liner is 11 gauge sheet steel from floor level to approximately 16 ft. up the wall and 16 gauge sheet steel from there to the top of the basin.

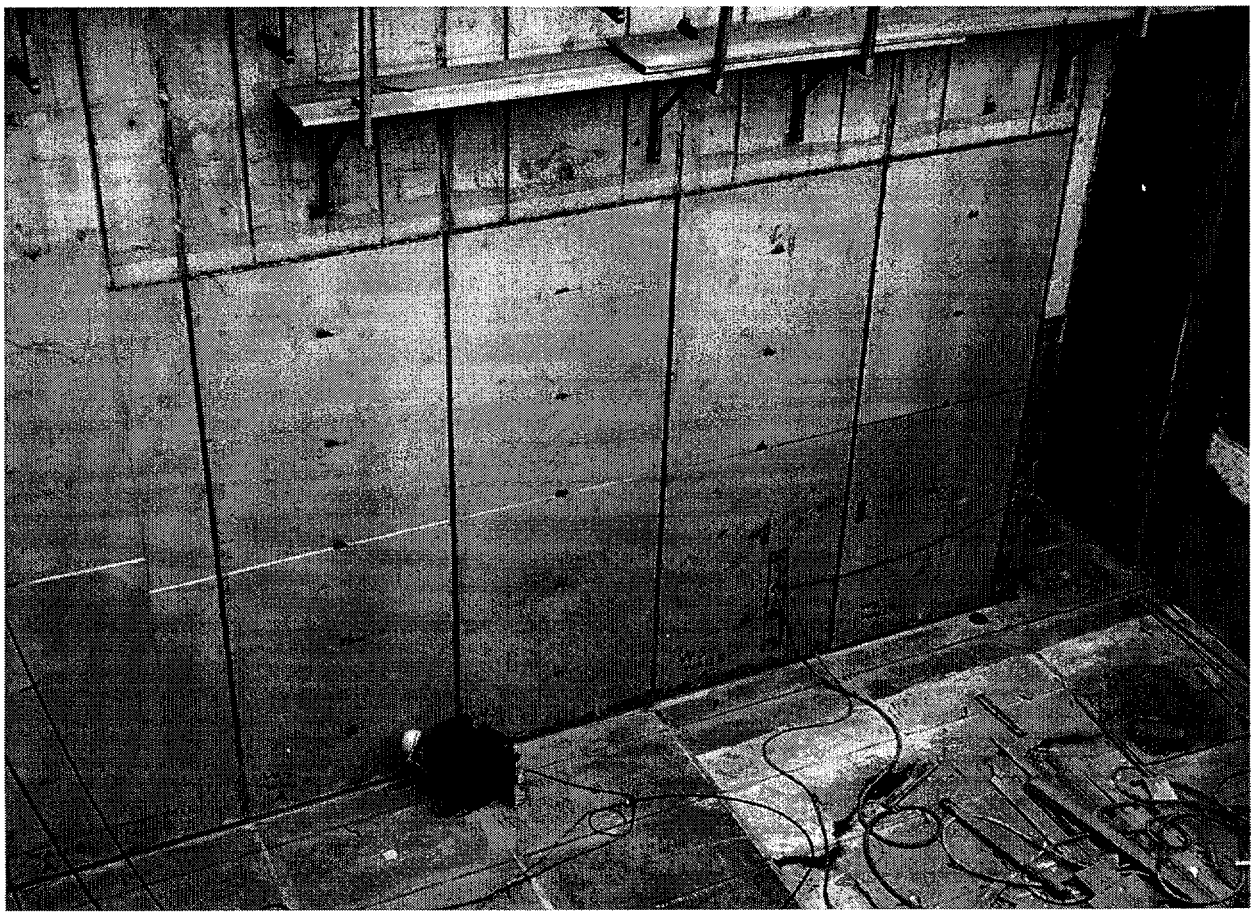


Figure 5-12. Stainless Steel Basin Liner. Both storage basins and the unloading basin are completely lined with stainless steel sheets (304L) placed flush against the concrete walls and floor, welded to a grid of stainless steel embedments.

Large liner sheets (generally on the order of 6 ft. by 16 ft.) were welded continuously along each edge to the gridwork of back-up bars and also were slot welded to embedded plates at intermediate locations so the liner is held against the concrete wall to reduce potential for puncture damage. To facilitate fit-up and ensure high integrity, liner sheets were welded to embedded stainless steel angles at wall-to-wall and floor-to-wall joints. The liner terminates on a stainless steel angle at the top of the basin. Specifications for liner installation include approved joint design welding procedures and welder qualification requirements. All welds were visually inspected and vacuum box tested to ensure leak tightness[13]. Final verification of liner integrity was provided during basin filling.

Because of the nonaggressive basin liner service environment, corrosion testing of 304L liner sheet steel was not required. However, a substantial quantity of 304L sheet steel material was subjected to corrosion tests, with few lots exceeding the 0.003 in. per month in Huey Tests as

specified in ASTM 262. Many rates were lower than 0.001 in. per month with no evidence of pitting or cracking.

The specified Huey Test is based on exposure to 65% HNO_3 at boiling temperatures whereas actual service is in neutral demineralized water at about 80 °F average temperature (maximum of 120 °F). In demineralized water at the lower temperatures, it is estimated that corrosion rates are reduced by a factor of more than 1,000 relative to those observed in accelerated tests. On this basis, a conservative rate of in-service liner corrosion would be 0.003 in. per month. For the thinnest (upper basin wall) liner, a 50% reduction in thickness from "one-side" corrosion at such a rate would require 10,000 months.

Basin liner corrosion, to the extent that it occurs, is expected to be a general attack with essentially no effects from galvanic corrosion. System pH is controlled and metal ions present in the system are minimized by use of demineralized water. Water purity is maintained by circulating basin water through a filtration and ion-exchange cleanup system.

5.5.1.4 Basin Liner Leakage Control

To facilitate drainage of water from between the concrete structure and the stainless steel liner (water that may seep in through the concrete as well as any liner leakage), 0.5 in. square drain slots on approximately 3 ft. centers are provided in concrete basin walls and floors back of the liner. These lead to a 1 in. square collection header located behind the floor-to-wall joint at the basin perimeter, which drains to a single sump at the bottom of the cask unloading pit. The sump consists of a 6 in. diameter vertical pipe embedded in the west (exterior) wall of the unloading pit, extending above water level to a point approximately 1 ft. below floor level and connecting to the perimeter collection header.

The sump contains a liquid level detector line and necessary piping for a removal system. Auxiliaries for the level detection and removal system are located in the basin pump room. The removal system employs an air-lift working in conjunction with an air operated pump. Operation of sump equipment has met design requirements.

5.5.1.5 Earthquake and Tornado Analyses

The basins were designed for earthquake and tornado conditions in accordance with criteria presented in Section 4. Earthquake, tornado and missile analyses are contained in Appendix B. Although much of the building is unused and not relevant to fuel storage, the structure does form a portion of the basin area east wall, as well as containing the ventilation tunnel, control room (SAS), and other support functions.

5.5.2 Basin Water Clean-Up System

The interconnected basins are supplied with demineralized water from the on-site well and water treatment facilities. These facilities include prefilters to control organic material in

incoming water. The basin water clean-up system includes a suction system for underwater "vacuum cleaning" and a resin-precoat filter system with associated equipment.

The purpose of the basin water treatment system is to maintain water clarity and quality, minimizing concentration of radioactive materials in the water. Basin water clarity is maintained such that objects at the bottom of the storage basin are visible from the pool surface with or without optical devices at the surface[14]. Radioactive material in basin water originating from fuel element surfaces and leakage from defective fuel rods is controlled to ensure that radiation and contamination levels are ALARA. Basin water quality is controlled to prevent potential corrosion attack and stress corrosion cracking of system components.

5.5.2.1 Water Quality and Characteristics

Water added to the basin has a minimum quality of 40,000 ohm-cm (≤ 25 μmho) with pH > 5.0 and < 9.0. Based on the operating experience of various reactors, and storage pools, conductivity and pH are the most important water quality indicators and are the only indicators of water quality commonly measured at such facilities.

Based on operating experience, factors of turbidity and organic material are not considered to be as important as conductivity and pH. Turbidity would present a temporarily inconvenient operating condition that would be remedied by adjusting filter media or procedures. Control of organic material by prefilters is considered adequate to maintain this contamination below acceptable limits.

5.5.2.2 Radioactive Materials in Basin Water

Experience with fuel storage at GE-MO indicates the concentration of radioactive material in the basin water is typically 2.6×10^{-5} and 4.2×10^{-4} μCi/ml for radiocobalt and radiocesium, respectively.

Principal radioactive contaminants in the storage basin water include fission products Cesium-134 and Cesium-137 with typical concentrations of 3.3×10^{-7} and 4.2×10^{-4} μCi/ml, respectively. Activation product Cobalt-60 is present in a typical concentration of 2.6×10^{-5} μCi/ml. A maximum concentration of 5×10^{-3} μCi/ml was measured at the end of a 3-week period during which the filter was purposely not operated. Similar levels of contamination have occurred in recent years.

Since removal mechanisms and relative proportions of the two principal contaminants differ, operational controls for the basin are based on exposure levels. Technical Specifications include two limits on concentration of radioactive material in basin water for which special corrective actions are required. If the gross β concentration reaches 0.02 μCi/ml a cleanup campaign will be initiated. Should the gross β concentration reach 0.1 μCi/ml, normal basin operations will be interrupted and water cleanup performed on an emergency basis.

5.5.2.3 Basin Water Filter System

The filter system maintains water clarity and removes dissolved materials. A 250 gpm pump delivers water from the skimmers or vacuum hoses to the coated tube filter (a 115 square-foot DeLaval unit, about 2.5 ft. in diameter by 6 ft.) and back to the basin. The filter is precoated with Solka Floc, a cellulose filter base. This base can be overlain with diatomaceous earth, Powdex resins, or Zeolon as desired. Sludge from the filter is collected in a small tank (approximately 600 gal.) and ultimately transferred to the Radwaste System.

The basin clean-up filter is housed in a heavily shielded, restricted access room with electric lock entry control. A Special Work Permit (SWP) is required for entry. The filter is changed remotely by a programmed controller (Figure 5-15) that flushes filter media from the filter septums into the sludge tank. Therefore, personnel are not routinely exposed to radioactivity accumulated in the unit.

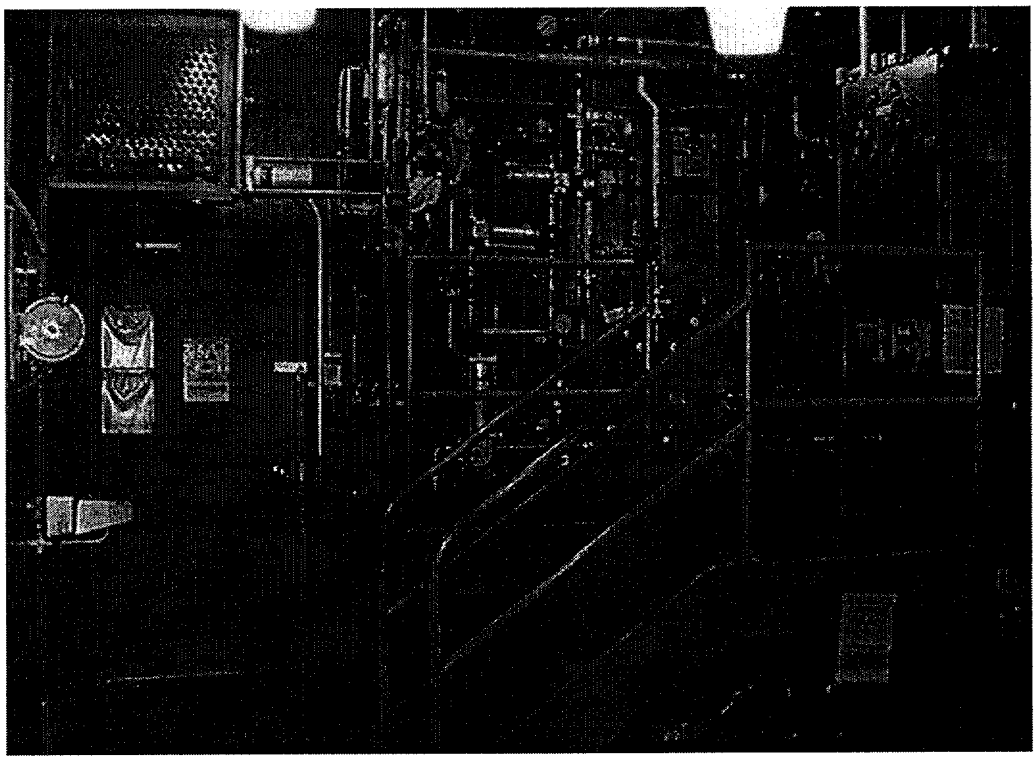


Figure 5-15. Basin Filter Controls: View shows basin filter programmed controller and associated instruments and piping. Filter is housed in shielded room behind locked door to the left.

An inherent advantage of the Powdex system is the ability to accommodate a variety of water purifiers. For example, a quantity of Zeolon[15], a synthetic alumino-silicate molecular sieve having a high affinity for cesium, may be added to the normal recharge used for the Powdex system. In practice, two kilograms of Zeolon-100 are added with the mixed cation-anion Powdex resins during filter make-up. The zeolite acts as a true ion exchanger and, under clean

basin water conditions, partitions radiocesium so that about 90% is sorbed by Zeolon and 10% remains in the water. This partitioning ratio remains constant irrespective of the radiocesium transfer rate (from fuel to basin water) since chemical concentration levels in the water do not measurably exhaust the chemical exchange capacity of the ion exchanger[16].

5.5.2.4 Safety Evaluation

Failure of the basin water treatment system is not critical to safety of the fuel storage system. Redundant or spare filters are not required. The system has been out of service for several weeks without marked deterioration of basin water quality. Typical basin water isotope concentration levels are shown in Table 5-1. Isotope concentrations vary, depending upon rate of addition of fuel to the basin and method of operation of the basin filter.

Data in Table 5-1 indicate that the activity levels in basin water do not contribute significantly to personnel exposure. There is little accumulation of contamination on the basin liner at waterline.

Table 5-1
TYPICAL ISOTOPE CONCENTRATIONS IN BASIN WATER

Isotope	Typical Concentration[a] (μCi/ml)
Cs-134	3.3×10^{-7}
Cs-137	4.2×10^{-4}
Co-60	2.6×10^{-5}
H-3	1.1×10^{-4}

[a] The concentration of other radionuclides which are low-energy beta emitters is less than the total radiocesium and cobalt. In terms of radiotoxicity they are insignificant compared with cesium and cobalt.

5.5.3 Basin Water Cooling System[17]

The heat load as of October 1996 is about 1×10^6 BTU/hr. At this point in the fuel and fission product decay cycle, the heat load should decrease about ten per cent each two years.

5.5.3.1 Equipment Description

Basin water heat dissipation is accomplished through the use of 2 parallel heat pumps and heat exchangers each forming a closed loop. Typically, one unit has the capacity to maintain the basin water temperature and the second unit will function as a back-up (Figure 1-19). Historically, basin water temperature has been maintained under 40 °C with typical temperatures less than 35 °C.

5.5.3.2 Safety Evaluation

Failure of the basin cooling system is not critical to safety of the fuel storage system. In the event that both heat exchanger units should fail, there is enough time to supply make-up water to the basin while the cooling system is repaired or replaced. If both heat exchanger units were inoperative and the storage basins were full of fuel with 44,000 MWd/TeU and a cooling time of 120 days, the temperature of the basin water would slowly rise (<2 °F/hr) and approach an equilibrium temperature of about 183 °F assuming the heat input is 6.4×10^6 BTU/hr. See Appendix A.9.3.

Potential leaks in the cooling system that could occur as a result of an accident have been analyzed and results are given in Section 8. It was concluded that the consequence of a leak in the system is insignificant[18]. Coolers are periodically inspected for leaks. Accumulation of radioactive contaminants in the cooling system components is monitored, and the system decontaminated when required (Section 7.3.2.3).

5.5.4 Ventilation Exhaust System

Facilities provided for filtration, monitoring, and release of effluent air are described in following sections (Figure 1-22 and Appendix A.14). Discussion of radioactive contaminants in effluent air is contained in Section 7.

5.5.4.1 Air Tunnel

A below-grade reinforced concrete tunnel runs the east-west length of the main building along its north wall. The tunnel was originally intended to collect all building ventilation exhaust air (via ducts from various cells, hoods, etc.) for routing to the ventilation exhaust filter. The rectangular cross section of the tunnel is on the order of 20 square feet, increasing in area toward its outlet at the sand filter. A 3 in. deep stainless steel floor pan is provided for collection of condensate. The floor slopes toward a collection point (41.5 ft. elevation) from which a drain line is routed to the off-gas cell sump. Instrument ports are located near the tunnel outlet for radiation off-gas monitors. Provisions are made for future extension of the tunnel to an additional sand filter, if ever required. Air from the basin area is drawn into the air tunnel via the canyon area.

5.5.4.2 Ventilation Exhaust Filter

A reinforced concrete structure, 75 ft. by 80 ft. in plan and 15 ft. in height, houses the low-velocity, upward flow sand filter through which effluent air is drawn before discharge from the stack. It is located immediately east of the main building and is connected to it by an underground extension of the ventilation air tunnel. The tunnel extension leads to a central air distribution duct at floor level (about 40 ft. reference elevation) of the filter structure.

The filter bed is about 8 ft. deep and is comprised of layers of graded gravel and sand. Openings are provided in the central duct to distribute incoming air laterally through the gravel bed which forms the bottom layer of the filter. The floor is sloped for positive drainage back through the air tunnel to the off-gas cell sump. Outlet from the upper, open portion of the filter structure is through ports in the west wall leading to an adjacent reinforced concrete structure (the equipment building) 24 ft. by 80 ft. in plan, housing exhaust blowers as well as a diesel-electric generator and associated switchgear, effluent air sampling system and two air compressors. This arrangement places all equipment and auxiliaries essential to exhaust system operation within reinforced concrete structures for protection against earthquake and tornado conditions.

5.5.4.3 Emergency **Equipment Building** (EEB)

The EEB is divided into three rooms:

a. Fan Room: Exhaust blowers are located in an area, 19 ft. by 35 ft. in plan, with a grade level concrete slab floor. Inlet ducting for the blowers connects directly to openings in the filter enclosure wall. Each blower unit consists of an electric motor and fan capable of providing 13,000 cfm of flow at 6 in. of water pressure differential. Normal system configuration is one unit operating with the second available for back up use. Other equipment in the fan room includes the system for continuous sampling of air entering the sand filter from the main building, and a sampling system for air being routed to the stack.

b. Compressor Room and Compressed Air System: Two air compressors, the primary air receiver and the dual bed air dryer are located in this area of the equipment building. Failure of the compressed air system is not critical to safety of the fuel storage system. The system is discussed here because it does perform some auxiliary non-safety related function involved with fuel storage, and are located in the emergency equipment building.

   Instrument air is supplied from the receiver to drying equipment in the equipment building from which it is delivered to an instrument air receiver in the main building. The air is used for instruments and air operated valves. The instrument air system is served preferentially upon loss of compressor air supplies to the main receiver; low pressure in the main receiver automatically valves off the process air system.

c. Generator Room[19]: (Not essential to fuel storage activities and discussed here because it is located in the emergency equipment building.) The remaining area of the filter building is 21 ft. by 23 ft. in plan and houses the diesel generator and auxiliaries. The diesel-driven 400 kVA unit is designed for automatic startup upon total loss of commercial power and is provided with both battery and air-pressure starting systems. Battery racks, with continuous charger, and two air bottles for the starting systems are located in the generator room as is all switchgear required for the secondary power supply system. A 1,000 gal diesel fuel tank is located adjacent to the generator room and both electrically driven and manual pumps are provided for transferring fuel from the storage tank to the 33 gallon day-

tank located in the generator room. The radiator for the diesel engine is mounted in a wall opening and is provided with a heavy grill for protection against wind borne missiles. This opening is in the west wall of the equipment enclosure and faces the main building, the east wall of which is about 30 ft. away, so that some additional protection against damage from wind borne missiles is provided.

5.5.4.4 Effluent Air Release

A 4 ft. diameter reinforced concrete pipe is provided for routing air from exhaust blowers to the main stack which is located approximately 350 ft. south of the sand filter and equipment building. The pipe is essentially at grade level and has a protective earth covering. It is equipped with a covered instrument enclosure to house monitoring equipment and a bolted-cover manhole to provide an alternative release point in the event flow through the stack is blocked due to stack failure. A stainless steel drain line is provided for routing condensate to the stack condensate collection system. The main stack is an all-welded steel unit which reaches a height of about 300 ft. (91.4 m.) above grade, and is supported on a reinforced concrete foundation by external cable guys. It is provided with an inner stainless steel liner. A spray nozzle system is located in the upper part of the stack for washdown purposes.

5.5.4.5 Earthquake and Tornado Protection

Provisions of earthquake and tornado protection for sand filter, exhaust duct, and stack are in accordance with design criteria and requirements stated in Section 4; also see Appendix B.

Earthquake and wind analysis of the main stack defines design wind velocity at 110 mph. This value is in accordance with Uniform Building Code recommendations and established engineering practice. Based on this velocity, the stack is capable of withstanding wind impressed loads and forces. Within the context of stack design the term "extreme conditions" is defined as conditions greater than design wind velocities. The stack is located sufficiently distant from other facilities so that structural failure would not result in damage to any fuel storage systems or structures. The earth-covered duct between exhaust fan enclosure and stack is provided with a port that can be opened to permit grade-level release of ventilation air in the unlikely event that structural failure resulted in severe restriction of stack flow.

5.5.5 Main (Process) Building Facilities

The main building contains certain facilities other than fuel basin areas that are directly or indirectly involved in fuel storage. Some of these have been discussed in preceding sections, such as the ventilation tunnel which extends almost the length of the building, passing underground to the ventilation filter building and servicing the Radwaste System. See illustrations, Appendix A.14.

5.5.5.1 Building Entrance Area

The main building entrance door, vestibule and lobby are located near the midpoint of the south service area, essentially at grade level. Between the gallery exterior wall and corridor which parallels the south canyon wall at this point are rest rooms, change room, shower room, and decontamination room required for control of personnel access to and exit from potentially contaminated areas of the main building. The corridor which services the change room complex leads to the mechanical cell operating gallery and fuel storage basin.

5.5.5.2 Gallery Area

Adjacent to the process canyon and structurally attached thereto are multi-level galleries which allow personnel access to the main building. The galleries extend the full length of the process canyon on the north and part way on the south sides and are connected by transverse corridors at the east end of the building. The gallery structure is of steel frame with reinforced concrete floors, walls and roof areas as was required for protection of equipment and functions under extreme conditions including tornado-generated missiles. Access to limited occupancy zones is provided by locked doors. Air locks are provided at major access points as required to maintain differential air pressure control during movement between ventilation zones.

5.5.5.4 Control Room, or Secondary Alarm Station (SAS)

The Control Room (or SAS), is located in the south gallery area intermediate level (65 ft. floor elevation). The room is about 75 ft. by 21 ft. in plan, with direct stairway access to the building lobby and secondary access to the unused computer room. Principal items of control room equipment include the main process control panel across one side of the room, and various monitoring equipment. Fuel storage functions monitored in the control room are listed in Table 5-2. Although some functions are normally controlled only from the control room (e.g., basin cooler pump and fan controls and well-water pump control), the noncritical nature of all control systems permits replacing controls with local control. The control room (SAS) is one of two Alarm Stations (other is in the Administration Building (CAS)). At least one (CAS or SAS) is continually staffed.

Table 5-2

CENTRAL ALARM STATION MONITORING OF FUEL STORAGE FUNCTIONS

The following functions are monitored in either the Control Room (SAS) or CAS..

<u>BASIN SYSTEMS</u>

- Filter System
  - Sludge Tank Level Indicator and Alarm
  - Filter Differential Pressure

- Water Chillers
  - Basin Cooling Unit CU102-8 Shutdown
  - Basin Cooling Unit CU102-9 Shutdown
  - Chiller Inlet Temperature
  - Chiller Outlet Temperature

- Basin Water
  - Water Temperature
  - Water Level Alarms
  - Leak Detection and Alarm
  - Water Addition Control and Measurement*

COMMUNICATIONS

- Radio Off, On Site
- Telephone
- Intercom - Public Address

SECURITY SYSTEMS

- Closed Circuit TV Systems
  - Main Gate Monitor
  - Basin Entry and Exit Monitors
  - Basin Area Monitors

VENTILATION SYSTEM

- Intake Plenum
  - Pressure, Temperature Indicators and Alarm
  - Controls and Indicators*

- Exhaust Plenum
  - Controls and Indicators*

- Stack
  - Air Flow
  - Sampler Pump Run Controls, Indicators and Alarms

STORAGE VAULTS

- Cladding Vault
  - Leak Detection Indication and Alarm

*Control Room Operation - Local Lockout Capability

- Low Activity Waste (LAW) Vault
  Intrusion System Indication and Alarm

- Dry Chemical Vault (DCV)
  Intrusion Detection and Alarm

UTILITY SYSTEMS

- Air Systems
  Pressure Indication and Alarms
  Compressor Run Controls*

- Water Systems
  Well Water Pump Run Controls*
  Water Tower Temperature Alarms and Indicators
  Demineralized Water Indicator and Alarm
  Utility Cooling Water Run Controls, Indicator and Alarms

ELECTRICAL SYSTEMS

- Diesel Generator
  Instrumentation, Indicators and Alarms

- Power Bus
  Indicators and Alarms
  Ground Faults and Malfunctions

HEAT PUMP SYSTEM

- Temperature, Flow, Condition Alarms and Indicators

SUMPS

- Basin Pump Room and Addition
  Alarms
- Hydraulic Equipment Room
  Alarms
- Canyon Areas
  Indicators for Decon Cell, Off Gas Cell and Mechanical Cell sumps.

*Control Room Operation - Local Lockout Capability

RADWASTE SYSTEM

- Evaporator Malfunction Alarm
- Tank Level Alarms
- High Filter Differential Pressure Alarms

MISCELLANEOUS

- Protected Area Door Controls and Indicators
- Evacuation, Take Cover Alarm Controls
- Fire Alarm Panel and Smoke Detectors
- Area Radiation Monitor (ARM) Indicators and Annunciators
- Criticality Alarm Indicators, Annunciator and Controls

*Control Room Operation - Local Lockout Capability

5.5.5.5 Off-Gas Cell

Process off-gas treatment facilities are located in the off-gas cell. It is roughly ell shaped, occupying the south side of the canyon opposite the anion exchange cell and spanning the full width of the canyon (19 ft.) at its east end. The cell floor is lined with stainless steel which extends up the cell walls to 3 ft. above the floor level. The lined sump is equipped for pumping collected liquids to the Radwaste System.

A vertical ventilation panel is provided near the canyon centerline to span the opening between the northside cell cover (42 ft. above the cell floor) and the southside cover (10 ft. lower). There are three equipment positions in the 19 ft. south wall of the cell. At one position is located the old low activity waste evaporator. Other equipment (process vent scrubber, etc.) is unused and not involved in fuel storage systems.

5.5.5.6 Radwaste System Evaporator

The new Radwaste System Evaporator is electrically heated. It is accessed through the PuNp Load Out Area on the 48' elevation in the Process Building. Evaporator bottoms are periodically transferred to steel barrels and stored in the UF^6 Room for subsequent shipment for treatment and subsequent burial. Steam vapor from the evaporator is demisted and routed to the air tunnel, then through the sand filter to the stack.

5.5.5.7 Ventilation Supply Room

Blowers and associated equipment to supply main building ventilation air are located in a room, 39 ft. x 21 ft. in plan, on the top floor of the south gallery area of the main building (81 ft. reference elevation). Personnel access is from the Control Room by way of the Computer Room and emergency power room with additional access from the Instrument Gallery. Two

hooded air intake openings are provided in the reinforced concrete roof (elevation approximately 93 ft.). Air conditioning system coolers also are located on the south gallery roof.

5.5.5.8 Basin Pump Room Addition (BPRA)

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

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

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

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

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

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

5.5.5.9 Basin Chiller Room

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

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

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

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

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

5.5.5.10 Electro-Decontamination Room (EDR)

The former ultrasonic cleaning area - Figure A.14-1, was modified for use as an electro-decontamination (EDR) (electro-polishing) facility for parts, assemblies, tools and fixtures. The facility is used to process items which have been previously cleaned by mechanical and/or chemical methods. Electro-polishing is then used to remove fixed or occluded contamination.

The room is equipped with ventilation systems, utilities, and other features required for electro-polishing. Walls and floors are coated with a non-permeable material to prevent contamination of concrete. Tanks are fabricated from stainless steel. The electro-polishing tank contains a phosphoric acid solution which can be heated to between 40 °C and 80 °C. Low watt/density electric heaters ensure slow heating of the acid solution. This tank and the first rinse tank are equipped with rear mounted lateral vent hoods, and a hood is provided for maintenance use.

An air cooled rectifier system provides direct current densities of 50 to 250 Amp/ft^2 at 8 to 12 volts. The rectifier has a capacity of 2,500 ampere, with current and voltage limiting and automatic shutdown within one half Hertz after a short circuit.

A safety shower and eye bath is a part of the facility.

5.6 WASTE VAULTS

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

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

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

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

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

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

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

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

5.6.1 Cladding Vault

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

5.6.2.1 Cladding Vault Construction

The cladding vault is constructed of reinforced concrete about 2 feet thick, and is lined with stainless steel. The top of its 2.5 ft. thick reinforced concrete cover is at 41.5 ft. elevation. The vault is located adjacent to the LAW vault on the south side of the main building (Figure 1-4). It

is connected to the mechanical cell in the canyon by a reinforced concrete waste disposal cart tunnel (top about at grade level) which extends across the top of the vault to a 235 sq. ft. cart equipment pit. The pit roof has two access openings with shield plugs. The vault is equipped with leak detection and sampling systems similar to those for the fuel storage basins, with level recorder and unit alarm in the control room (SAS) and local control in the mechanical cell operating area.

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

5.6.2.2 Cladding Vault Description

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

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

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

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

located in the mechanical cell gallery of the main building. Water from the pump-out system is routed to the Radwaste System.

5.7 SUPPORT FACILITIES

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

5.7.1 Utility and Service Building

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

5.7.1.1 Utility Section

The 1,700 sq. ft. utility section of the building is divided into two major rooms, the larger of which houses water demineralization and three smaller room partitions for training, an operations ready room, and a first aid room. The demineralizer system consists of ion exchange resin provided by a contact service. It is capable of treating 25 gpm continuously. Pumps required for operation and distribution are located nearby and a 1,000 gal. demineralized water surge tank is mounted on an overhead platform in the room.

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

5.7.1.2 Outside Facilities

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

a. A chain link fence surrounds a rectangular area 62 ft. by 30 ft. in size located on the east side of the building, and encloses the terminal structure of two 34,000 volt incoming overhead transmission lines and two CECo owned 1,500 kVA transformers which reduce

the incoming supply to 480V. The fenced area is locked to preclude accidental access to high voltage equipment.

5.8 UTILITY SYSTEMS

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

5.8.1 Water Supply

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

The pedestal is equipped to use a gas heater to prevent freezing of water in the sphere.

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

5.8.1.1 Utility Water Supply

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

5.8.1.2 Demineralized Water Supply

Demineralized water is used for fuel storage basin supply. This water is supplied from the series cation-anion demineralizer located in the utility building which is capable of treating 25 gpm continuously (50 gpm instantaneously) from the utility water supply system. Distribution to points of use is via a pump-pressurized header system. There is a 1 in. line to the basin to furnish make-up water. Basin water level is maintained under manual control of the basin operator, who would normally add water when basin water level dropped 2 in., which is low enough to affect basin cleanup system operation. A back-up-low-level alarm in the CAS/SAS activates if basin water level drops 6 in. below normal.

5.8.1.3 Fire-Fighting Water Supply

Potable and utility water usage is limited by location of outlet piping to the topmost 8,000 gal. of water sphere capacity, with the remaining 42,000 gal. reserved for fire protection. Distribution is

via a standard underground piping system located beneath historical frost penetration in accordance with underwriter and building codes.

5.8.1.4 Backup Water Supplies

Parallel fuel storage basin pumps and heat exchangers reduce the likelihood of complete loss of basin cooling capability. In the highly unlikely event that cooling system capability could not be restored within 30 hours (or more, depending on circumstances), makeup can be provided from demineralized or utility water storage or from other emergency sources, including water pumped from the DNPS cooling lake, or even from the river[22]. Emergency pumping equipment could be brought to the site and placed in operation within the 30 hr. period with no impact on public health or safety from stored fuel.

5.8.2 Electrical Supply

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

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

5.8.2.1 Normal Electrical Power Source

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

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

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

5.8.2.2 Essential Services Power Facilities

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

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

5.8.2.3 Distribution System

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

5.8.2.4 Operating Characteristics

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

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

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

a. The standby diesel generator will start.

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

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

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

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

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

5.8.3 Site Natural Gas Supply

Low pressure natural gas for site use is supplied from the underground main located in an easement at the north edge of the GE property, along the east-west county road. The 2 in. supply line from the main terminates within the plant security area at a meter owned by Northern Illinois Gas Company, which is mounted on a concrete slab located between the general warehouse and elevated watersphere (Figure 1-4). From the meter station, underground distribution lines are routed to the utility and service building, water tower boiler system, main building, CSF and CRA for space heating.

5.8.4 Sewer Systems

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

5.8.5 Rail Transportation Facilities

Rail service to the site is provided by a spur track from the DNPS siding, approximately 0.5 mile north of the plant site, which connects to the Elgin, Joliet and Eastern right-of-way serving the general area. The spur track is designed to carry heavy cask car loads at low speed (ASCE 100 lb. rails, appropriately limited curves and grades). After crossing the county road, the track is divided into three spurs and enters the OCA.

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

5.9 ITEMS REQUIRING FURTHER DEVELOPMENT

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

5.10 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

15. Proprietary product of Norton Co.

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

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

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

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

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

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

6.0 WASTE MANAGEMENT

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

6.1 UNDERGROUND WASTE VAULTS

6.1.1 Dry Chemical Vault

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

6.1.2 Low Activity Waste (LAW) Vault

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

6.1.3 Cladding Vault

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

6.2 RADWASTE SYSTEM

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

6.3 SOLID RADIOACTIVE WASTE

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

6.4 NONRADIOACTIVE WASTE

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

7.0 RADIATION PROTECTION

7.1 INTRODUCTION

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

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

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

7.3 RADIATION SOURCES

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

7.3.1 Irradiated Fuel

General characteristics of irradiated fuel are given in Section 4. However, for purposes of estimating dose rates, calculations are based on parameters that more realistically reflect fuel in storage or expected to be stored. Although most fuel currently in storage has cooled much longer than a year (10 to 26 years), and has an average exposure of about 20,000 MWd/TeU, it is conservatively assumed that all fuel in the basin has the following characteristics for radiation protection calculations under normal operation:

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

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

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

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

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

Calculations are performed in Section 7.4.2.1.

7.3.2 Storage Basin Water

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

Shipment to GE-MO of fuel known to be defective and leaking is not permitted, unless the fuel is canned or otherwise contained.

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

		Activity (Ci/TeU)[a]		
Isotope	Half-Life	1 Year	5 Years	10 Years
Kr-85	10.701y	7.62 E3	5.88 E3	4.25E3
Rb-86	18.82d	6.93 E-4	0	0
Sr-89	50.55d	9.41 E3	1.88 E-5	0
Sr-90	28.82y	6.47 E4	5.88 E4	5.21E4
Y-90	64.06h	6.48 E4	5.87 E-4	5.21E4
Y-91	59d	2.08 E4	6.33 E-4	2.54E-13
Zr-93	1.53E6y	2.31	2.31	2.31
Zr-95	63.98d	4.15 E4	5.55E-3	1.42E-11
Nb-95m	86.6h	5.27 E2	0	0
Nb-95	354.97	8.78 E4	2.33 E-8	0
Ru-103	39.35d	2.68 E3	1.78E-8	0
Ru-106	366.5y	1.72 E5	1.09 E4	3.43E2
Rh-103m	56.116m	2.68 E3	0	0
Rh-106	29.8s	1.72 E5	0	0
Ag-110m	252.2d	1.23 E4	2.22 E2	1.47
Ag-111	7.5d	8.53 E-11	0	0
Cd-115m	44.8d	3.49	5.32 E-10	0
Sn-119m	250d	2.64 E1	4.6 E-1	2.91E-3
Sn-121m	55y	9.12 E-2	8.67 E-2	8.14E-2
Sn-123	129d	6.13 E2	2.39 E-2	1.31E-5

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

Isotope	Half Life	Activity (Ci/TeU)[a] 1 Year	5 Years	10 Years
Sn-125	9.625d	3.93 E-8	0	0
Sb-124	60.2d	3.23	1.61 E-7	0
Sb-125	2.71y	4.84 E3	1.74 E3	4.84E2
Sb-126	12.4d	4.74 E-2	0	0
Te-125m	58d	1.18 E3	3.08 E5	1.02E-14
Te-127m	109d	1.32 E3	1.22 E-1	1.1E-6
Te-127	9.35h	1.30 E3	0	0
Te-129m	33.52d	4.31 E1	3.00 E-12	0
Te-129	69.5m	2.74 E1	1.90 E-12	0
I-129	1.57E7y	2.1 E-2	2.1 E-2	2.1E-2
I-131	8.04d	2.37 E-8	0	0
Xe-131m	11.77d	1.59 E-5	0	0
Xe-133	5.245d	4.71 E-15	0	0
Cs-134	2.062y	8.89 E4	2.32 E4	4.32E3
Cs-136	13d	1.48 E-4	0	0
Cs-137	30.174y	7.79 E4	7.11 E4	6.34E4
Ba-137m	2.5513m	7.34 E4	6.72 E4	6E4
Ba-140	12.789d	5.13 E-3	0	0
La-140	40.27h	5.90 E-3	0	0
Ce-141	32.55d	8.00 E2	2.46 E-11	0
Ce-144	284.5d	5.30 E5	1.51 E4	1.76E2
Pr-143	13.59d	1.56 E-2	0	0
Pr-144	17.3m	5.30 E5	1.51 E4	1.76E2
Nd-147	10.98.1d	7.69 E-5	0	0
Pm-147	2.62344y	1.04 E5	3.61 E4	9.65E3
Pm-148m	41.29d	9.45 E1	2.11 E-9	0
Pm-148	5.37d	6.51	1.48 E-10	0
Sm-151	87y	9.36 E2	9.07 E2	8.71E2
Eu-154	8.59y	4.39 E3	3.17 E3	2.11E3
Eu-155	4.96y	1.02 E3	5.83 E2	2.9E2
Eu-156	15.11d	6.64 E-3	0	0
Tb-160	72.1d	1.66 E1	1.32 E-5	3.13E-13
All Remaining Fission Products		1.175 E1	1.169 E1	1.163E1

[a] N is the power of 10, from the expression EN, by which the number must be multiplied; e.g., 4.047 E3 = 4.047×10^3 and 5.13 E-3 = 5.13×10^{-3}.

Table 7-2

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

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

Cask flush water (if applicable) may be sampled upon receipt to detect fuel damaged in transit (or damaged fuel not identified prior to shipment). If damaged fuel should be discovered special handling procedures will be followed during unloading operations. Because of passive storage conditions, if any defects occurred during storage they would likely be minor perforations (or "pin holes") in the fuel cladding rather than gross cladding failure.

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

7.3.2.1 History of Radioactive Material Concentration

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

7.3.2.2 Contaminants

The principal dissolved radioactive contaminant in basin water is Cesium-137 with concentrations (typically now 4.14×10^{-4} μCi/ml) ranging up to 1.2×10^{-3} μCi/ml. A means of cesium removal has been found that makes reduction and control of this contaminant relatively simple. For example, over a 10-week period in 1974, radiocesium concentration was reduced to one-third of that at the beginning of the period. The basin water inventory was correspondingly reduced from about 29 Ci to 11 Ci. In 1975, during a 4-week period, the radiocesium concentration was reduced by a factor of six and the basin water inventory reduced from 14 Ci to 2.3 Ci. At the end of the latter period, the radiocesium concentration was 9×10^{-4} μCi/ml.

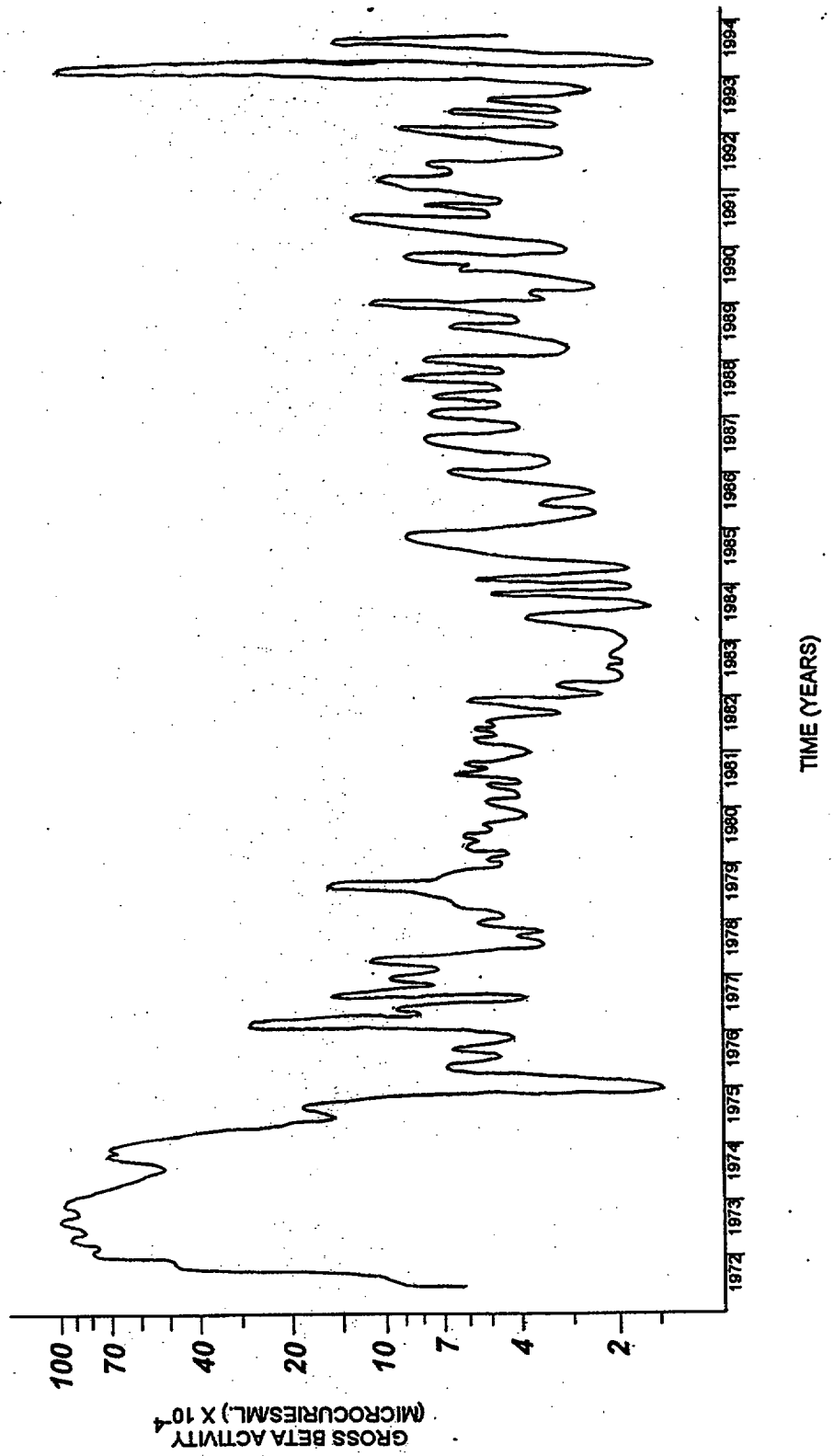


Figure 7-1. History of Morris Operation Basin Water Activity.

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

In addition to radiocesium, the radionuclide contributing most significantly to basin water contamination is cobalt-60. Concentrations in basin water (typically now 2.58 x 10^{-5} µCi/ml) are attributed to corrosion products on fuel bundle surfaces released to water. Normal filtration and ion exchange reduces cobalt concentrations without special effort.

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

7.3.2.3 Basin Chiller Decontamination

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

7.3.3 Airborne Radioactive Material Sources

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

a. Effluent from the Radwaste System evaporator

b. Vented gasses from shipping casks

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

d. Decontamination activities

e. Uranium used in MFRP testing

Although release of radioactive material in the demisted effluent from the evaporator is possible, such occurrences would be rare and the amount released would be very small. Casks are vented to ventilation air which is exhausted to the air tunnel and sand filter. In more than 27 years of fuel storage experience, there has been no apparent leaking of gases from stored fuel. Incidental airborne contamination from decontamination activities could occur. Use of special enclosures ("greenhouses") and other techniques limit such releases to very small amounts, and these activities are infrequent. Natural uranium was used in MFRP testing and some

contamination is present within the canyon that could become dislodged and subsequently exhausted via the air tunnel.

Actual measurements of particulate radioactive materials in air exhausted via the stack are made routinely at GE-MO. In 1999 less than 3.18 µCi of beta emitting nuclides were released. The resulting dose to the public was less than 2.9×10^{-6} mRem.

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

Annual basin air samples indicate fuel basin Kr-85 source term is about 0.8 Ci per year[5]. Kr-85 could result from cask venting in the unlikely event fuel undergoes cladding failure in transit.

This phenomenon is not considered viable since the fuel storage basin is near capacity.

7.4 RADIATION PROTECTION DESIGN FEATURES

7.4.1 Facility Design Features

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

GE-MO has provisions for controlling personnel access to areas of the plant having actual or potential levels of radiation or radioactive contamination that exceed levels specified in plant procedures. There is little potential for high radiation dose rates or contamination levels in most areas. However, since venting and flushing of arriving casks has potential for releasing Kr-85 to a small air stream (approximately 700 cfm) during a brief period of time, personnel are excluded from relatively high-potential-level areas such as the ventilation air tunnel while arriving casks are vented and flushed.

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

Basic procedures and criteria for controlling personnel exposures are specified in the GE-Morris Operation Environmental, Safety and Health Plan (ESHP). Programs adopted for controlling radiation exposure are the result of previous experience at other installations. The program uses modern equipment and techniques proven effective for control of exposures. Such an approach has effectively maintained exposures well below 10 CFR 20 limits.

7.4.2 Shielding

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

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

7.4.2.1 Estimated Direct Radiation from Fuel in Storage

The direct radiation dose rate from fuel in storage has been calculated. Results are compared with actual measurements of dose rates. Calculations are based on the following assumptions:

a. 500 metric tons of irradiated fuel are stored. This is the maximum amount of fuel that can be stored in the larger of the two storage basins (the former waste storage basin, now designated as Basin 2).

b. Fuel burnup is 24,000 MWd/TeU and specific power is 40 kW/kgU.

c. Cooling time is 1 year for all fuel. This is a conservative figure since, by the time the storage basin is full, the majority of fuel will have cooled over 2 years.

The source is characterized as follows:

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

b. The source volume is 14.5 ft. below the pool surface. This distance is the approximate depth to the top of the fuel portion of the fuel bundles. Credit is taken for self shielding in the source volume assuming the source medium is only water (not a mixture of water, fuel, steel, etc.). Gamma buildup is accounted for in flux calculations for source and shield material (i.e., fuel and water).

The gamma flux at the pool surface is approximated by the equation[6] for each energy group:

$$\phi = \left(\frac{BS_V}{2\mu_s}\right)\left(E_2(b_1) - E_2(b_3)\right)$$

where:

ϕ = scalar flux (MeV/cm^2-sec);

B = build-up factor;

S_v = source strength (MeV/cm^3-sec);

b_1 = μt;

b_3 = $b_1 + \mu_s h$;

μ_s = macroscopic cross section of source material, water (cm^{-1});

μ = macroscopic cross section of shield material, water (cm^{-1});

t = shield thickness (441.96 cm); and

h = source thickness (426.72 cm).

Fission product activity (Ci/TeU) for fuel with a burnup of 24,000 MWd/TeU and specific power of 40 kW/kgU is given in Table 7-1. Values given for 1 year cooling and 5 years cooling were calculated with the aid of The ORNL Isotope Generation and Depletion Code, ORIGEN-1[7].

The source term is specified for four energy groups as given in Table 7-2. Gamma flux at the pool surface was calculated for each energy group. Using the curve given in Figure 2-2 of Rockwell (page 20), energy-dependent conversion factors were determined for expressing flux in MeV/cm^2-sec as dose rate in Roentgen/hr. Dose rates for each energy group were added to yield a total of 1.748E-5 mRem/hr. (This number is based on calculations using Microshield)

Actual dose rate measurements were taken in November 1975. A dose rate of 3 mRem/hr was found to be constant with depth when measured below the surface of the pool to within 7 ft. of the top of the fuel. The 3 mRem/hr dose rate is due to radioactive contamination in the pool water. Underwater, about 4 ft. above fuel bundles, dose rates were 150 mR/hr to 200 mR/hr.

Routine dose rate measurements are taken throughout the storage basin area. For example, during the year 1999, the average dose rate on the basin crane was about 1.77 mRem/hr as measured by TLD.

7.4.3 Ventilation

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

7.4.3.1 Primary Safety Considerations

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

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

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

7.4.3.2 Principal Mechanisms for Ensuring Safety

a. Confinement of mobile radioactive material is ensured by:

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

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

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

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

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

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

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

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

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

7.4.4 Airborne Effluent Monitoring Instrumentation

7.4.4.1 Functional Description

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

7.4.4.2 Major Components and Operating Characteristics

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

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

7.4.4.3 Sampling Considerations

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

a. Representative air sampling is ensured by:

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

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

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

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

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

b. Continuous sampling of effluent is ensured by:

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

(2) Employing sample pumps of high reliability.

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

(4) An effective system maintenance program.

7.4.5 Radiation Monitors

7.4.5.1 Functional Description

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

7.4.5.2 Major Components and Operating Characteristics

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

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

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

c. Criticality Monitors: A detection system (two ARMs) is provided in the fuel storage pool enclosure to warn personnel in the highly unlikely event of a criticality incident and to initiate evacuation to staging areas. Detectors are set at a trip point high enough to lessen

potential for false alarms. Two detectors ensure monitoring continuity. Criticality alarms are unique, intermittent klaxons so situated that they can be heard throughout the main building, auxiliary buildings, and in outside areas.

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

7.4.5.3 Radiation Monitor Considerations

Location of basin area radiation monitors are depicted in Figure 7-2. ARMs are designed to be fail-safe in that they alarm both audibly and visually in the event of an upscale reading. On a downscale reading a warning light will signify instrument malfunction.

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Figure 7-2. Radiation monitor locations.

a. Adequacy of protection system coverage is ensured by:

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

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

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

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

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

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

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

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

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

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

(2) Providing redundant power supply systems.

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

(4) Source check portable instruments before use

(5) Use of self-reading pencils

(6) Use of film badges

(7) An effective system maintenance program

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

7.5 PERSONNEL EXPOSURE ASSESSMENT

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

Management controls include operating limits for radioactive material concentration in basin water requiring special corrective actions. The gross β concentration values are 0.02 μCi/ml and 0.1 μCi/ml, respectively. If the lower rate is exceeded, a cleanup campaign is initiated. If the upper rate is exceeded, normal basin operations are interrupted and cleanup performed.

7.6 HEALTH PHYSICS PROGRAM

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

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

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

The safety program is conducted according to the ESHP. Plant operations are conducted under procedures consistent with the manual. Operations and maintenance procedures having safety significance are reviewed by the Safety Committee.

7.7 ESTIMATED MAN-REM OFF-SITE DOSE ASSESSMENT

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

7.7.1 Effluent and Environmental Monitoring Program

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

Monitoring program results from 1968 to 1994 confirm the absence of detectable off-site radioactive contamination. Off-site exposure resulting from fuel storage is a very small fraction of regulatory limits. In addition, Illinois Department of Public Health measured radiation dose

rates in populated areas around the DNPS/GE-MO sites and in 13 central Illinois counties from 1971 to 1976 indicate no significant difference in radiation exposure between the two areas even though the joint site consists of two reactors and a fuel storage facility[9].

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

Particulate radioactive material in air consists of residual radioactive fallout from weapons testing and other man made events plus cosmic and natural sources. Cosmic and background sources result in a dose rate of 2 to 3 mRem/week. River water concentrations show a natural background of about $1x10^{-8}$ µCi/ml due to natural radium, uranium and radiopotassium.

The program meets USNRC requirements.

Table 7-3
MORRIS OPERATION RADIOLOGICAL MONITORING PROGRAM

Particulates in Air

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

1. The stack air monitoring system and back-up system fails or is out of service for a time period greater than 24 hours.

2. License specification 4.1.1 "Effluent Air" gross beta activity exceeds $4x10^{-8}$ µCi/ml.

3. An airborne activity release alert is declared as defined by the GE-MO Emergency Plan.

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

This Figure Withheld under 10 CFR 2.390

Figure 7-3. TLD Sampling Locations

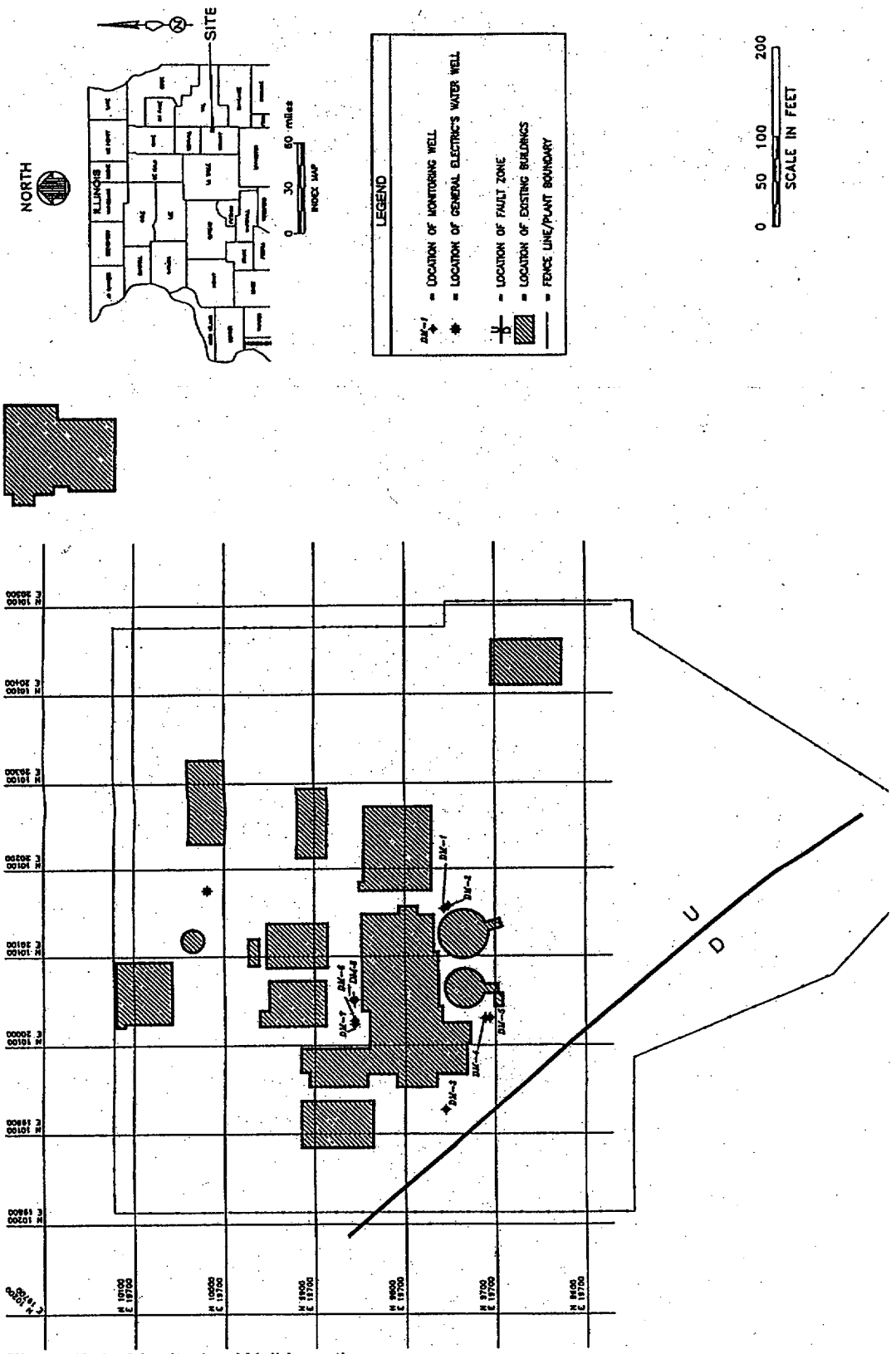

Figure 7-4. Monitoring Well Locations

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Figure 7-5. Environmental Water Sample Locations

7.7.2 Estimated Exposures

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

7.7.2.1 External Exposure

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

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

7.7.2.2 Internal Exposure from Inhalation

Estimated internal exposures were calculated based on equations and dose conversion factors given in DOE/EH 0071[11].

Using an annual $\overline{X}/Q$ value of 3.1 x 10^{-8} sec/m^3 (see Appendix A.5), the maximum off-site whole body dose rate was calculated to be 6.6 x 10^{-7} mRem/yr.

7.7.2.3 Man-Rem Calculations

Man-Rem calculations were estimated for annual whole body exposure due to inhalation of released beta emitters and skin dose due to release of Kr-85. Averages of exposures were calculated for concentric circles with radii of multiples of 10 miles. These average values were multiplied by the population within each area which gives an average annual whole body man-Rem. The sum of these values for each area out to a radius of 50 miles gives a total of less than 2 x 10^{-6} man-Rem/yr whole body and less than 0.12 man-Rem skin dose for the period from 1970 to the year 2100.

For comparison, the population exposure from normal background radiation (taken at 100 mRem/yr) in the same area is about 665,000 man-Rem for 1970, to 750,000 man-Rem for the year 2000. Therefore, the radiological impact from the GE-MO fuel storage operations is relatively insignificant.

7.7.3 Liquid Releases

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

7.8 REFERENCES

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

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

3. A proprietary product of the Norton Co.

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

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

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

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

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

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

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

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

8.0 ACCIDENT SAFETY ANALYSIS

8.1 INTRODUCTION

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

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

8.1.1 Release Pathways

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

8.1.1.1 Direct Radiation

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

8.1.1.2 Airborne Release

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

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

baskets into a critical array by a tornado-generated missile. In reality, however, the above events have an extremely low probability of occurring and the impact of either would be substantially less than the limits of Part 72.104.

8.1.1.3 Waterborne Release

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

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

8.1.2 Accident Description/Discussion

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

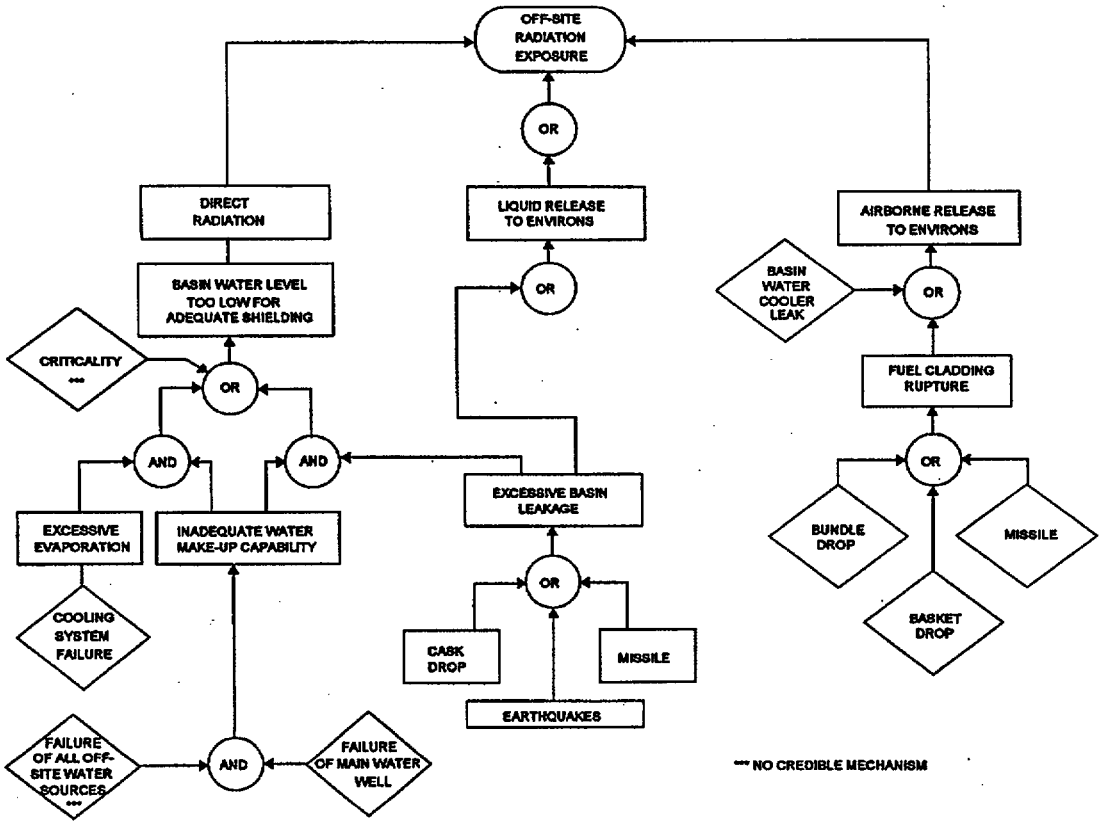

Figure 8-1. Event Diagram of Postulated Accidents

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

8.1.3 Exposure Paths

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

8.2 LOSS OF FUEL BASIN COOLING

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

The time available to provide make-up water if the cooling and water make-up system are out of service has been determined by actual fuel storage conditions. Based on actual storage, the maximum heat load is not more than 1×10^6 BTU/hr. On this basis, the minimum time available to provide make-up water is more than 54 days.

8.2.1 Basin Water Temperature

Maximum basin water temperature can be derived using the following assumptions:

a. Uniform water temperature throughout the basins

b. Ambient air at 70 °F and 70% relative humidity in contact with the basin water surface

c. Basin enclosure removed with zero air velocity across the basin water surface (worst case)

Under these assumptions, the temperature of the basin water would slowly rise (< 2 °F/hr) for about 3 days and even slower thereafter (a nonlinear function of time). The maximum temperature would be about 166 °F, and more than 39,000 ft^3 of water would have to evaporate before the tops of the fuel bundles would be exposed. This would require more than 9 days.

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

8.3 DRAINAGE OF FUEL BASINS

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

8.3.1 Basin Liner rupture Experience

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

8.4 CASK DROP INTO THE CASK UNLOADING BASIN

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

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

8.5 FUEL DROP ACCIDENTS

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

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

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

a. Fission Gas Inventory in the Fuel

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

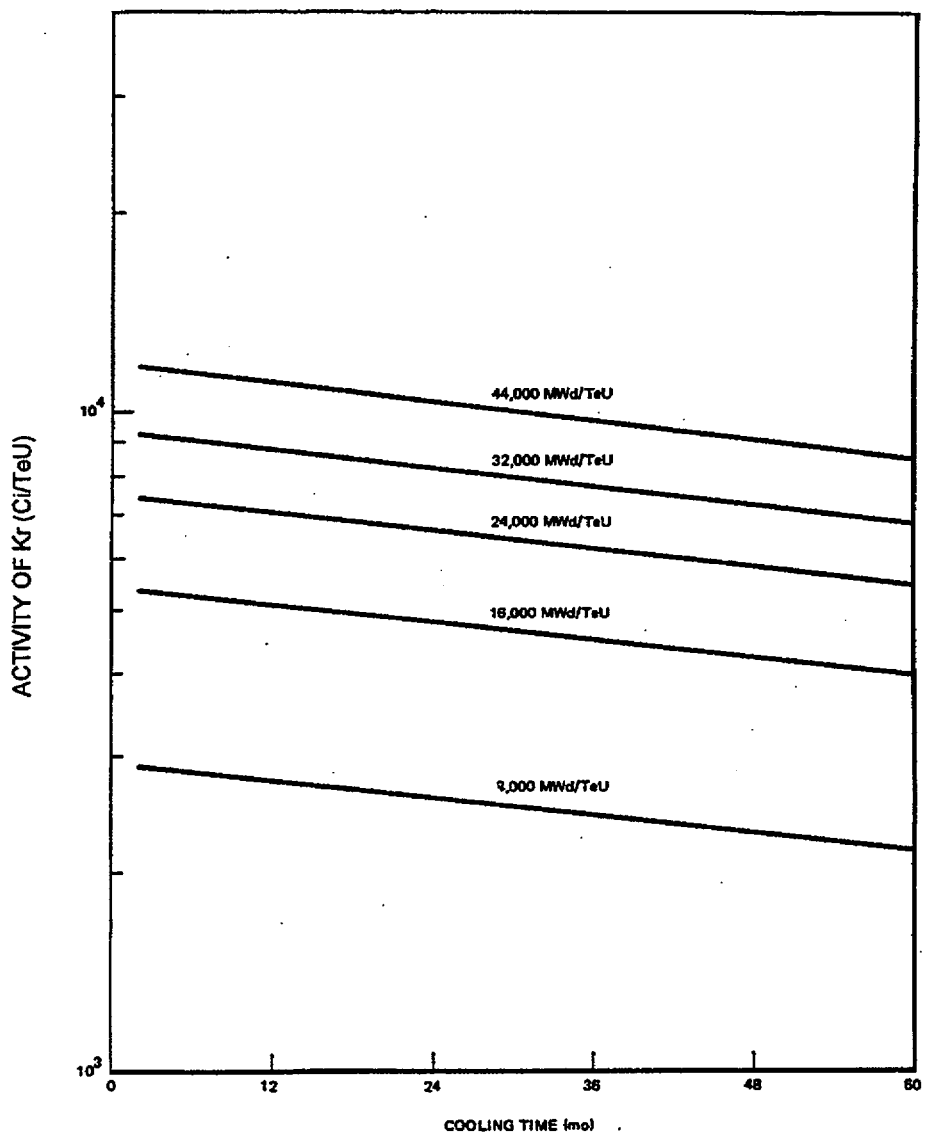

Figure 8-2. Kr-85 Activity as function of cooling time for different fuel exposures. (Total inventory in fuel rod.)

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

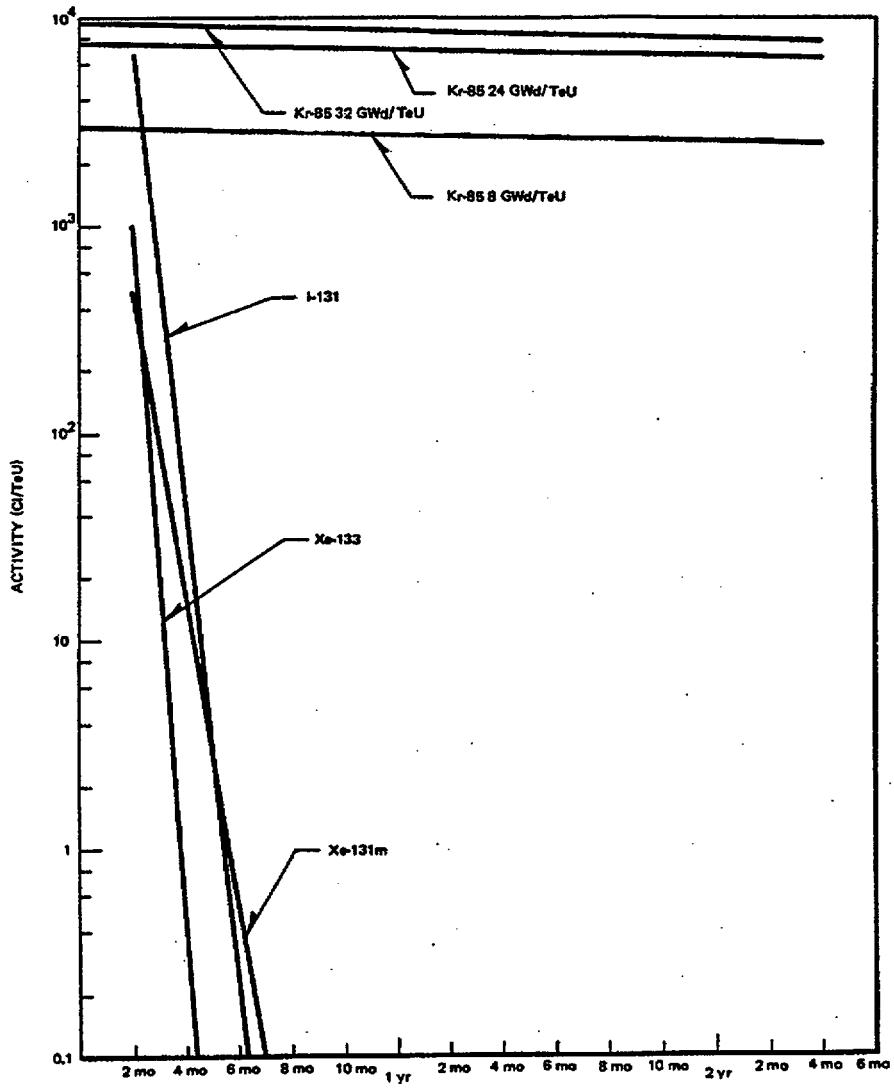

Figure 8-3. Iodine, Krypton and Xenon Decay

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

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

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

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

b. Water Decontamination Factor

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

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

Decontamination Factor = (7.3) exp [0.313 t/d]

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

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

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

c. Assumptions

The following assumptions are made for the safety analysis:

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

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

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

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

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

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

7. Dose conversion factors are:

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

8.5.1 Fuel Bundle Drop Accident

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

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

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

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

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

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

b. Actual Bundle Drop Experience

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

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

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

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

8.5.2 Fuel Basket Drop Accident

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

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

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

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

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

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

8.5.2.1 Accident Analysis

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

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

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

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

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

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

8.5.3 Recovery Practice

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

8.6 TORNADO-GENERATED MISSILE ACCIDENT

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

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

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

8.6.1 Accident Analysis

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

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

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

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

Table 8-1
LIST OF TORNADO-GENERATED MISSILES

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

- Defined as rotational plus translational velocity.

Table 8-2

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

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

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

8.6.2 Assumptions

Assumptions used in the safety analysis include the following

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

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

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

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

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

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

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

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

8.6.3 Dose Rate Calculations

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

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

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

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

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

8.7 CHILLER SYSTEM LEAK

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

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

8.8 CRITICALITY ACCIDENT

The safety margin against an accidental criticality could potentially be reduced by receiving fuel that is more reactive than assumed in the design analyses or by mechanical damage to the

storage basket or fuel sufficient to cause the stored fuel bundles to be forced into a critical configuration.

8.8.1 Fuel Handling Procedures

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

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

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

8.8.2 Reactivity Calculations

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

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

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

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

PWR fuel having an initial k_∞ of 1.35 (2.8% U-235) and having undergone one cycle of irradiation (10,000 MWd/TeU) would have a post-irradiation k_∞ based on BNWL calculations using the LEOPARD code, of approximately 1.19. Calculations of uniform arrays of PWR fuel were made by GE personnel using proprietary reactor design codes, to describe the

relationships between k_∞ spacing and K_{eff}. These calculations did not include the poisoning effect of the stainless steel in the baskets, which BNWL calculations indicated would reduce k_{eff} by 2.5%. Figure 8-4 depicts the relationship between k_∞ and K_{eff} for PWR fuel bundle arrays with 2 in. separation. A 2.5% reduction in k_{eff} is included for the effect of stainless steel. The data shows that k_∞ would have to exceed 1.21 for the array to be critical.

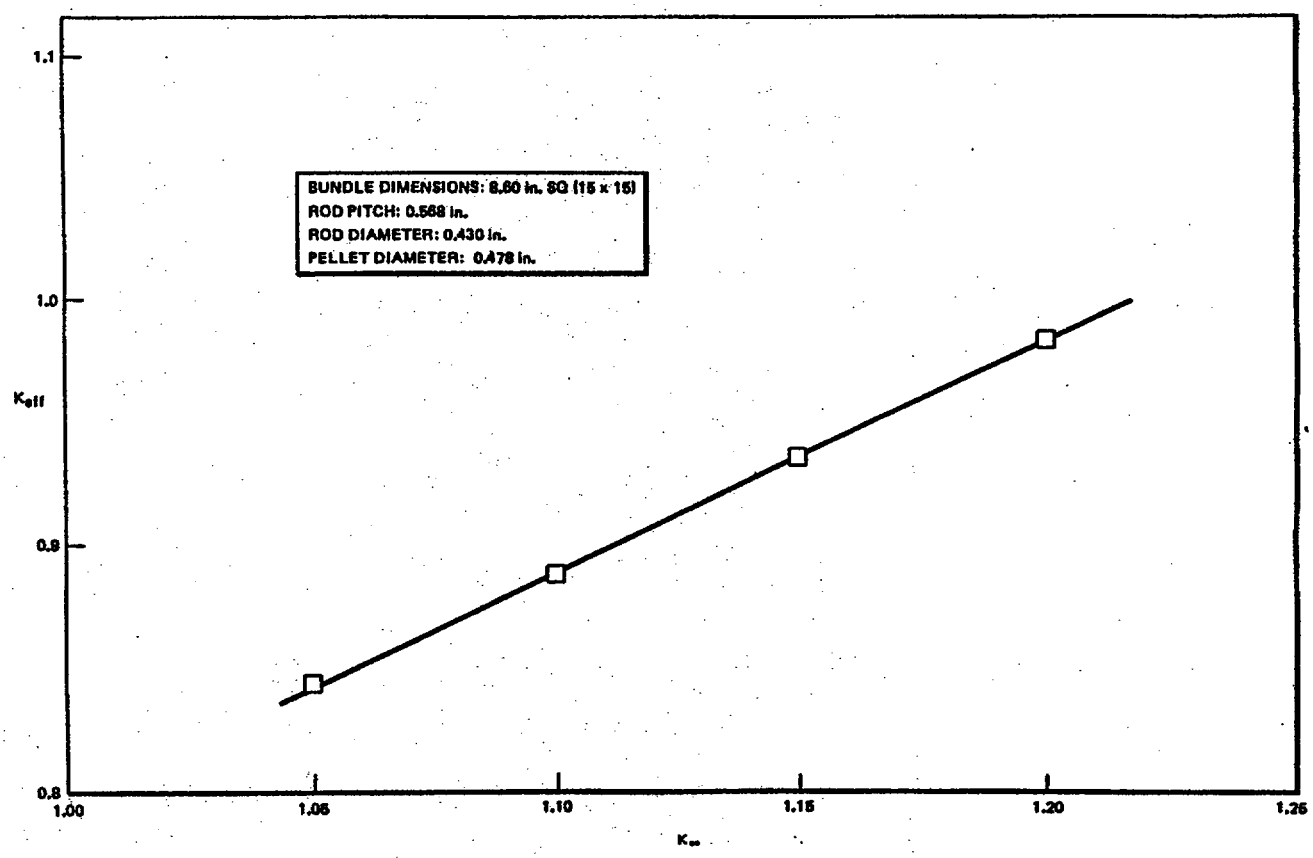

Figure 8-4. PWR fuel bundle array at 2-inch separation.

8.8.3 Missile Impact

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

Accurate predictions of the effects of the impact of a tornado-borne missile on a system as complex as an array of the fuel storage baskets would be difficult to make or to prove. To provide insight into the potential increase in neutron multiplication that could arise from reduced spacing, an analysis of three PWR bundles in a "T" configuration, closely spaced over their

entire length, was done to estimate the effect of driving three assemblies into a corner. Since this example does not provide consideration of the fourth bundle in the basket, an example of reduced spacing involving four PWR bundles is provided. Such a condition represents an extremely improbable event since the fuel would have to be compacted into a corner from two directions 90° apart over a substantial portion of its length. Because such a compaction would result in separation of the fuel in the compacted array by more than 10 inches of water from the fuel in the closest baskets, the four-bundle array can be considered isolated. The results of calculations performed by GE personnel for a water-reflected, close-packed, square array of four PWR fuel bundles are shown in Figure 8-5.

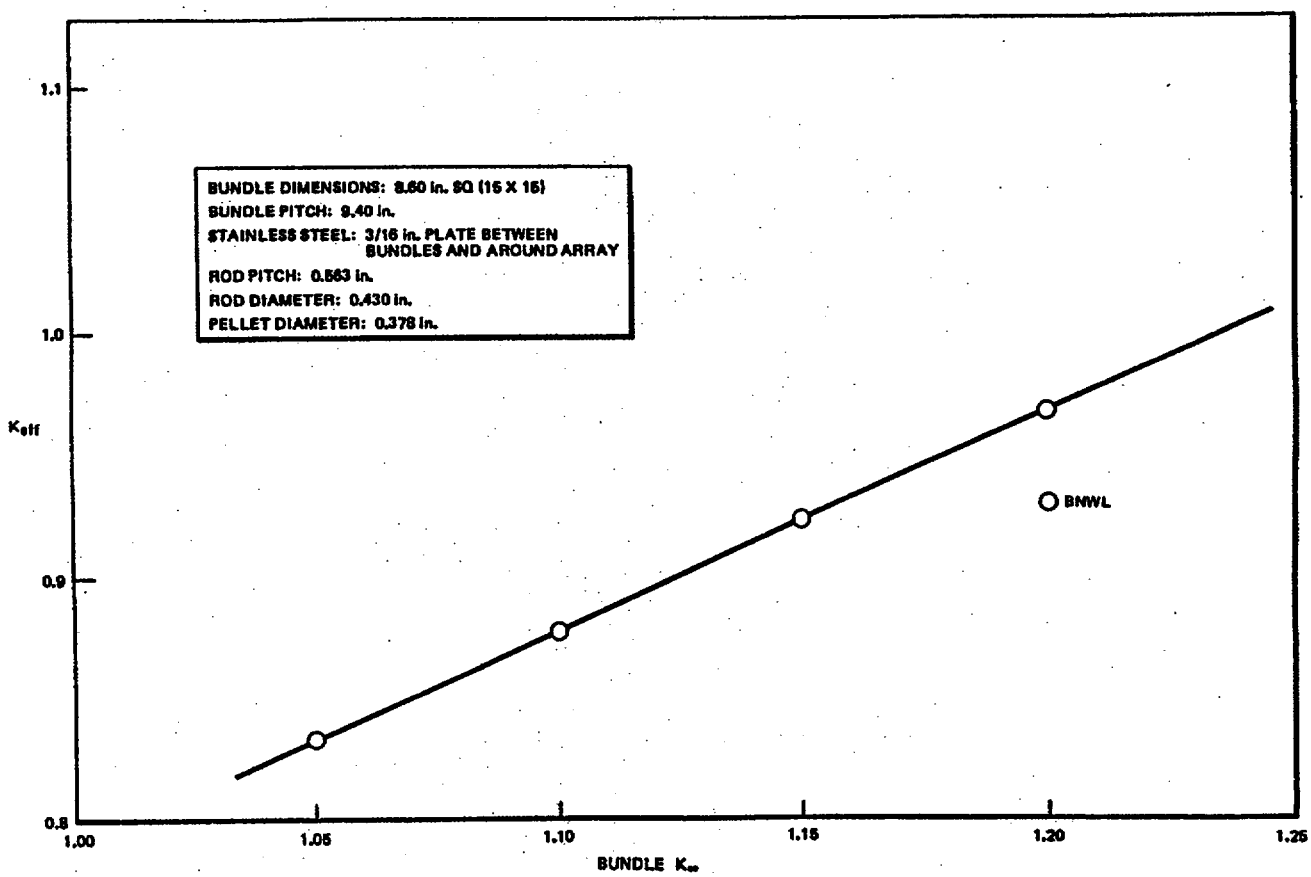

Figure 8-5. Close-Packed array of four PWR bundles.

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

8.8.4 Consequences of a Criticality Accident

No criticality accidents have occurred in low enriched LWR bundle systems. Accidents have occurred in chemical reprocessing or critical assemblies involving plutonium or highly enriched uranium. Historical criticality incidents in nuclear separation facilities have had fission

magnitudes estimated at 1.3×10^{17} to 4×10^{19} fissions. In no case has the reaction been of an explosive nature.

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

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

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

b. Design and operation of the storage system

c. Administrative procedures for fuel receiving and storage

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

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

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

8.8.4.1 Assumptions

Primary assumptions used to evaluate a criticality accident include:

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

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

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

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

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

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

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

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

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

The dose rate is:

$$D' = \phi / c$$

where

D' = dose rate mR/hr

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

Values for c for each energy group are:

$c_1 = 5.2 \times 10^2$

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

$C_3 = 7.8 \times 10^2$

$C_4 = 8.6 \times 10^2$

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

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

where

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

Table 8-3

PROMPT FISSION GAMMA-RAY SPECTRA

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

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

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

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

erg/gm - hr watt for approximately 175 cm. The fast neutron dose at a distance of about 488 cm is negligible.

Table 8-4

DOSE, mR, PER FISSION,
AT BASIN SURFACE

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

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

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

8.9 REFERENCES

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

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

3. RESSAR-41, April 1974.

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

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

9.0 CONDUCT OF OPERATIONS

9.1 INTRODUCTION

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

9.2 CORPORATE ORGANIZATION

Principal organizational levels of General Electric Company in effect as of January 2000 are shown in Figure 9-1.

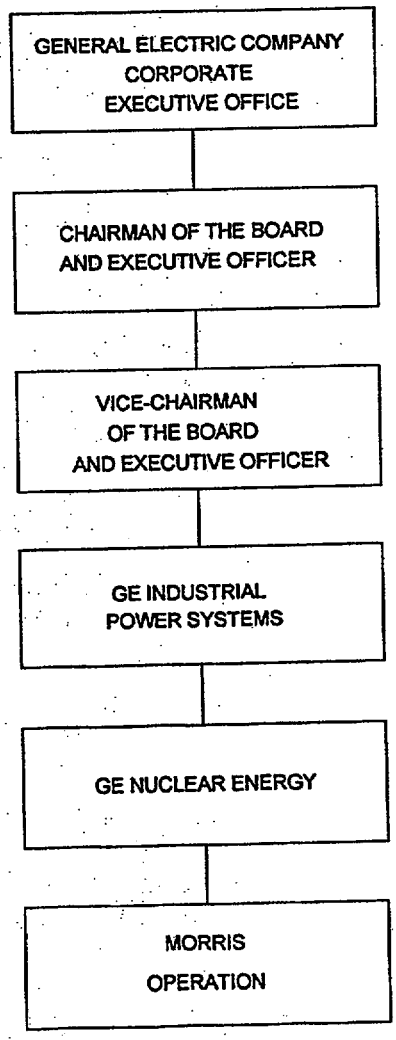

Figure 9-1. GE Morris Operation relationship to the GE Corporate Offices.

9.2.1 Organization Functions, Responsibilities, and Authorities

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

9.2.1.1 Company Policies

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

9.2.1.2 Nuclear Energy Policies

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

9.2.1.3 Operation Policies

Morris and Vallecitos Operation (MVO) focuses Company and GENE policies to specifically address the Operation's requirements.

9.2.1.4 Irradiation Processing Operation

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

9.2.2 GENE MVO Components

Morris Operation is a sub-section of MVO.

9.2.2.1 Morris Operation

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

9.2.2.2 Regulatory Compliance

GE-MO Regulatory Compliance is responsible for directing and coordinating activities related to obtaining and support of licenses and permits including developing practices and procedures

and compliance verification in accordance with applicable Company and Government requirements.

9.2.3 Morris Operation Organization

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

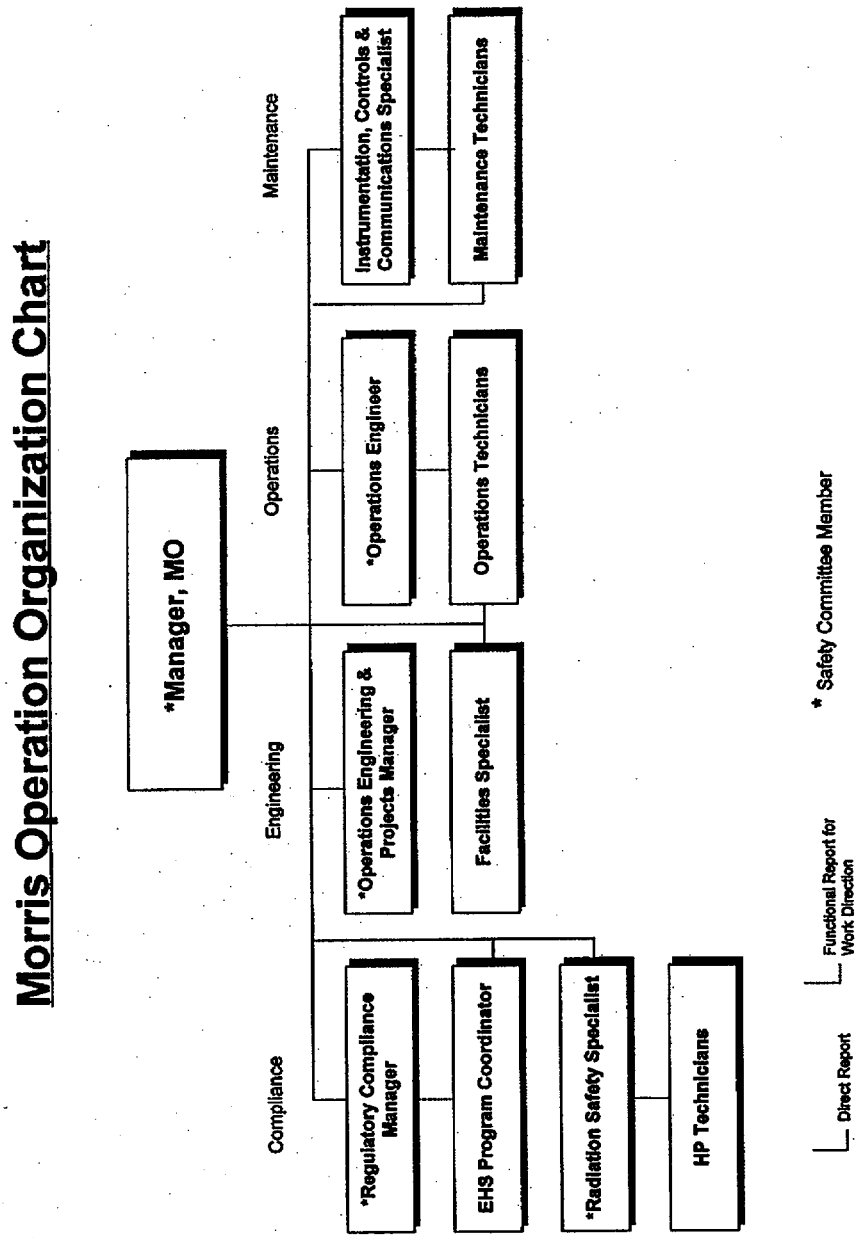

Figure 9-2. GE Morris Operation Organization Chart.

and compliance verification in accordance with applicable Company and Government requirements.

9.2.3 Morris Operation Organization

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

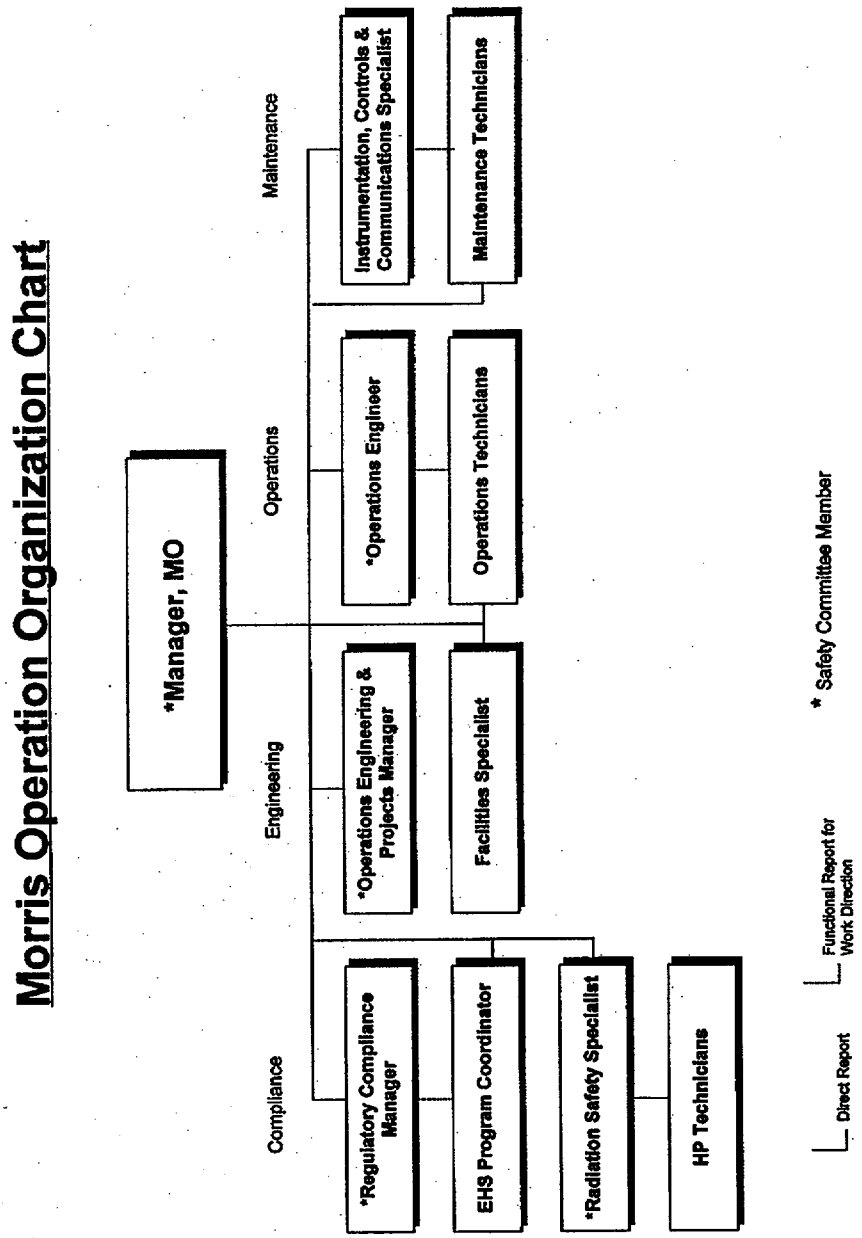

Figure 9-2. GE Morris Operation Organization Chart.

9.2.3.1 Manager - Morris Operation

The Manager - MO is responsible for safe operation and maintenance of facilities, including compliance with license conditions and applicable Federal, State, and local regulations to ensure protection of health and safety of public and plant personnel.

9.2.3.2 Operations, Engineering, and Projects Manager

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

9.2.3.3 Regulatory Compliance Manager

The Regulatory Compliance Manager (RCM) is responsible to the Manager - MO for licensing compliance activities including special nuclear material accountability and plant physical security. In addition, the RCM is responsible for providing industrial and radiological safety support, coordinating site regulatory matters with local, State, and Federal regulatory agencies, and directing site environmental activities. The RCM reviews facility and operating procedure changes to determine need for nuclear safety review and reviews fuel data to ensure conformance with criteria for fuel storage.

9.2.4 Safety Committee

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

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

9.3 TRAINING PROGRAMS

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

a. Orientation and Indoctrination

b. Radiation and Industrial Safety

c. Security/Safeguards

d. Emergency Response

e. Quality Assurance

f. Basic Plant Facilities and organization

g. Fuel Receiving and Storage Operations

h. Utilities and Operating Systems

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

9.3.1 Operator Qualification, Training, and Certification

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

9.3.2 Trained and Certified Personnel

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

9.4 NORMAL OPERATIONS

9.4.1 Facility Procedures

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

9.4.1.1 Morris Operation Instructions (MOIs)

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

9.4.1.2 Standard Operating Procedures (SOPs)

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

9.4.1.3 Environmental Health and Safety Plan (EHSP)

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

9.4.1.4 Special Work Permits (SWPs)

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

9.4.1.5 Regulated Work Permits (RWPs)

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

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

9.4.1.6 Equipment Maintenance Programs

A Work Request (WR) system is employed at GE-MO for initiating requests for maintenance, repairs, modifications, alterations and new installations. WRs are reviewed by the Operations Engineer (OE) or delegate and Quality Assurance for conformance to facility procedures and

instructions. Equipment maintenance is performed in accordance with manufacturer's recommended practices and operating experience. Overall responsibility for equipment maintenance is assigned to the OE&PM. Assistance is provided by other components, as required, to ensure safety and operability criteria are correctly interpreted and performance capability maintained.

9.4.2 Records and Reports

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

9.4.2.1 Record Retention

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

9.4.3 Facility Modification

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

9.4.3.1 Project Design Activity

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

9.4.3.2 Licensing Activity

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

9.4.3.3 Project Implementation

The Manager, MO, may at his discretion, designate a Project Manager who is assigned responsibility for construction, installation, testing, startup, and related activities. The Manager -

MO retains full responsibility for project safety and normal concurrent activities involving operation of the facility during modification.

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

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

9.4.3.5 Audits and Reviews

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

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

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

9.4.4 Changes, Tests, and Experiments

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

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

9.4.4.1 Unreviewed Safety and Environmental Issues Criteria

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

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

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

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

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

9.4.4.2 Records and Reports for Changes, Tests and Experiments

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

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

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

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

9.5 EMERGENCY PLAN

9.5.1 Purpose and Scope

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

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

9.5.2 Responsibilities

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

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

9.5.3 Action Procedures

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

9.5.3.1 Emergency Classification

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

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

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

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

d. Major Equipment Failures or Operational Accidents: Defined as any component failure or malfunction having significant potential for personnel injury or major damage to plant facilities. Detection systems are provided for certain conditions; detection of others will be by direct observation or by indication that operating characteristics have changed. All

such incidents are reported immediately to the EC on duty for prompt assessment and initiation of corrective procedures.

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

9.5.4 Activation of Emergency Organization

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

9.5.4.1 Communication Methods

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

9.5.4.2 Notification of Off-Site Agencies

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

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

9.6 DECOMMISSIONING

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

10.0 OPERATION SPECIFICATIONS

1.0 INTRODUCTION

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

1.1 DEFINITIONS

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

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

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

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

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

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

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

2.0 FUNCTIONAL AND OPERATING LIMITS

2.1 AUTHORIZED MATERIALS

2.1.1 Specification

a. Light-water reactor nuclear fuel to be received and stored at GE-MO shall meet the

following requirements:

(1) Fuel shall contain uranium as uranium dioxide (UO2).
(2) Fuel shall be clad with stainless steel, zirconium or zirconium alloys.
(3) Maximum average exposure of reactor discharge batch (fuel) shall be 44,000 megawatt-days per TeU.
(4) Fuel shall be cooled a minimum of one year after reactor shutdown and prior to receipt at GE-MO.
(5) Rod lattice k_∞ limits without allowance for burnup shall not exceed:
- 1.37 for 15x15 PWR (<8.55 inches square)
- 1.38 for 10x10 BWR (<5.65 inches square)
- 1.40 for 7x7 or 8x8 BWR
- 1.41 for 14x14 PWR (<7.80 inches square)

b. Fuel parameters shall be within the ranges specified in Figures 2-1 (a and b) and 2-2 (a and b), or as otherwise specified in this specification.

GE-MO is authorized to store stainless steel clad LaCrosse 10x10 BWR fuel, pellet diameter of 0.35 inch, a pitch of 0.565 and enriched to a maximum of 3.9% U-235.

c. Radioactively contaminated tools and equipment that are incidental to the conduct of General Electric's nuclear and nuclear related business may be possessed, repaired and decontaminated. The total contamination of all tools and equipment shall not exceed 15 Ci as determined by external exposure from the items as received. Items containing smearable contamination shall be packaged for storage.

d. Tools and equipment specifically related to the conduct of fuel storage operations, such as shipping cask internals, that have become contaminated with radioactive materials may be possessed.

2.1.2 Basis

The design criteria and subsequent safety analysis of GE-MO assumed certain characteristics and limitations for fuels that are to be received and stored. Specification 2.1.1a assures that these bases remain valid by defining the allowable fuel form, cladding, k_∞ and irradiation history. Specification 2.1.1b establishes fuel parameters referencing graphical and other criteria. The fuel requirements establish criteria (including k_∞) for fuel to be stored to protect against an accidental criticality. For the most reactive conditions credible, k_{eff} for any array of stored fuel must be less than 0.95 at the 95% confidence level.

The design bases for criticality analyses were selected from detailed analytical studies which were based on the physical parameters of specific fuel designs (See Table A.10-1, Appendix A.10). The largest bundle cross section area and infinite bundle length were assumed in the calculations. These limits were based on unirradiated, clean fuel and include allowance for the

poisoning effect of the stainless steel baskets. Fuel centerline locations and other orientations were assumed to be those giving the maximum system reactivity.

Figures 2-1 (a and b) and 2-2 (a and b) provide k_∞ as a function of fuel enrichment and reactor type, as well as correction factors for principal variables affecting k_∞: pellet diameter, the water-to-fuel ration, and the cladding material. Other fuel configurations that have been analyzed and reviewed separately may be excepted from the limitation of Figures 2-1 and 2-2, as referenced in Specification 2.1.1b.

Specification 2.1.1c provides for storage of tools and equipment incidental to the conduct of General Electric's nuclear businesses while awaiting decontamination, reuse, or ultimate disposal. Activity will be calculated from exposure rate measurements from a package, assuming the radiation originates from a uniform volumetric source having approximately the same dimensions as the package. Unless otherwise determined, gamma emissions of 1 MeV/disintegration will be assumed.

Specification 2.1.1d provides for storage of tools and equipment specifically related to the conduct of General Electric fuel storage operations, such as cask internals and yokes, while awaiting decontamination, reuse, or ultimate disposal.

2.2 FUEL STORAGE PROVISIONS

2.2.1 Specification

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

2.2.2 Basis

The design criteria and subsequent analysis for GE-MO assume irradiated fuel is stored under water in fuel storage baskets, mounted in a support grid in a fuel storage basin. Specification 2.2.1 assures that these assumptions remain valid. The fuel storage baskets and support grid are those described in Section 5.

2.3 VENTILATION EXHAUST VACUUM

2.3.1 Specification

a. A negative air tunnel vacuum is required whenever the low activity waste evaporator is operating.

b. If the air tunnel vacuum drops below 0.5 inches of water during low activity waste evaporator operation, evaporator shutdown will be initiated promptly.

2.3.2 Basis

A negative water pressure in the air tunnel is required relative to the equipment cells to assure adequate air flow to carry radioactive material released during low activity waste evaporator operation through the air tunnel to the sand filter and up the stack. Air tunnel vacuum measurement is indicated In the Control Room (as shown in Table 5-2).

3.0 LIM ITING CONDITIONS

3.1 LIMITING CONDITION – WATER SHIELD

3.1.1 Specification

The depth of water between the uppermost part of a fuel bundle and the surface of the basin water shall be a minimum of nine (9) feet.

3.1.2 Basis

This specification establishes a minimum thickness of water shielding to limit the radiation dose rate in the basin area. This specification applies to all fuel in storage or being transferred from cask to storage location (see also Section 5.2).

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

3.2 LIMITING CONDITION – CRITICALITY

3.2.1 Specification

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

3.2.2 Basis

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

4.0 SURVEILLANCE REQUIREMENTS

Requirements for surveillance of various radiation levels, water levels, and other physical quantities, as well as inspections and other periodic activities to provide assurance of specification compliance are contained in specific Morris Operation Compliance and Operability Tests. Included among these are:

4.1 EFFLUENT AIR

Ventilation air leaving the sand filter is monitored to provide assurance that offsite concentrations will be within the 10CFR 20 limits. The sampling and analysis program provides data for estimating the amounts of radioactive material released to the environment during routing or accident conditions

4.2 HOLDING BASINS

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

4.3 SEALED SOURCES

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

4.4 INSTRUMENTATION

4.4.1 Specification

Systems and equipment shall be tested for operability and calibrated in accordance with manufacturers recommendations, and operational tests shall be performed to check alarm functions and demonstrate other operational features of the system or equipment.

4.5 CHILLERS

4.5.1 Specification

Basin water chiller heat exchangers that are in service shall be inspected at least once each month:

a. The equipment shall be visually inspected for signs of leakage.

4.5.2 Basis

Leakage could occur in the tubes of the heat exchangers, releasing contaminated basin water to the heat exchanger sump. Routine visual tests are made to detect leakage.

4.6 CASK FLUSH

4.6.1 Specification

The concentration of radiocesium in the first flush of a shipping cask containing spent fuel shall be less than one microcurie of radiocesium per milliliter. If this limit is exceeded, the fuel in the cask shall be assumed to have failed and action shall be taken in accordance with established procedures.

4.6.2 Basis

Specification 4.7.1 provides for detection of off-standard conditions within a cask so the need for special handling or other considerations can be evaluated.

4.7 BASIN WATER CHEMICAL CHARACTERISTICS

4.7.1 Specification

Basin water chemistry shall be maintained as follows:

Item	Acceptable Analysis
pH	4.5 to 9.0 or equivalent conductivity measurement less than 2.5 uMho/cm
$NaNO_3$	<200 ppm
Cl	<10 ppm

4.7.2 Basis

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

4.8 BASIN WATER RADIOACTIVE CONTAMINANTS

4.8.1 Specification

Additional basin water cleanup measures shall be initiated if the concentration of radioactive

materials in the water exceeds 0.02 μCi/ml beta. The USNRC shall be notified, and immediate measures taken to reduce concentrations below 0.1 μCi/ml prior to continuation of fuel receiving operations.

4.8.2 Basis

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

5.0 DESIGN FEATURES

The energy-absorbing pad on the cask set-off shelf shall not be altered without appropriate safety review and documentation.

5.1 FUEL STORAGE BASIN

5.1.1 Basis

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

5.2 FUEL STORAGE SYSTEM

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

5.2.1 Fuel Storage Baskets

5.2.1.1 Basis

a. The neutron attenuation properties of stainless steel are considered in the nuclear safety analysis.

b. The structural strength, as fabricated, is considered in seismic accident analyses and is related to nuclear safety.

c. The heat transfer properties are considered in fuel cooling thermal analyses and are related to nuclear safety.

5.2.2 Basket Support Grids

5.2.2.1 Basis

a. The spacing of the grids determines the spacing of fuel that was used in the nuclear safety analysis.

b. The structural strength of the grids and the grid-to-wall intertie are integral to the strength of the system during the design seismic conditions, and therefore related to nuclear safety.

5.2.3 Fuel Grapples

5.2.3.1 Basis

Fuel grapples used with the fuel handling crane and those used with the basin crane are designed to preclude lifting a fuel bundle closer than 9 feet to the normal basin water level.

5.2.4 Fuel Basket Grapples

5.2.4.1 Basis

Basket grapples are designed for use with the basin crane, and are designed to preclude lifting a basket such that the fuel bundles are closer than 9 feet to the normal basin water level.

6.0 ADMINISTRATIVE CONTROLS

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

6.2 ORGANIZATION

6.2.1 The facility staff organization is shown in Figure 9-2.

6.3 PLANS AND PROCEDURES

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

6.3.1 Changes to Plans and Procedures

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

6.3.2 Plans and Procedures – Minimum Requirements

Plans and procedures required by this section shall include:

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

b. Requirements for analysis of cask drop accident consequences prior to receipt of spent nuclear fuel in types of casks not previously received or unloaded.

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

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

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

6.4 REVIEW AND AUDIT

6.4.1 Safety Committee

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

6.4.2 Audits

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

6.5 ACTION REQUIRED FOR SPECIFICATION NONCOMPLIANCE

6.5.1 Functional and Operating Limits

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

a. When feasible, prompt action shall be taken to assure timely return of operations to specification compliance.

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

c. NRC Inspection and Enforcement Regional Offices, Region III, shall be notified within 24 hours, advising them of events that resulted in a noncompliance condition.

d. A review of the incident shall be made by the Safety Committee to establish the cause and to define means to prevent reoccurrence.

6.5.2 Limiting Conditions

The following actions shall be taken if a limiting condition is found to have been exceeded:

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

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

c. NRC Inspection and Enforcement Regional Office, Region III, shall be notified at the next inspection to advise them of events resulting in limiting conditions being exceeded.

d. Whenever a given limiting condition has been exceeded more than once in a 3 month period, or more than twice in any 12-month period, the Safety Committee shall establish the cause and define means to eliminate or reduce the frequency of occurrence.

6.5.3 Surveillance Requirements

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

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

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

c. NRC Inspection and Enforcement Regional Office, Region III, shall be notified at the next inspection, advising them of events that resulted in a surveillance requirement being violated.

6.5.4 Design Features

Design features shall only be changes in accordance with specification 6.3.1, and Section 9. Unauthorized modifications of specified design features, or unauthorized introduction of unapproved tools, fixtures or other equipment shall require action as specified for functional and operating conditions in Specification 6.5.1.[2]

6.6 LOGS, RECORDS AND REPORTS

6.6.1 Logs and Records

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

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

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

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

7.0 REFERENCES AND NOTES

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

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

8.1 ENVIRONMENTAL MONITORING PROGRAM

8.1.1 Specification

The licensee will maintain the effectiveness of the environmental monitoring program. Changes in frequency or collection sites by the licensee shall be evaluated against the experience of acquired data and reported with the information required by Specification 8.2.

8.1.2 Basis

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

8.2 ANNUAL ENVIRONMENTAL REPORT

8.2.1 Specification

An annual report will be submitted to the NRC Region III office with a copy to the Director, Office of Nuclear Material Safety and Safeguards, within 60 days after January 1 of each year, specifying the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents during the previous 12 months of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent release and direct radiation at the site property protection area.

8.2.2 Basis

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

11.0 QUALITY ASSURANCE

11.1 INTRODUCTION

Activities at Morris Operation (GE-MO) are conducted in accordance with a quality assurance plan reviewed and accepted by the USNRC and implemented by instructions and procedures at GE-MO. A microfiche copy of this plan (NEDE-31559)is included in this report.

11.2 QUALITY ASSURANCE (QA) HISTORY

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

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

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

11.3 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

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

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

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

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

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

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

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

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

APPENDIX A INDEX

Appendix	Title
A.1	Licensing Action History, Midwest Fuel reprocessing Plant and Morris Operation
A.2	Reference Publications
A.3	Estimation of Ground-Level Radiation Dose Rates for Stack Emission of Radioactive Materials (Continuous Releases)
A.4	Estimation of Ground-Level Radiation Doses from Release of Airborne Radioactive Materials
A.5	Atmospheric Diffusion Calculations
A.6	Flood Potential - Elevation/Discharge Curve (Des Plaines and Kankakee Rivers)
A.7	Decommissioning Plan
A.8	Aging Management
A.9	Fuel Storage System Heat Transfer
A.10	Fuel Basket System Nuclear Design Criteria and Bases
A.11	Fuel To Be Stored - Administrative and Technical Controls
A.12	Fuel Basket System Design Analyses
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A.14	List of Engineering Drawings
A.15	Analysis of Tornado Missile Generation and Impact on the Morris Operation Fuel Storage Basin

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

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

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

A.2 REFERENCE PUBLICATIONS

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

Docket Information[a]	Date[b]	Title or Subject	Docket
(GE document - no number)	11/66	Design and Analysis - Midwest Fuel Recovery Plant	50-268
NEDO-14503	4/71	MFRP Technical Specifications (Proposed)	50-268
NEDO-14504	6/71	Applicants Environmental Report - Midwest Fuel Recovery Plant	50-268
NEDO-14506	7/71	Midwest Fuel Recovery Plant Emergency Plan	50-268
NEDO-10178	12/70	Final Safety Analysis Report - Midwest Fuel Recovery Plant	50-268
NEDO-10178-1 through NEDO-10178-17	7/71	Response to USAEC questions and Amendments through 36	50-268
Letter: B. F. Judson (GE) to H. J. Larson, Director, Div. of Materials and Fuel Cycle Facility Licensing, USNRC	4/74	License SNM-1265, Docket 70-1308 Request to Increase Storage Capacity w/Preliminary Safety Evaluation Report	70-1308
NEDO-20825	3/75	Safety Evaluation Report for Morris Operation Fuel Storage Expansion	70-1308
P&RS 74766[1]	4/75	Fuel Storage System Design Report	70-1308
P&RS 74766[1] Supplement 1	5/75	Supplement 1 to Fuel Storage System Design Report	70-1308
(None)[2]	5/75	Criticality Safety Basis for the MFRP Project-1 Fuel Storage Baskets	70-1308
NEDO-20776	1/75	Fuel Recovery Operation Quality Assurance Plan	70-1308
NEDO-20969	8/75	Operating Experience - Irradiated Fuel Storage - GE Morris Operation	70-1308
NEDO-20825-1	9/75	Response to NRC Staff Questions	70-1308
NEDO-21326-1 NEDO-21326-2	1/77	Consolidated Safety Analysis Report (basic issue)[3]	70-1308
NEDO-21326-1A NEDO-21326-2A	4/77	Incorporate changes and correction re USNRC REVIEW	70-1308
NEDO-21326-2A1	4/77	Incorporate Proposed Operating Specifications	70-1308
NEDO-21326-1A2 NEDO-21326-2A2	8/77	Incorporate new geological information and appendices	70-1308
NEDO-21326-1a3 NEDO-213262a3	2/78	Minor changes, corrections	70-1308
NEDO-21326-2a4	1/79	Incorporate Operating Specifications	70-1308

Docket Information[a]	Date[b]	Title or Subject	Docket
NEDO-21326-c	1/79	General revision - license renewal application	70-1308

[a] General Electric publication number unless noted otherwise

[b] Month/year

REFERENCES

1. Programmed & Remote Systems Corp., St. Paul, Minn. (job Number)
2. Battelle Pacific Northwest Laboratories, Richland, Wash.
3. B-series revisions not related to existing plant.

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

ABSTRACT

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

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

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

1.0 INTRODUCTION

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

2.0 METEOROLOGICAL FACTORS

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

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

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

where:

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

Q_0 = Release rate (Ci/sec);

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

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

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

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

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

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

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

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

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

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

where:

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

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

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

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

where:

θx = Sector width.

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

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

where:

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

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

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

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

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

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

where:

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

C_z = Suttons's vertical diffusion coefficient.

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

Table C-1
DEPOSITION VELOCITIES

Condition	$\left(\frac{V_d}{\bar{\mu}_0}\right)$ [a] Particulates	Halogens
Very Stable	1.5×10^{-4}	2.4×10^{-3}
Moderately Stable	2.2×10^{-4}	3.4×10^{-3}
Neutral	3×10^{-4}	4.6×10^{-3}
Unstable	6×10^{-4}	8×10^{-3}

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

Table C-2
DIFFUSION COEFFICIENTS

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

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

very stable $\geq +1.5$ °C
moderately stable ≥ -0.5 °C but $< +1.5$ °C
neutral ≥ -1.5 °C but < -0.5 °C

unstable < - 1.5 °C

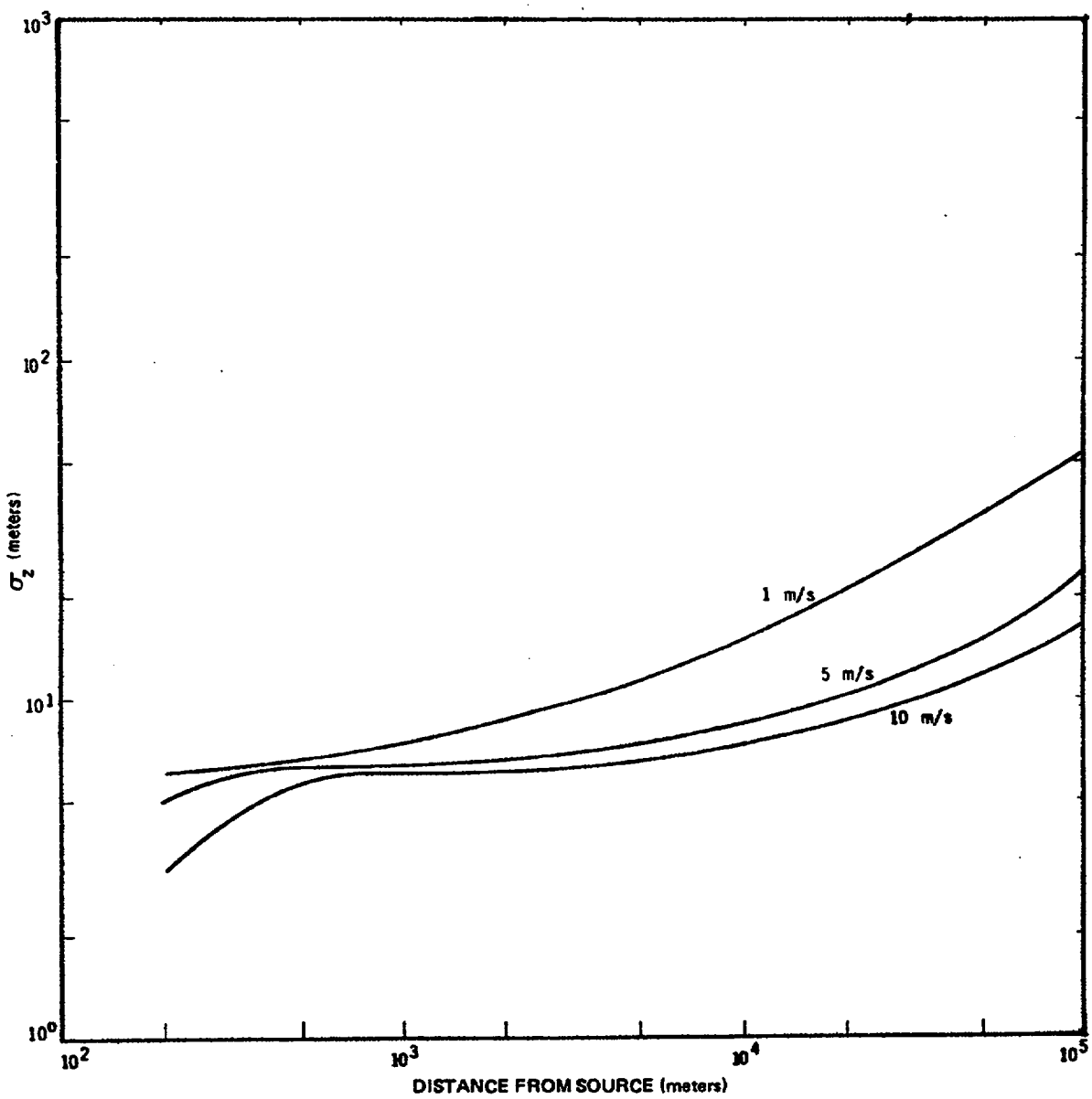


Figure C-1 Vertical Cloud Width – Very Stable

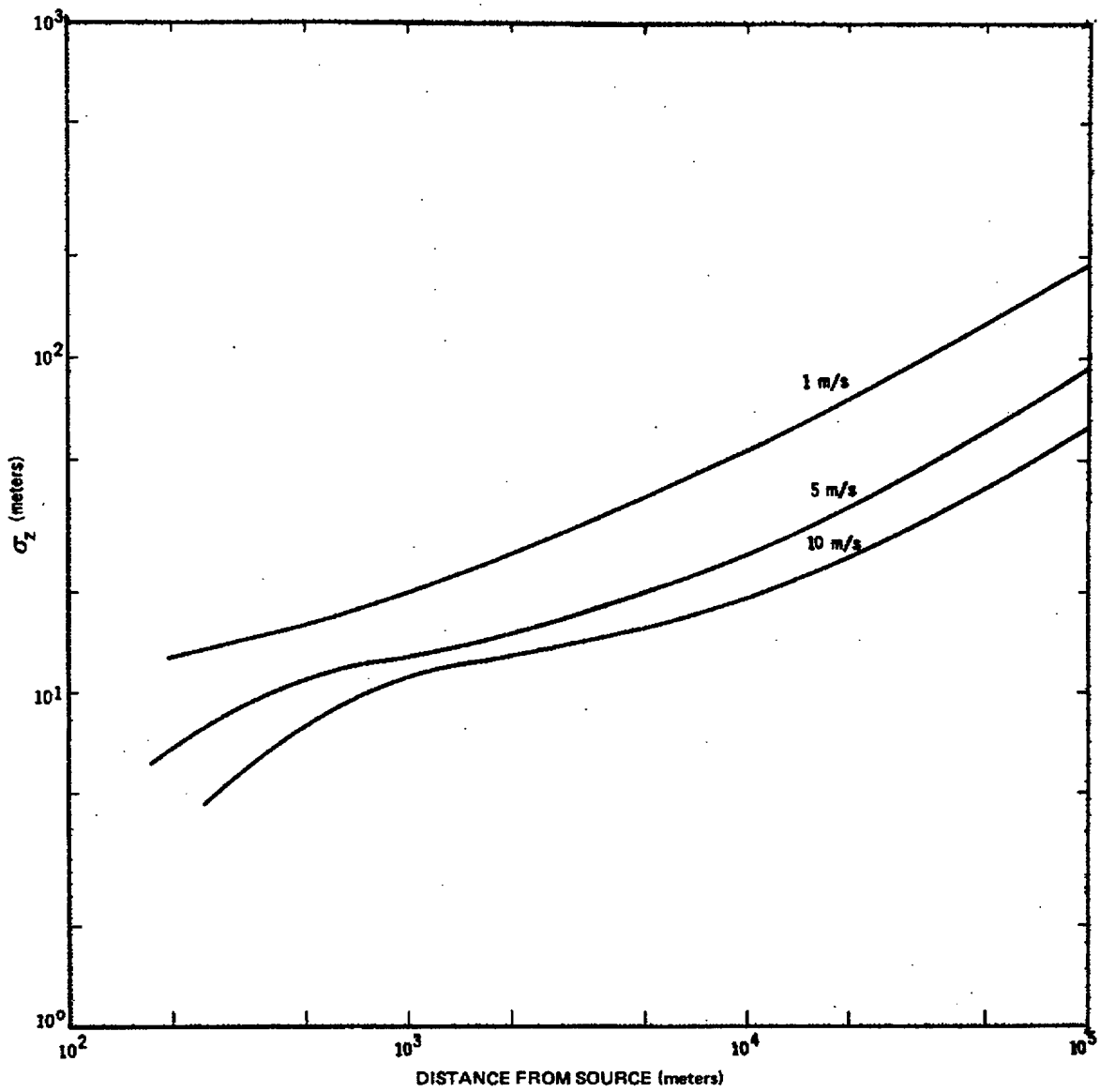


Figure C-2. Vertical Cloud Width - Stable

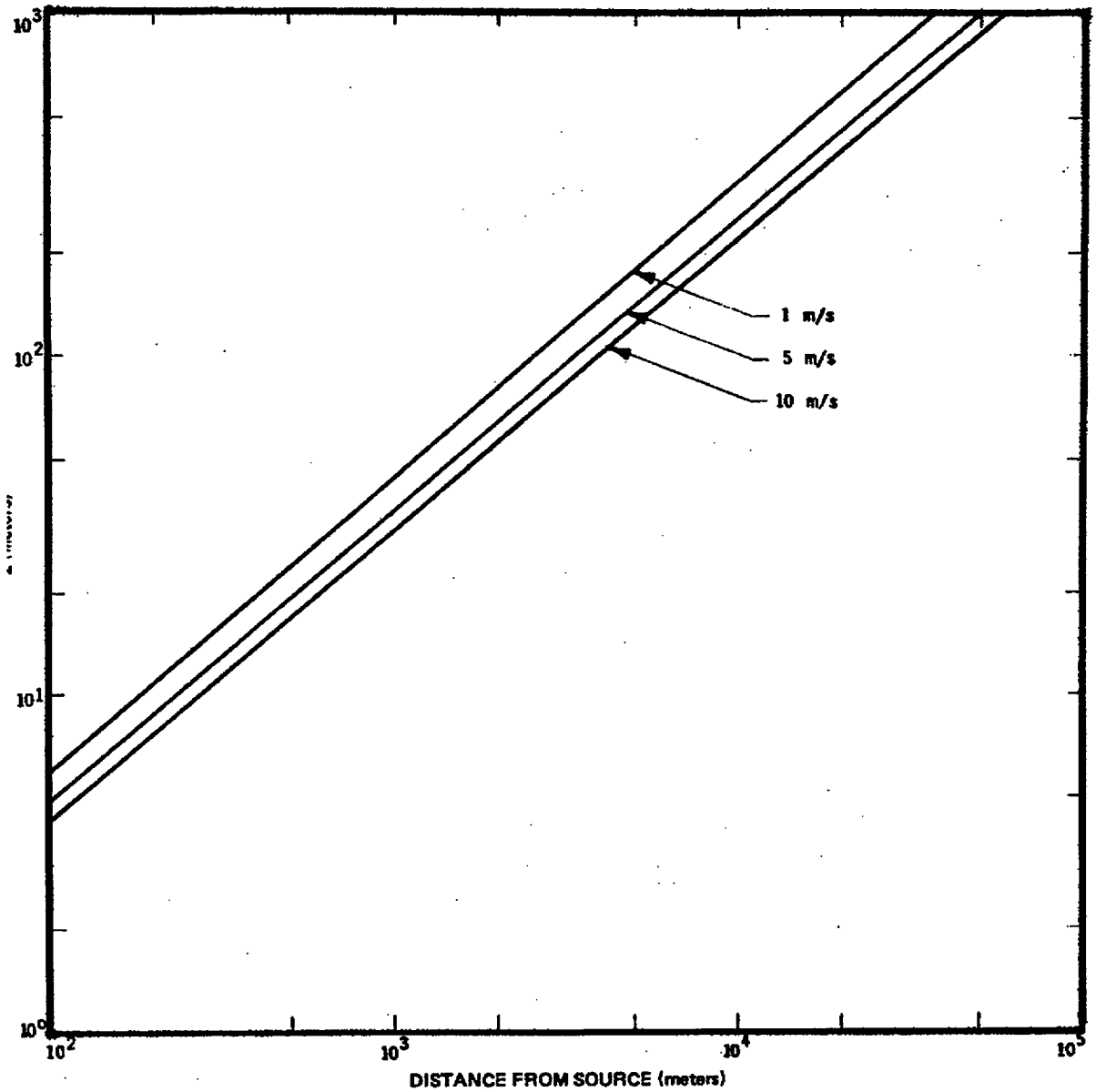


Figure C-3. Vertical Cloud Width – Neutral Stability.

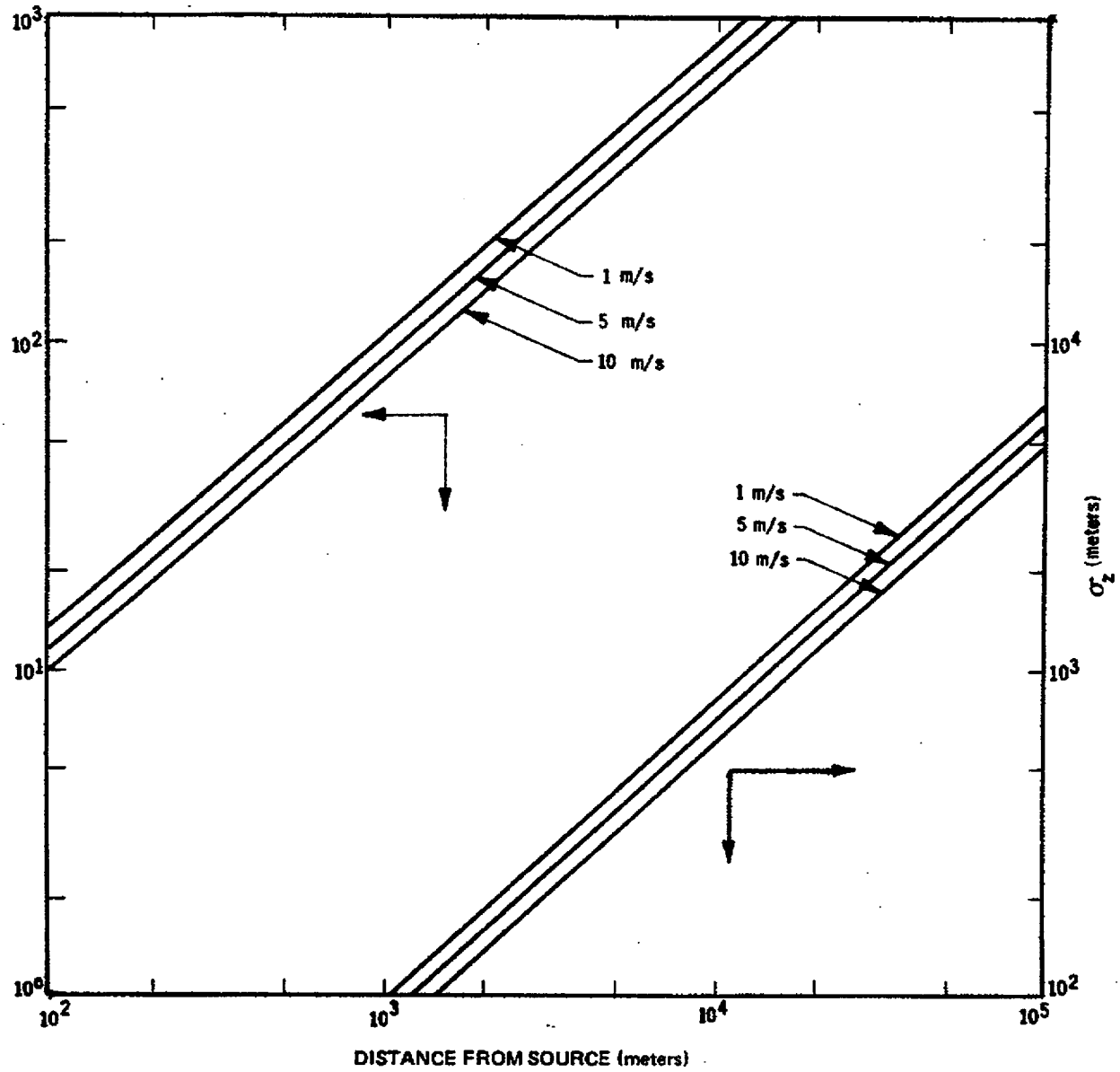


Figure C-4. Vertical Cloud Width - Unstable

3.0 RADIOLOGICAL FACTORS

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

1. External radiation to persons on the ground who are not in the plume but who receive radiation (principally gamma radiation) from the plume. This is the case for persons located near the stack.

2. External radiation to persons on the ground and in the plume (gamma and beta radiation). This is pertinent only at distances where the plume has reached the ground.

3. Internal radiation exposure to persons in the plume as a consequence of inhalation.

4. Internal radiation exposure from ingestion of agricultural products affected by deposition of radioactive materials on vegetation.

Each type of exposure is considered separately below.

3.1 External Dose (Gamma)

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

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

where:

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

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

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

where:

B = Buildup factor = 1 + KμT;

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

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

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

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

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

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

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

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

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

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

3.2 External Dose (Beta β)

The range of β particles in air is only a few meters. Hence, for β calculations, a cloud of material released via a stack and which expands to large dimensions at downwind distances where the cloud has reached ground level, is frequently considered an "infinite" cloud. In such a cloud, the air dose rate is calculated by assuming that the rate of energy release per unit

volume in the cloud is equal to the rate of absorption in that volume (no buildup). The body is considered a small volume within the flux in the cloud and therefore causes no perturbation in the flux.

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

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

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

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

3.3 Internal Dose From Inhalation

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

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

In the discussion of internal dose rates, the air concentration designated by Equation (C-4) is used. Equation (C-4) is not correct in describing the plume after it has diffused to ground level.

The ground represents a barrier to vertical (downward) diffusion. Accordingly, some treat the ground as a perfect reflector, and estimate near-ground-level concentrations on the basis of doubling those otherwise calculated. Whether this is done, or some other factor or method is used to account for this boundary effect. Equation (C-4) needs an appropriate adjustment.

3.4 Internal Dose From Ingestion

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

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

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

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

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

this effect is concerned. In the case of the milk chain, biological transfer within the cow must also be considered. Watson and Gamertsfelder[1] estimate the I-131/Sr-90 ratio to be 10 for transfer into the milk.

4.0 ENGINEERING FACTORS

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

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

where:

V_s = Exit velocity (meters/sec);

d = Stack diameter (Meters);

Q_h = Heat emission of effluent (cal/sec);

$\bar{\mu}_h$ = Wind speed at stack exit (meters/sec); and

c = Correction factor from Moses.

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

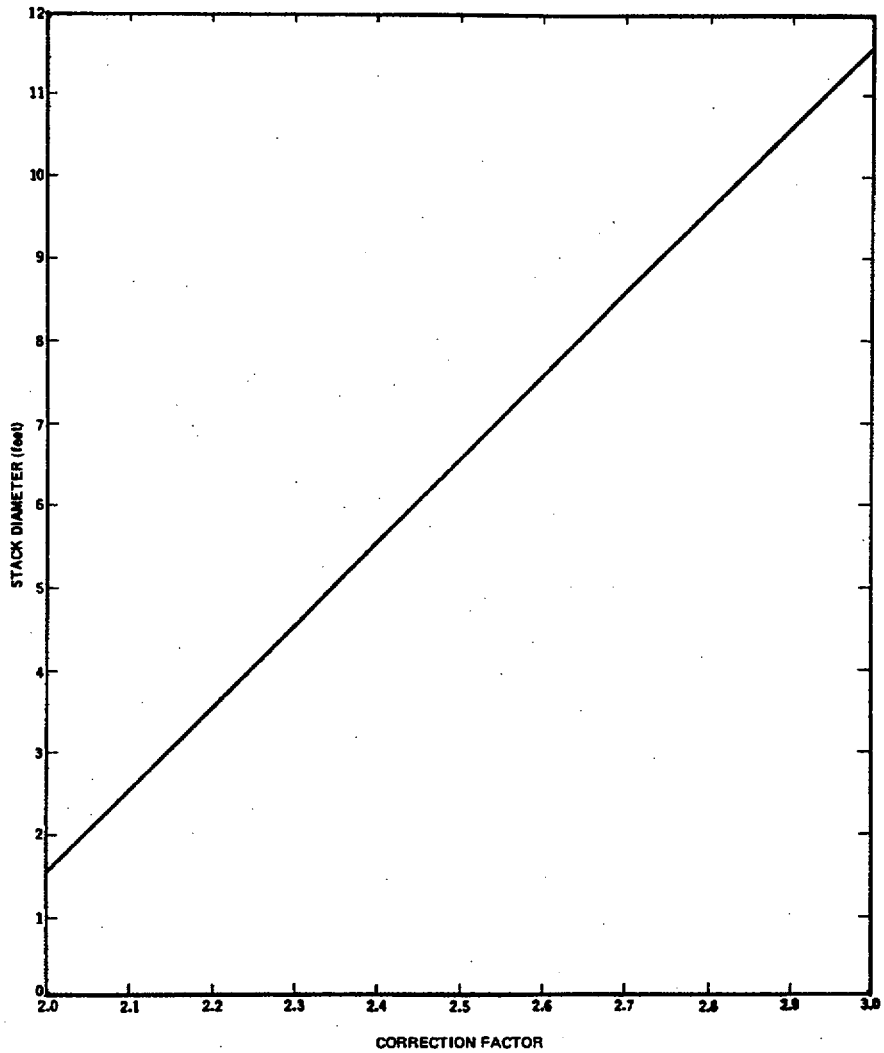


Figure C-5. Holland Plume Rise Formula Correction Factor

5.0 CONCLUSION

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

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

REFERENCES

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

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

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

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

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

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

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

ABSTRACT

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

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

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

1.0 INTRODUCTION

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

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

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

2.0 ATMOSPHERIC DIFFUSION MODEL

In the calculation of the transport and dilution of an airborne cloud, the time period of release of the cloud is very significant. This is so, primarily because the wind does not tend to remain fixed direction-wise, but rather it meanders and fluctuates to a considerable extent. Thus, if a

cloud is formed during a long release period, portions of it will tend to be transported in different directions. On the other hand, if the cloud is formed from an explosive release or "puff" it will all tend to be transported in the same direction. This variability of the wind refers principally to the horizontal changes as opposed to vertical changes, since the former is often very significant while the latter is much more subdued.

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

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

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

where:

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

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

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

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

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

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

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

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

where:

exp [-λt] = Radioactive decay function.

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

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

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

$$\sigma_y^2 = \frac{C_y^2 X^{2-n}}{2} \text{ , and} \qquad \text{(D-3)}$$

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

where:

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

TABLE D-1
VALUES FOR VARIABLES

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

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

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

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

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

where:

A,α = Diffusion coefficients;

and

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

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

σθ = Standard deviation of horizontal wind direction variation during release.

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

3.0 RADIATION DOSE MODEL

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

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

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

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

Each type of exposure is considered separately below.

3.1 External Passing Cloud Dose (Gamma)

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

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

where:

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

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

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

where:

B = Buildup factor = 1 + KμT;

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

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

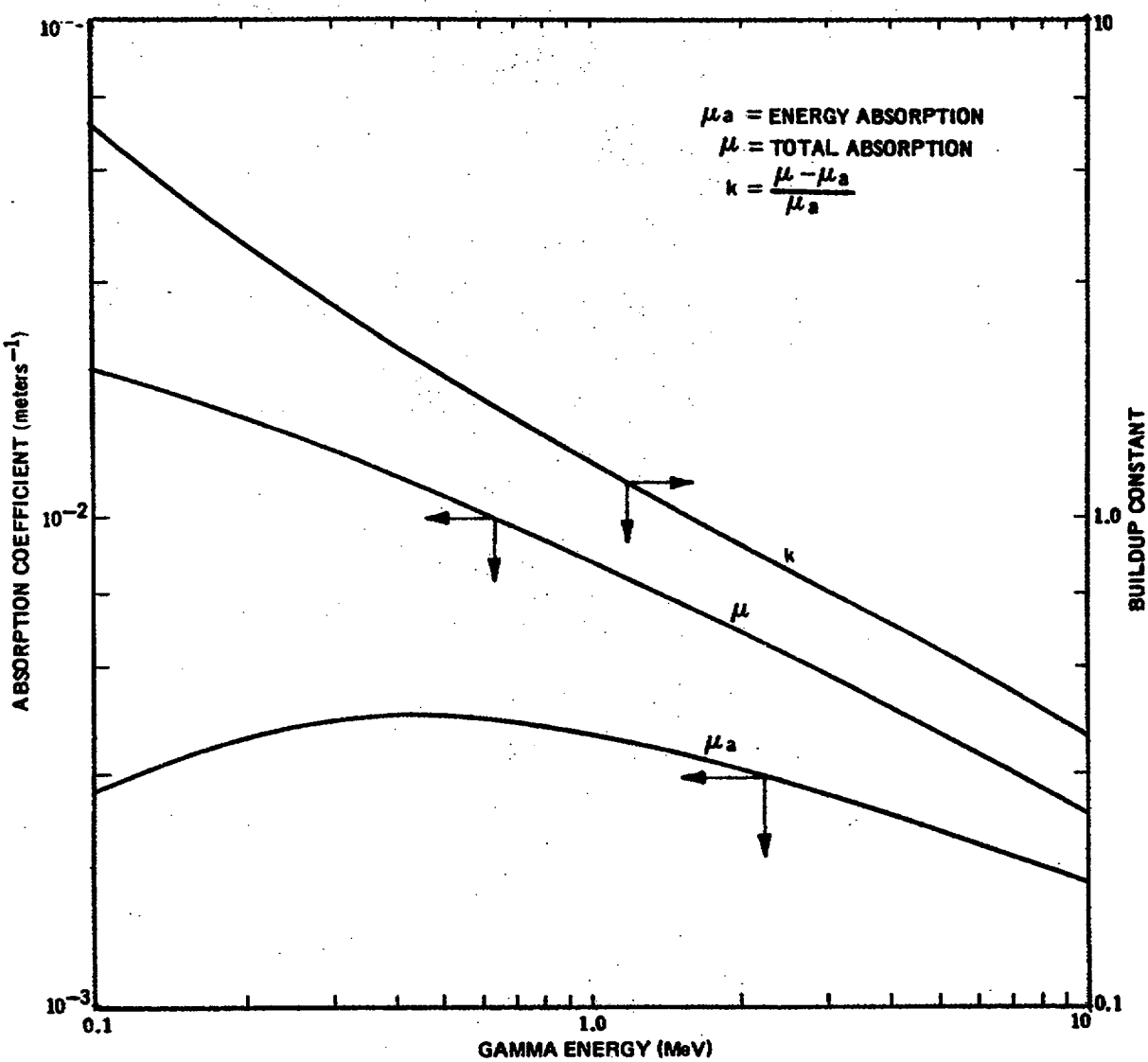


Figure D-1. Gamma Radiation Absorption Coefficients and Buildup Constants for Air (STP)

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

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

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

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

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

TABLE D-2
RADIOBIOLOGICAL FACTORS -- NOBLE GASES

Isotope		Disintegration Gammas Emitted	
Name	**Half-Life**	**Number**	**Energy**
Noble Gases			
Kr-83m	1.86 h	1	0.032
		2	0.009
Kr-85m	4.4 h	1	0.15
		2	0.305
Kr-85	10.76 y	1	0.522
Kr-87	76 m	1	2.05
		2	2.57
		3	0.847
		4	0.347
Kr-88	2.8 h	1	2.4
		2	2.21
		3	0.19
		4	1.55
		5	0.85
		6	0.17
		7	0.02
Xe-131m	12 d	1	0.164
Xe-133m	2.3 m	1	0.233
Xe-133	5.27 d	1	0.081
Xe-135m	16 m	1	0.53
Xe-135	9.2 h	1	0.604
		2	0.36
		3	0.244
Xe-138	14 m	1	0.42
Particulate Daughters[a]			
Rb-88	18 m	1	0.91
		2	1.28

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

[a] Significant particulate daughters only

3.2 External Dose (Beta β)

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

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

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

3.3 Internal Dose from Inhalation

Internal dose from inhalation may be related directly to ground-level air concentration. The air concentration at ground level is as given in Equation (D-2) for any specific meteorological

condition. The dose due to inhalation of the cloud is calculated by first determining the quantity inhaled and then multiplying by the conversion factor of dose per unit amount inhaled. The Quantity inhaled (Q_i) is calculated from

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

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

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

$$D_i = 230(X) q f_2 f_a \frac{t_{1/2}}{0.693} \quad \text{(Rem)}, \qquad \text{(D-12)}$$

where:

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

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

$q f_2 f_a \frac{t_{1/2}}{0.693}$ = k Rem/μCi inhaled.

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

3.4 Fallout Dose (Gamma Dose)

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

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

where:

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

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

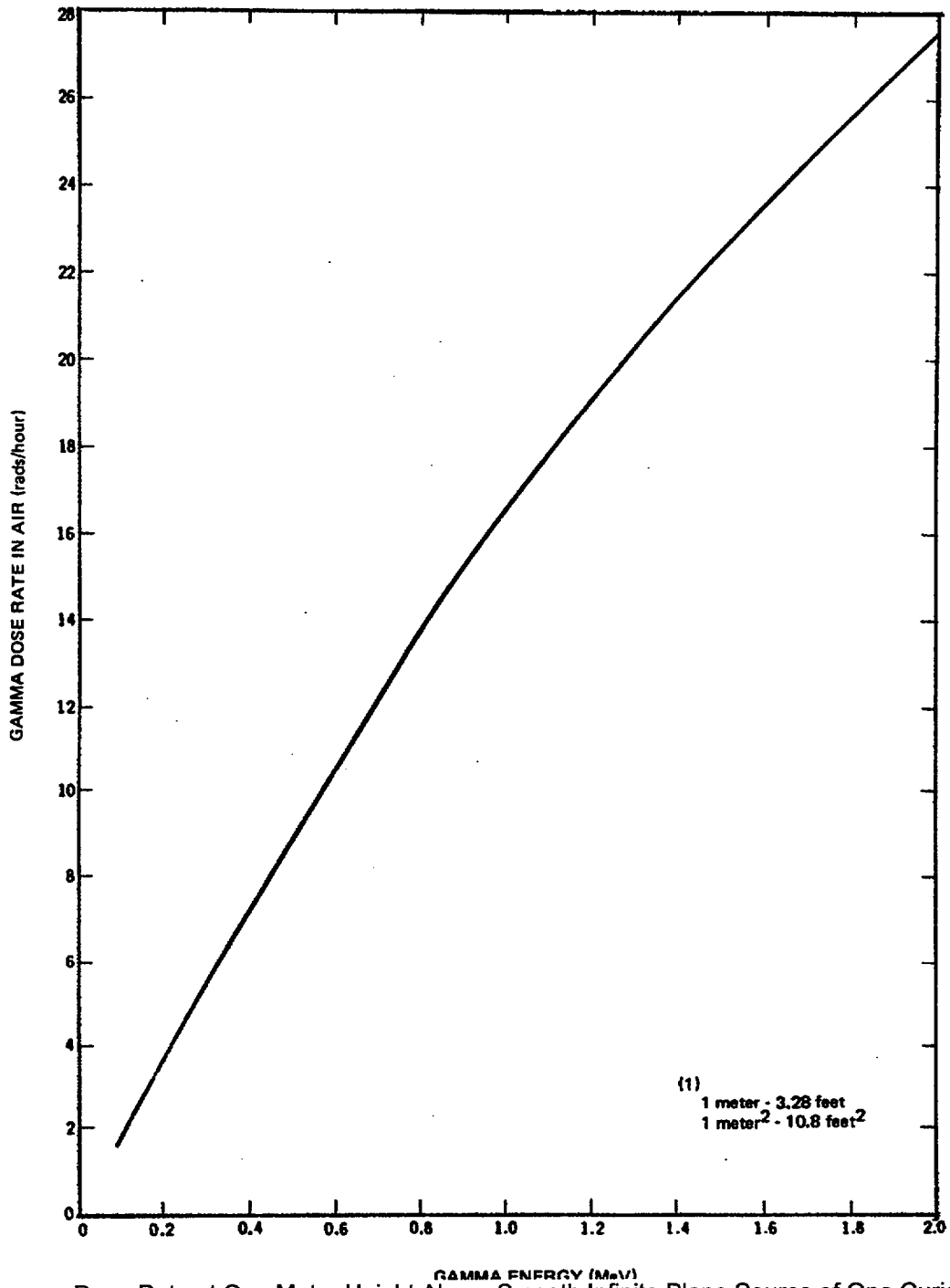


Figure D-2. Gamma Dose Rate at One Meter Height Above Smooth Infinite Plane Source of One Curie Per Square Meter.

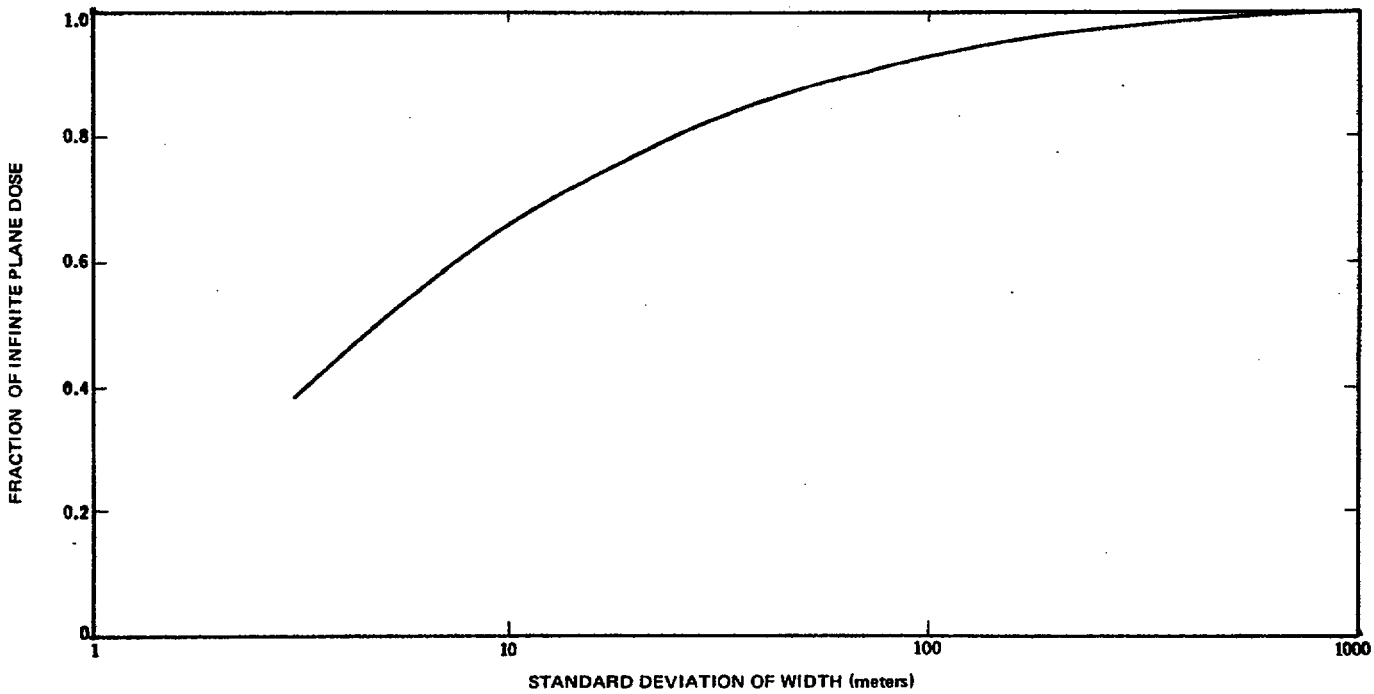


Figure D-3. Ratio of Gamma Dose from Finite Pattern to Infinite Plane Dose.

TABLE D-3
RADIOBIOLOGICAL FACTORS -- HALOGEN RADIOISOTOPES

Isotope Name	Isotope Half-Life[a]	Eff Half-Life[b]	$\overline{E}_\gamma$ (MeV)	$\overline{E}_\beta$ (MeV)	$\overline{E}_{Eff}$ (MeV)[c]	k (Rem/μCi)[d]
*I-131	8.05 d	7.0 d	0.39	0.191	0.23	1.6
*I-132	2.3 h	2.3 h	1.992	0.434	0.65	4.5×10^{-2}
*I-133	21 h	21 h	0.444	0.45	0.14	4.0×10^{-1}
*I-134	53 m	53 m	1.27	0.6	--	2.6×10^{-2}
*I-135	6.7 h	6.7 h	1.54	0.308	0.066	1.3×10^{-1}

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

TABLE D-4
RADIOBIOLOGICAL FACTORS -- VOLATILE SOLID RADIOISOTOPES

Isotope Name	Isotope Half-Life[a]	Eff Half-Life[b]	$\overline{E}_\gamma$ (MeV)	$\overline{E}_\beta$ (MeV)	$\overline{E}_{Eff}$ (MeV)[c]	k (Rem/μCi)[d]

*Mo-99	66 h	66 h	0.24	0.376	0.45	2.6×10^{-2}
*Te-127m	105 d	105 d	0.0885	0	0.083	1.7×10^{-1}
*Te-127	9.3 h	9.3 h	--	0.23	0.24	4.6×10^{-3}
*Te-131	25 m	25 m	0.475	0.577	0.73	--
*Te-132	78 h	78 h	0.231	0.073	0.13	6.4×10^{-2}
*Cs-134	2.1 y	120 d	1.41	0.52	0.074	5.6×10^{-1}
*Cs-137	30 y	138 d	0	0.192	0.192	4.6×10^{-1}

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

TABLE D-5
RADIOBIOLOGICAL FACTORS -- NONVOLATILE SOLID RADIOISOTOPES

Isotope		Eff	$\overline{E}_\gamma$	$\overline{E}_\beta$	$\overline{E}_{Eff}$	k
Name	**Half-Life[a]**	**Half-Life[b]**	**(MeV)**	**(MeV)**	**(MeV)[c]**	**(Rem/μCi)[d]**
*Sr-89	50.4 d	50.4 d	0	0.487	0.49	4×10^{-1}
*Sr-90	28 y	17.53 y	0	0.2	1.1	36
*Sr-91	9.7 h	9.7 h	0.845	0.523	3.3	5.0×10^{-3}
*Y-90	64.2 h	64.2 h	--	0.73	4.4	2.6×10^{-2}
*Y-91	59 d	59 d	0.551	0	2.9	3.3×10^{-1}
*Zr-95	65 d	59.5 d	0.733	0.127	0.57	5.5×10^{-2}
*Nb-95m	90 h	59.5 d	0.235	0	3.8	--
*Nb-95	35 d	33.8 d	0.745	0.053	0.36	1.2×10^{-2}
*Ru-103	40 d	2.4 d	0.473	0.08	0.43	--
*Ru-106	1.0 y	15 d	--	0.013	0.013	--
*Rh-105	36 h	1.39 d	0.032	0.183	0.86	--
*Ba-140	12.8 d	10.7 d	0.237	0.268	1.5	8×10^{-2}
*La-140	40.2 m	1.68 d	2.11	0.495	2.7	5.0×10^{-3}
*Ce-141	32.5 d	31 d	0.097	0.163	0.17	2.2×10^{-2}
*Ce-143	33 h	1.33 d	0.344	0.355	2.2	3.8×10^{-3}
*Ce-144	285 d	243 d	0.043	0.087	1.3	1.1
*Pr-143	13.7 d	13.7 d	0	0.311	1.6	2.0×10^{-2}
*Nd-147	11.1 d	11.1 d	0.286	0.228	1.2	1.8×10^{-2}
*Pm-147	2.7 y	570 d	--	0.074	0.22	2×10^{-1}
*Pm-149	53 h	2.2 d	0.285	0.35	1.9	3.3×10^{-1}
*Pu-240	6.7×10^{3} y	1.95×10^{3} y	0.011	0	0.88	$7 \times 10^{+3}$

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

4.0 APPLICATION OF METHODS

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

4.1 Height of Release

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

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

where:

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

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

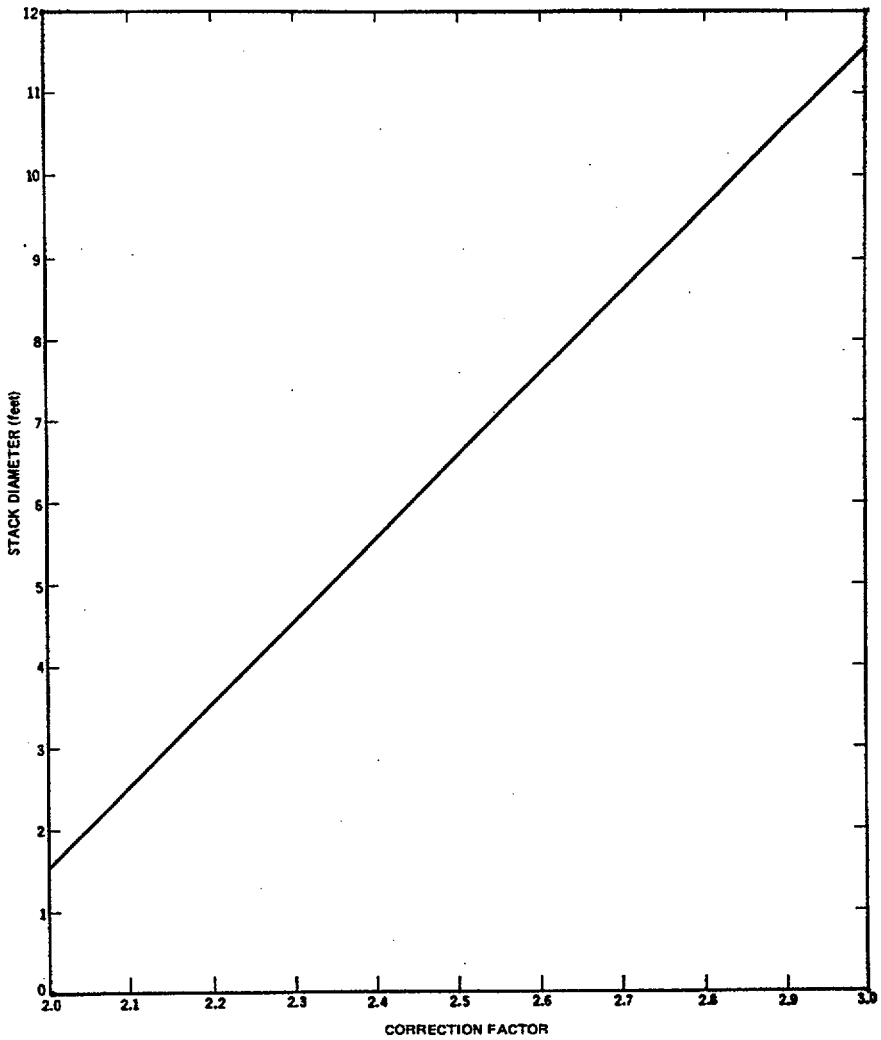


Figure D-4. Holland Plume Rise Formula Correction Factor

4.2 Prolonged Release

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

Wind direction persistence data have been summarized by the Weather Bureau for several locations. The data are partially shown in Table D-6 for ten locations including valley, desert, coastal, and lake-shore locations. These data do not differentiate between stability conditions

or wind speed (see Table D-7 for typical wind speed frequencies). However, the distribution of various periods of persistent wind direction is indicated. From these data the amount of persistence applicable to an analysis can be chosen on the basis of the probability level deemed appropriate.

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

4.3 Cloud Depletion

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

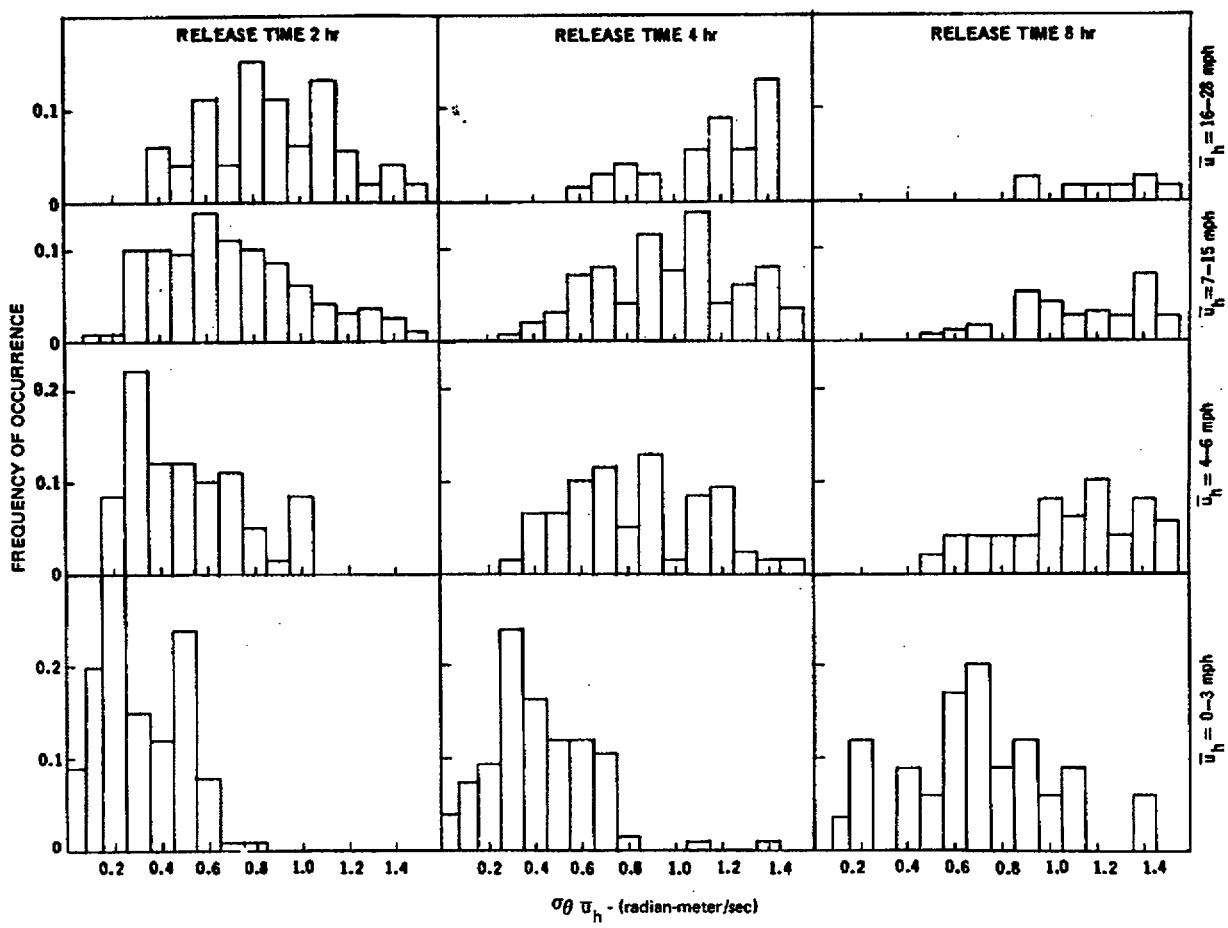


Figure D-5.

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

where:

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

4.4 Sample Calculation

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

Assumptions:

1. Quantities of materials released are:

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

λ = 1×10^{-4} sec^{-1}, and

B. Halogens - 1 curie I-131.

2. Release period of 2 hrs.

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

4. Meteorological conditions are:

   A. Inversion (moderately stable);

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

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

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

5. Radiation effects to be calculated:

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

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

Calculations:

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

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

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

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

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

From Equation (D-11):

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

From Table D-3:

(k) for I-131 = 1.48 Rem/µCi inhaled

Therefore, the lifetime thyroid dose is:

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

From Equation (D-13):

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

where:

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

V_d = 3.4×10^{-3} m/sec (Table D-1);

R = 7.0 rad/h per Ci/m^2 (Figure D-2; and
$\lambda = \frac{0.693}{t_{1/2}} = \frac{0.693}{8.05 \times 86{,}400} = 9.9 \times 10^{-7}\ \text{sec}^{-1}$ (Table D-3).

Therefore;

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

= 7.1×10^{-7} rad.

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

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

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

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

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

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

Site	Wind Speed (mph) 0-3[b]	4-7	8-12	13-18	19-24	25
Albany, New York	23	24	27	21	4	1

Chicago, Illinois	7	26	36	25	5	1
Jacksonville, Florida	10	33	35	18	3	1
Kansas City, Missouri	7	25	37	25	6	1
Los Angeles, California	28	33	27	11	1	1
Miami, Florida	14	30	34	20	2	1
New York, New York	6	15	30	31	12	5
Philadelphia, Pennsylvania	11	27	35	21	5	1
Springfield, Missouri	4	13	34	32	13	4
Tulsa, Oklahoma	9	24	34	26	7	1
Average	12	25	33	23	6	1

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

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

5.0 CONCLUSION

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

6.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

[11] A digital computer program was used for this calculation.

A.5 ATMOSPHERIC DIFFUSION CALCULATIONS

The atmospheric diffusion methods reported by Watson and Gamertsfelder[1] were used as a basis for these calculations.

a. **Height of Release.** Effluents from the 300 foot stack take advantage of increased atmospheric dispersion from an elevated point. The effective height of release is the sum of the stack height plus any effluent rise due to momentum and buoyancy. However, momentum and buoyancy are small in this case, and calculations were made by assuming the height of release to be only the stack height.

b. **Cloud Dispersion Calculations.** Horizontal cloud growth, as expressed by the standard deviation of width σ_y, is given by

$$\sigma_y^2 = \frac{C_y^2 x^{2-n}}{2} \qquad \text{(A.5-1)}$$

Vertical cloud growth, as defined by the standard deviation of width, is given by

$$\sigma_z^2 = a\left(1 - e^{-k^2 t^2}\right) + bt \quad \text{for the stable case} \qquad \text{(A.5-2)}$$

and

$$\sigma_z^2 = \frac{C_z^2 x^{2-n}}{2} \quad \text{for the neutral and unstable case} \qquad \text{(A.5-3)}$$

The values of the constants in Equations A.5-1, A.5-2, and A.5-3 used in each case are given in Appendix A.4, Table D-1.

The calculated values for σ_y and σ_z were used in the Gaussian equation to calculate concentrations in air at various downwind distances.

$$\frac{X}{Q_0} = \frac{Q/Q_0}{2\pi\sigma_y\sigma_z\bar{\mu}_h} e^{-1/2\left(\frac{y^2}{\sigma_y^2} + \frac{z^2}{\sigma_z^2}\right)} \qquad \text{(A.5-4)}$$

where

X/Q_0	=	integrated air concentration (X) per unit activity release (Q_0)
y	=	Distance from centerline crosswind (since plume centerline used, y = 0
z	=	Height of plume above ground

Q/Q_0 = Correction for depletion (halogens and particulates only)[2].

c. **Cloud Depletion and Ground Deposition**. The fallout concentrations of radioactive materials were determined on the bases of particle settling by eddy diffusion only, since settling by gravity is expected to be negligible in this case.

The extent of halogen and solid fission product deposition on the ground is a function of the apparent deposition velocity, which, in turn, is considered to be a function of the diffusion condition and wind speed. Deposition velocities used in this evaluation were based on British results cited in HW-SA-2809 and are given in Appendix A.4, Table D-1. These values of the deposition velocity are used in the calculation of the cloud depletion term defined as Q/Q_0 above.

d. **Precipitation Washout**. Cloud depletion as a result of precipitation washout could cause ground deposition from an otherwise elevated cloud. washout rates commonly used give the same results as from the dry deposition rates, discussed above, from a ground release in the stable case. Thus, the calculation of deposited concentrations from washout were made using the same diffusion conditions as in the other dose calculations, but assuming a ground release. This is not to say that rain occurs during stable conditions or that a ground level release actually is assumed, but merely that this approach was taken as a way of calculating deposited quantities using the same diffusion model.

A.5.1 DIFFUSION ESTIMATES

Using the methods described above, integrated air concentrations, X (μCi/cc), were calculated for a release rate, Q, of 1 Ci/sec. Following are estimates of integrated air concentrations and X/Q values for both long-term (routine) releases and short-term (accident) releases.

a. **Long-Term (Routine) Diffusion Estimates**. Annual average integrated air concentrations are given in Table A.5-1 as a function of distance. Values are listed for 16 sectors around the main stack. As indicated, the maximum concentration (3.1×10^{-8} μCi/cc) occurs at 800 meters (nearest occupancy) and the north-northeast sector. It follows that the maximum $\overline{X}/Q$ value is 3.1×10^{-8} sec/m^3 which is in reasonable agreement with that determined from the environmental monitoring program. The maximum $\overline{X}/Q$ averaged over the past 3 1/2 years of environment monitoring is 7.3×10^{-8} sec/m^3.

The $\overline{X}/Q$ value used in off-site dose assessments for routine operation is 3.1 x 10-8 sec/m3.

b. **Short-Term (Accident) Diffusion Estimates**. Table A.5-2 lists integrated air concentrations for six different meteorological conditions and for a release height of 300 feet. The meteorological conditions include:

Symbol	Atmospheric Condition	Wind Speed (miles/hour)
VS-2	Very Stable	2
MS-2	Moderately Stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

To conservatively calculate the consequences of a postulated accident, the "worst-case" X/Q value is used. As indicated in Table A.5-2, the worst case X/Q value (2.8×10^{-5} sec/m^3) occurs at the nearest occupancy (1/2 mi) and U-2 meteorological conditions.

To analyze potential consequences of postulated ground-level releases, maximum X/Q of 4.5×10^{-4} sec/m^3 and 4.0×10^{-4} sec/m^3 are used for particles and halogens, respectively. These values are taken from Table A.5-3.

The worst-case X/Q values given above are extremely conservative. The greatest X/Q measured in the environmental monitoring program is 20.3×10^{-8} sec/m^3 (1-month avg).

TABLE A.5-1
ANNUAL AVERAGE INTEGRATED AIR CONCENTRATION (μCi/cc)
(Based on a Continuous Unit Release Rate = 1 Ci/sec)

Downwind Distance (meters)	SECTOR N	NNE	NE	ENE	E	ESE	SE	SSE
400	1.114E-08[a]	1.134E-08	8.231E-09	1.157E-08	9.858E-08	7.667E-09	6.363E-09	5.641E-09
600	2.764E-08	2.813E-08	1.975E-08	2.766E-08	2.442E-08	1.910E-08	1.586E-08	1.412E-08
800	3.031E-08	3.089E-08	2.160E-08	3.006E-08	2.683E-08	2.099E-08	1.740E-08	1.548E-08
1,200	2.427E-08	2.507E-08	1.828E-08	2.473E-08	2.225E-08	1.717E-08	1.393E-08	1.219E-08
1,600	1.869E-08	1.968E-08	1.507E-08	1.992E-08	1.813E-08	1.376E-08	1.076E-08	9.189E-09
2,000	1.467E-08	1.569E-08	1.240E-08	1.620E-08	1.491E-08	1.120E-08	8.480E-09	7.104E-09
2,400	1.172E-08	1.268E-08	1.021E-08	1.325E-08	1.231E-08	9.182E-09	6.799E-09	5.620E-09
3,200	7.848E-09	8.608E-09	7.069E-09	9.127E-09	8.573E-09	6.351E-09	4.577E-09	3.723E-09
4,800	4.134E-09	4.586E-09	3.827E-09	4.923E-09	4.670E-09	3.439E-09	2.421E-09	1.943E-09
8,000	1.753E-09	1.954E-09	1.681E-09	2.139E-09	2.030E-09	1.471E-09	1.017E-09	8.138E-10
16,000	7.454E-10	8.120E-10	8.108E-10	9.733E-10	8.847E-10	5.754E-10	3.861E-10	3.183E-10

[a] 1.114E-08 = 1.114×10^{-08}

TABLE A.5-1 (Continued)

Downwind Distance (meter)	SECTOR S	SSW	SW	WSW	W	WNW	NW	NNW
400	6.434E-09	5.711E-09	4.993E-09	5.224E-09	5.722E-09	6.616E-09	7.918E-09	2.250E-09
600	1.620E-08	1.420E-08	1.274E-08	1.295E-08	1.407E-08	1.622E-08	1.910E-08	2.274E-08
800	1.783E-08	1.555E-08	1.407E-08	1.415E-08	1.535E-08	1.768E-08	2.068E-08	2.491E-08
1,200	1.435E-08	1.239E-08	1.110E-08	1.108E-08	1.210E-08	1.410E-08	1.628E-08	2.031E-08
1,600	1.120E-08	9.581E-09	8.368E-09	8.296E-09	9.111E-09	1.093E-08	1.235E-08	1.605E-08
2,000	8.925E-09	7.583E-09	6.468E-09	6.374E-09	7.017E-09	8.659E-09	9.599E-09	1.286E-08
2,400	7.216E-09	6.105E-09	5.116E-09	5.018E-09	5.529E-09	6.968E-09	7.617E-09	1.041E-08
3,200	4.910E-09	4.134E-09	3.389E-09	3.304E-09	3.640E-09	4.711E-09	5.062E-09	7.087E-09
4,800	2.621E-09	2.198E-09	1.767E-09	1.715E-09	1.888E-09	2.502E-09	2.650E-09	3.788E-09
8,000	1.091E-09	9.126E-10	7.253E-10	7.146E-10	7.865E-10	1.066E-09	1.134E-09	1.654E-09
16,000	3.482E-10	2.872E-10	2.282E-10	2.788E-10	3.135E-10	4.851E-10	5.662E-10	8.496E-10

TABLE A.5-2
UNIT INTEGRATED AIR CONCENTRATION[a]
(μCi-sec/cc per Ci Released)
H = 300 feet

Distance (miles)		VS-2	MS-2	N-2	N-10	U-2	U-10
1/2	Noble Gases	< 1 E-20	2.4 E-9	5.6 E-6	1.4 E-7	2.8 E-5	3.1 E-6
	Particles	< 1 E-20	2.4 E-9	5.6 E-6	1.4 E-7	2.8 E-5	3.0 E-6
	Halogens	< 1 E-20	2.4 E-9	5.6 E-6	1.4 E-7	2.8 E-5	3.0 E-6
1	Noble Gases	< 1 E-20	1.2 E-7	1.4 E-5	1.3 E-6	1.2 E-5	1.6 E-6
	Particles	< 1 E-20	1.2 E-7	1.4 E-5	1.3 E-6	1.2 E-5	1.6 E-6
	Halogens	< 1 E-20	1.2 E-7	1.3 E-5	1.3 E-6	1.2 E-5	1.5 E-6
3	Noble Gases	1.8 E-16	2.0 E-6	4.6 E-6	8.0 E-7	2.1 E-6	3.6 E-7
	Particles	1.8 E-16	2.0 E-6	4.6 E-6	8.0 E-7	2.1 E-6	3.6 E-7
	Halogens	1.8 E-16	2.0 E-6	4.2 E-6	8.0 E-7	2.0 E-6	3.3 E-7
5	Noble Gases	1.0 E-12	2.8 E-6	2.3 E-6	4.4 E-7	9.5 E-7	1.7 E-7
	Particles	1.0 E-12	2.8 E-6	2.2 E-6	4.4 E-7	9.5 E-7	1.7 E-7
	Halogens	1.0 E-12	2.7 E-6	2.0 E-6	4.0 E-7	8.6 E-7	1.5 E-7

[a] Multiply by 2.0 to include effects of reflection factor

TABLE A.5-3
UNIT INTEGRATED AIR CONCENTRATION[a]
(μCi-sec/cc per Ci Released)
H = 0[a]

Distance (miles)	Particles	Halogens
1/2	4.5 E-04	4.0 E-04
1	1.9 E-04	1.5 E-04
3	4.4 E-05	2.9 E-05
5	2.2 E-05	1.2 E-05

[a] Used to calculate fallout dose from precipitation washout case. Moderately stable 2 miles/hour used. This does not mean that a ground release is assumed, but rather that this is used to obtain ground deposition values from washout.

A.5.2 REFERENCES

1 E. C. Watson and C. C. Gamertsfelder, **Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria,** February 14, 1963 (HW-SA-2809).
2 **Meteorology and Atomic Energy,** U.S. Weather Bureau, July 1955 (AECU-3066).

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

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

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

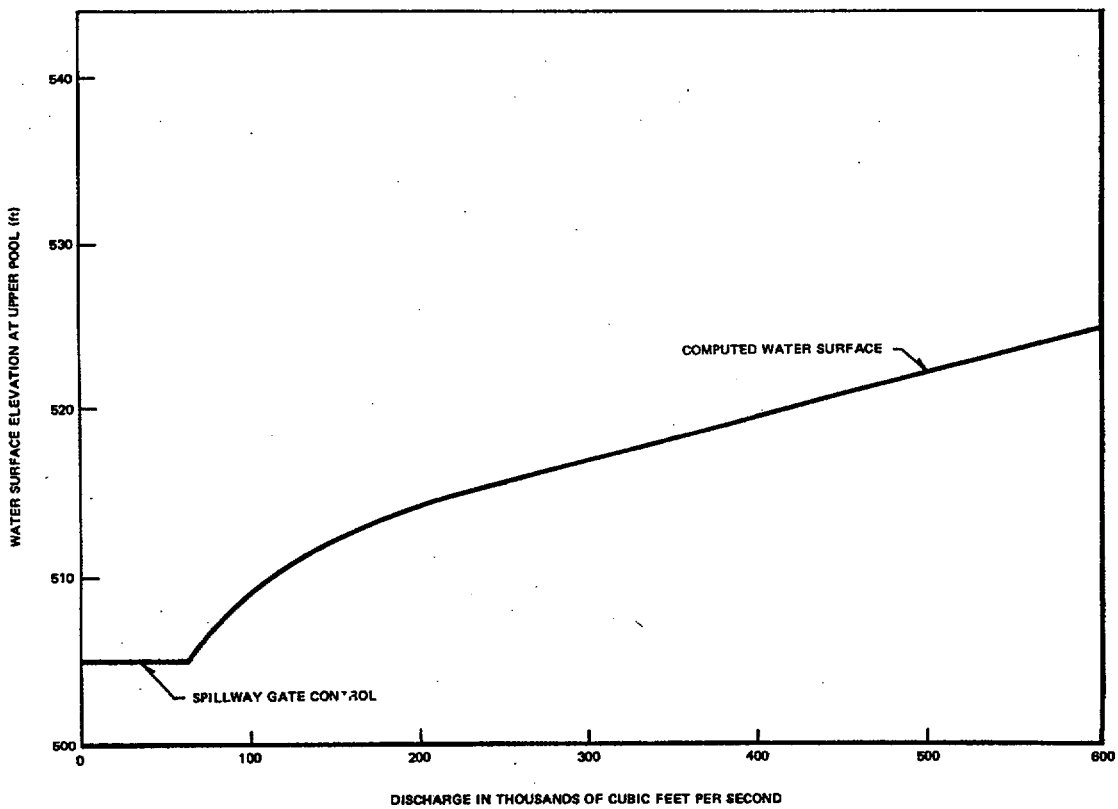


Figure A.6-1. Dresden Lock and Dam, Upper Pool – Preliminary Water Level Discharge Rating Curve.

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

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

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

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

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

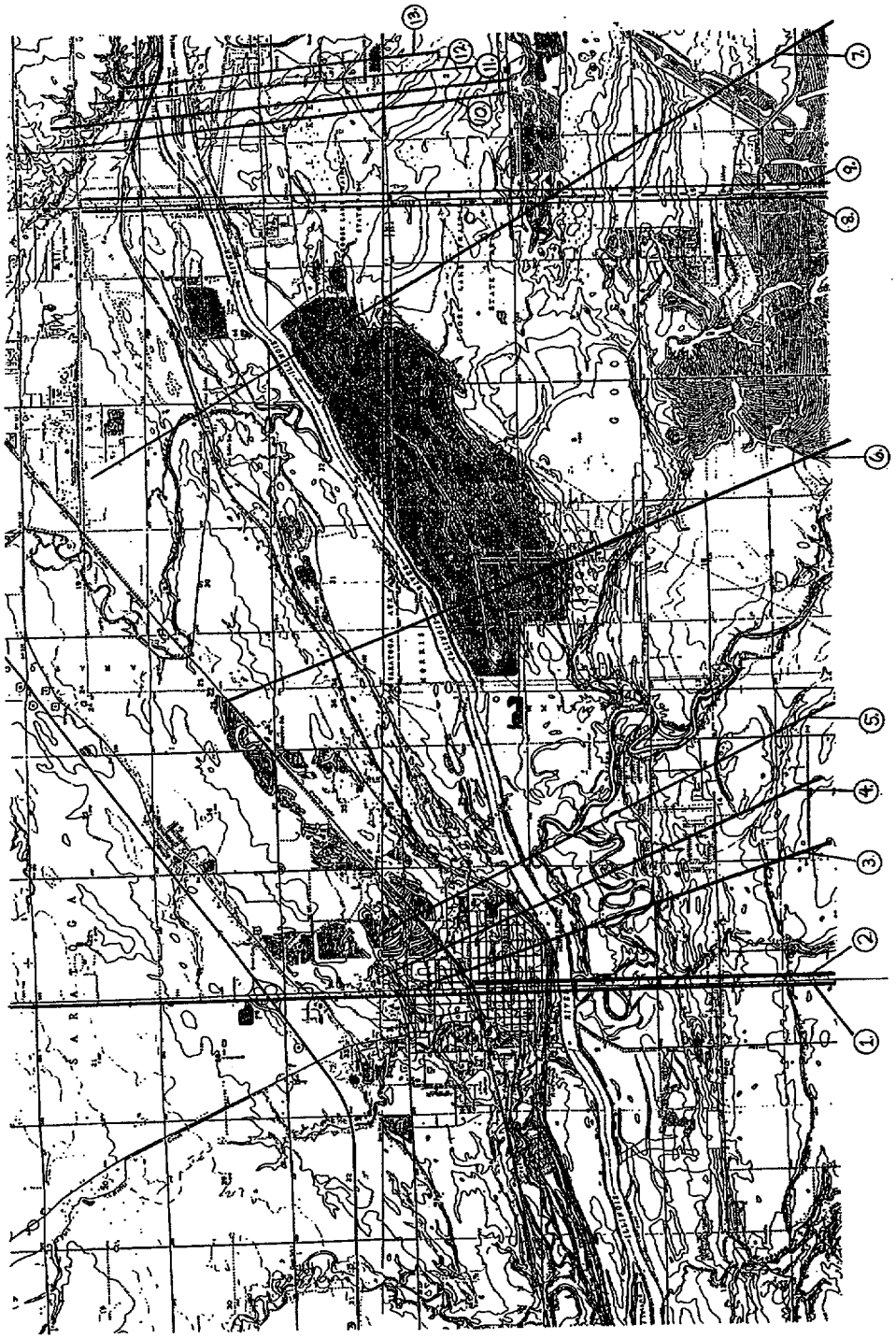


Figure A.6-2. GE-Morris Operation Site Study Reach.

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

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

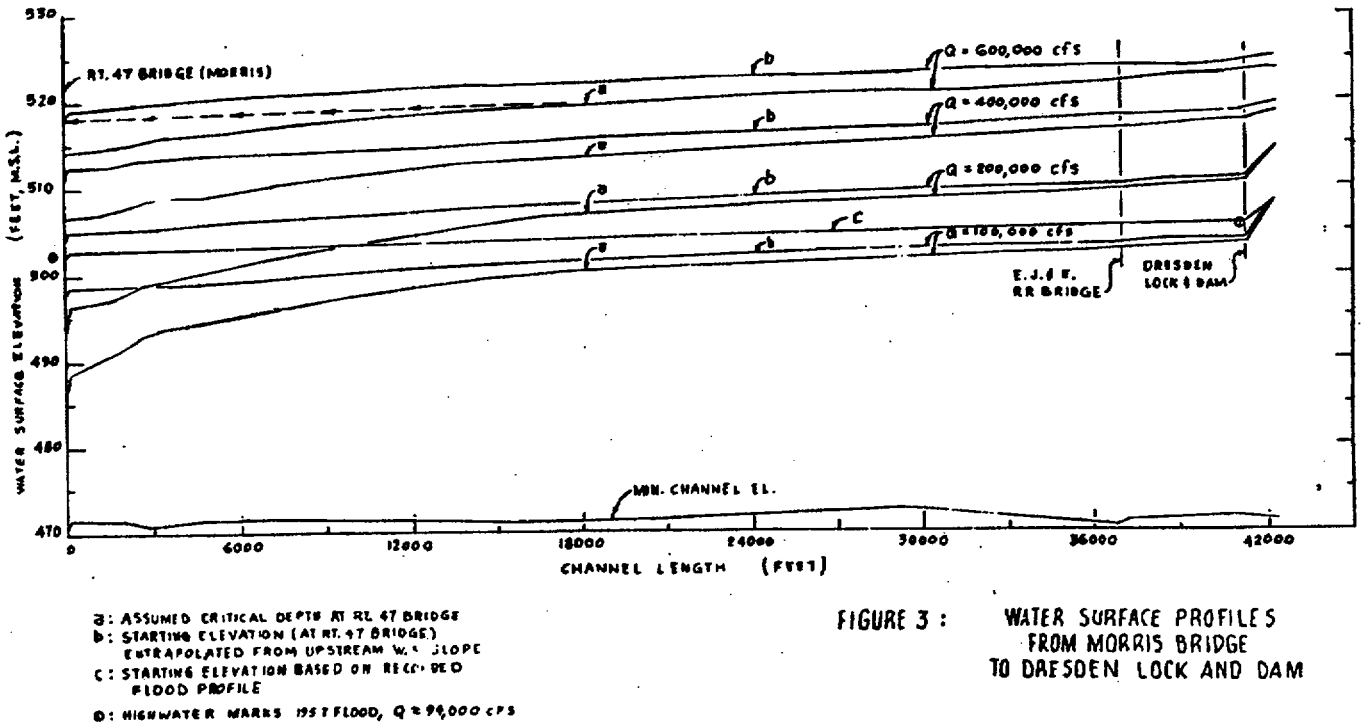


Figure A.6-3. Water Surface Profiles from Morris Beidge to Dresden Lock and Dam.

A.7 DECOMMISSIONING PLAN

A.7.1 INTRODUCTION

A.7.1.1 Purpose and Scope of Plan

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

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

A.7.1.2 History of Operations

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

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

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

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

A layaway program was initiated in February 1975 to place reprocessing equipment, instruments and certain facilities in protective status to minimize deterioration. Concurrent with

fuel receipt and storage operations, procedures were developed and implemented to flush and purge vessels and piping to "mothball" mechanical and electrical equipment. As of December 1978, all reprocessing equipment is in layaway status at the site except for the uncontaminated fluorine production equipment which was sold and has been removed.

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

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

A.7.2 PLAN ASSUMPTIONS AND BASES

A.7.2.1 Site Status

This decommissioning plan is based on the following assumptions:

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

A.7.2.2 Performance Objectives

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

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

    a. Remove the contaminated surface from the site for authorized disposition.
    b. Entomb on-site if such action is supported by evaluation of potential risk exposure and accepted by regulatory authority.
    c. Apply surface covering (paint, etc.) only if contamination levels are as low as can be obtained by reasonable effort or if such action is approved by regulatory authority.

- Remove piping, ducting and vessels for authorized salvage or disposal if their interior surfaces cannot be ensured of meeting unrestricted release limits. (Entomb on site if supported by evaluation of potential risk exposure and accepted by regulatory authority.)
- Dispose of scrap, rubble and other waste materials from site clean-up operations in accordance with applicable provisions of the Code of Federal Regulations, 10 CFR 72.130.

A.7.2.3 Other Considerations

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

A.7.3 PLANNED TASKS

A.7.3.1 Radiation Survey

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

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

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

A.7.3.2 Supplementary Systems

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

A.7.3.3 Bulk Materials Removed

A.7.3.3.1 Waste Vault Contents

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

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

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

A.7.3.3.2 Contaminated Equipment

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

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

A.7.3.4. Residual Contamination Survey/Assessment

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

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

A.7.3.5 System Decontamination and Dismantling

A.7.3.5.1 Fuel Receipt and Storage Facilities

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

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

A.7.3.5.2 Canyon

The plan for the canyon cells is to:

- Remove all fixed piping (other than imbedded) and instrument and electric cables.
- Decontaminate all surfaces. Remove stainless cell liners if the performance objectives (Section A.7.2.2) cannot be met with them in place.
- Decontaminate or package canyon cell covers and the canyon crane.
- Decontaminate imbedded piping and fill with grout.
- Leave the main building concrete structure including the canyon area in place after decontamination.
- Package and ship contaminated waste to off-site disposal facilities.

A.7.3.5.3 Other Main Building Areas

Several areas are not used for fuel storage operations. Other areas used during fuel storage operations may be minimally contaminated. The plan for these areas is to:

- Remove contaminated equipment.
- Remove and package other contaminated items such as instruments, piping ducts, services.
- Decontaminate area surfaces with techniques employed in the canyon cells.
- Package and ship contaminated materials to off-site disposal facilities.

A.7.3.5.4 Waste Storage Vaults

It is anticipated that minimal contamination (principally natural uranium) remains in the DCV. Assuming successful decontamination, the DCV will be backfilled with dirt and sealed, leaving the concrete walls and liners intact.

Radioactive contamination in the LAW and Cladding Vault consists almost exclusively of radiocobalt, radiocesium and radionickel. The plan for these vaults is to:

- Investigate the feasibility of further decontamination of inner walls.
- Backfill and seal the maintenance pit and off-gas cell openings, leaving the walls, inner tank and liners intact.
- If residual contamination levels prove unacceptable, the inner tank/liner shall be removed and shipped for burial or metal melt.

These structures will be decommissioned last permitting use of main building ventilation for the majority of the decommissioning work. The plan for these structures is to:

- Flush the floor of the air tunnel. Route the flush solution to the radwaste system.
- Either fill the air tunnel with concrete over its entire length or decontaminate to acceptable limits and fill. Seal the cell openings to the air tunnel.
- Remove the exhaust blowers and duct work located next to the sand filter.
- Remove the contaminated sand and gravel from the sand filter as required. Package it and ship it to an off-site disposal facility.
- Decontaminate and backfill the sand filter concrete structure and seal the filter openings.
- Decontaminate and backfill or package the horizontal duct between the sand filter and stack.
- Decontaminate and cap (ground level and top) or dispose of the 300 ft. stack.
- Package and ship contaminated materials to an off-site disposal facility.

A.7.3.5.6 Final Waste Removal

The remaining items to be considered are:

- Decontaminate potentially contaminated underground piping and fill with grout or dig up and package for disposal.
- Decontaminate special equipment used in decommissioning work and package for disposal.
- Package miscellaneous tools that are no longer useful for disposal.

A.7.3.6 Final Survey

A comprehensive final survey similar to the initial one described in Section A.7.3.1 will be performed. The survey report will include:

- Description of scope and general procedures used in the survey.
- Description of remaining contamination.
- Results of survey for comparison with performance objectives.
- Surveillance recommendations and future use restrictions.

A.7.3.7 Inspection and Acceptance

A final survey report will be submitted to the USNRC.

It is anticipated that the USNRC will terminate Materials License No. SNM-2500 and release the facility for unrestricted use following their review and inspection.

A.7.4 PLAN ENVIRONMENTAL EFFECTS

A.7.4.1 Balancing of Effects

The decommissioning plan described in this document presents what is believed to be the most balanced approach to limiting environmental effects as they relate to potential risks to the public and site personnel. In summary, the approach involves evaluating each task of the plan at the time of implementation, and making the final decision for disposition based on a comparison of the alternatives below:

- Decontamination to unrestricted limits
- Removal to off-site disposal facilities
- Fixation and isolation

This approach ensures an optimization of effects.

A.7.4.2 Conclusions

Dispersal of significant radioactivity as a result of the implementation of this plan is highly unlikely. The main building ventilation system will be operated to provide normal filtration of particulate and aerosol matter. There are no radioactive liquid effluents from the site during normal license operations and there will be none during decommissioning activities. Radioactive wastes will be disposed of by transporting to licensed repositories in approved containers. Approved shipping practices shall be followed, thereby creating no significant impact on the environment.

After the performance objectives of the plan have been attained, the site will be available for unrestricted use with no impact on the environment.

A.7.5 RESOURCE REQUIREMENTS

A.7.5.1 Manpower Estimates

General Electric will carry out the specific tasks defined in Section A.7.3, utilizing Company personnel, contractor personnel, or a combination of both. Table A.7-1 depicts cost estimates for the various tasks using General Electric 1992 manpower rates for the on-site work. The removal of vault bulk materials was assumed to be carried out by subcontractors.

In estimating manpower requirements, it is anticipated that total implementation of the decommissioning plan will take 3 years. Some tasks will be performed in parallel but the general sequence of tasks is that described in Section A.7.1.

A.7.5.2 Shipping and Disposal Costs

Shipping and burial cost estimates include 1996 costs of shipping containers (nonreusable), transportation fees, and burial charges at a low-level waste disposal site. The cost estimate includes weights and volumes of materials based on past experience of GE-MO. The transportation costs assume that the waste will be transported to an unspecified out of state burial facility. (In reality, waste will be disposed in the designated Midwest Compact Commission disposal site scheduled to open in 1999.)

Disposal of "clean" materials is not included in the costs shown in Table A.7-1 since noncontaminated items are not addressed in this plan. (See Section A.7.2.2.)

A contingency of 25% of the decommissioning cost (Table A.7-1) was included in the total cost shown.

A.7.5.3 Financial Assurance

Decommissioning costs for the GE-MO facility are small compared to the total assets of the General Electric Company. Therefore, it is unlikely that General Electric would be unable to meet the financial commitments generally associated with the decommissioning activities as outlined and estimated.

The General Electric Company fulfills the requirements of 10 CFR30 Appendix C, "Criteria Pertaining to Use of Financial Tests and Self Guarantees for Providing Reasonable Assurance of Funds for Decommissioning".

By action of the Board of Directors in meeting on April 27, 1979, (Minute #9640, April 27, 1979), a Vice President of General Electric Company may execute such an obligation on behalf of the Company.

TABLE A.7-1
MORRIS DECOMMISSIONING -- TOTAL COST SUMMARY

	TASK	Total GE Labor (Man-hrs)	(000 $)	LSA Waste Pkg/ Transport/Burial Volume (Cu-ft)	Cost $/Cu-ft $530.00)	Other Purchased Services (000 $)	Mat'l & Equip (000 $)	Total Cost (000 $)
1.	Integration work including Licensing, Engineering, and Health Physics	37,240	2,086	N/A	N/A	N/A	41	2,127
2.	Energy Costs (Utilities, Etc.)	3,587	157	N/A	N/A	N/A	690	847
3.	Special Services (Include Security, Site, Monitoring, Janitorial, & Landlord)	N/A	N/A	N/A	N/A	1,190	N/A	1,190
4.	Insurance, Taxes, & Special Fees	N/A	N/A	N/A	N/A	1,415	N/A	1,415
5.	Bulk Materials Removal for Vaults & Basins including Labor, Consumable Materials, Equipment, Shipping, & Burial	17,116	773	N/A	N/A	2,393	77	3,243
6.	System Decontamination & Dismantling -- Total Task #6	58,575	2,646	8,325	4,412	1,534	398	8,990
	6A -- Fuel Receipt & Storage Facilities	16,610	750	4,100	2,173	332	120	3,375
	6B -- Canyon Cells	9,350	422	1,725	914	382	34	1,752
	6C -- Other Main Building Area	6,985	316	1,200	636	170	20	1,142
	6D -- Vaults	16,830	760	300	159	570	139	1,628
	6E -- Air Tunnel, Sand Filter, & Stack	8,800	398	1,000	530	80	85	1,093
7.	Subtotal (Tasks #1 thru #6)	116,518	5,662	8,325	4,412	6,532	1,206	17,812
8.	Contingency (25% - same as NUREG 0278							4,453
9.	TOTAL COST							22,265

A.8 AGING MANAGEMENT

Structures, systems and components at GE-MO that, while not performing a safety-related function, but do perform a function that demonstrates compliance with NRC regulations on environmental qualification, are identified in the CSAR, section 11, paragraph 11.3.

11.3 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY
No credible event, planned discharge or design basis accident at GE-MO is identified that would expose a member of the public to radiation in excess of limits specified in 10 CFR 72.104 or 10 CFR 72.106.

It is therefore, the position of GE-MO that the term "basic components" in the sense defined by 10 CFR 21.3(a)(2) and 10 CFR 21.3 (m) is not applicable to GEMO.

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

a. Fuel storage basin - concrete walls, floors, and expansion gate are principal elements in protection of stored fuel, and in isolation of basin water from the environment.
b. Fuel storage basin - stainless steel liner forms a second element in fuel protection and basin water isolation, facilitating decontamination.
c. Fuel storage system, including baskets and supporting grids is a principal element in protection of stored fuel.
d. Unloading pit doorway guard - is designed to prevent a loaded fuel basket from being tipped so that fuel bundles could fall into the cask unloading pit. The unloading pit doorway guard is an element in protection of fuel during movement of a loaded basket.
e. Filter cell structure - the concrete cell part of the basin pump room area provides radiation shielding to reduce occupational exposure.

However, since these systems do contain the stored fuel or provide support functions, they have been reviewed for aging management.

In June 1993, the fuel storage basin was inspected to confirm expectations of continued structural integrity, as well as confirm the absence of microbe induced corrosion (MIC). To confirm and document the integrity of the liner, a routine inspection plan was developed in accordance with ASME Boiler and Pressure Vessel Code and other industry approved IVVI procedures. The inspection plan included use of underwater TV cameras to inspect the basin welds.

The results of this inspection showed, that based on high resolution visual inspection and surface examination, the basin liner is judged to have continued integrity, with no environmental degradation associated with 20+ years of fuel storage experience. Also, considering the continuous maintenance of high purity water flow in the fuel storage basins, continued long term service is indicated.

The above is detailed in report GENE 689-013-0893, "Morris Fuel Recovery Center Fuel Storage Basin Liner Visual Examination Summary Report", dated September 1993.

Additionally, in 1994 an approximately 1.5" x 3.5" coupon was cut from the basin liner in the cask unloading pit. This area was then had a patch welded over it. The sample was sectioned for optical metallography and scanning electron microscopy (SEM). Cross sectional views did not find evidence of significant surface attack, and the maximum surface penetration was 0.4 mils. SEM examination of the surface found oxide deposits, which is expected for a stainless steel that has been exposed to a water environment for 20+ years. Chemical analysis of the deposits determined the composition to be mostly iron oxide. No detrimental chemical species were found. No evidence of a MIC phenomena was observed.

The nominal liner wall thickness in the unloading pit is 0.125 inches. Assuming the degradation occurred over 20 years and the corrosion rate remained constant, the line would not be penetrated until 2050.

See report number GENE-689-003-0494, "Morris Fuel Recovery Center Fuel Storage Basin Liner Metallurgical Evaluation", dated May 1994.

While the above reports speak specifically to the basin liner, all SSC's in the basin are 304 Stainless Steel. Therefore, logically the same corrosion degradation could be applied to the entire fuel storage system, including baskets and supporting grids as a principal element in protection of stored fuel. All these items have been in a static mode since the last fuel receipt in 1988, so there also hasn't been any mechanical wear.

A.9 FUEL STORAGE SYSTEM HEAT TRANSFER

A.9.1 INTRODUCTION

Heat transfer from fuel storage baskets has been calculated for both BWR and PWR fuels. The basket assembly design is such that even with some hole plugging (considered an incredible event) fuel temperatures remain conservatively satisfactory. Even with the basin water cooling system inoperative, maximum water temperature would be about 183 °F (83.9 °C).

The calculation of heat transfer from maximum energy bundles to the pool water was conservatively based on 44,000 MWd/TeU exposure and a cooling time of 120 days, for a basin water temperature of 120 °F (48.9 °C), and with other considerations as described in the analyses[1]. Three different cases involving heat transfer from bundles in the basket assembly to water for three flow paths were analyzed. Figures A.9-1 and A.9-2 show dimensions and cooling water flow paths in and around the bundles and baskets. The three cases are:

Case I - Center BWR Tube
Water flows up through the latch rod guide area of the BWR basket base plate through the 1 1/2 in. diameter hole in the tube, and either up through the bundle or in the space between the bundle and the tube wall.

Case 2 - Outside BWR Tube
Water flows between the basket base plates of a BWR basket, through the 1 1/2 in. diameter hole in the tube, and either up through the bundle or in the space between the bundle and the tube wall.

Case 3 - Outside PWR Tube
Water flows between the basket base plates of a PWR basket, through the 1 1/2 in. diameter hole in the pipe, and either up through the bundle or in the space between the bundle and the tube wall.

A computer program was written to calculate heat transfer coefficients and water temperatures for the bundle-basket assemblies. The program divides bundle length into 20 nodes or sections. Heat and mass transfer calculations were done for each node to balance heat transfer with water flow. The results of this program are summarized in Tables A.9-1, A.9-2, and A.9-3, which indicate that temperature rise for water flowing around a BWR and PWR bundle is 13 °F (7.2 °C) and 22 °F (12.2 °C), respectively. The respective maximum rod surface temperatures are 130 °F (54.4°C) and 146°F (63.3°C). There was little difference between Case 1 and Case 2, since most of the pressure drop is across the 1 1/2 in. diameter hole. This pressure drop for BWR tubes is 0.58 and 0.61 lb./ft.^{2} (Case 1 and Case 2) and 1.41 lb./ft.^{2} for PWR tubes.

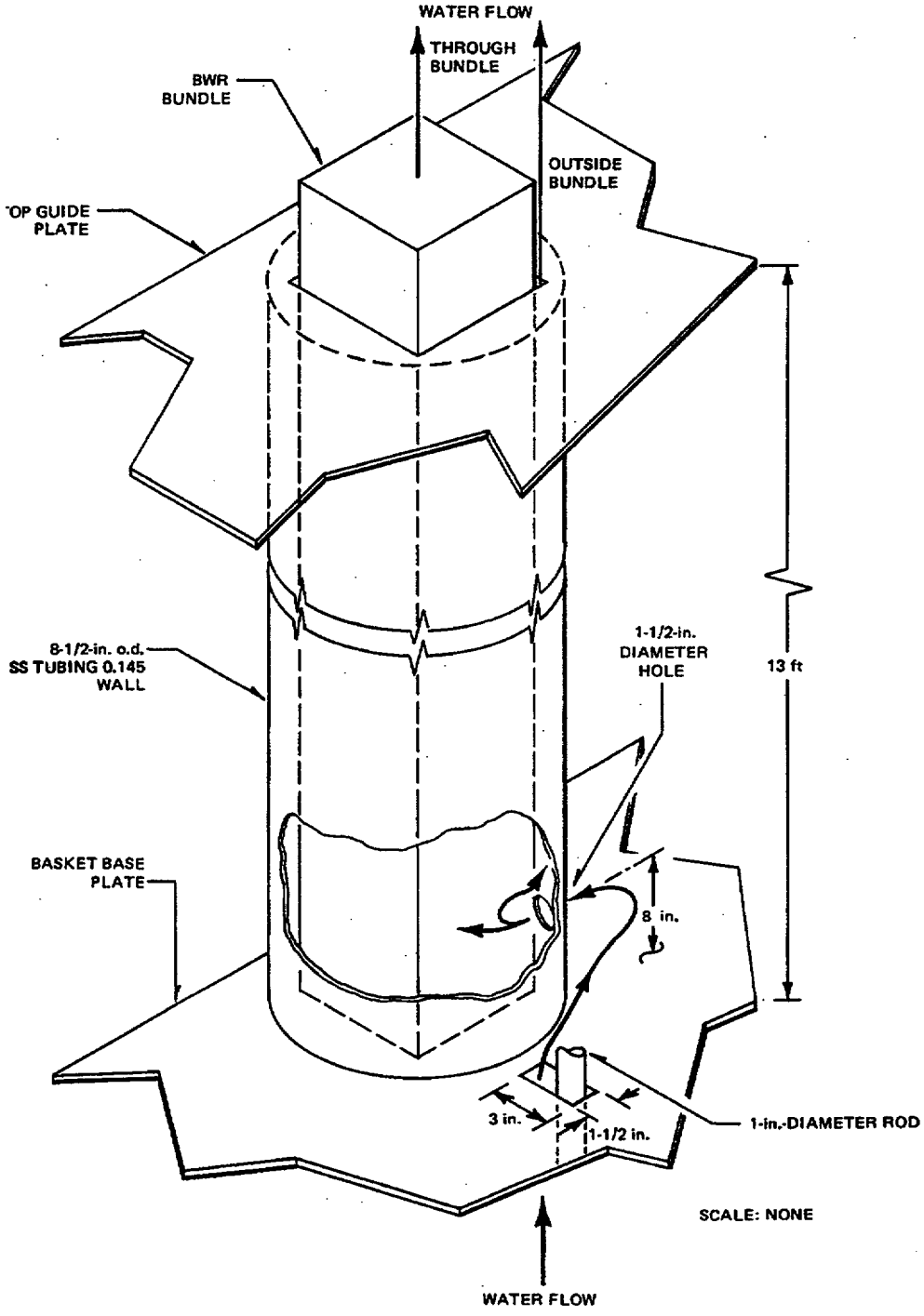
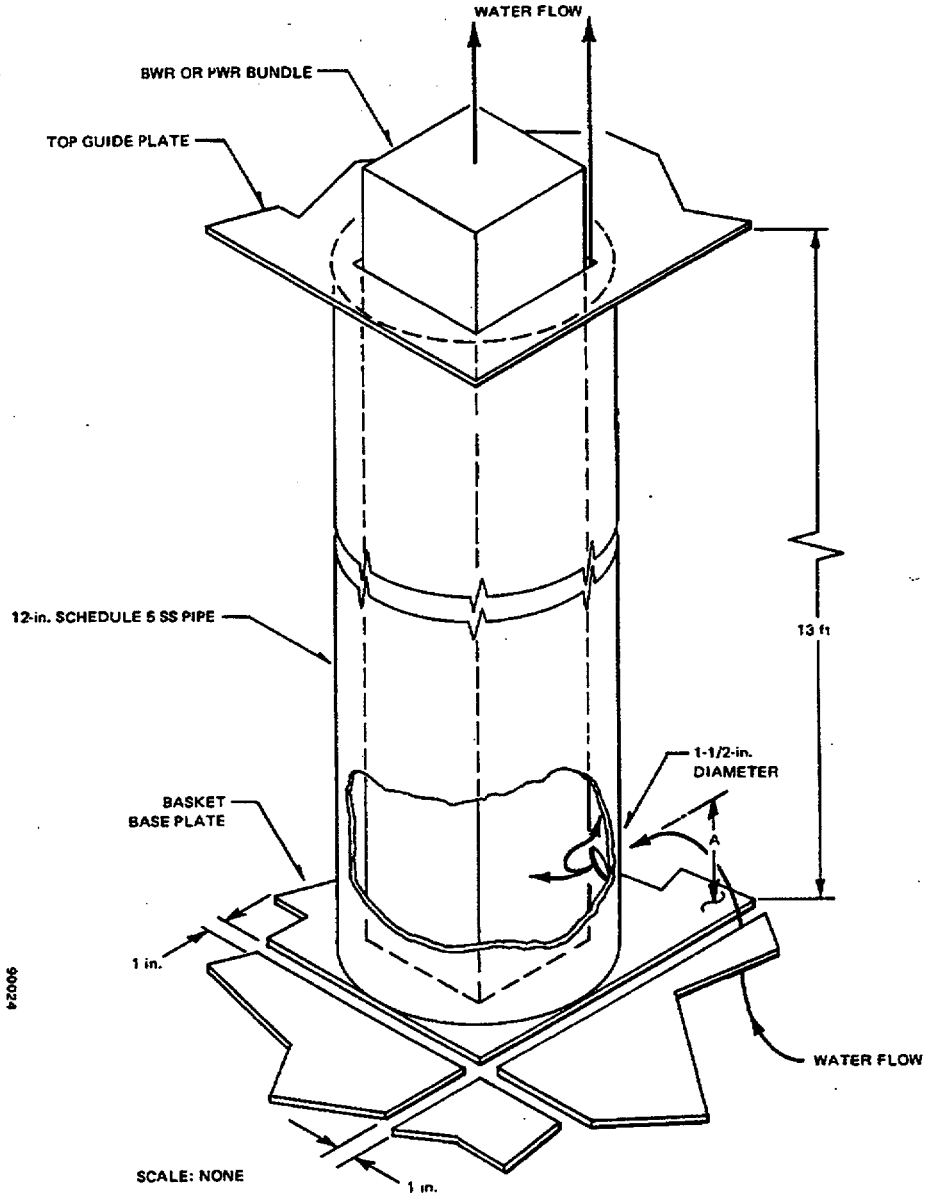


Figure A.9-1. Case 1: Flow of Water Through BWR Bundle by Internal Cooling.



BUNDLE DIMENSIONS

TYPE	ROD o.d. (in.)	ARRAY	ROD C/C DIST (in.)	ACTIVE LENGTH (ft)	A (in.)
BWR	0.563	7x7	0.738	13	8
PWR	0.422	15x15	0.562	13	5

Figure A.9-2. Cases 2 and 3: Flow of Water Through BWR or PWR Bundle by External Cooling Hole.

The model used in making the above calculations is based on water flowing up through the bundle and in the space between the bundle and tube. An iterative approach was used to solve the problem by assuming an outlet water temperature (thereby establishing the water flow rate), tube wall to water temperature difference and fraction of bundle heat transmitted to water inside tube. Heat and mass transfer calculations were made for each node, and balances were made, shown in flow chart Figure A.9-3. The total bundle pressure drop was then compared to a water density change (driving force) resulting from water temperature increase. Any differences between total pressure drop and driving force were adjusted until pressure drop due to system interference (spacer, tie plate, orifices, etc.) equaled driving force pressure drop. At this point, a printout was made of node temperatures, heat transfer coefficients, pressure drop, fraction of total heat, water flow rate and wall/fluid temperature difference (Tables A.9-1, A.9-2 and A.9-3).

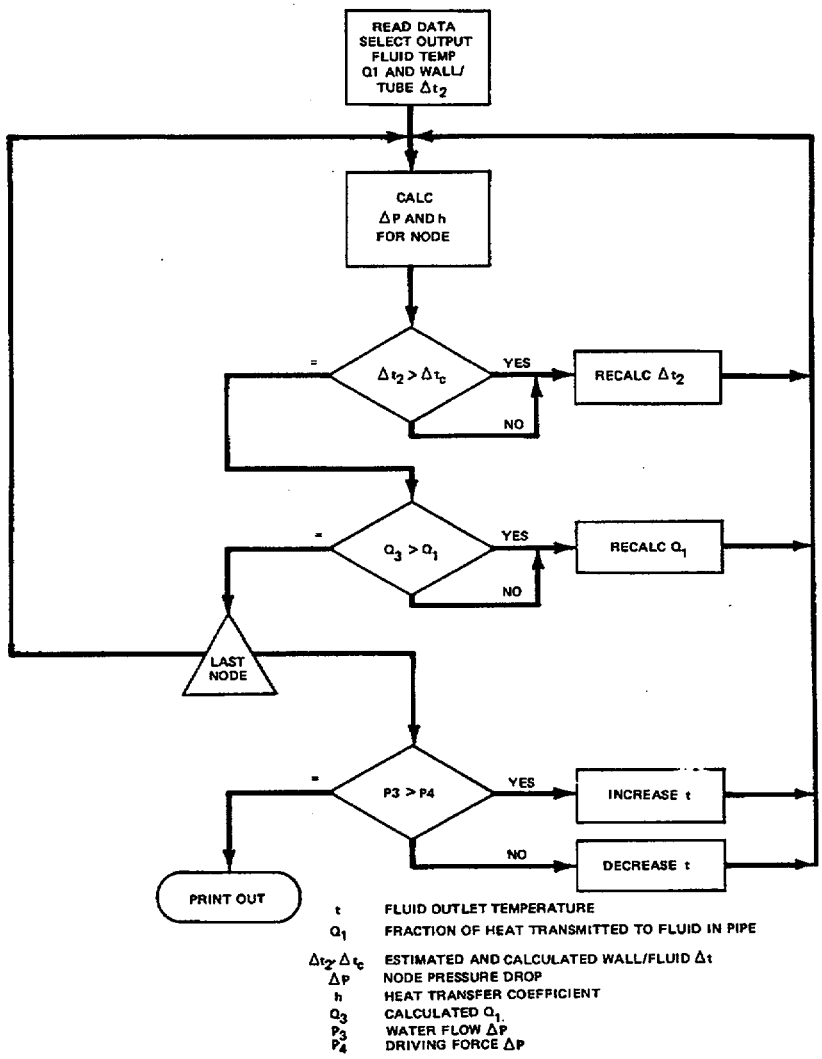


Figure A.9-3. Flow Chart, Fluid Cooling of Irradiated Fuel Bundles.

The basic equations referred to in the following are listed in Table A.9-4. Nomenclature is defined in Table A.9-5.

TABLE A.9-1
Case 1
BWR BUNDLE STORED IN BASKET UNDERWATER
Tube Hole is Internal
44,000 MWd/Te 120-day Cooled

H_2O in °F	Exit °F	Flow ΔP	H_2O ΔP	ΔT °F
1.20 E+02	1.33 E+02	1.62 E+00	1.63 E+00	1.35 E+01

Node	Temp °F	H-B/HF2F	ΔP lb/ft²	Q/Fraction	Flow lb/hr	ΔT R_w/W °F
1.00 E+00	1.20 E+02	3.16 E+01	7.38 E-01	9.98 E-01	1.27 E+03	5.76 E+00
2.00 E+00	1.21 E+02	3.16 E+01	4.73 E-02	9.98 E-01	1.27 E+03	5.76 E+00
3.00 E+00	1.22 E+02	3.16 E+01	5.00 E-02	9.89 E-01	1.27 E+03	5.76 E+00
4.00 E+00	1.23 E+02	3.16 E+01	4.67 E-02	9.85 E-01	1.27 E+03	5.76 E+00
5.00 E+00	1.23 E+02	3.16 E+01	4.65 E-02	9.81 E-01	1.27 E+03	5.75 E+00
6.00 E+00	1.24 E+02	3.17 E+01	4.91 E-02	9.76 E-01	1.27 E+03	5.75 E+00
7.00 E+00	1.24 E+02	3.17 E+01	4.54 E-02	9.72 E-01	1.27 E+03	5.75 E+00
8.00 E+00	1.25 E+02	3.17 E+01	4.54 E-02	9.68 E-01	1.27 E+03	5.75 E+00
9.00 E+00	1.26 E+02	3.17 E+01	4.54 E-02	9.64 E-01	1.27 E+03	5.75 E+00
1.00 E+01	1.26 E+02	3.17 E+01	4.51 E-02	9.59 E-01	1.27 E+03	5.75 E+00
1.10 E+01	1.27 E+02	3.17 E+01	4.78 E-02	9.54 E-01	1.27 E+03	5.75 E+00
1.20 E+01	1.28 E+02	3.17 E+01	4.46 E-02	9.50 E-01	1.27 E+03	5.74 E+00
1.30 E+01	1.29 E+02	3.17 E+01	4.73 E-02	9.48 E-01	1.27 E+03	5.74 E+00
1.40 E+01	1.29 E+02	3.17 E+01	4.41 E-02	9.41 E-01	1.27 E+03	5.74 E+00
1.50 E+01	1.30 E+02	3.17 E+01	4.39 E-02	9.37 E-01	1.27 E+03	5.74 E+00
1.60 E+01	1.30 E+02	3.17 E+01	4.66 E-02	9.33 E-01	1.27 E+03	5.74 E+00
1.70 E+01	1.31 E+02	3.17 E+01	4.34 E-02	9.28 E-01	1.27 E+03	5.74 E+00
1.80 E+01	1.32 E+02	3.18 E+01	4.61 E-02	9.24 E-01	1.27 E+03	5.74 E+00
1.90 E+01	1.32 E+02	3.18 E+01	4.30 E-02	9.20 E-01	1.27 E+03	5.73 E+00
2.00 E+01	1.33 E+02	3.18 E+01	4.93 E-02	9.15 E-01	1.27 E+03	5.73 E+00

TABLE HEADINGS

H_2O in °F	Temperature of Water Supply (°F)
Exit °F	Water Temperature After Passing Through Tube (°F)
Flow ΔP	Total Water Pressure Drop (lb/ft²)
H_2O ΔP	Driving Force Pressure Drop (lb/ft²)
ΔT °F	Temperature Rise of Water Passing Through Tube (°F)
H-B/HF2F	Node Heat Transfer Coefficient (Btu/hr ft² °F)
ΔP lb/ft²	Node Pressure Drop (lb/ft²)
Q/Fraction	Fraction of Heat Absorbed by Water Inside of Tube
Flow lb/hr	Flow Rate of Water (lb/hr)
ΔT R_w/W °F	Rod Wall to Water Temperature Difference (°F)

TABLE A.9-2
Case 2
BWR BUNDLE STORED IN BASKET UNDERWATER
Tube Hole is External
44,000 MWd/Te 120-day Cooled

H_2O in °F	Exit °F	Flow ΔP	H_2O ΔP	ΔT °F
1.20 E+02	1.33 E+02	1.59 E+00	1.60 E+00	1.32 E+01

Node	Temp °F	H-B/HF2F	ΔP lb/ft^2	Q/Fraction	Flow lb/hr	ΔT R_w/W °F
1.00 E+00	1.20 E+02	3.18 E+01	6.97 E-01	9.98 E-01	1.30 E+03	5.72 E+00
2.00 E+00	1.21 E+02	3.18 E+01	4.83 E-02	9.94 E-01	1.30 E+03	5.72 E+00
3.00 E+00	1.22 E+02	3.19 E+01	5.11 E-02	9.89 E-01	1.30 E+03	5.72 E+00
4.00 E+00	1.22 E+02	3.19 E+01	4.77 E-02	9.85 E-01	1.30 E+03	5.72 E+00
5.00 E+00	1.23 E+02	3.19 E+01	4.74 E-02	9.81 E-01	1.30 E+03	5.72 E+00
6.00 E+00	1.24 E+02	3.19 E+01	5.02 E-02	9.76 E-01	1.30 E+03	5.71 E+00
7.00 E+00	1.24 E+02	3.19 E+01	4.69 E-02	9.72 E-01	1.30 E+03	5.71 E+00
8.00 E+00	1.25 E+02	3.19 E+01	4.87 E-02	9.68 E-01	1.30 E+03	5.71 E+00
9.00 E+00	1.26 E+02	3.19 E+01	4.64 E-02	9.64 E-01	1.30 E+03	5.71 E+00
1.00 E+01	1.26 E+02	3.19 E+01	4.61 E-02	9.59 E-01	1.30 E+03	5.71 E+00
1.10 E+01	1.27 E+02	3.19 E+01	4.59 E-02	9.55 E-01	1.30 E+03	5.71 E+00
1.20 E+01	1.28 E+02	3.19 E+01	4.56 E-02	9.51 E-01	1.30 E+03	5.71 E+00
1.30 E+01	1.28 E+02	3.19 E+01	4.94 E-02	9.46 E-01	1.30 E+03	5.70 E+00
1.40 E+01	1.29 E+02	3.19 E+01	4.51 E-02	9.42 E-01	1.30 E+03	5.70 E+00
1.50 E+01	1.30 E+02	3.19 E+01	4.48 E-02	9.38 E-01	1.30 E+03	5.70 E+00
1.60 E+01	1.30 E+02	3.19 E+01	4.77 E-02	9.34 E-01	1.30 E+03	5.70 E+00
1.70 E+01	1.31 E+02	3.20 E+01	4.44 E-02	9.29 E-01	1.30 E+03	5.70 E+00
1.80 E+01	1.32 E+02	3.20 E+01	4.72 E-02	9.25 E-01	1.30 E+03	5.70 E+00
1.90 E+01	1.32 E+02	3.20 E+01	4.39 E-02	9.21 E-01	1.30 E+03	5.70 E+00
2.00 E+01	1.33 E+02	3.20 E+01	5.05 E-02	9.16 E-01	1.30 E+03	5.69 E+00

TABLE HEADINGS

H_2O in °F	Temperature of Water Supply (°F)
Exit °F	Water Temperature After Passing Through Tube (°F)
Flow ΔP	Total Water Pressure Drop (lb/ft^2)
H_2O ΔP	Driving Force Pressure Drop (lb/ft^2)
ΔT °F	Temperature Rise of Water Passing Through Tube (°F)
H-B/HF2F	Node Heat Transfer Coefficient (Btu/hr ft^2 °F)
ΔP lb/ft^2	Node Pressure Drop (lb/ft^2)
Q/Fraction	Fraction of Heat Absorbed by Water Inside of Tube
Flow lb/hr	Flow Rate of Water (lb/hr)
ΔT R_w/W °F	Rod Wall to Water Temperature Difference (°F)

TABLE A.9-3
Case 3
PWR BUNDLE STORED IN BASKET UNDERWATER
44,000 MWd/Te 120-day Cooled

H_2O in °F	Exit °F	Flow ΔP	H_2O ΔP	ΔT °F
1.20 E+02	1.42 E+02	2.62 E+00	2.64 E+00	2.19 E+01

Node	Temp °F	H-B/HF2F	ΔP lb/ft^2	Q/Fraction	Flow lb/hr	ΔT R_w/W °F
1.00 E+00	1.21 E+02	3.11 E+01	1.50 E+00	9.98 E-01	1.98 E+03	4.75 E+00
2.00 E+00	1.22 E+02	3.11 E+01	6.05 E-02	9.95 E-01	1.98 E+03	4.74 E+00
3.00 E+00	1.23 E+02	3.11 E+01	6.71 E-02	9.92 E-01	1.98 E+03	4.74 E+00
4.00 E+00	1.24 E+02	3.11 E+01	5.91 E-02	9.89 E-01	1.98 E+03	4.74 E+00
5.00 E+00	1.25 E+02	3.11 E+01	5.91 E-02	9.85 E-01	1.98 E+03	4.74 E+00
6.00 E+00	1.26 E+02	3.11 E+01	5.54 E-02	9.82 E-01	1.98 E+03	4.73 E+00
7.00 E+00	1.27 E+02	3.12 E+01	5.77 E-02	9.79 E-01	1.98 E+03	4.73 E+00
8.00 E+00	1.28 E+02	3.12 E+01	6.41 E-02	9.76 E-01	1.98 E+03	4.73 E+00
9.00 E+00	1.29 E+02	3.12 E+01	5.67 E-02	9.72 E-01	1.98 E+03	4.73 E+00
1.00 E+01	1.30 E+02	3.12 E+01	5.43 E-02	9.69 E-01	1.98 E+03	4.73 E+00
1.10 E+01	1.31 E+02	3.12 E+01	6.29 E-02	9.66 E-01	1.98 E+03	4.72 E+00
1.20 E+01	1.32 E+02	3.12 E+01	5.52 E-02	9.63 E-01	1.98 E+03	4.72 E+00
1.30 E+01	1.34 E+02	3.12 E+01	5.12 E-02	9.59 E-01	1.98 E+03	4.72 E+00
1.40 E+01	1.35 E+02	3.13 E+01	5.47 E-02	9.56 E-01	1.98 E+03	4.72 E+00
1.50 E+01	1.36 E+02	3.13 E+01	5.39 E-02	9.52 E-01	1.98 E+03	4.72 E+00
1.60 E+01	1.37 E+02	3.13 E+01	6.09 E-02	9.49 E-01	1.98 E+03	4.72 E+00
1.70 E+01	1.38 E+02	3.13 E+01	5.30 E-02	9.46 E-01	1.98 E+03	4.71 E+00
1.80 E+01	1.39 E+02	3.13 E+01	5.99 E-02	9.43 E-01	1.98 E+03	4.71 E+00
1.90 E+01	1.40 E+02	3.13 E+01	5.22 E-02	9.40 E-01	1.98 E+03	4.71 E+00
2.00 E+01	1.41 E+02	3.13 E+01	5.01 E-02	9.36 E-01	1.98 E+03	4.71 E+00

TABLE HEADINGS

H_2O in °F	Temperature of Water Supply (°F)
Exit °F	Water Temperature After Passing Through Tube (°F)
Flow ΔP	Total Water Pressure Drop (lb/ft^2)
H_2O ΔP	Driving Force Pressure Drop (lb/ft^2)
ΔT °F	Temperature Rise of Water Passing Through Tube (°F)
H-B/HF2F	Node Heat Transfer **Coefficient** (Btu/hr ft^2 °F)
ΔP lb/ft^2	Node Pressure Drop (lb/ft^2)
Q/Fraction	Fraction of Heat Absorbed by Water Inside of Tube
Flow lb/hr	Flow Rate of Water (lb/hr)
ΔT R_w/W °F	Rod Wall to Water Temperature Difference (°F)

TABLE A.9-4
BASIC EQUATIONS

Pressure Drop

1. $$\Delta P_{orifice} = \frac{w^2}{2 g_c K^2 A^2 \rho} \quad (1)$$

2. $$\Delta P_{Friction} = \frac{32 w \mu L}{\rho g_c D_H^2 A} \quad (2)$$

3. $$\Delta P_{acc} = \frac{w^2}{g_c A^2}\left(\frac{1}{\rho_{out}} - \frac{1}{\rho_{in}}\right) \quad (3)$$

4. $$\Delta P_{local} = \left(\frac{K_2}{A_K^2}\right)\left(\frac{w^2}{2 g_c \rho}\right) \quad (3)$$

Heat Transfer

$$Q = h A_1 \Delta t_1 = w C_p \Delta t_2$$

$$h = \frac{1.86 c G}{\left(\frac{c\mu}{k}\right)^{\frac{2}{3}}\left(\frac{L}{D}\right)^{\frac{1}{3}}\left(\frac{\mu_w}{\mu}\right)^{0.14}\left(\frac{DG}{\mu}\right)^{\frac{2}{3}}} \quad (4)$$

(1) Crane Technical Paper 410, Flow of Fluids Through Valves, Fittings & Pipe, P. 2-14.

(2) John H. Perry, Ed., Chemical Engineer's Handbook, 4th edition, 1963, Tables 5-11, pp. 5-21.

(3) J. M. Healzer, D. R. Nelson, and H. S. Sakasegawa, COFCOR-ISCOR User's Manual A Digital Computer Program for the Steady-State Thermal Hydraulic Analysis of a Nuclear Reactor Core, General Electric Company, June 1970 (NEDE-10063).

(4) loc. cit., pp. 10-13. See Reference (1).

TABLE A.9-5
NOMENCLATURE

A	Cross-section Area (ft^2)
A_K	Area associated with K^2
A_1	Surface area (ft^2)
C_p, c	Specific heat (Btu/lb- °F)
D	Diameter (ft)
D_H	Hydraulic diameter (ft)
G	Mass velocity (lb/hr-ft^2 °F)
g_c	Gravitational constant 32.17 ($lb_m ft/lb_F sec^2$)
h	Heat transfer coefficient (Btu/hr-ft^2 °F)
k	Thermal conductivity (Btu/hr-ft °F)
K	Flow coefficient for square-edge orifices
K_2	Experimentally determined flow factor
L	Length (ft)
Q	Quantity of heat (Btu/hr)
Δt_1	Wall-air temperature (°F)
Δt_2	Outlet-inlet temperature (°F)
w	Mass flow (lb/sec)
Δp	Outlet-inlet pressure (lb/ft^2)
μ	Viscosity (lb/sec-ft)
μ_w	Wall viscosity (lb/hr-ft)
ρ	Density (lb/ft^3)

A.9.2 DISCUSSION OF CALCULATIONS

Equation 1 (Table A.9-4) is the equation used for pressure drop of square-edged orifices. A value of K = 0.6 was chosen since it represents a conservative value for flow through the areas defined. The real value probably lies somewhere in the 0.6 to 0.66 range. Selecting 0.6, the pressure drop calculated is slightly higher than might be expected. Equation 1 was used for water flow through the base plate latch rod hole, the 1 in. space between baskets, the 1 1/2 in. diameter hole in each basket tube and the area at the top of the bundle where water flows through the bundle opening of the top guide plate. Where the opening geometry is not circular, a comparable hydraulic radius was used.

Equation 2 (Table A.9-4) predicts pressure drop due to friction on irregularly-shaped surfaces. This equation was used to calculate pressure drop of water passing up through the bundle and area between bundle and tube. The symbol D_H is the hydraulic diameter for the flow area through bundle and area between bundle and tube. This cross-sectional area is designated as (A). A primary assumption of this equation is that the flow is laminar. Because the Reynolds numbers are less than 40, Equation 2 can be used.

As fluid density changes due to temperature changes, velocity of the fluid increases proportionally to volume increases. The increase in velocity by this mechanism creates a pressure drop termed "acceleration pressure drop" defined by Equation 3 (Table A.9-4). Each node in the program has an inlet and outlet density value which was used in the equation to calculate acceleration pressure drop between each node.

Extensive experiments by engineers in BWR reactor core design and core hydraulic analysis have produced data which describe pressure drop resulting from fluid passing through and around bundle tie plates and spacers[2]. Values derived from experiments are known as the "K factors" (K_2). That is, an arbitrary value for cross-sectional flow area (A_K) was assigned to the local disturbance (tie plate or spacer) and then pressure drops were measured when flow passed through and around the disturbance. The "K factor" was then calculated by rearranging Equation 4 and solving for K_2. Once K_2 is known, it is then used to calculate pressure drop for local disturbances. The K_2 values obtained were used in the program to calculate heat transfer coefficients and water temperatures around the upper tie plate and seven spacers in each BWR bundle. The lower tie plate K factor was not used since inlet water comes through a 1 1/2 in. diameter hole and bypasses this plate. Whenever a tie plate or spacer was within a node area, additional pressure drop was calculated using Equation 4 for flow through that node.

The heat transfer equations were used to calculate heat transferred from bundle to fluid and from tube wall to water outside of the tube. The Colburn equation is a laminar flow correlation relating Reynolds and Prandtl numbers to heat transfer coefficients.

A heat balance was established for each node by making the following simplifying assumptions:

- constant and uniform heat generation within the rod;
- water freely flows from within the bundle to the area between bundle and tube wall and vice versa;
- temperature of the fluid inside of bundle and outside of bundle is uniform due to large exchange of fluid from in-bundle to bundle-tube wall area;
- average node temperatures were used to determine fluid flow properties;
- average density change of the fluid was used to compute driving force of the fluid; and
- laminar flow form of the Colburn equation is applicable.

The two heat transfer equations were used to compute heat transfer coefficients, water flow rates and tube wall to fluid temperature differences. The term "Q/Fraction" in the output is the amount of bundle heat absorbed by water flowing on the inside of the tube. The remaining heat is transferred through the wall to basin water circulating outside of the tube. The program constantly compares and changes selected values with calculated values and iterates until selected versus calculated values are within 1% of each other. When this occurs, the program continues to the next set of calculations.

The effect of 1% comparison value can be noted from Tables A.9-1 through A.9-3 and by comparing Flow ΔP with H_2O ΔP values. These values are within 1% of each other but are not

exactly equal. A closer agreement than 1% was not necessary since it would not significantly increase accuracy of the answers.

Water temperature and maximum rod surface temperature for each case are shown in Table A.9-6.

Hole plugging is not considered a credible event. The pool water is filtered continuously and is free of sediment. The holes in the latchrod guide area are under the top basket plate such that nothing can sink from the surface and cover them. The 1.5 in. diameter holes in the side of the tubes are 7 to 8 in. off the bottom of the tube such that settling sediment, if any, could not plug them. The design of the square tube BWR storage baskets retains the same fuel geometry and flow paths of the round tube design with additional flow through the storage tubes and fuel bundles provided by holes in the bottom of the tubes as well as in the sides. The analysis of the round tube design assumed storage of 120 day cooled fuel which was allowed by the original GE-MO license. The current license limits storage to fuel bundles with a minimum of 1 year cooling. This reduces the maximum heat load in a tube by about a factor of 2. The analysis performed for the round tube design is therefore a conservative upper limit for the square tube baskets.

TABLE A.9-6
WATER TEMPERATURES AND MAXIMUM ROD SURFACE TEMPERATURES

Case	Description	Water Temperature Increase (°F/°C)	Maximum Rod Surface Temperature (°F/°C)
I	BWR Basket -- Internal Hole Flow	13.5/7.5	139/59.4
II	BWR Basket -- External Hole Flow	13.2/7.3	139/59.4
III	PWR Basket -- External Hole Flow	21.8/12.1	146/63.3

A.9.3 FUEL STORAGE BASIN WATER-HEAT TRANSFER WITH LOSS OF CIRCULATION

An analysis was made of pool water heating when the basin water cooling system is inoperative. Assuming the basin if filled to proposed capacity of fuel now in storage and that projected to be received, the heat load would be approximately 6.4 x 10^6 Btu/hr. The maximum pool temperature was determined to be 183 °F (83.9 °C).

The flow rates of the water circulating through passages in and around the baskets, up through the fuel assemblies and between the assemblies and tubes would be greater at higher pool temperatures than at normal temperatures due to the lower viscosities and the consequent

slightly lower pressure drops. Thus, it is conservative to assume that the temperature rise of the water flowing through and around the fuel assembly is the same as that given in Table A.9-6. Flow rates per assembly will be very close to those previously developed (Tables A.9-1, A.9-2, and A.9-3) or 1300 lb/hr for BWR and 1980 lb/hr for PWR fuel.

For BWR fuel in the basin, the water would enter the bottom of the tube at 183 °F (83.9 °C), flow through and around the assembly, and out the top of the tube at 196 °F (91.1 °C). The water would mix with cooler water as it rises to the pool surface, cool to 183 °F (83.9 °C) as it mixes in the pool water above the baskets, circulate down near the basin wall, under the basket bottom plates, and up through the tubes again.

The flow path around PWR fuel assemblies would be the same. However, the temperature of the exit water would be approximately 205 °F (96.1 °C). This would also mix with the upper pool water and reach the average pool temperature of 183 °F (83.9 °C).

Maximum temperatures reached by the rod surfaces can be obtained by adding the $\Delta T\ R_w/W$ °F from Case I and Case III to the maximum water temperatures. For BWR fuel, the rod temperatures would be approximately 202 °F (94.4 °C) and for PWR, 210 °F (98.9 °C).

Depending on the initial temperature (81 °F to 110 °F), the time it would take the temperature of the basin water to reach a maximum of 183 °F (83.9 °C) would be 140 to 190 hours. If something happened to the cooling system to cause it to be inoperative, there would be almost a week before the pool temperature would reach a maximum and most repairs could be effected in less than a week. Even if repairs would take longer than a week, the evaporation loss would be offset by adding make-up water as required from normal or emergency sources.

The heat transfer analysis was done with the aid of a computer program. The program was used to simulate transient heat transfer characteristics of the water, basin walls and soil. Subdividing the walls and soil into small increments, a steady-state heat transfer was assumed for small increments of time. The program was designed to handle a variety of heat capacities, heat transfer coefficients and initial temperatures to calculate the time required to reach steady-state conditions for a given transient.

Results of the calculations are given in Figure A.9-4 and Table A.9-7 for the different cases analyzed. The analysis shows that the time required to reach an equilibrium basin water temperature is dependent on initial temperature. In addition, this time period is relatively insensitive to concrete thermal conductivity or soil temperature. The largest heat loss occurs at the water surface by evaporation. Heat losses through concrete walls and floor are minimal.

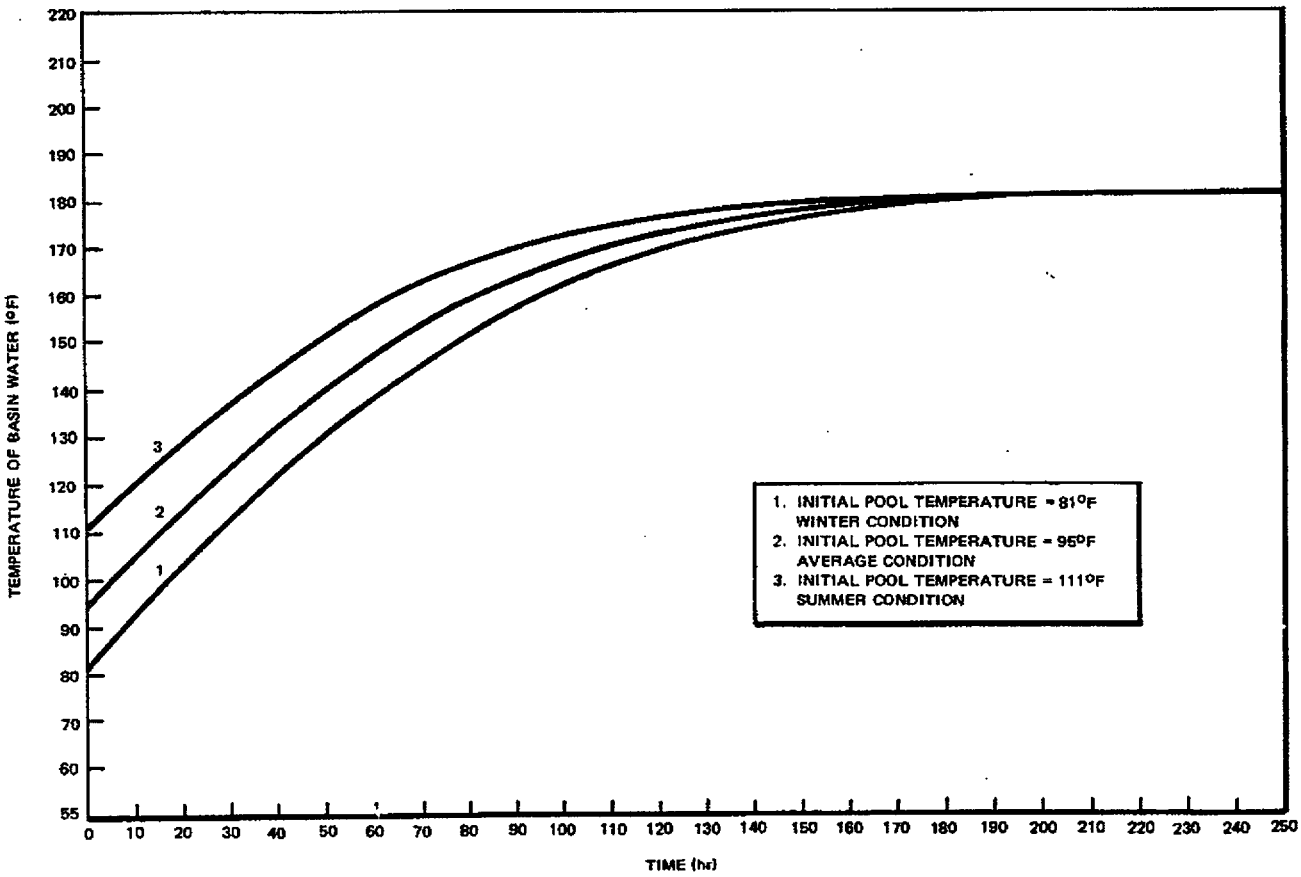


Figure A.9-4. Temperature of Basin Water versus Time for Loss-of-Coolant Accident.

TABLE A.9-7
TIME REQUIRED TO REACH EQUILIBRIUM TEMPERATURE IN FUEL BASINS

Initial Water Temperature (°F)	Soil Temperature (°F)	Concrete Thermal Conductivity (Btu-ft/hr-ft^2-°F)	Time (hr)
95	56	0.5	240
95	50	0.76	240
95	50	1.00	240
95	57	0.50	240
95	57	0.76	240
95	57	1.00	240
81	50	0.76	260
81	57	0.76	260
111	50	0.76	220
111	57	0.76	220

The calculation method is given as follows:

A.9.3.1 Calculational Method

Basic Equations

Heat Balance:

$$Q = Q_1 + Q_2 + Q_3$$

where:

Q	=	heat generated in pool by irradiated assemblies (Btu/hr);
Q_1	=	heat added to water (Btu/hr);
Q_2	=	heat transmitted through walls (Btu/hr); and
Q_3	=	heat loss by evaporation (Btu/hr).

Heat added to water, Q_1:

$$Q_1 = \frac{WC_p(\Delta t)_w}{\theta}$$

where:

W	=	Weight of water in pool (5.42×10^6 lb) (structures, baskets and fuel assemblies in pool are neglected);
C_p	=	specific heat of water (1.0 Btu/lb °F);
$(\Delta t)_w$	=	temperature increase of water (°F); and
θ	=	time in hours.

Heat transmitted through walls, Q_2:

$$Q_2 = hA\Delta t_D = \frac{k_1 A\Delta t_1}{X_1} + \frac{AX_1(C_p\rho)_c \Delta tm_c}{\theta} = \frac{k_2 A\Delta t_2}{X_2} + \frac{AX_2(C_p\rho)_s \Delta tm_s}{\theta}$$

where:

h	=	heat transfer coefficient water-concrete (35 Btu/hr - ft^2 °F for free convection);

A	=	area (ft^2);
Δt	=	temperature difference water to wall (°F);
k_1	=	thermal conductivity of concrete (Btu-ft/hr-ft^2 °F
Δt_1	=	temperature differential across concrete (°F);
X_1	=	thickness of concrete (ft);
k_2	=	thermal conductivity of soil (Btu-ft/hr-ft^2 °F);
$(C_p\rho)_{s,c}$	=	specific heat x density of soil, concrete;
Δt_2	=	temperature difference outer concrete surface to reference point in soil (°F);
X_2	=	soil thickness outer concrete surface to soil reference point;
Δtm_c	=	mean temperature increase of concrete with thickness, X_1 (°F); and
Δtm_s	=	mean temperature increase of soil with thickness, X_2 (°F).

Heat loss by evaporation, Q_3:

Data fit to the information given in **Handbook of Air Conditioning and Ventilating,** by Strock and Koral, Industrial Press, 2nd ed., p. 2-140 for 70°F air, 70% relative humidity, and zero air velocity gives the following equation:

$$Q_3 = (A_2 + B_1 t_w + B_2 t_w^2 + B_3 t_w^3)\, A_1$$

where:

A_1	=	pool surface area (2840 ft^2);
A_2	=	-968.29;
B_1	=	30.735;
B_2	=	-0.33104;
B_3	=	0.001409; and
t_w	=	water temperature (°F).

A.9.4 EQUATION DERIVATION

A.9.4.1 Pool Water Temperature

Combining the equivalent terms for the Q's in the heat balance and rearranging terms, the equation for water temperature is obtained:

$$t_w^3 + \left(\frac{B_2}{B_3}\right)t_w^2 + \left(\frac{wC_p}{\theta AB_3} + \frac{B_1}{B_2}\right)t_w + \left(\frac{A}{B_3} - \frac{wC_p t_s}{\theta AB_3} - \frac{Q}{AB_3} + \frac{Q_2}{AB_3}\right) = 0$$

where

ts = soil temperature (°F).

Once Q_2 is determined, this cubic can be solved for the water temperature, t_w. This temperature will allow for heat loss through the walls to floor of the basin and for evaporation loss from the pool surface.

A.9.4.2 Transient Equation

From the relationship:

(heat entering element) - (heat retained in element) = heat leaving element,

one can write the following equation:

$$\frac{k_1 A\left(t_{n-1}^{\theta} - t_n^0\right)}{\Delta X} - A\Delta X\left(C_p\rho\right)_c\left(t_n^{\theta+1} - t_n^{\theta}\right) = \frac{K_1 A\left(t_n^{\theta+1} - t_n^{\theta}\right)}{\Delta X}$$

where:

superscript θ, θ+1	=	present time increment, previous time increment, etc.
subscript n-1, n, n+1	=	previous increment, present increment, and next increment of concrete thickness;
ΔX	=	incremental thickness of concrete (or soil if in the soil region);
$(C_p\rho)_c$	=	concrete specific heat x concrete density (or the same constants for soil if in the soil region); and
t	=	temperature.

Rearranging terms and solving for $t_n^{\theta+1}$, the following equation results:

$$t_n^{\theta+1} = \frac{t_{n-1}^{\theta} + (M-2)t_n^{\theta} + t_{n+1}^{\theta}}{M}$$

where:

$M = \frac{\Delta X^2}{\alpha\theta}$, and

$$\alpha = \frac{k}{(C_p\rho)_c}$$

Solution of this equation involves the following steps:

a. Establish an initial steady-state temperature gradient from the pool water through the concrete and out into the soil at some reference distance. In the cases calculated, this reference distance was set at 7.0 ft.

b. Divide the concrete or soil thickness into thickness increments. In this case, a 3 in. thickness was used for concrete (i.e., X = 0.25 ft.) and a 6 in. thickness for the soil.

c. Set a time increment suitable for the problem involved (θ = 1 hr. for this problem).

d. Solve for water temperature.

e. Solve for concrete and soil temperatures for various thicknesses assumed.

f. Compute heat losses through concrete and evaporation heat loss.

g. Repeat (a) through (e) until stable water temperature is reached, or for as long a time period as desired.

The above procedure has been incorporated into a computer code which permits calculation of many different cases quickly.

A.9.5 CONCRETE THERMAL CONDUCTIVITY

Literature values for concrete vary widely and range from 0.4 to 1.0 Btu-ft/hr-ft^2 °F. The most probable conductivity for the Morris pool was arrived at by taking the concrete mix data, calculating a density from this data, and reading a thermal conductivity from the curve shown in Figure A.9-5.

Representative concrete mix data are shown below:

	wt/yd^3
Cement	470 lb
Fine Aggregate	1,450 lb
Coarse Aggregate	1,770 lb
Water	5 gal
Total wt/yd^3 =	3,732 lb

Shrinkage is negligible, so the density - (3732/27) = 138 lb/ft^3.

This is without air content, which runs 5%; thus, density used is 0.95 x 138 = 131 lb/ft^3. The thermal conductivity for this density is:

9.1 Btu-in/hr-ft^2 °F (0.76 Btu-ft/hr-ft^2 °F).

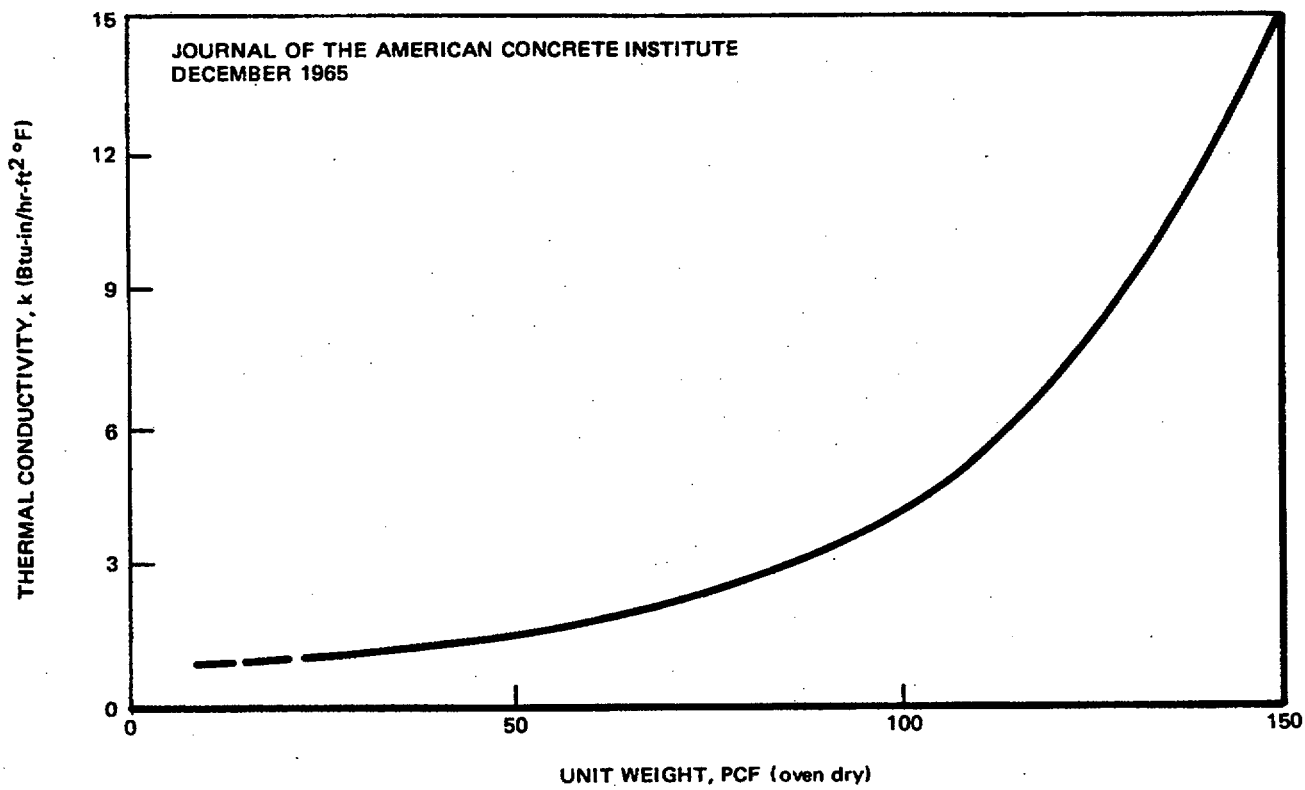


Figure A.9-5. Thermal Conductivity k of Concrete.

A.9.5.1 Assumptions and Conditions

The assumptions made are listed below:

a. Heat Losses by Conduction

- Pool water temperature is uniform throughout.
- There is no gap between the concrete and the rock it was poured against.
- No allowance is made for structural components in the pool such as framework, baskets, fuel assemblies, and interior concrete walls in the sensible heat losses. The pool was assumed to contain only water.
- Liner temperature is the same as pool water temperature.
- Initial pool water and surrounding strata was assumed to be 95 and 57 °F, respectively.
- Weight of water in pool, 5.42×10^6 lb.
- Water surface area available for evaporation, 2,840 ft^2.
- Inner concrete surface area available for heat loss to soil, not including inner walls but including floor area, 9,655 ft^2.

b. Heat Losses by Evaporation

- Enclosure walls are gone due to high winds or other action.
- Ambient air is at 70 °F and 70% relative humidity.
- Velocity of air is zero over pool surface but there is enough air movement that the surface is contacting essentially infinite air at the above conditions.

A.9.6 REFERENCES

1. The basis adopted in October 1976 for fuel exposure was 32,000 MWd/TeU, at specific power levels of 40 kW/kgU, and cooled for 180 days. The heat transfer calculations have not been changed from the old basis.

2. J. M. Healzer, D. R. Nelson, and H. S. Sakasegawa, **COFCOR-ISCOR User's Manual A Digital Computer Program for the Steady-State Thermal Hydraulic Analysis of a Nuclear Reactor Core,** General Electric Company, June 1970 (NEDE-10063).

A.10 FUEL BASKET SYSTEM NUCLEAR DESIGN CRITERIA AND BASES

A.10.1 INTRODUCTION

The design criteria for the fuel basket system are as follows:

a. In determination of subcritical limits, the k_{eff} calculated for the most reactive credible conditions shall be less than 0.95 at the 95% confidence level.

b. The initial k_∞ value of fuel to be stored without restrictions other than on the k_∞ value shall not exceed a rod lattice k_∞ of:

   1.37 for 15 x15 PWR fuel bundles (< 8.55 in.^{2})
   1.41 for 14 x 14 PWR fuel bundles (< 7.80 in.^{2})
   1.40 for 7 x 7 or 8 x 8 BWR fuel bundles
   1.38 for 10 x 10 BWR fuel bundles (< 5.65 in.^{2})

c. The k_∞ limit for BWR fuel shall be based on the initial design value of k_∞ (cold, clean fuel) as determined by the fuel designer.

d. The reactivity limit for specification PWR fuel shall be based on the initial cold, clean k_∞, including the poisoning effect of any stainless steel cladding, as determined by the fuel designer or utility customer.

e. For PWR fuel having k_∞ values in excess of the limits for unrestricted storage, the fuel shall have undergone sufficient irradiation to reduce the reactivity to a level below the storage limit taking into account the uncertainties in the calculations of burnup effects.

The k_{eff} for the basin filled with 15 x 15 PWR fuel at k_∞ of 1.37 would be 0.933 as calculated using equations developed by Battelle (Section 5.3.5.3). A k_∞ limit of 1.37 will also allow storage of stainless steel clad fuel enriched to 4.0% U-235 for which k_∞ cold clean would be 1.353.

An additional k_∞ limit has been established for the 14 x 14 PWR fuel since the smaller bundle size results in a lower k_{eff} for a given value of k. A k_∞ limit of 1.41 was established for this fuel since k_{eff} for the basin if filled with fuel at this k_∞ value would be approximately 0.920 at the 95% confidence level.

The rod lattice k_∞ limit of 1.40 for 7 x 7 or 8 x 7 BWR fuel was left unchanged from the original basis for the MFRP to avoid unnecessary changes. The basis for determining k_∞ for BWR fuel (criterion c) considers only cold, clean fuel to avoid the complexity of assessing the reduction in maximum k_∞ value caused by the burnable poison in the fuel. The cold, clean rod lattice k_∞ limit

of 1.40 covers any 7 x 7 or 8 x 8 BWR fuel that might be stored in the fuel storage facilities at Morris Operation (GE-MO).

These design criteria result in limiting the administrative control of fuel receipt largely to fuel identification and evaluation of the cold, clean rod lattice k_∞. The need for determination of the effects of irradiation on the k_∞ value of fuel to be stored should be very infrequent.

The design bases for the fuel basket system are as follows:

a. Criticality evaluations are based on the physical dimensions of specific fuel designs using the largest assembly widths and considering the length to be infinite.

b. The initial U-235 enrichment corresponding to various values of k_∞ was calculated.

c. The poisoning effect of the stainless steel (iron 74%, chromium 18%, nickel 8%, manganese neglected) in the storage basket was included in the criticality evaluation.

d. Fuel centerline location within the storage tube was assumed to be that giving the maximum system reactivity and fuel was assumed to be oriented with the horizontal axes in square array and parallel to the basket axes.

e. The principal criticality calculations were made using a water temperature of 20 °C since the codes employed for the calculations had been most extensively validated at this temperature.

A.10.2 FUEL BASKET SYSTEM - NUCLEAR DESIGN ANALYSIS

The nuclear design analysis was performed by Battelle Pacific Northwest Laboratories[1] using the preceding design criteria and bases and EGGNIT, GAMTEC-11, and KENO-II Monte Carlo computer codes.

Results of the calculations of critical systems to provide validation for the KENO-II code show the code to be slightly conservative (approximately 1.75%). Fuel characteristics are shown in Table A.10-1. The critical systems and calculated results are summarized in Table A.10-2.

Summarized below are the results of calculations used in establishing the bases for the fuel basket designs.

TABLE A.10-1
PHYSICAL CHARACTERISTICS OF REPRESENTATIVE LWR FUEL ASSEMBLIES

	1	2	3	4	5	6	7	8	9
Reactor Type	BWR	BWR	BWR	PWR	PWR	PWR	PWR	PWR	PWR
Fuel Designer	GE	GE	GE	B&W	CE	W	W	W	W
Active Fuel Length (in.)	144	144	144	144	137	120	122	144	144
Nominal Envelope (in.)	$(5.438)^2$	$(5.438)^2$	$(5.47)^2$	$(8.536)^2$	$(8.18)^2$	$(7.763)^2$	$(8.426)^2$	$(7.763)^2$	$(8.426)^2$
Rod Array	7x7	7x7	8x8	15x15	14x14	14x14	15x15	14x14	15x15
Rod Pitch (in.)	0.738	0.738	0.640	0.568	0.580	0.556	0.563	0.566	0.563
Rod o.d. (in.)	0.570	0.563	0.493	0.430	0.440	0.422	0.422	0.422	0.422
Clad Material	Zirc-2	Zirc-2	Zirc-2	Zirc-4	Zirc-4	SS	SS	Zirc-4	Zirc-4
Clad Thickness (in.)	0.0355	0.032	0.034	0.0265	0.026	0.0165	0.0165	0.0243	0.0243
Pellet o.d (in.)	0.488	0.487	0.416	0.370	0.3795	0.3835	0.3835	0.3659	0.3659
Radial Gap (in.)	0.0055	0.006	0.0045	0.0035	0.0043	0.0028	0.0028	0.0038	0.0038
H_2O/UO_2 Vol Ratio	1.5476	1.5874	1.691	1.650	1.630	1.4675	1.533	1.717	1.684
Poison Material	Gd_2O_3	Gd_2O_3	Gd_2O_3						

a. k_{eff} = 0.889 for an infinite system of PWR fuel bundles having the physical dimensions indicated in Column 4 of Table A.10-1, an initial enrichment of 2.6% U-235 (fuel rod lattice k_∞ = approximately 1.33) and in a close-packed square array of full-length, 12 in., Schedule 5, stainless steel pipe canisters at 20 °C water temperature.

b. k_{eff} = 0.774 for an infinite system of BWR fuel assemblies having physical dimensions indicated in Column 2 of Table A.10-1, an initial enrichment of 2.6% U-235 (fuel rod lattice k_∞ = approximately 1.34) and in a close-packed square array of full-length, 8 in., Schedule 5, stainless steel pipe canisters at 20 °C water temperature.

c. For an infinite system of PWR fuel assemblies in four-element fuel baskets, consisting of four 12 in., Schedule 5 pipes in close-packed square array, located on 26.25 in. centers, the effect of fuel assembly location within the pipe canister did not have a significant effect on the system reactivity (ΔK <0.3 of the standard deviation of the calculational method for array reactivity).

d. For infinite systems of PWR fuel baskets as defined in c above, the following relationships among enrichment, fuel lattice k_∞ and system k_{eff} were calculated:

Enrichment (% U-235)	Lattice k_∞	System keff
1.625	1.2003	0.788 ± 0.006
1.920	1.2504	0.824 ± 0.006
2.295	1.2993	0.864 ± 0.005
2.825	1.3500	0.912 ± 0.006

e. For an infinite system of BWR fuel assemblies in nine-element fuel baskets, consisting of nine 8 in., Schedule 10 pipes in close-packed square array, located on 26.25 in. centers, it was calculated that locating the eight peripheral bundles as close to the central bundle as possible resulted in a maximum increase in k_{eff} of 4.5% over that for fuel at the centerlines for fuel with a lattice k_∞ in the range 1.20 to 1.40.

f. For an infinite system of BWR baskets as defined in e. above, the following relationships between enrichment, fuel lattice k_∞ and system k_{eff} were calculated:

Enrichment (% U-235)	Lattice k_∞	System k_{eff}
1.570	1.2001	0.652 ± 0.005
1.850	1.2500	0.688 ± 0.006
2.210	1.2994	0.732 ± 0.006
3.420	1.3996	0.792 ± 0.007

g. Effects of burnup (fissile material depletion and long-lived fission product buildup) were calculated for BWR and PWR fuels using the LEOPARD code.

A detailed nuclear safety evaluation was made which includes:

- Validation of the correlation between initial U-235 enrichment and rod lattice k_∞ which has been made using the EGGNIT code.
- Correlation of rod lattice k_∞ with bundle array k_∞ (at the reactor bundle lattice pitch). For PWR fuel arrays this difference is very small as the additional water layer at the fuel bundle boundary is approximately 1/16 in. For BWR fuel arrays the effects are somewhat greater as the water layer at the fuel bundle boundary is approximately 0.75 in. Since the safety margins for BWR fuel storage arrays are substantial, the effect does not significantly change the safety of the system.
- Extension of the array calculations to k_∞ of 1.40 for BWR fuel and 1.35 for PWR fuel.
- Evaluation of the effect of elevated fuel and water temperatures.

Additional KENO-II calculations were made to evaluate k_{eff} for PWR arrays at k_∞ (cold) of 1.35 and for temperatures of 20 °C, 50 °C, and 115 °C. It was concluded that temperature does not significantly affect the fuel reactivity.

For BWR fuel containing burnable poison (Gd_2O_3), the value of k_∞ (cold) rises from approximately 1.15 to < 1.25 and declines to < 1.20 by the end of one cycle of irradiation. Thus the presence of poison in the BWR fuel adds to the safety margins in the event of early discharge of the fuel.

Nuclear design analysis for the square tube BWR storage basket was performed by GE[2] to demonstrate the k_{eff} is maintained less than 0.95 with the new square tube geometry. The results of these analyses show that for the worst case abnormal storage condition the maximum k_∞ of 0.836 which is considerably less than the allowed limit of 0.9 k_{eff}.

A.10.3 REFERENCES

1. BPNL, Basin Criticality Safety for MFRP Project-1 Fuel Bundle Storage Baskets, May 1975. (Appendix B.5)

2. GE, Criticality Safety Analysis for Square Tube Fuel Storage Baskets at Morris Operation, May 1987. (Appendix B.15)

A.11 FUEL TO BE STORED -- ADMINISTRATIVE AND TECHNICAL CONTROLS

A.11.1 INTRODUCTION

Administrative control of the k_∞ limits for fuel to be stored at the Morris facility depends primarily on correctly identifying the fuel bundles by number and on assuring that the pre-irradiation k_∞, cold, is less than the limits set by design criterion b[1]. The value for k_∞ is determined principally by the initial U-235 enrichment and to a much smaller degree by the pellet diameter (±0.25%) and the water/fuel volume ratio (±1.3%).

Figures A.11-1 and A.11-2 are used to evaluate the k_∞ value. They were prepared from data provided by Battelle Pacific Northwest Laboratories (BPNL). The form of these charts was designed to avoid the necessity for interpolation and to minimize potential for error in use of the data. When using these charts, the correction factors for variation in water-to-fuel ratio are slightly more conservative (approximately 0.12%) at the higher water-to-fuel ratio than the average value that would be obtained from calculations.

In addition to fuel evaluated as described above, other LWR fuel may be accepted for storage after specific analysis of nuclear characteristics and regulatory approval. For example, fuel from the LaCrosse BWR has been approved for storage after evaluation for storage in the fuel storage system (Figure A.11-1), and for rod lattice k_∞ (Figure A.11-2). Special storage authorizations are included in Chapter 10.

A.11.2 GENERAL PRACTICES

Prior to any transfer of fuel from a reactor site to Morris Operation (GE-MO), a utility transmits sufficient data on the fuel to be stored to calculate the rod lattice k_∞. The validity of this transmitted data is certified by two qualified individuals from that utility, one being from that organization's quality assurance component. General Electric Company determines the acceptability of that fuel in accordance with Materials License No. SNM-2500 as amended.

A separate confirmation of the fuel identity and initial enrichment is provided by documents required by government regulations. Current NRC policy requires that all transfers of nuclear material be documented on a NRC-741 form, which is initiated by the shipper and completed by the receiver. Copies of the completed NRC-741 form are transmitted to the shipper and appropriate NRC branch within 10 days of receipt, thus verifying the transfer of the material. In order to provide a separate verification of the initial enrichment of each fuel bundle, copies of NRC-741 forms covering shipment of the fuel from the fabricator to the utility will be provided to GE by the utility concurrently with transmittal of the Data for Storage Compliance (Fig. A.11-3).

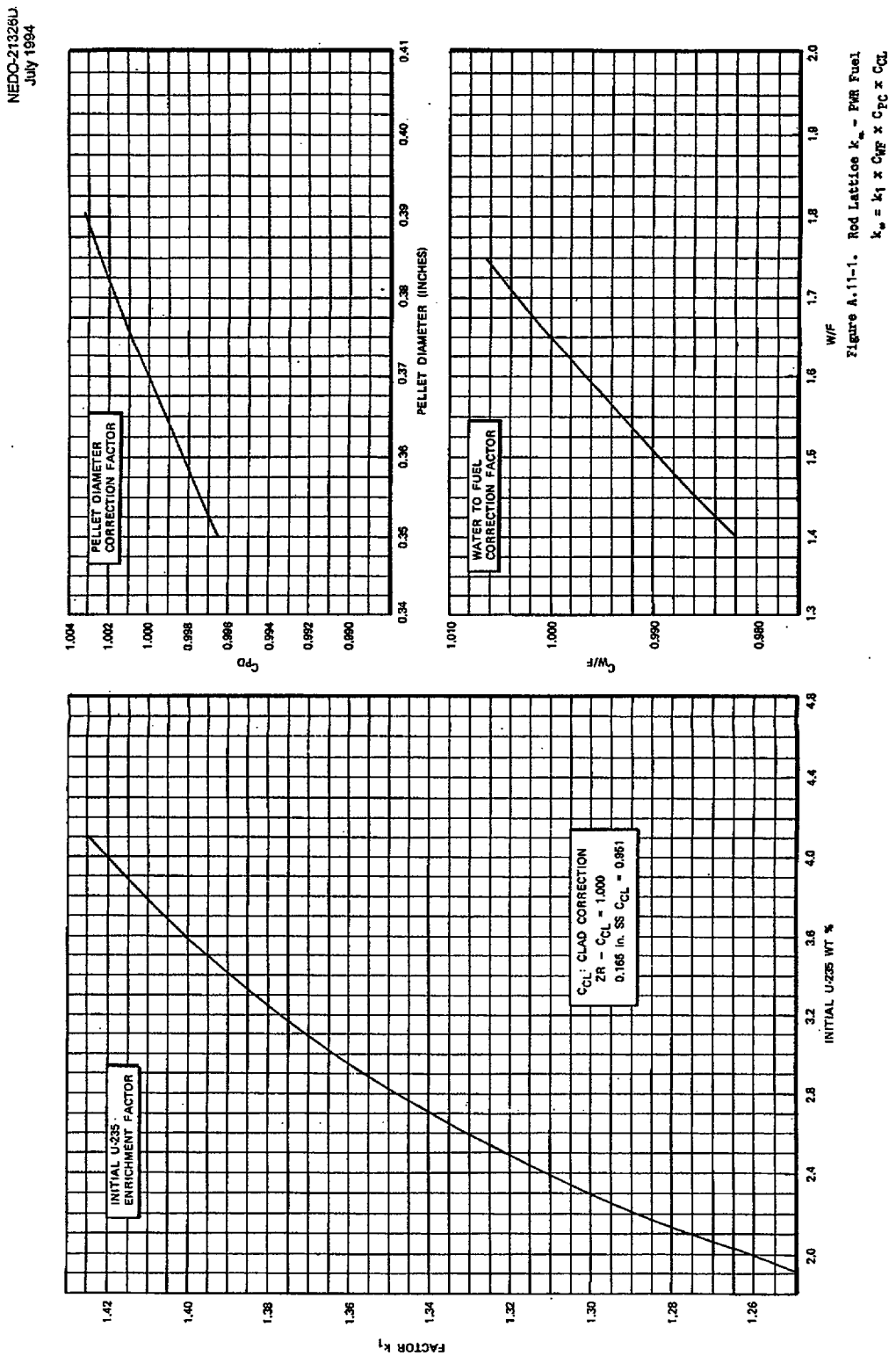


Figure A.11-1. Rod Lattice k_∞ - PWR Fuel
$k_\infty = k_1 \times C_{WF} \times C_{PC} \times C_{CL}$

Figure A.11-2

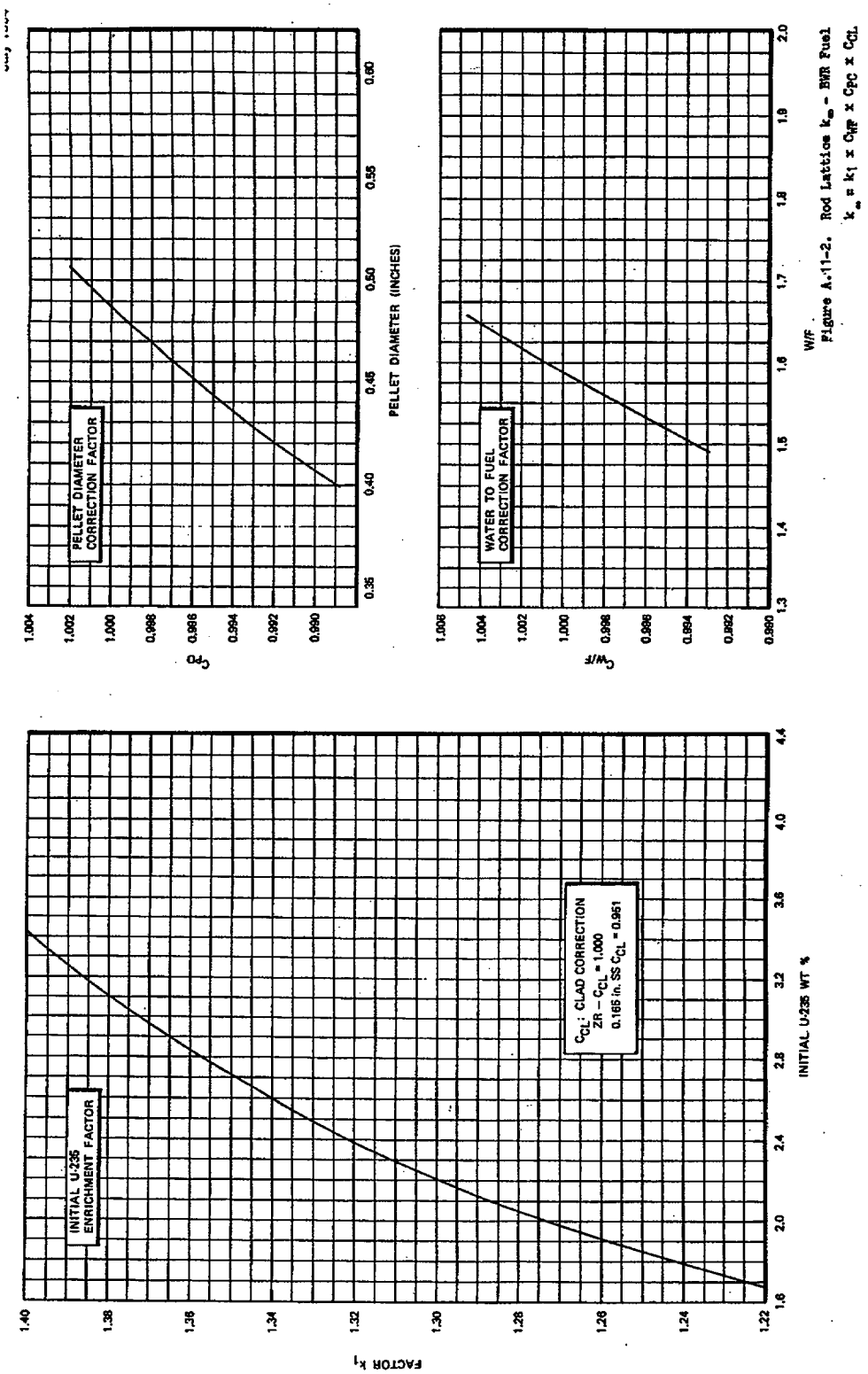


Figure A.11-2. Rod Lattice k_∞ - BWR Fuel
$k_\infty = k_1 \times C_{WF} \times C_{PD} \times C_{CL}$

Figure A.11-3

FIGURE A.11-3
DATA FOR FUEL STORAGE COMPLIANCE
MORRIS OPERATION

==========

Reactor: _____ Bundle Rod Array: _____ x _____
Batch Discharge #: _____ Bundle Dimension: _____ (in.)
Date of Discharge: _____ Cladding Material: _____
Nom. Cladding Thickness: _____ (in.)

==========

Bundle I.D. No.	Total U (kg)	Initial %U-235	Pellet O.D. (in.)	Rod O.D. (in.)	Rod Pitch (in.)	Average Burnup (GWD/MTU)	Final %U-235	Initial Cold, Clean k_{eff} (by Mfg.)	Final k_{eff} Cold, no Xe (if avail)

==========

Data Compiled by:
Signature: _____
Title: _____
Company: _____
Date: _____

Data Verified by QA:
Signature: _____
Title: _____
Company: _____
Date: _____

Prior to shipment, the Regulatory Compliance Manager (RCM), or designee, will determine the acceptability of each fuel bundle for storage. This determination will include, but not be limited to, the evaluation of k_∞, using Figures A.11-1 and A.11-2. The rod lattice k_∞ value determined from this evaluation is compared with the bundle k_∞ value received from the contracting utility.

The initial cold, clean k_∞ values determined from the evaluation must be less than or equal to the limit set by design criterion b[1] and in agreement with the k_∞ value from the shipper to within 2%. For BWR bundles, the calculated rod lattice k_∞ is compared to the bundle k_∞ from the shipper plus 0.052. For PWR bundles, the calculated rod lattice k_∞ is compared to the bundle k_∞ from the shipper since the rod lattice and bundle k_∞s are essentially the same.

Should General Electric's evaluation determine that the k_∞ of any fuel bundle differs from that value stated by the contracting utility by more than 2%, shipment of that fuel bundle shall be deferred until such time as the difference is resolved and its acceptability established in a manner equivalent to that outlined above. Upon determination that the fuel is acceptable, the General Electric Company will notify the contracting utility that the fuel bundle is acceptable and that it can be shipped.

At the time a fuel bundle is to be shipped to GE-MO, its identity is checked and verified against the approved list by two individuals of the contracting utility, and documented on the shipping release forms. A copy of this list is maintained in the permanent records at GE-MO.

Upon receipt at GE-MO, the Operations Engineer (OE), or designee, verifies that the bundle listed on the Shipping Report form is one of the approved bundles for receipt. This verification is documented and maintained in permanent files at GE-MO. The cask is then released to the cask receiving area.

During cask unloading operations, the identity of the fuel is determined and verified by the OE or designee. The fuel bundles are then transferred to their assigned locations in the fuel storage basin. The identity and locations of the bundles in the basin are documented in a computer data base.

The procedures described above provide sufficient control to ensure fulfillment of the double contingency policy. Each action or transaction is verified by two competent representatives of the organization primarily responsible for that act. The independent review and analysis by General Electric personnel provides further checks on the validity of the data transmitted by the contracting utility and the ultimate acceptability of each fuel bundle. The bundle identity is verified by a minimum of four individuals and documented on at least three forms. As the General Electric Company's evaluation of rod lattice k_∞ is most sensitive to initial enrichment of the fuel bundle, copies of the NRC-741 forms, initiated by the fabricator, will be provided by the contracting utility to assure that the initial enrichment value used as a base for k_∞ is correct.

A.11.3 BWR AND PWR FUEL QUANTITIES

To permit some flexibility in the relative amounts of BWR and PWR fuel to be stored at the Morris facility, the fuel baskets are designed to have a common base and hold-down mechanism. The fuel basket designs accommodate either nine BWR bundles in 8 in. stainless steel pipe or four PWR bundles in 12 in. stainless steel pipe.

Preliminary calculations by BPNL showed that 15 x 15 PWR fuel having k_∞ of 1.35 would give an array k_{eff} of approximately 0.90. A k_∞ limit of 1.35 was used as the basis for the basket detailed design to allow some margin for dimensional tolerances and for any uncertainty in the final design calculations. The completed analysis showed that for k_∞ of 1.35, k_{eff} at the 95% confidence level would be 0.917. At k_∞ of 1.4008, k_{eff} would be 0.952 at the 95% confidence level. Thus the entire basin could therefore be used to store 15 x 15 PWR fuel limited to a k_∞ of 1.37 in an "unrestricted manner."

The k_∞ limits set by design criterion b[1] provide reasonable assurance of meeting near-term utility needs without restrictions other than reactivity. Should a need arise for storage of a limited amount of slightly more reactive fuel, it could be accommodated safely by requiring the fuel have undergone sufficient burnup to assure that k_∞ is below the limit set by design criterion b[1].

A.11.4 CRITICALITY PREVENTION

Protection against accidental criticality in the fuel storage system is provided by:

a. Administrative controls limiting the enrichment and reactivity of the fuel as fabricated.

b. comparison of fuel identity upon receipt to shipping data to ensure that it meets specified limits on enrichment and reactivity.

c. fuel basket design which assures safe spacing between fuel bundles and between fuel baskets even in the unlikely event that fuel basket should be dropped; and

d. moving fuel between the fuel unloading basin and the storage basins only in fuel storage baskets and by handling individual fuel bundles one at a time.

Before a fuel shipment is scheduled for shipment to the GE-MO facility, the serial number and initial or maximum reactivity (cold k_∞) for each fuel bundle will be stated and certified by the utility. These values will be reviewed and compared to correlations provided by BPNL. (See Section 5.3.5.6.)

PWR fuel having a cold, clean k_∞ in excess of the limits established by design criterion b[1] is classified as non-specification fuel in the standard fuel storage contract, which is the basis for establishing the conditions for fuel storage at the Morris Operation. Presently, there is no PWR fuel contemplated for storage which would have a k_∞ in excess of the specified limits. For such non-specification fuel to be included under the contractual arrangement for storage, it will be necessary to establish that the post-irradiation value for k_∞ is confirmed to be less than the limiting value set by design criterion b. The evaluation of pre-irradiation k_∞ will be made based on the BPNL correlation of enrichment versus k_∞ adjusted as appropriate for pellet diameter and water-to-fuel ratio. The amount of irradiation required to assure that the post-irradiation k_∞ is

less than the limit set by design criterion b[1] will be ascertained using the pre-irradiation k_∞ and BPNL correlations.

A.11.5 REFERENCES

1. Refer to A.10.1, a through e.

A.12 FUEL BASKET SYSTEM DESIGN ANALYSES

The fuel unloading and storage basins at Morris Operation (GE-MO) are designed in accordance with earthquake and tornado criteria as described in Chapter 4. General criteria also apply to the design of the fuel basket system:

a. No deformation or damage shall occur to the concrete or to the liner that would result in significant leakage.

b. There shall be no piping or penetration failure that would lower water level significantly.

c. Cranes may be derailed, but must not fall into the basins.

d. The enclosure framework above the basin must remain essentially intact.

e. Handling and storage areas shall withstand contact or impact with stored materials.

The fuel basket system design is consistent with the response spectra specified in Regulatory Guide 1.60 and dampening values specified in regulatory Guide 1.61.

Because the supporting grid system transmits earthquake forces to the basin walls, the basin structure has been analyzed to ensure that these forces are adequately carried.

Design analyses have been performed under General Electric direction as follows:

a[1]. Manual static analyses of
  (1) Grid support structure
  (2) Latch Mechanism
  (3) Fuel basket

b[1]. Computer analysis of the grid:
  (1) Natural frequency analysis
  (2) Dynamic model analysis
  (3) Static load analysis

c[1]. Thermal analysis of the grid
d[1]. Analysis of friction loading on the latch mechanism
e[1]. Static load test of the latch mechanism
f[2]. Dynamic load test of the latch mechanism
g[3]. Effects from pilot model spacing and section changes
h[3]. Load effects on basin walls and liner from pilot model changes
i[3]. Unloading pit basket retainer frame
j[3]. Basket lifting tools (yokes)

A.12.1 REFERENCES

1. Programmed & Remote Systems Corporation (PaR), Fuel Storage System Design Report - GE Morris Operation, April 1975. (See Appendix B.16)

2. Construction Engineering Research Lab, Seismic Shock Environment Test of Simulated Nuclear Fuel Storage Basket, Department of Army, August 1975 (Technical report M-150). See Appendix B.

3. Supplement 1, Fuel Storage System Design Report - GE Morris Operation, General Electric, May 1975 (no publication number). See Appendix B.

A.13 CASK DROP ANALYSES

A.13.1 INTRODUCTION

Two analyses were made to assess the effects of a cask drop: a cask drop on the cask unloading pit shelf and a cask drop to the floor of the unloading pit.

In considering the integrity of basin structures, it should be noted that the cask unloading pit area of the main process building rests directly on an underlying shale bed. Tests of this rock structure indicate ultimate compressive strengths of 6,000 to 11,000 lb./sq. in. Therefore, the limiting material in regard to ability to absorb cask drop forces is the 3 ft. 10 in. thick foundation which is constructed of 3,000 psi design concrete having 28-day break test values in excess of 4,500 psi. The floor of the unloading pit is lined with 1/4 in. thick stainless steel sheet supported on a steel plate 1 3/4 in. thick to resist puncture and to distribute cask forces over the concrete surface. The unloading pit shelf (refer to Section 5.3.4) is lined with 1/4 in. thick stainless steel sheet, directly on the concrete, over which is located a 2 in. thick load distribution plate and an energy-absorbing pad.

A.13.2 CASK DROP ON THE SHELF

Analyses of the potential dropping of a shipping container onto either the floor of the unloading pit or the floor of the unloading pit shelf considered the effect of such an accident on both the container and the basin structure. For the purpose of the analysis it was assumed that the accident would involve the IF-300 shipping cask, which was the largest shipping container for irradiated LWR fuel then in use. Further, it was assumed that the cask would strike in such a manner as to allow minimum energy absorption by the shipping container fins and therefore the highest loading on the floor.

NOTE: It should be noted that when the use of casks to ship fuel is again considered, these analyses for cask drop should be reviewed, based on the cask proposed for shipment.

A.13.2.1 Impact Pad

The floor of the shelf in the cask unloading pit is protected by a pad that consists of a 1 in. thick stainless steel plate welded to 4 in. high x 1/2 in. thick stainless steel fins designed to crush at a predicted force, thereby limiting the force imposed on the floor to acceptable values. The pad is designed to crush at a force of 1.2×10^7 lb., where a force of 1.8×10^7 lb. is required to deform the fins on the IF-300 cask. Thus, the total energy of the drop must be absorbed by the pad. The pad is placed on a 2 in. thick floor plate consisting of two 1 in. thick stainless steel plates.

A.13.2.2 Drop Height and Energy

The maximum lift height and therefore drop height assumed is 1 ft. above the wall between the decontamination area and the unloading pit. The impact height (h_w) will be equivalent to 21.5 ft.

of water (2 ft. in air, equivalent to 3 ft. in water, plus 18.5 ft. in water). The final velocity of impact (v_2) is found by conservation of energy:

$$(F_n)(h_w) = \left(\frac{1}{2}\right)(m)(v_2)^2 \qquad \text{(A.13-1)}$$

where:

F_n = net force, and

m = mass.

The net force, F_n, can be calculated by summing the forces of gravity, buoyancy and drag in the vertical direction. The buoyant force, F_B, is calculated from the equation.

$F_B = \rho V$

where

ρ = density of water

V = volume of cask.

The drag force (F_D) is calculated from an equation given in Mark's **Handbook of Mechanical Engineering**, Section 11, page 72.

$$F_D = (C_D)\left(\frac{1}{2}\right)(\rho)(\bar{v})^2(A)$$

where

C_D = drag coefficient;

ρ = density of water;

$\bar{v}$ = average velocity and

A = cross-sectional area of cask.

The value of C_D is 1.1, which is found in Vennard's **Fluid Mechanics,** pages 516-517. The average velocity ($\bar{v}$) is calculated as follows:

$\bar{v}$ = 1/2 ($v_0 + v_2$). Since $v_0 = 0 = \bar{v} = 1/2\ (v_2)$.

v_2 = impact velocity

Substituting the dimensions and weight of the IF-300 container into Equation A.13-1 gives v_2 = 33.6 ft./sec. The equivalent height in air, h_e, is:

$h_e = (v_2)^2/2g = 17.5$ ft.

The total impact energy (E) is described by the equation:

$E = Wh_e$

where:

W = weight of the cask = 146,000 lb. (includes the yoke).

The total impact energy of the cask is found to be 2,555,000 ft-lb. (3.07×10^7 in.-lb.).

A.13.2.3 Fin Bending Data Analysis

In 1970-71, ORNL conducted a series of tests to determine the energy absorbing capability of steel fins under impact, large deformation conditions. The results of his work are reported in ORNL TM-1312 Vol. 9. This work is the source of fin deflection and impact force calculations used in the General Electric analysis.

General Electric applied details of the 0° tests for use in designing the energy absorbing fins for the IF-300 cask. A correlation was developed from the tests which permitted GE to estimate cask stopping distance (hence deceleration) given cask kinetic energy, fin material and fin geometry. This same correlation was also used to estimate the deflection of the impact pad fins used to protect the shelf in the GE-MO unloading basin. A summary of this correlation and the method used for the analysis follows:

In tests, specimens were mounted on an instrumented load cell and impacted by guided falling weights dropped from various heights. Test data was recorded on an oscilloscope and photographed, from which force-time relationship graphs were plotted.

Test specimens mounted vertically always formed two hinges. (See Figure A.13-1.) Specimens inclined 10° with the vertical formed two hinges with about 85% frequency; the remainder only one hinge. At angles somewhat greater than 10°, one hinge was always the case. Test specimens tabulated in Table A.13-1 were all mounted vertically and formed two hinges.

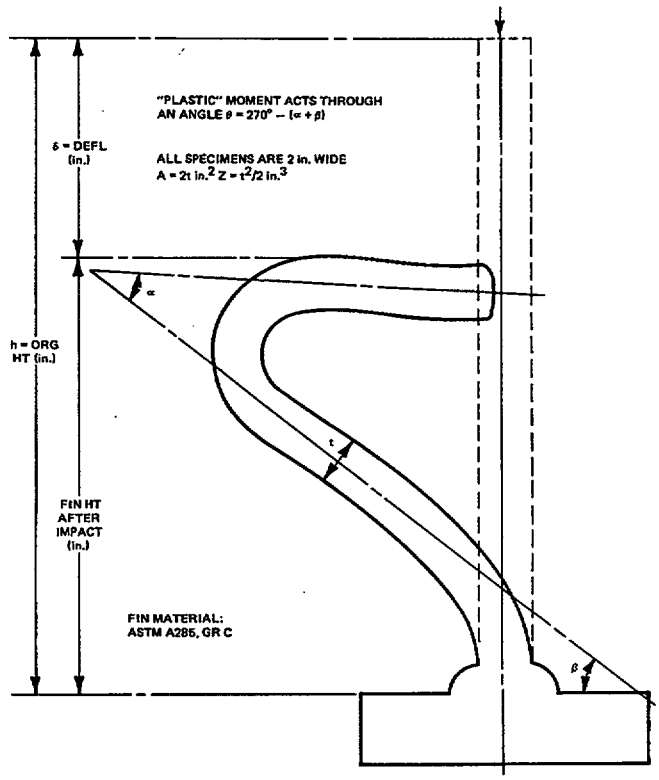


Figure A.13-1. Traced Profile of Specimen No. 5 After Impact (Typical)

In evaluating the test results, reference was made to **NACA Technical Note No. 868,** Figures 25 and 35 (copies of which are included as Figures A.13-2 and A.13-3, respectively) to determine the "hinge" stress level.

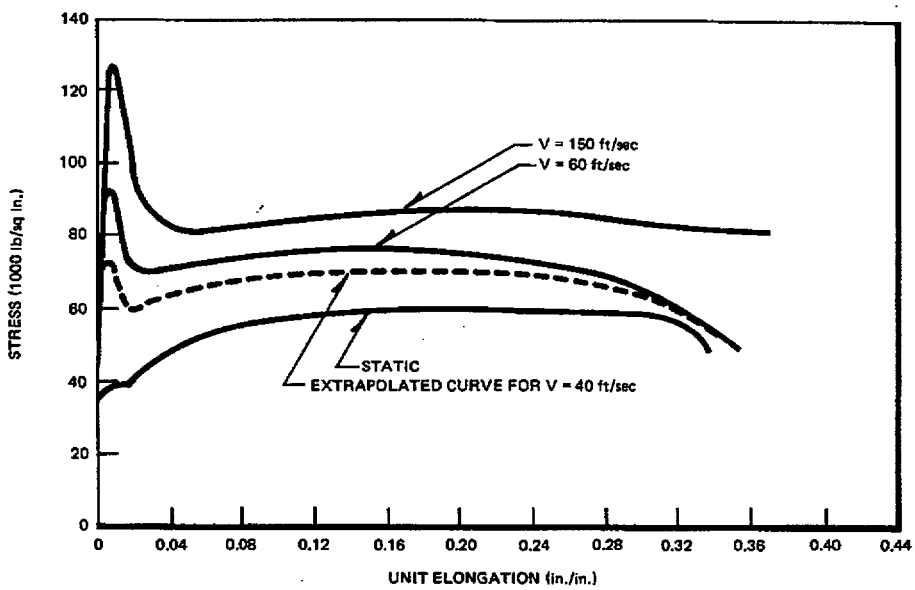


Figure A.13-2. Stress Strain Curves, Hot Rolled Steel

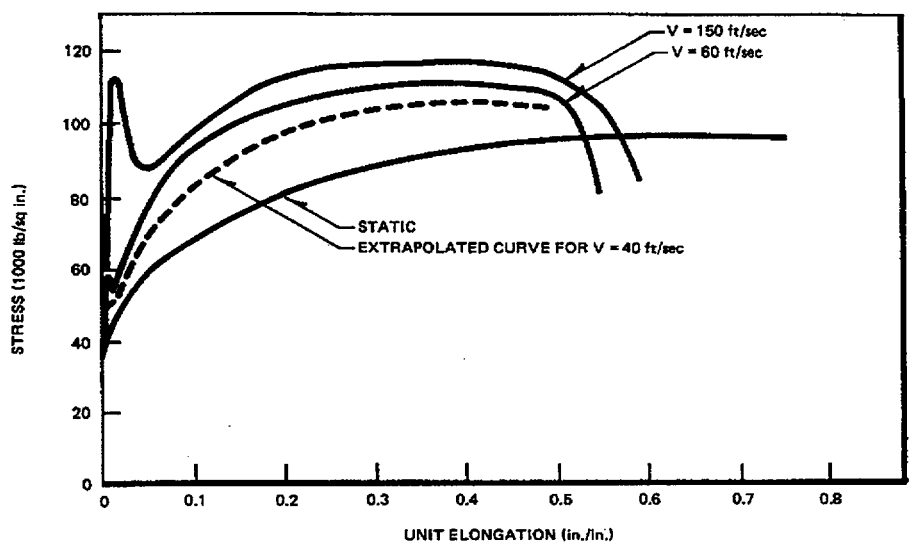


Figure A.13-3. Stress Strain Curves, 18-8 Stainless Steel

Referring to Figure A.13-2 for hot-rolled steel, the properties of which closely resemble those of ASTM A285, Gr C, of which the test fins were made, a hinge stress of σ_H = 65.0 ksi[a] was chosen as representing a reasonable value for the velocities involved, Likewise, for ASTM A240, Type 304L (18-8 stainless steel), σ_H = 90.0 ksi (Figure A.13-3).

a Thousand pounds = kip, Thousand pounds per square inch = ksi

Energy of Bending:

$$E_m = M\theta \qquad M = \sigma_h z \qquad Z = \frac{bt^2}{4}$$

where:

M is the plastic moment

θ is defined in Figure A.13-1

b is fin width (inches)

t is fin thickness (inches)

and $$E_m = \frac{\sigma_H b t^2 \theta}{4} inch - kip$$

For A285, Gr C: $$\theta = \frac{E_m}{16.25bt^2}$$ (test fins)

For A240, type 304L: $$\theta = \frac{E_m}{22.5bt^2}$$ (cask fins and pad fins)

Referring to Table A.13-1 and the columns headed E, E_m and E_p ($E_p = E - E_m$), it is noted that E_p (absorbed energy not accounted for by calculated bending) represents, with only one exception, more than 50% of the total external drop energy, "E". In evaluating the fins, it was conservatively assumed that "E_p" accounts for only one-half of the total energy.

In order to determine the fin height after impact, it was necessary to establish the empirical relationship between θ, δ, and h. (See Figure A.13-1.) This was done by calculating the percentage of δ to h and plotting against θ. As noted in Figure A.13-4, reasonable correlation was developed.

Table A.13-1 TEST SPECIMENS DATA

No.	Fin Size[a] h x t(in.)	δ in.	θ Deg	Rad	Drop Wt (kip)	Drop Ht (in.)	Energy[b] E(in.-k)	Energy of Bndg[b] E_m(in.-k)	$(E-E_m)$[b] E_p(in.-k)	$\frac{100E_p}{E}$	$\frac{100\delta}{h}$	$\frac{l}{r}$	Impact Vel FPS
1	6 x 0.75	2.63			0.472	354.3							
2	6 x 0.75	2.00	177	3.09	0.472	345.3	167.3	56.5	110.8	66.2	33.3	27.7	43.6
3	6 x 0.75	2.56	184	3.22	0.472	354.3	167.3	58.9	108.4	64.8	42.7	27.7	43.6
4	6 x 0.75	1.75			0.472	354.3							
5	9 x 0.75	6.00	268	4.68	0.472	351.3	165.8	85.5	80.3	48.4	66.7	41.5	43.4
6	9 x 0.75	1.75	126	2.20	0.304	351.3	106.7	40.2	66.5	61.4	109.5	41.5	43.4
7	8 x 0.50	3.44	197	3.44	0.178	352.0	62.7	28.0	34.7	55.3	43.0	55.4	43.5
8	8 x 0.50	2.25	151	2.64	0.157	352.0	55.3	21.4	33.9	61.3	28.1	55.4	43.5
9	6 x 0.50	1.25	134	2.34	0.157	345.0	55.6	19.0	36.6	65.8	20.8	41.5	43.6
10	6 x 0.50	1.00	124	2.16	0.157	354.0	55.6	17.5	38.1	68.5	16.7	41.5	43.6
11	6 x 0.50	0.81	101	1.76	0.157	354.0	55.6	14.3	41.3	74.3	13.5	41.5	43.6
12	6 x 0.50	0.69	88	1.54	0.157	354.0	55.6	12.5	43.1	77.5	11.5	41.5	43.6
13	6 x 0.50	0.94	117	2.04	0.157	354.0	55.6	16.6	39.0	70.1	15.7	41.5	43.6
14	3.5 x 0.50	0.13	64	1.12	0.157	356.0	55.9	9.1	46.8	83.7	3.7	24.2	43.7
15	3.5 x 0.50	0.25	107	1.87	0.199	356.0	70.8	15.2	55.6	78.5	7.1	24.2	43.7
16	3.5 x 0.50	0.69	156	2.72	0.241	356.0	85.8	22.1	63.7	74.3	19.7	24.2	43.7
17	3.5 x 0.50	0.75	162	2.84	0.241	356.0	85.8	23.1	62.7	73.2	21.4	24.2	43.7
18	3.5 x 0.50	0.63	147	2.57	0.241	356.0	85.8	20.9	64.9	75.6	18.0	24.4	43.7
20	6 x 0.25	3.12	230	4.02	0.094	180.0	16.92	8.16	8.76	51.8	52.0	83.0	31.1
22	6 x 0.25	3.00	148	2.59	0.094	144.0	13.53	5.26	8.27	61.1	50.0	83.0	27.8
23	4 x 0.25	0.81	131	2.29	0.094	120.0	11.28	4.65	6.63	58.8	20.3	55.4	25.4
24	4 x 0.25	0.69	123	2.15	0.094	144.0	13.53	4.37	9.16	67.7	17.3	55.4	27.8
25	4 x 0.25	1.94	174	3.04	0.094	180.0	16.92	6.17	10.75	63.5	48.5	55.4	31.1
26	4 x 0.25	1.56	187	3.27	0.094	168.0	15.80	6.64	9.16	58.0	39.0	55.4	30.0
27	4 x 0.25	1.44	170	2.97	0.094	168.0	15.80	6.03	9.77	61.9	36.0	55.4	30.0
28	2.5 x 0.25	0.75	170	2.97	0.094	180.0	16.92	6.03	10.89	64.3	30.0	34.6	31.1
30	2.5 x 0.25	1.12	203	3.54	0.094	216.0	20.30	7.19	13.11	64.5	44.8	34.6	34.1
31	2.5 x 0.25	0.87	190	3.32	0.094	216.0	20.30	6.74	13.56	65.8	34.8	34.6	34.1

[a] All test fins are 2 in. wide.
[b] in.-k = thousand inch-pounds

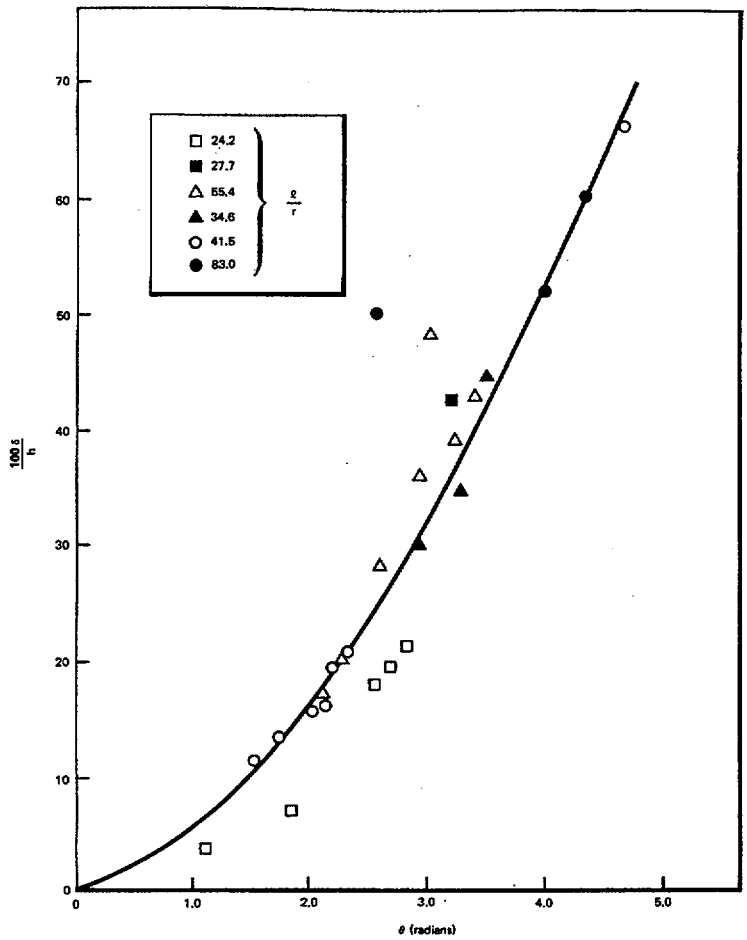


Figure A.13-4. Empirical Relationship Between θ, δ and h

Use of the Correlation

Using Figure A.13-1 as an example:

$$\theta = \frac{E_m}{16.25bt^2}$$

E = 165.8 inch-kip

E_m = 1/2 (165.8) = 89.2 inch-kip

b = 2 in. fin width

t = 0.75 in. fin thickness

$$\theta = \frac{(82.9)\ inch-kip}{(16.25)(2)(0.75)(2)} = \quad = 4.53 \text{ radians}$$

From Figure A.13-4 at θ = 4.53

$$\frac{100\delta}{h} = 62.5$$

Since h = 9 in. $\delta = \frac{62.5}{100}(9) = 5.63$ in.

This correlates very well with the measured deflection of 6 in. for fin No. 5.

The g loading for this fin would be defined as:

$$g = \frac{\text{Drop Height}}{\text{Stopping Distance}}$$

$$= \frac{351.3 \text{ in.}}{5.63 \text{ in.}}$$

$$= 62.4$$

This is compared to 59 g based on actual deflection and therefore the correlation is somewhat conservative. It is very conservative based on measured average forces, Figure A.13-5.

The method described above was applied to the design of the GE-MO unloading pit shelf impact absorbing pad.

A.13.2.4 Impact on Step Corner

The impact absorbing pad on the floor of the step of the unloading pool has been designed to limit the forces of a falling cask and distribute these forces over a large area. The pad on the step extends to the front edge and to a point 6 in. from each wall. The space between the pad and wall is not large enough to allow the cask to hit an unprotected part of the floor.

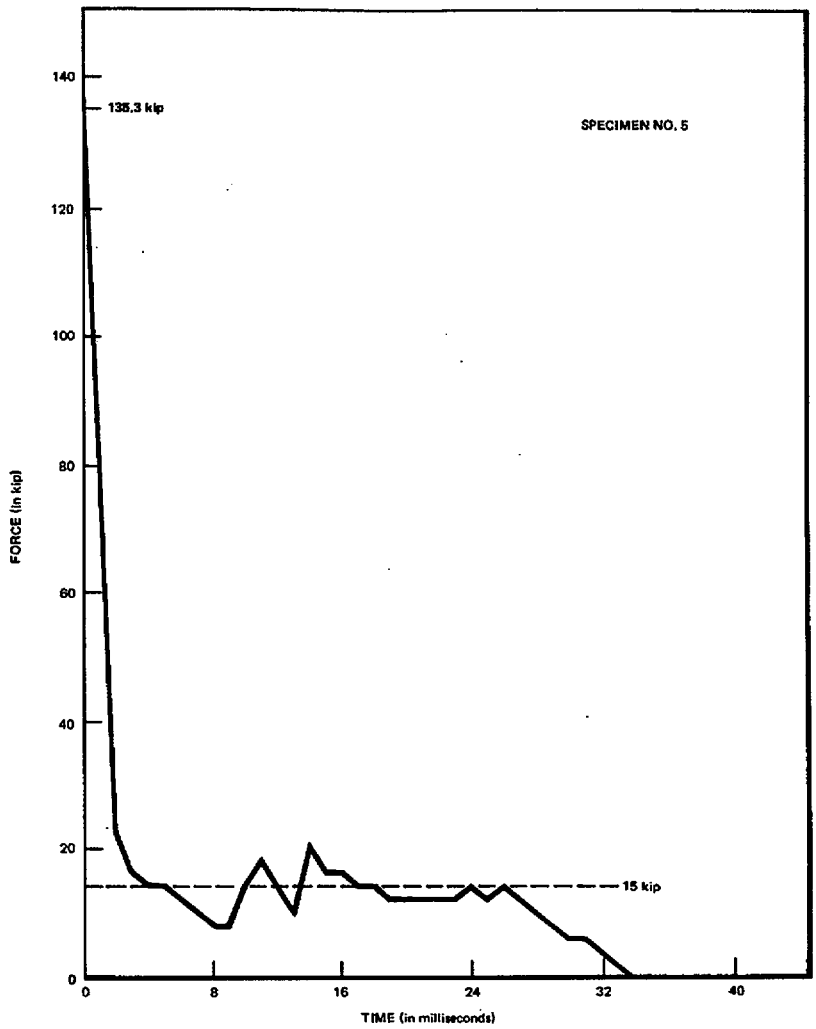


Figure A.13-5. Force-Time Curve for Specimen No. 5

The maximum load that the corner of the step could experience from a falling cask is when the cask's center of gravity is located directly above the edge at the time of impact. Stresses in the concrete foundation that result from such an accident are analyzed by calculating the forces developed as the energy is absorbed by the impact absorbing pad. As the kinetic energy is absorbed the load on the concrete from the resultant force is distributed, by the impact pad and the 2 in. floor plate, over an area that is considerably larger than half the cross-sectional area of the cask.

The impact absorbing pad is constructed of a top plate, 1 in. thick, welded to fins that are 1/2 in. thick and 4 in. long (see fin orientations in Figure A.13-6). The pad is placed on a 2 in. thick floor plate consisting of two sheets, each of which is 1 in. thick. All the construction material is 304L stainless steel.

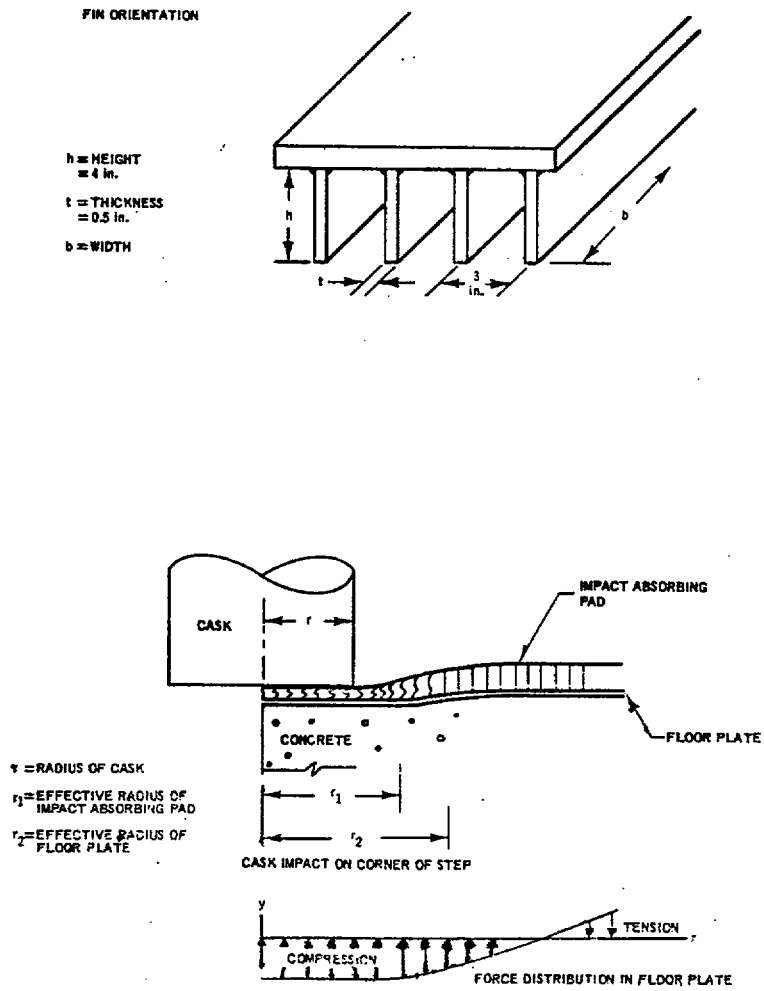


Figure A.13-6. Fin Orientation

When the cask hits the pad there is a radius on the flat plate beyond the cask where the compressive forces change to tension[1] (Figure A.13-6). At that point the force is zero. By taking a weighted average of fin deflection as a function of force, the effective radius is found to be 4.75 inches more than the cask radius or 39.86 inches. From this effective radius, the total width of affected fins is calculated to be 697.6 in.

The angle θ through which the plastic moment acts when a fin bends is given by the equations:

$$\theta = \frac{E_m}{22.5t^2} = 3.90 \text{ radians}$$

where

E_m = half the total drop energy (1.53×10^4 inch-kips)

t = fin thickness (0.5 inch)

The deflection (δ) is calculated by using the correlation given in Figure A.13-4. For θ = 3.90, 100 (δ)/h = 50.5 and h = 4 inches:

δ = 4(50.5)/100 = 2.02 in.

The g-loading then for a 17.5 ft drop is:

$$g = \frac{H_e}{\delta_i} = \frac{17.5 \times 12}{2.02} = 104g$$

This means that a force of 104 g is imparted to the IF-300 container as a result of the 17.5 ft. drop. Since a force of 272 g is required to bend the IF-300 fins in an end drop on an unyielding surface, the fins on the IF-300 will not deform as a result of the drop onto the impact pad.

Results of tests conducted by Atchely and Furr[2] indicate that the ultimate dynamic load for concrete is 1.5 times greater than the ultimate static load. The ultimate static load indicated by the 28-day test[3] is 4,634 psi. Therefore, the ultimate dynamic load is 6,951 psi. Under this load, maximum deflection of reinforced concrete is approximately 2.317 x 10^{-3} inches. From "flat-plate" theory, the maximum effective radius that results from the 2 in. floor plate is 2.96 inches more than the effective radius of top plate of the pad (39.86 inches). By taking a weighted average of the deflection as a function of force, the average effective radius is 42.795 inches. The effective area, A_e, on the concrete is:

A_e = $(\pi/2)(42.795)^2$ = 2,876.7 in.^{2}

The load experienced by the concrete that results from the impact force (F_I) is:

L = F_I/A_e

$$F_I = \frac{E}{\delta} = \frac{3.07 \times 10^7 \text{ in.-lb.}}{2.02 \text{ in.}} = 1.52 \times 10^7 \text{lb.}$$

E = Total impact energy

then:

$$L = \frac{1.52 \times 10^7 \text{ lb.}}{2876.7 \text{ in.}^2} = 5{,}283 \text{ psi}$$

Because the load on the concrete is less than its ultimate dynamic load, the integrity of the concrete is protected.

It should be noted that the probability of this postulated accident occurring is extremely low. Two failures must occur before the cask could be dropped. The hoist operator must fail to observe operating procedure and move a cask containing a design basis load over the corner of the shelf while suspended in air above the pool. Then the equipment must fail in such a way that the cask is released. The falling cask must land on the corner of the shelf with its center of gravity directly over the edge. The calculations reflect further conservatism by assuming that the concrete is not reinforced by steel rebar (it is reinforced) and the impact absorbing pad absorbs all the energy. Also, the fins of the cask will absorb some energy.

A.13.2.5 Fin Weld Analysis

The welding of the fins of the impact pad to the horizontal plates was also analyzed. The static plastic moment of the fin weld (M_p) is given by

$$M_p = \sigma_y \left(\frac{bt^2}{4} \right)$$

where

σ_y = yield stress of 304L (25,000 psi);

b = 1 in. unit length; and

t = fin thickness of 0.5 in.

Then

$$M_p = 25{,}000 \left(\frac{1 \times 0.5^2}{4} \right) = 1{,}560 \text{ psi per unit length of weld}$$

Weld stress (S) is given by:

$$S = \frac{(1.414) M_p}{(b)(L)(h+b)}$$

where

b = 0.25 in., weld size,

h = 0.5 in. fin thickness; and

L = in., weld length.

Then

$$S = \frac{(1.414) \times 1560}{(0.25)(1)(0.5 + 0.25)} = 11{,}764 \text{ psi}$$

which is less than the yield stress for 304L stainless steel.

This analysis assumes the fin is held firmly by the base plate. The fins will be attached with a fillet weld using 308L stainless steel rods. According to AWS-ASTM classification of corrosion-resisting chromium and chromium-nickel steel welding rods, the tensile strength of 308L stainless steel rod is 75,000 psi. The stress is also less than the permissible stress for welded joints as given in the **Code for Arc and Gas Welding in Building Construction** of the American Welding Society. The permissible shear stress on the section through the throat of a 308L fillet weld is 13,600 psi.

A.13.3 CASK DROP IN DEEP PIT

A.13.3.1 Drop Height and Energy

The fuel unloading pit has been analyzed for a postulated shipping cask drop accident. When a shipping cask is placed in the fuel unloading pit, first the cask is lowered to a shelf 18.5 ft. below the water level. A cask extension yoke is then employed to lower the cask to the unloading pit floor 30 ft. below the step. Assuming the cask is raised 1 ft. above the step surface and then moved horizontally over the unloading pit, the height of the postulated drop is 31 ft. The cask will be underwater during the postulated drop.

The vertical forces acting on the cask (assume downward is the positive direction) are positive gravity, negative buoyancy force and negative drag force. The equations for these forces are:

g = force of gravity = 32.2 ft/sec^2

F_B = buoyancy force = ρV

where

ρ = density of water

V = volume of cask

F_D = drag force = $0.5 C_d \rho \bar{u}^2 A$

where

C_D = drag coefficient = 1.1

ρ = density of water

$\bar{u}$ = average of velocity

A = cross-sectional area of cask

Assuming the cask is a IF-300 shipping cask, acceleration, velocity, drag force, and kinetic energy were calculated in 1 ft. increments throughout the 31 ft. height. The acceleration dropped from 32.2 to 20.5 ft./sec.^{2} due to the drag force, the impact velocity was 38.8 ft./sec. and the kinetic energy was 3,362,484 (4.035×10^7 in.-lb.). This energy is less than a postulated 30 ft. drop in air, which is 5.04×10^7 in.-lb., and therefore, the consequences will be less than those experienced in a 30 ft. drop in air as far as the shipping cask is concerned.

A.13.3.2 Floor Construction

As indicated in the FSAR (GE document No. NEDO-10178-2, July 1971), the floor of the cask unloading pit rests directly on a shale bed. The ultimate compressive strength of this bed was tested and found to be from 6,000 to 11,000 psi. The floor is made of reinforced concrete 3.83 ft thick and covered with a steel plate 2 in. thick.

A.13.3.3 Floor Loading Analyses

An accidental cask drop in the unloading pit was analyzed for a perpendicular drop and a corner drop. It was found that the corner drop (axis of the cask inclination equal to 14.23°) has the greatest potential for damage to the floor of the unloading pit.

The cask corner drop was analyzed using the modified National Defense Research Committee (NDRC) formula for missile penetration calculations[4]. The analysis was made for the IF-300 shipping cask which weighs 146,000 lb. in air and 126,000 lb. in water, with an impact velocity of 38.8 ft./sec.

A calculation using the modified NDRC formulation showed that the penetration depth is less than 16 in. The foundation mat thickness required to prevent perforation was calculated as 42.3 in. using the NDRC formulation. The total thickness of the concrete floor in the unloading pit is a minimum of 46 in., indicating that there will be no perforation.

The calculations neglect the energy required to deform the cask fins. The total energy of the cask was accounted for in perforation of the steel plate and penetration of the concrete floor. Thus the penetration is a maximum value.

This analysis did not consider any material below the concrete mat. Since the floor of the unloading pit rests directly on a shale bed, there can be no scabbing of the lower surface of the floor. This adds additional conservatism to the calculation of mat thickness to prevent perforation and thus to the conclusion that no perforation of the concrete mat will occur.

Since perforation of the concrete floor is not expected, the only consequence of a cask drop accident would be penetration of the pit liner with release of small quantities of basin water to the region between the liner and concrete wall. Experience at GE-MO with a cask tipping incident[5] has shown that leakage due to a breach of the pit liner can be handled with no measurable release of basin water from the facility to the local perched aquifers and liner repair can be made in a short time with no serious impact on the operation of the fuel storage facility.

According to R. P. Kennedy (Reference 4) the modified NDRC formula is applicable to this case since it adequately predicts test results for large-diameter, low-velocity missiles.

Even if the results of the penetration/perforation analysis are ignored and it is very conservatively assumed that the corner cask drop results in a breach of the concrete mat such that there is leakage of pool water to the local perched aquifers; there would be no significant release of radioactivity to accessible water sources.

Analyses of the leakage paths of water from the fuel basin is contained in Dames & Moore's, "Transport Modeling for Accidentally Released Water from Spent Fuel Storage Basin at Morris, Illinois Facility of General Electric Company", October 26, 1993.

A.13.4. REFERENCES

1. D. Hartog, "Flat Plate Theory," **Advanced Strength of Materials.**

2. Atchley and Furr, "Strength and Energy Absorption Capacities of Plain Concrete Under Dynamic and Static Loading," **ACI Journal**, November 1967. **Discussions,** 65: 414-16, May 1968.

3. **Loading Pit Concrete Tests**, by H. H. Holmes Testing Lab., Inc., April 1969, under AE Contract No. 4204.

4. R. P. Kennedy, **A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,** Nuclear Engineering and Design, Volume 37, 1976 pp. 183-203.

5. B. F. Judson, Plant Manager, General Electric Co., MFRP, Morris, Illinois, letter to B. Grier, Regional Director, USAEC, Division of Compliance - Region III, 799 Roosevelt Road, Glen Ellyn, Illinois, June 27, 1972.

A.14 LIST OF ENGINEERING DRAWINGS

Figure	Title
A.14-1	Main Building Below Elevation 45' 0" and 48' 0"
A.14-2	Main Building - Sections A, B, J, K and L
A.14-3	Main Building - Sections E and F
A.14-4	Main Building - Longitudinal Section G
A.14-5	Main Building - Longitudinal Section H
A.14-6	Sand Filter Building - Floor Plan
A.14-7	Sand Filter Building - Isometric

Drawing	Title
C5483E-2351	Grid Assembly Detail 'J.'
V3874 17988-D	Basket Ass'y BWR Fuel
V3887 17987-D	Basket Ass'y PWR Fuel
C6221 17988-E	Basket Ass'y BWR Fuel - Square Tube Design

This Figure Withheld under 10 CFR 2.390

Figure A.14-1. Main Building Below El 45'0" and 48'0"

This Figure Withheld under 10 CFR 2.390

Figure A.14-2. Main Building - Sections A, B, J K and L

This Figure Withheld under 10 CFR 2.390

Figure A.14-3. Main Building - Sections E and F

This Figure Withheld under 10 CFR 2.390

Figure A.14-4. Main Building - Longitudinal Section G

This Figure Withheld under 10 CFR 2.390

Figure A.14-5. Main Building - Longitudinal Section H

This Figure Withheld under 10 CFR 2.390

Figure A.14-6. Sand Filter Building - Floor Plan

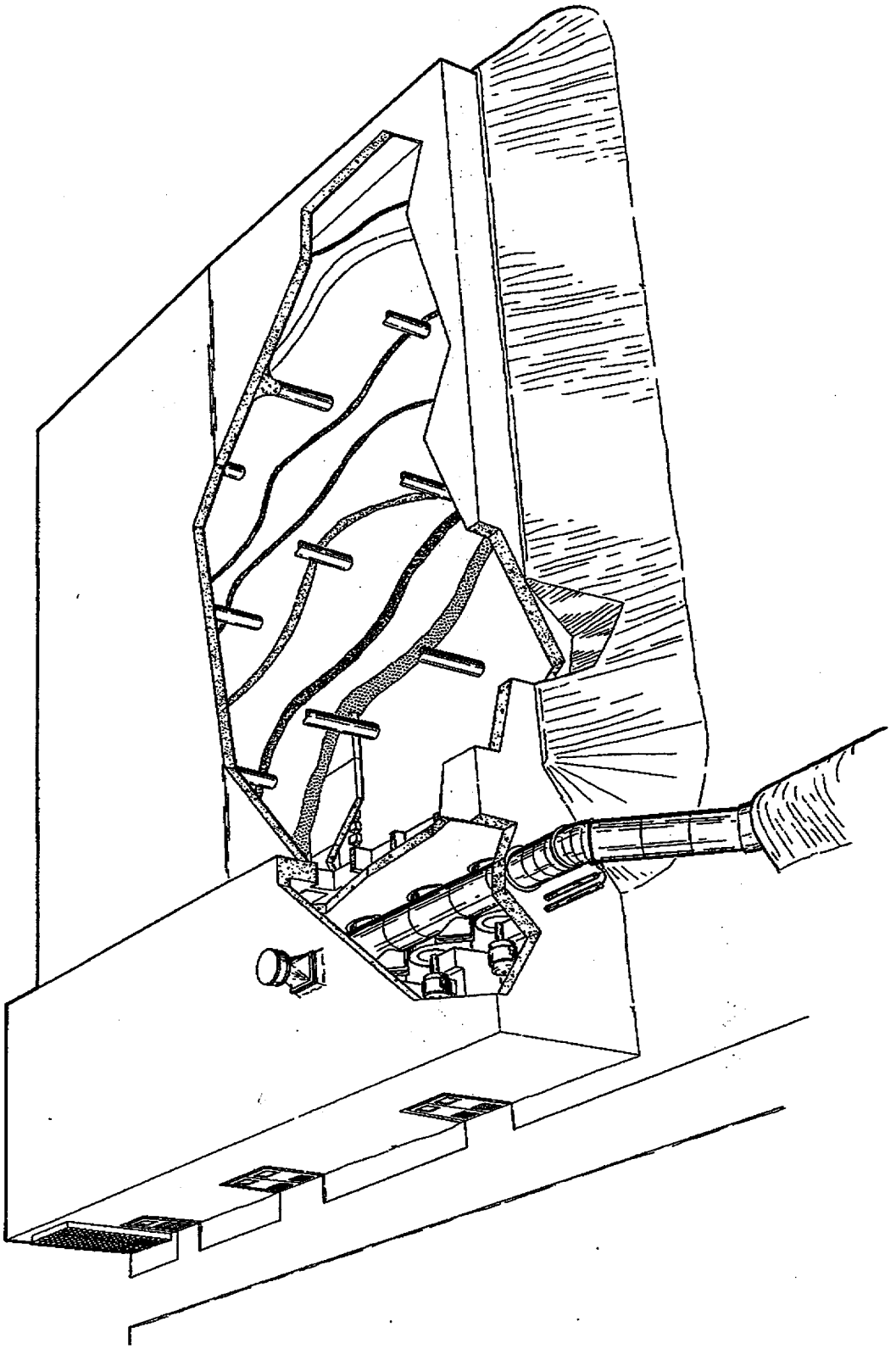


Figure A.14-7. Sand Filter Building - Isometric

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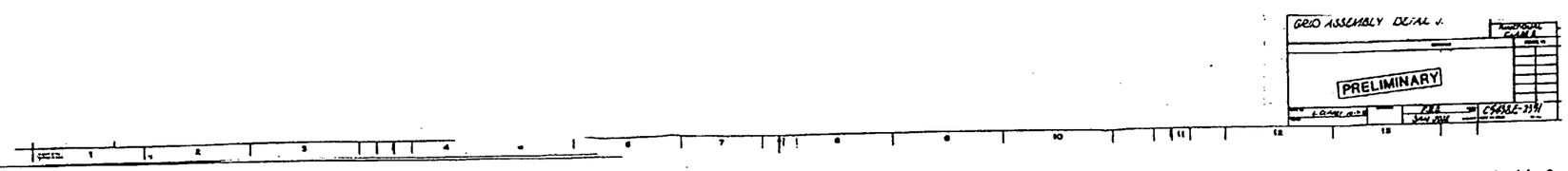

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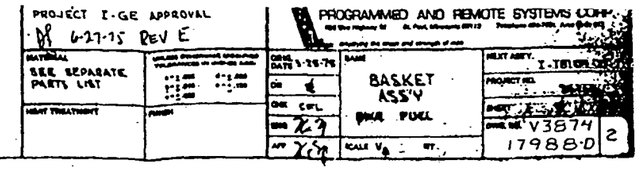

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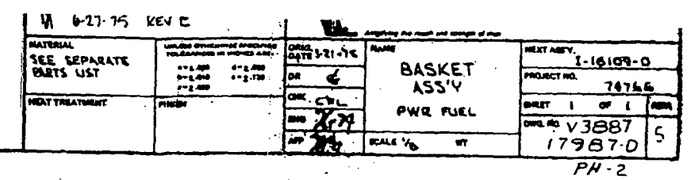

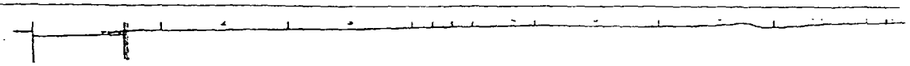

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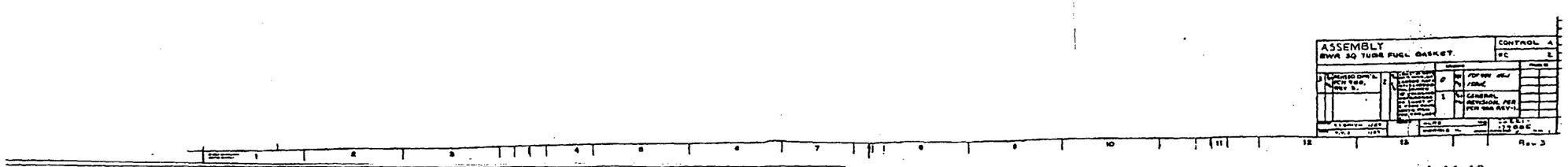

A.15 ANALYSIS OF TORNADO MISSILE GENERATION AND IMPACT ON THE MORRIS OPERATION FUEL STORAGE BASIN

A.15.1 INTRODUCTION

Only those windborne objects which could have a significant downward velocity on entry into the water-filled basin have the potential for causing damage to basin contents. Such objects must have been at a significant elevation above ground level, prior to entry, to develop the required vertical velocity component to result in damage.

A.15.2 POTENTIAL MISSILES

Potential missiles can be classified in regard to their relative elevation, as follows:

1. Objects in the immediate area which, when the tornado strikes, are at elevations above the level of the basin surface (operating equipment and auxiliaries, components of the enclosing structure, etc.).

2. Objects in the general vicinity, which are of such shape and density that they become airborne by aerodynamic lift, are carried by the tornado for a distance and then are dropped into the basin (roofs, doors, etc.).

3. Objects in the general vicinity which are too heavy to be lifted by aerodynamic forces but which conceivably could be deflected upward into a ballistic trajectory after being accelerated by the tornado winds at ground level (small automobiles, boulders, etc.)

4. Objects in the general vicinity which are too heavy to be lifted but, when the tornado strikes, are already at a location above ground level (tops of telephone poles, etc.) so that they could be carried by the tornado and dropped into the basin.

Fuel handling tools and equipment, as well as building siding and roof decking, are of low mass and could not be accelerated over the distance required to achieve the potential velocity at which damage could occur, since they are located within the immediate vicinity of the basins. Heavier items, such as fuel shipping casks, are capable of withstanding tornado winds without displacement.

To become airborne by aerodynamic lift, objects in the second category must be relatively light and of large surface area. Thus, high impact velocities would be required to cause damage but deceleration would be rapid upon entry into the water. For these reasons, damage potential from such objects is not significant.

Although the likelihood of actual occurrence is very low, objects in the third category must be considered because they are relatively dense and conceivably could arrive at the basin location with a high downward velocity.

Objects in the fourth category do not have significant damage potential because of their limited initial elevation, except as they may be deflected upward, after initial acceleration, in which case they become similar to the missiles described in the third category. In summary, only dense objects which achieve significant elevations by the mechanism described for the third category appear to have potential for inflicting damage to the basins or fuel.

In recognition of the fact that sufficient data are not available on which exact characterization can be based, four different methods of calculating potential missile velocities are considered. Three of these are derived from sources in the literature and the fourth from discussions with U. S. Weather Bureau Personnel. These methods are then applied to two simple geometrical bodies typical of potentially damaging missile objects, as described above; viz., a 12 in. diameter by 20 ft. long section of telephone pole weighing 630 lb. and a small automobile, 5 ft. by 5 ft. by 8 ft. in dimension and weighing 1,800 lb.[1]. The most conservative conditions of acceleration and ramp deflection are used in evaluating potential missile effects, although the analysis is based on assumptions regarding missile behavior which have a very low probability of actual occurrence.

A.15.3 TORNADO WIND VELOCITIES

A tornado is a violent whirlwind usually accompanied by a funnel cloud produced by low pressure inside the storm. Estimates of wind speed within the tornado funnel have been made directly from the shape of the funnel cloud, moving and still pictures of funnels and debris, and the extent of damage and patterns on the ground resulting from flying debris. Estimates of tangential wind speeds from damage can be significantly in error due to the many assumptions which must be made. Studies by Fujita, et. al. indicate that minimum wind speeds ranging from 55 to 217 mph are required to effect the typical damage wrought by Midwestern tornadoes.

Measurements of the Fargo tornadoes show a maximum tangential wind speed of about 230 mph with a translational speed of about 30 mph. The Dallas tornado measurements show a maximum tangential velocity of 170 mph and an average translational speed of 27 mph. Goldman calculated vertical velocities of 126 mph at a 750 ft. radius and about 900 ft. above the ground in studies of the Illinois tornadoes of April 1963. The tangential speed along the funnel edge of a Texas tornado of March 1956 was computed as about 230 mph at a radius of 200 ft. and 2,700 ft. above the ground. Some old estimates of 500 mph have been largely discounted over the last few years as more knowledge has been compiled on the subject, and it can be concluded that a maximum tangential speed of 300 mph is a conservative speed to be used in design of nuclear power facilities[2].

A.15.4 ANALYTICAL CRITERIA

The analyses reported herein were based on the following criteria:

Maximum Wind Velocity - 300 mph

Missile No. 1	Telephone pole, 12 in. diameter x 20 ft., weight assumed as 40 lb./ft.^{3}
Missile No. 2	Small automobile, 5 ft. x 5 ft. x 8 ft. long, weight assumed as 1,800 lb.
Drag Coefficient	1.3

A.15.5 ANALYTICAL METHODS

In several of the analysis methods used the trajectory of the missiles is considered, while in other methods the energy produced in the missile is translated to velocity or height and combinations of both.

For an object to become a missile, it is necessary for it to be aerodynamically lifted and set in motion by the winds of the tornado. The three modes of injection are:

a. Explosive injection into the suddenly imposed pressure differential of the tornado. Here there must be a sufficient volume of air below the object injected to cause the explosion (for example, roofs on poorly vented building).

b. Aerodynamic injection of an object having some configuration which produces lift in the horizontal flow.

c. Ramp injection, where the object is accelerated horizontally and deflected upward.

Aerodynamically, it is impossible for a 300 mph wind to generate missiles approaching that speed because the object has to be accelerated and is subject to the influence of its shape, weight and friction relative to the air.

Four methods are used to determine the speed of the missiles under consideration. Method 1 assumes the object is accelerated and deflected upward at an angle of 45° while constantly exposing a maximum area perpendicular to the direction of the wind. Method 2 is similar to Method 1 in that the distance through which it is acted upon and the manner of acceleration are the same, but the object is considered to tumble as it travels with the tornado winds. In Method 3, an initial elevation is assumed and the missile is acted upon by simultaneous horizontal and vertical wind forces. In Method 4, a tumbling object is acted upon by the maximum winds over an average period of time.

A.15.5.1 Method 1

The following assumptions are made in this procedure[2]:

a. The velocity of the tornado winds at ground level is 300 mph.

b. The force associated with a 300 mph tornado acts on the object over a horizontal distance equal to a 90° chord of the diameter of the maximum velocity of the tornado. The linear horizontal distance (the chord) from the point at which the tornado picks up the object, to the point where it leaves the tornado, is 348 ft.

c. The maximum area of the missile remains perpendicular to the winds for the entire distance over which it is accelerated.

d. The object is deflected upward at an angle of 45° without loss of energy.

e. No drag force acts on the object once it leaves the tornado.

Horizontal acceleration may be expressed as:

$$\ddot{X}\left(\frac{W}{g}\right) = \frac{C_d A_m d\left(V_w - \dot{X}\right)^2}{2g}$$

and

$$\ddot{X} = \frac{C_d A_m d\left(V_w - \dot{X}\right)^2}{2W}$$

where

C_d = drag coefficient

A_m = maximum cross-sectional area of the object

d = density of air

V_w = wind velocity

$\dot{X}$ = horizontal velocity of object

W = weight of object

This equation can be solved for $\dot{X}$ as expressed in the following form:

$$\frac{V_W}{\left(V_W - \dot{X}\right)} - \log_e\left(V_W - \dot{X}\right) = \left(C_d A_m d\left(\frac{X}{2W}\right)\right) + 1$$

where X is the chord distance described in assumption b., above, Once the object has left the tornado area, it is acted upon by gravity only. The equations of motion are:

$$Y = \left(\frac{-gt^2}{2}\right) + \dot{X}_0 \sin 45° t$$

$$\dot{Y} = -gt + \dot{X}_0 \sin 45°$$

$$X = \dot{X}_0 \sin 45° t$$

$\dot{X} = \text{constant} = \dot{X}_0 \sin 45° =$ horizontal velocity after the object is deflected 45° upward

$\dot{X}_0 =$ horizontal velocity of object when leaving tornado

The vertical and horizontal velocities and displacements of the missile after it has left the tornado may be calculated using $\dot{X}_0$ and $\dot{X}_0 \sin 45°$ as initial conditions.

Several assumptions for this method are very conservative. A tangential wind velocity of 300 mph is a conservative maximum value and is common design practice[2]. Furthermore, maximum winds normally occur better than 100 ft. above the ground, so that a ground level assumption is very conservative. The distance over which the object is acted upon by maximum winds is also conservative. Since the momentum of the object will cause it to be hurled in a straight line, the distance over which an object would be accelerated by winds would necessarily be limited to something less than the assumed 90° chord. Furthermore, the object would most certainly bounce several times, thereby slowing the missile down. The assumption that the maximum area of the missile remains perpendicular to the winds for the entire distance over which it is accelerated is very conservative. Some objects, such as roofs and trees, can sail and soar in the winds, but objects which present the most serious potential hazards to the fuel in storage are not aerodynamically stable, and will turn in the wind.

A.15.5.2 Method 2

Method 1 assumes that the maximum area of an object remains constantly perpendicular to the wind. Method 2 is largely predicated on the same assumption as is Method 1, except the object is assumed to tumble in the wind, and its energy may be expressed as velocity or height or combinations of both. The force exerted by wind on an object is expressed as:

$$D_f = 0.5 C_d d \left(V_w - \dot{X}\right)^2 \left(A_+ \cos^2 a + A_= \sin^2 a\right)$$

where a is the angle of the wind with respect to an orthogonal axis of the object, A_+ is the cross-sectional area perpendicular to the wind, and $A_=$ is the cross-sectional area parallel to the wind.

Most frequently, the object tumbles in such a manner that the wind makes a random angle with respect to the orthogonal axes. The average values of $\cos^2 a$ and $\sin^2 a$ are therefore 1/4, obtained by squaring their values integrated over all angles from 0 to π, and the equation becomes:

$$F_{ave} = 0.125 C_d d \left(V_w - \dot{X}\right)^2 \left(A_+ + A_=\right)$$

Very short increments of time are used to determine the velocity of the object at any instant:

$$\dot{X}_i = \dot{X}_{i-1} + F_{ave} / M(dt)$$

A step-by-step integration is then used to determine the final velocity.

A.15.5.3 Method 3

This method was presented by Bates and Swanson[3], and later included in a paper by Doan[4]. As in the previous method, tumbling of the object is assumed. The average force on the object is assumed to act for an average time of application, and the difference in velocities between the wind and the missile is not considered. The force acting on the object is approximately:

$$F_d = q\, C_d\, (A_+ \cos^2 a + A_= \sin^2 a)$$

where:

$$q = 1/2\, d\, V_w^2$$

Again, the values of $\cos^2 a$ and $\sin^2 a$ are determined to be 1/4, and the equation becomes:

$$F_{ave} = 1/4\, C_d\, q\, (A_+ + A_=)$$

The speed and kinetic energy of the missile are:

$$\dot{X} = F_{ave} t_{ave} / M$$

$$E = F_{ave}^2 t_{ave}^2 / 2M$$

where t_{ave} is the average time of force application. This average is estimated to be on the order of 0.2 second.

If all of the energy acquired is used to lift the object vertically, the maximum height attained is:

$$H_{\max} = \frac{F_{ave}^2 t_{ave}^2}{2Mg}$$

Bates and Swanson observe that the force exerted on a fixed object (conserved angle of attack) is of short duration because by the time the aerodynamic force has increased to a value sufficient to lift most objects, the moments which produce tumbling are also large. The mean time interval of action is estimated as 0.2 second. Doan does not discuss the merits of his time interval for a tumbling object, but simply estimates it as 0.2 second. The relatively low values obtained from this method reflects this short period of time.

A.15.5.4 Method 4

This method was developed after discussions by telephone with several offices of the Weather Bureau concerned with tornadoes and is based on the following assumptions:

a. The object is acted upon by the maximum winds for a distance equal to the radius of the tornado.

b. A maximum horizontal wind of 300 mph and a maximum vertical wind of 300 sin 45° mph act constantly on the vertical and horizontal faces of the object.

c. Since vertical velocities are small at the ground surface, it is assumed that the object is initially at a height of 30 feet above the ground.

The two basic equations of motion for objects within the tornado are:

$$M\ddot{Y}_1 = F_y - Mg$$

$$M\ddot{X}_1 = F_x$$

where $\ddot{Y}_1$ and $\ddot{X}_1$ are the accelerations within the tornado; F_y and F_x are the forces due to the tornado-induced pressures in the vertical and horizontal directions, respectively. The initial motion of the object when encountered by the tornado is zero. Upon leaving the tornado area, the missile is acted upon by gravitational force alone, and the equations become:

$$M\ddot{Y}_2 = -Mg$$

$$M\ddot{X}_2 = 0$$

Here, the initial velocity conditions are the maximum attained within the tornado.

A.15.6 Discussion and Results

Results of these analyses are listed in Table A.15-1, and applied in Section 8. Method 1 proved the most severe, the second being Method 2. Principal differences of all of the methods are: a. constant exposure of maximum missile area to wind versus a tumbling action, and b. the duration of time of wind acting on object. While the time element of Method 3 or 4 may be more nearly correct, the lack of pertinent information on the effective time of attack rules in favor of Methods 1 and 2. Of these, method 2 is more realistic but impact analyses were performed for velocities calculated by Method 1 to be more conservative.

Table A.15-1
RESULTS OF WIND ANALYSES

Missile	Telephone Pole 12-in. Diameter x 20 in. (630 lb)				Automobile 5 ft x 5 ft x 8 ft long (1,800 lb)			
	Maximum Velocity (ft/sec)		Maximum Distance (feet)		Maximum Velocity (ft/sec)		Maximum Distance (feet)	
	Horizontal	Vertical	Horizontal	Vertical	Horizontal	Vertical	Horizontal	Vertical
Method 1	264[a]	187	2174	543	242[a]	171	1820	455
Method 2	171[a]	121	899	228	187[a]	132	1083	271
Method 3	16	-	6	4	17.5	-	7	5
Method 4	71	53.5	195	44	49	50.5	112	36

a Horizontal velocity before the object is deflected upward at 45°

A.15.6.1 Impact Analysis

For analysis of impact effect within the water-filled basins, it is further assumed that the object enters the water vertically at the velocities calculated from the above assumptions (187 ft./sec. for the telephone pole and 171 ft./sec. for the automobile), as shown in Table A.15-1, in an "end on" orientation.

Upon entering the basin water, forces acting on a missile are:

(1) Mass inertia $(m\dot{x})$

(2) Weight of the missile (mg)

(3) Buoyancy (ρvg)

(4) Drag $\left(\frac{C_d A\dot{x}^2}{2}\right)$

where:

m = mass of the missile

v = submerged volume of the missile

A = area of the missile perpendicular to the direction of movement

C_d = drag coefficient (assumed in all cased to be 1.0)

$\ddot{x}$ = missile acceleration at time t

$\dot{x}$ = missile velocity at time t

ρ = mass per unit volume of the basin water

For passage of a missile through the basin water,

$$m\ddot{x} = mg - \left(\frac{C_d A\dot{x}^2}{2}\right) - \rho v g,$$

or

$$\ddot{x} = g - \left(\frac{C_d A\dot{x}^2}{2m}\right) - \frac{\rho v g}{m} = \left(1 - \frac{\rho v}{m}\right) - \left(\frac{C_d A}{2m}\right)\dot{x}^2,$$

and

$$\ddot{x} = \frac{d\dot{x}}{dt} = \left(\frac{d\dot{x}}{dx}\right)\left(\frac{dx}{dt}\right) = \frac{\dot{x}d\dot{x}}{dx}$$

Letting $\theta = C_d A/m$ and $\phi = g\,(1 - \rho v/m)$ and noting that $\dot{x} = V_0$ at x = 0,

$$\dot{x}^2 = \frac{2\phi}{\theta} + \left(V_0^2 - \frac{2\phi}{\theta}\right)e^{-\theta x}$$

In the case of the telephone pole, a step-by-step solution was developed to evaluate its velocity at different depths of penetration, assuming constant end-on orientation. On this basis, velocity

after penetrating to a depth of 8 ft. is 138 ft./sec.; after 14 ft., 111 ft./sec.; and after 21 ft., 88 ft./sec.. Total penetration required to stop the pole exceeded the depth of the basin. On striking the bottom liner, missile kinetic energy would be approximately 5 x 10^4 ft.-lb.

For the automobile, no buoyancy was assumed until after it had penetrated 2 ft. into the basin water and its submerged volume then was assumed to remain constant to account for leaks. On this basis, total penetration for deceleration to terminal velocity (< 6 ft./sec.) was 7.3 ft.

A.15.6.2 Effects of Missile Impact on Basin Structure

The potential penetration of the basin liners and wall by the postulated missiles was evaluated[5]. The penetration of a steel plate is described by the equation:

$$\frac{E}{D} = U\left(0.344t^2 + 0.00806wt\right)$$

where:

E = critical kinetic energy required for penetration;

D = diameter of missile (in.);

U = ultimate tensile strength of steel;

t = thickness of steel plate (in.); and

w = distance between supports of the plate (in.).

The penetration and perforation of concrete, masonry and sand is similarly described by the equation:

$$D' = KAV'R$$

where

D' = depth of penetration (ft);

K = penetration coefficient for reinforced concrete = 4.76 ft./lb.;

A = sectional mass of missile (lb./ft.^{3});

R = thickness ratio of the penetration of a slab of thickness T to the penetration of a slab of infinite thickness;

V' = velocity factor for impact penetration = $\log_{10}\left[1+\frac{V^2}{215{,}000}\right]$

where:

V = missile velocity.

Material properties and structural dimensions used in the analysis were:

t = 0.125 in. for basin liners up to 16 ft. elevation

= 0.0625 in. for basin liners above 10 ft. elevation

= 2 in. for deep pit floor

U = 75,000 psi (70,000 for deep pit)

D (telephone pole) = 13.5 in. diameter

D (small automobile) = 61.2 in. diameter

W = 3 ft.

Analyses were performed, for each postulated missile, for potential penetration of each type of material (i.e., assuming no concrete backing for the steel plate and for concrete penetration assuming no liner). Both the walls and floor of the basin were analyzed for potential penetration.

A.15.6.2.2 Analysis

Wood planks, sections of steel pipe, a telephone pole and a small automobile have been analyzed as potential missiles. Of these missiles, the telephone pole and the automobile represent equivalent or greater potential damage than the others. The analysis of the automobile missile indicates that it does not have sufficient energy to penetrate the 1/16 in. thick wall liner even at its maximum horizontal velocity of 440 ft./sec. due to the large impact cross-sectional area of the automobile. At the maximum horizontal velocity, the kinetic energy of the automobile (5.4×10^6 ft.-lb.), ignoring the fact that the liner is backed by reinforced concrete. The automobile would be traveling in a trajectory and thus would not strike the wall perpendicularly. There is no possibility of penetrating the 3/16 in. thick floor liner as the automobile would be traveling at its settling velocity (< 6 fps) and the kinetic energy is only about 1,000 ft.-lb.

The analysis of the impact of the telephone pole missile indicates that puncture of the basin liner is extremely improbable. For example, the energy required to penetrate the floor liner in the basin, ignoring the backup strength of the concrete, is in excess of 1.6×10^6 ft.-lb. for an impact perpendicular to the liner. At that depth, the kinetic energy of the telephone pole is less than 5×10^4 ft.-lb., and, thus, there will be no penetration of the floor liner in the basins. A recent report[6] indicates that telephone poles (utility poles) are ineffective in producing significant local and structural damage even under the most improbable missile impact conditions.

The telephone pole cannot strike the walls at any angle that is nearly perpendicular at a depth sufficient to cause significant leakage even if the liner should be penetrated. Penetration of the liner near the top of the pool would not be of concern and penetration of a vertical wall deep in the pool would require more energy than bottom penetration (1.6×10^6 ft.-lb.) due to the angle of impact. For example, after travel through 21 ft. of water, the impact kinetic energy would be only about 4.29×10^5 ft.-lb. Even for a perpendicular impact, penetration can occur only if the concrete backing is ignored. The compressive strength of the concrete might be exceeded in local areas, but due to the low void fraction of structural concrete and its confinement, there would be no significant crushing. Therefore, the telephone pole will not penetrate the wall liners based on the strength and ductility of the liner, on the possible angle of impact, and on the relative crushing strengths of the pole and the concrete.

It is concluded that penetrations of the basin liners caused by the telephone pole missile are very unlikely and that the leaks resulting from such penetrations, if any, are well within the repair capability of GE-MO.

A.15.7 REFERENCES

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APPENDIX B MICROFICHE INDEX

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B.1	Report of Preliminary Site Exploration - September 1966, Dames and Moore, Chicago, Illinois
B.2	Foundation Investigation - December 1967, Dames and Moore, Chicago, Illinois
B.3	Earthquake and Wind Analysis - Plant Effluent Stack, November 1968, H. J. Saxon & Associates, Engineers, San Francisco, California
B.4	Earthquake and Wind Analysis - Underground Structures, March 1969, H. J. Sexton & Associates, Engineers, San Francisco, California
B.5	Criticality Safety Basis for Project I Fuel Storage, May 1975, Battelle Pacific Northwest Laboratories, Richland, Washington
B.6	Fuel Storage System Design Report - Project I Fuel Storage, April-May 1975, Programmed & Remote Systems Corporation, St. Paul, Minnesota
B.7	Seismic Shock Environment Test of Simulated Nuclear Fuel Storage Basket, August 1975, Department of the Army, Construction Engineering Research Laboratory, Champaign, Illinois
B.8	Spent fuel Services Operation Quality Assurance Plan, NEDO-20776, General Electric Company, San Jose, California
B.9	Report of Fault Investigation at the Midwest Fuel Reprocessing Plant near Morris, Illinois, October 1974; Dames & Moore, Chicago, Illinois
B.10	Report of Geologic and Ground Water Investigation, Proposed Spent Fuel Storage Facility near Morris, Illinois, September 1975; Dames & Moore, Chicago, Illinois
B.11	Evaluation of Foundation Recommendations - Project IV -Fuel Storage Capacity Expansion near Morris, Illinois (letter report) May 1977; Dames & Moore, Chicago, Illinois
B.12	Report of Ground Water Investigations - Project IV - Fuel Storage Capacity Expansion near Morris, Illinois, August 1977; Dames & Moore, Chicago, Illinois
B.13	Report of Geophysical Investigations - Project IV - Fuel Storage Capacity Expansion near Morris, Illinois, July 1977; Dames & Moore, Chicago, Illinois

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B.14	Report of Geologic Investigations - Project IV - Storage Capacity Expansion near Morris, Illinois, August 1977; Dames & Moore, Chicago, Illinois
B.15	Criticality Safety Analysis for Square Tube Storage Baskets at the Morris Operation, May 1987; G.E.
B.16	Morris Square Tube Basket Stress Report Revision, June 1987; G.E.
B.17	Proposed Approach to Evaluate Adequacy of Ground Water Monitoring System, February 1993; Dames & Moore, Chicago, Illinois
B.18	Ground Water Modeling & Specs for Monitoring Wells, August 1993; Dames & Moore, Chicago, Illinois
B.19	Quality Assurance Plan Monitoring Well Installation, October 1993; Dames & Moore, Chicago, Illinois
B.20	Final Report Transport Modeling for Accidentally Released Water from Spent Fuel Storage Basin, October 1993; Dames & Moore, Chicago, Illinois
B.21	Ground Water Monitoring Well Network Summary & Installation Report, January 1994; Dames & Moore, Chicago, Illinois
B.22	Offsite Dose Calculation Manual, July 1994; GE
B.23	Radwaste System Description, July 1994; GE

APPENDIX B.22
OFFSITE DOSE CALCULATION MANUAL FOR GENERAL ELECTRIC - MORRIS OPERATION

B.22.1 INTRODUCTION

This manual presents methods for calculating doses to members of the public from releases of radioactive material from General Electric Morris Operation (GE-MO). The methods employ values of wind speed, stability, and average X/Q taken from 1992 site meteorological data as typical of local conditions. This is simpler than using real time meteorological conditions and on-line calculations, and can be justified because the doses which could occur from credible releases are a small fraction of those allowed by regulation. For doses as small as these, the effort required to obtain the more sophisticated on-line analyses is not cost effective.

The low value of credible doses which could be received from accidental releases from GE-MO is illustrated by information taken from the Consolidated Safety Analysis Report[1](CSAR). This is shown as follows:

Type of Accident	Whole Body Dose (mRem)
Fuel Bundle Drop	2.0×10^{-2}
Fuel Basket Drop	8.1×10^{-2}
Tornado Missile	8.0×10^{-1}

In addition, the doses from normal operations have been calculated, based on the annual quantities of radionuclides released from the site over the past 10 years (1983 through 1992). These values are shown as follows:

Radionuclide	Maximum Annual range	Off-Site Dose (mRem) average
H-3	2.4 to 6.8×10^{-7}	4.4×10^{-7}
Co-60	0.39 to 2.0×10^{-8}	1.2×10^{-8}
Kr-85	1.0 to 1.6×10^{-5}	1.4×10^{-5}
Cs-134	0.18 to 1.6×10^{-9}	5.6×10^{-10}
Cs-137	0.01 to 5.8×10^{-8}	1.9×10^{-8}

These values are also many orders of magnitude below both regulatory limits and default values used to evaluate the achievement of ALARA goals (10 mRem/year according to Draft Regulatory Guide DG-8013[2]).

The methods are divided into three categories. The first category includes the methods for determining releases during normal operations and is based on measurement of stack samples

or calculations of volumes of air released. Doses found using these methods are the annual doses which would be received by a member of the public at the worst off-site location.

The second category includes the methods used to calculate doses from an accidental ground level release. These methods require a measurement or estimate of the quantity of each radionuclide released, and give the dose to the nearest resident (assuming conservatively that the wind is in the direction of that nearest resident at the time of the accident). For a ground level release the nearest resident down wind receives the highest dose. The dose to that resident can be determined for worst case meteorological conditions, or more typical conditions.

The third category consists of similar methods for determining the dose from an accidental release through the 300 ft. stack. Again, a determination of the amount of material released is needed. With this given, the methods allow the calculation of the dose to a member of the public who remains at the worst off-site location for the duration of the release. The dose can be determined for the worst meteorological conditions, or average conditions.

Appendices of the manual give a tabulation of the parameters used to calculate values of X/Q for accidental releases, the variation of stability classes and wind speeds seen on-site in 1992, and justification for using neutral condition X/Q values for more stable conditions when considering releases via the 300 ft. stack.

B.22.2 NORMAL CONDITIONS

The off-site dose under normal conditions is considered to be the result of chronic releases from the 300 ft. stack. This is approximated by assuming a uniform release rate over a period of a year. Three factors must be known to compute the dose, the atmospheric dispersion (X/Q in sec/m^3), the dose per μCi inhaled (Rem/μCi), and the average rate of release.

The atmospheric dispersion has been determined experimentally over a number of years as part of the joint GE-CECO meteorological monitoring program conducted at the adjacent Dresden Nuclear Power Station (DNPS). The value of the atmospheric dispersion used in this manual is the maximum offsite relative concentration (X/Q) for 1992. That value, 7.88×10^{-8} sec/m^3, is typical of those seen for the annual periods over the past 20 years.

The dose per μCi inhaled (dose conversion factor) (Rem/μCi) is taken directly from the "Internal Dose Conversion Factors for Calculation of Dose to the Public", DOE/EH 0071[3] for the individual radionuclide inhaled. The concentration in the air breathed is multiplied by the quantity breathed (22,800 liters per day for the Standard Man) and this dose conversion factor.

The average activity released over a year is determined differently depending on whether the radionuclides released are particulates, krypton-85 (Kr-85), or tritium (H-3) (as HTO).

Particulates: The basis for determining the dose from the chronic release of particulates is the result of the stack sampler composite analysis. Weekly air samples are collected from Loops 1

and 2 of the stack sampler, and composited. At the end of six months each composite is analyzed for gamma emitting radionuclides using a germanium detector. The highest activity of a radionuclide (from analysis of the Loop 1 or Loop 2 composite) during the first six month period is added to the highest value for that radionuclide during the second six months. The total (referred to as the total quantity sampled) is a conservative assessment of the amount of that radionuclide that passed through a sample loop during the year.

The activity of the radionuclide released is found by multiplying that total by the ratio of flows, ((in the stack) to (in a sample loop)). This number is typically about 24,000, the stack flow rate (12,000 ft^3/min.) divided by the loop flow rate (0.5 ft^3/min.). The product of the total quantity sampled and this ratio gives the quantity of that radionuclide released in the year (in μCi). The average release rate in Ci/sec. is determined by dividing this value by the number of seconds per year and converting μCi to Ci.

Equation 1 gives the average concentration (Ci/m^3 or μCi/cm^3) of any radionuclide released in particulate form at the worst off-site location.

$$\text{Average Concentration} = (A_1+A_2)(\text{Flow ratio})(X/Q)/3.15 \times 10^{13} \qquad (1)$$

Here A_1 is the activity in μCi of the radionuclide of interest on the stack sampler composite for the first half of the year (The highest value of Loop 1 and Loop 2). A_2 is the same value for the second half. 3.15×10^{13} is the conversion from μCi/yr. to Ci/sec.

Equation 2 gives the annual committed effective dose equivalent (in Rem) which could result from this radionuclide if a person were to occupy the worst off-site location for the year's duration.

$$\text{Annual CEDE} = 8.32 \times 10^9 (\text{Aver. conc.})(\text{Dose conversion factor}) \qquad (2)$$

The average concentration is taken from Equation 1, and dose conversion factor (in Rem/μCi) is from DOE/EH 0071. 8.32×10^9 is the cm^3 of air breathed in a year by the standard man.

Kr-85: The basis for determining the dose from the chronic release of Kr-85 is the measurement of that radionuclide in the air over the basin. This was originally done in 1980. The concentration[4] was found to be 5.8×10^{-8} μCi/cm^3. More recently, in 1992, two additional measurements were made, one over each basin. They averaged 3.8×10^{-8} μCi/cm^3. The agreement between the two indicates that the release rate can be considered to be constant. For this manual the higher of the two values is used.

The concentration in the basin air is multiplied by the flow rate into the basin exhaust plenum to get the rate of release, and this value is multiplied by X/Q to get the concentration at the worst off-site location. The calculation of this worst average off-site concentration (Ci/m^3 or μCi/cm^3) is shown in Equation 3.

Kr-85 off-site concentration = 4.72×10^{-4} (C) (Flow rate) (X/Q) (3)

Here "C" is the concentration of Kr-85 over the basin in $\mu Ci/cm^3$, and "Flow rate" is the air flow rate through the basin exhaust plenum in $ft.^3/min.$, (typically 7,500 $ft.^3/min.$). 4.72×10^{-4} converts the product from ($\mu Ci/cm^3$)($ft.^3/min.$) to Ci/sec. The whole body and skin doses which would be incurred from occupying the area of highest off-site concentration are determined from Equations 4 and 5 respectively.

Kr-85 skin dose (mRem/yr.) = 10^6 (Off-site conc.) (Skin dose factor) (4)

The skin dose factor is taken directly from the "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," DOE/EH 0070[5]. It is 1.58×10^3 (mRem/yr.)/($\mu Ci/m^3$) for Kr-85.

Kr-85 deep dose (mRem/yr.) = 10^6 (Off-site conc.) (Deep dose factor) (5)

The deep dose factor is taken directly from DOE/EH 0070. The value is 1.12×10^1 (mRem/yr.)/($\mu Ci/m^3$) for Kr-85.

H-3: The basis for determining the annual dose from the chronic release of H-3 is the measured concentration of H-3 in the basin and the amounts of water released from this reservoir during the year. Concentrations of H-3 in the basin are determined periodically as part of the Operability Test/Compliance Test system. The volumes of water released are determined from operating records for basin make-up water added during the year. Equation 6 gives the average concentration of tritium (Ci/m^3 or $\mu Ci/cm^3$) at the worst off-site location.

H-3 off-site concentration = $(C_1V_1)(X/Q)/3.15 \times 10^{13}$ (6)

Here C_1 is the average H-3 concentration in basin water ($\mu Ci/cm^3$) and V_1 is the volume of basin make-up water added in the year in cm^3. . 3.15×10^{13} is the same conversion used in Equation 1. The whole body committed effective dose equivalent (Rem) which would be incurred from occupying the area of highest off-site concentration is determined from Equation 7.

Annual CEDE = 8.32×10^9 (Off-site conc.)(Dose conversion factor) (7)

The dose conversion factor for H-3, taken from DOE/EH 0071, is 6.3×10^{-5} Rem/μCi.

B.22.3 Accident Conditions (General)

The off-site dose under accident conditions depends on the quantity of each radionuclide released, the release point (ground level or 300 ft.), and the meteorological conditions. The amount of each radionuclide released is estimated or measured at the time of release.

Analyses made to support the Consolidated Safety Analysis Report (CSAR) give some conservative estimates for releases which could occur as the result of credible accidents at GE-MO. Examples of some of the amounts which could be released according to the CSAR are shown as follows:

Fuel Bundle Drop	1.471×10^{3} Ci noble gases and 8.94×10^{-3} Ci iodines
Fuel Basket Drop	6.111×10^{3} Ci noble gases and 3.717×10^{-2} Ci iodines
Tornado Generated Missile	3.5×10^{3} Ci noble gases and 7.7×10^{-4} Ci iodines
Basin Cooler Leak	2×10^{-6} Ci/sec Cs-134

Because of the age of the fuel stored at GE-MO the noble gases are assumed to be all Kr-85, and the iodines all I-129. For a similar reason Cs-137 will be used instead of Cs-134.

The release point for the various kinds of accidents is known because of the features of the facility. Radionuclides released from events which occur in the basin or the process building such as the fuel bundle and basket drops exit through the sand filter and the 300 ft. stack. Radionuclides released by evaporation or by tornado (which would destroy the integrity of the metal building covering the basin) were assumed to be ground level releases.

The dispersion of the released material due to air movement is given by Equation 8.

$$X = \left(\frac{Q}{2\pi\sigma_z\sigma_y\mu_h}\right)\exp-\left(\frac{z^2}{\sigma_z^2}+\frac{y^2}{\sigma_y^2}\right) \qquad (8)$$

Here "X" is the average air concentration (Ci/m^3 or μCi/cm^3) at any selected point, and "Q" is the release rate in Ci/sec. "u_h" is the average wind speed at the height of the release, and "σ" is the standard deviation of the cloud width in the horizontal (y) direction and vertical (z) direction. "y" and "z" are horizontal and vertical distances from the centerline of the plume, and "t" is the time after the release. Therefore the distance (x) from the point of release to the selected point is the product of u_h and t.

For purposes of this manual cloud depletion is not considered and the dose is figured to a person directly downwind of the release. Therefore "y = 0". For a ground level release "z = 0" also. Under these conditions Equation 8 can be simplified. This simplified expression is shown as Equation 9.

$$\frac{X}{Q} = \frac{1}{\left(2\pi\sigma_z\sigma_y\mu_h\right)} \tag{9}$$

Watson and Gamertsfelder[6] also give equations to determine σ_y and σ_z. These are repeated as Equations 10, 11, and 12.

$$\sigma_y^2 = \left(C_y\right)^2 x^{(2-n)/2} \tag{10}$$

$$\sigma_z^2 = \left(C_z\right)^2 x^{(2-n)/2} \quad \text{(neutral and unstable case)} \tag{11}$$

$$\sigma_z^2 = a\left(1-\exp\left(-k^2t^2\right)\right)+bt \quad \text{(stable case)} \tag{12}$$

Parameters needed to solve these equations are given for four stability classes and several wind speeds in the FSAR[7], and repeated in Appendix 1 to this manual. The fraction of the time that each condition (wind speed and stability class) was observed during the reference period (1992) according to Murray and Trettel[8] is reproduced in Appendix 2. As shown in the following table the range in the lapse rate for each class was used to establish the relationship between the four classes of the FSAR and these seven classes:

FSAR Stability Class	Lapse Rate	M&T Stability Class
Unstable	< - 1.5 °C	EU + MU + SU
Neutral	- 1.5 to - 0.5 °C	N
Moderately Stable	- 0.5 to 1.5 °C	SS
Very Stable	> 1.5 °C	MS + ES

Appendix 2 to this manual gives the fractional time that each FSAR stability class and each wind speed were prevalent during 1992.

No consideration is given in this manual to variations in wind direction. It is assumed that when the accidental conditions exist, the wind is blowing straight towards the most likely exposed member of the public. For ground level releases, this is the closest individual. For elevated releases this is the individual at the distance where (X/Q) is maximum.

The following sections determine the atmospheric dispersion (X/Q) based on the parametric values originally provided in the FSAR for different release points (ground level and 300 ft.), wind speeds, and stability classes. These values are then matched with the corresponding prevalence of each atmospheric condition (from M&T) to determine the most likely (X/Q), the median (X/Q), and the maximum (X/Q) that would be expected.

B.22.4 Accident Conditions - (Ground Level Release)

The maximum concentration from a ground level release is to the nearest neighbor, and his residence is about 508 m east of the process building. The concentration at this worst location was determined by using the parameters in Appendix 1 to compute σ_y and σ_z for each of the four stability classes, and for wind speeds of 2.25, 5.5, 10, 15, 21, and >24.5 mph. Then Equation 9 was used to compute X/Q. Table B.22-1 gives the X/Q values for each atmospheric condition.

TABLE B.22-1
MINIMUM ATMOSPHERIC DISPERSION for a GROUND LEVEL RELEASE

Wind Speed (m/h)	Unstable	Neutral	Moderately Stable	Very Stable
1 - 3.5	3.47 E-5	1.63 E-4	3.83 E-4	9.17 E-4
3.6 - 7.5	1.42 E-5	6.68 E-5	1.97 E-4	4.07 E-4
7.6 - 12.5	1.07 E-5	6.24 E-5	1.23 E-4	2.31 E-4
12.6 - 18.5	7.11 E-6	4.16 E-5	9.39 E-5	1.56 E-4
18.6 - 24.5	5.83 E-6	3.43 E-5	8.08 E-5	1.12 E-4
> 24.5	4.53 E-6	2.66 E-5	7.44 E-5	8.74 E-5

When this table is combined with the data in Appendix 2 (which has been condensed to match the four stability classes found in the FSAR) one can determine the distribution of expected X/Q values. This is shown in Table B.22-2.

TABLE B.22-2
DISTRIBUTION OF EXPECTED VALUES OF ATMOSPHERIC DISPERSION
for a GROUND LEVEL RELEASE

X/Q	Cum %	X/Q	Cum %	X/Q	Cum %
9.17 E-4	00.00	1.12 E-4	20.33	3.47 E-5	76.19
4.07 E-4	00.62	9.39 E-5	21.38	3.43 E-5	76.25
3.83 E-4	02.49	8.74 E-5	31.87	2.66 E-5	83.92
2.31 E-4	02.91	8.08 E-5	31.88	1.42 E-5	86.79
1.97 E-4	06.10	7.44 E-5	36.12	1.07 E-5	89.53
1.63 E-4	08.00	6.68 E-5	36.96	7.11 E-6	94.23
1.56 E-4	08.95	6.24 E-5	43.18	5.83 E-6	98.51
1.23 E-4	13.28	4.16 E-5	59.06	4.53 E-6	99.80

In Table 2 "Cum %" refers to the percent of the time that a larger value of X/Q would be expected based on the 1992 data. A review of this table shows, based on 1992 data, that the

worst case X/Q for a ground level release is 9.17×10^{-4} sec/m^3. In addition, the median and most frequent X/Qs are found to be 4.16×10^{-5} sec/m^3.

The values from Table 2, when combined with the credible releases identified in the CSAR allow one to calculate the doses that the person occupying the residence at the worst off-site location would receive. In each case it is assumed that person is exposed for the entire duration of the release. The necessary calculations are made using Equations 13 through 16.

Deep dose from Kr-85 (Rem) = 3.17×10^{-5} (X/Q)(Ci released)(1.12×10^{1}) (13)

Here 3.17×10^{-5} converts mRem to Rem, years to seconds and Ci to µCi; and 1.12×10^{1} gives the deep dose in mRem per yr. from immersion in a cloud of Kr-85 with a concentration of 1.0 µCi/m^3 (from DOE/EH 0070).

Skin dose from Kr-85 (Rem) = 3.17×10^{-5} (X/Q) (Ci released) (1.58×10^{3}) (14)

This equation is the same as Equation 13 except for the dose conversion factor of 1.58×10^{3} (mRem/yr.)/(µCi/m^3) to the skin (from DOE/EH 0070).

CEDE from I-129 (Rem) = 2.64×10^{2} (X/Q)(Ci released)(0.18) (15)

The factor 2.64×10^{2} is the volume of air breathed per second by the Standard Man (in cm^3). It comes from 22,800 L/day usually quoted in tables. 0.18 is the dose conversion factor for iodine-129 in Rem/µCi from DOE/EH 0071.

CEDE from Cs-137 (Rem) = 2.64×10^{2} (X/Q)(Ci released)(3.2×10^{-2}) (16)

This equation is similar to Equation 15, except that the dose conversion factor (3.2×10^{-2} Rem/µCi) is for Cs-137.

Conservative values of the maximum dose, the most likely dose, or the median dose can be calculated from these formulae depending on the value of X/Q chosen from Table 2.

For cases other than those quantified in the CSAR, the amount and kind of the radionuclides released needs to be measured or estimated. The amount of each radionuclide released is substituted into Equation 13 to determine the external deep dose from immersion in the plume. Similarly, the use of Equation 14 will give the skin dose, and substitution into Equation 15 or 16 is for determining the CEDE. For radionuclides other than Kr-85, I-129 and Cs-137 appropriate dose conversion factors need to be found by referring to DOE/EH 0070 or DOE/EH 0071.

B.22.5 Accident Conditions -- (release at 300 ft.)

The maximum concentration from a release at an elevation of 300 ft. occurs at the center of the plume, downwind, at a distance that depends on the meteorological conditions. The distance

and the maximum X/Q can be determined from Equation 8 for unstable and neutral conditions as shown in the following text. It is subsequently shown that the maximum X/Q for stable and very stable conditions is lower than X/Q for neutral or unstable conditions. Therefore, the value for neutral conditions can be used as a conservative estimate of X/Q for more stable conditions.

The distance where X/Q is maximum is determined by differentiating Equation 8 with respect to "x" and setting the result equal to 0. Since one is interested in the value at the plume centerline, y = 0, and half of the exponential term drops out. Equation 17 gives the expression for determining the downwind distance (in meters) where X/Q is maximum.

$$x^{(n-2)} = \left(\frac{C_z}{z}\right)^2 \qquad (17)$$

"z" is the stack height (no credit is taken for plume rise), and "C_z" is the value from Appendix 1 for the meteorological condition considered.

Once "x" is determined, it can be substituted into Equation 8 to calculate X/Q. This calculation was done for each of the wind speed categories shown in Table 1 for both neutral and unstable conditions. The results are shown in Table 3.

TABLE B.22-3
MINIMUM ATMOSPHERIC DISPERSION for a RELEASE at 300 feet

Wind Speed (m/h)	Unstable X/Q (sec/m^3)	Distance (m)	Neutral X/Q (sec/m^3)	Distance (m)
1 - 3.5	6.94 E-6	575	6.94 E-6	1523
3.6 - 7.5	2.85 E-6	575	2.85 E-6	1523
7.6 - 12.5	1.57 E-6	674	1.57 E-6	1965
12.6 - 18.5	1.04 E-6	674	1.04 E-6	1965
18.6 - 24.5	7.47 E-7	737	7.47 E-7	2170
> 24.5	5.80 E-7	737	5.80 E-7	2170

For moderately stable and very stable conditions the differentiation of Equation 8 is more difficult, since Equation 12 must be used to express σ_z. However, it can be shown that X/Q for a release at 300 ft. for the stable cases is always less than the X/Q for the neutral case. Appendix 3 gives an example showing that this relationship exists. This relationship allows the X/Q for a given wind speed and neutral conditions to be used conservatively for the moderately stable and very stable conditions. Table 4 gives the cumulative distribution of X/Q values for the 300' release based on using neutral condition X/Q values for more stable weather.

TABLE B.22-4
DISTRIBUTION of EXPECTED VALUES of ATMOSPHERIC DISPERSION for a RELEASE at 300 feet

X/Q	Cum %	X/Q	Cum %	X/Q	Cum %
6.94 E-6	00.00	1.57 E-6	14.78	7.47 E-7	81.80
2.85 E-6	02.05	1.04 E-6	45.59	5.80 E-7	96.25

In this table as well as in Table 2 "Cum %" refers to the percent of the time that a larger value of X/Q would be expected based on the 1992 data. A review of Table 4 shows, based on 1992 data, that the worst case X/Q for a release at 300 ft. is 6.94 x 10^{-6} sec/m^3. In addition, the median X/Q and the most frequent X/Q are found to be 1.04 x 10^{-6} sec/m^3.

To calculate the dose that a person occupying the residence at the worst off-site location would receive from the release of Kr-85, I-129, Cs-137 at 300 ft. -- one measures or estimates the quantity released, selects the appropriate X/Q value from Table 4, and uses Equations 13 through 16. In each case it is assumed that the person is exposed for the entire duration of the release.

B.22.6 Accident Conditions (Summary)

Three steps are needed to compute the dose to a member of the public from accidents other that the four listed under "Accident Conditions (General)". First, one should decide if the release is from ground level or via the 300 ft. stack and select an X/Q value from Table 2 or 4 respectively. Secondly, the type and amount of radioactive material released should be determined (by measurement or estimation). Finally, these values should be substituted into Equations 13 or 14 (for doses via immersion) or Equations 15 or 16 (for doses via inhalation). For radionuclides other than Kr-85, I-129, and Cs-137 the appropriate dose conversion factors need to be found by referring to DOE/EH 0070 or DOE/EH 0071.

B.22.7 REFERENCES

1. Consolidated Safety Analysis Report, General Electric Co., NEDO 21326D, July, 1983.

2. U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-8013, "ALARA Levels for Effluents from Materials Facilities," October, 1992.

3. Internal Dose Conversion Factors for Calculation of Dose to the Public, Department of Energy, DOE/EH 0071, July, 1988.

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6. E. C. Watson and C. C. Gamertsfelder, Environmental Radioactive Contamination as a factor in Nuclear Plant Siting Criteria, February 14, 1963, (HW-SA-2809).

7. Final Safety Analysis Report, Midwest Fuel Recovery Plant, Morris, Illinois, General Electric Co., NEDO 10178, December, 1970.

8. "Semi-Annual Report on the Meteorological Monitoring Program at the General Electric - Morris Operation", Murray and Trettel, Inc., January-June, 1992.

APPENDICES

Appendix 1
Values of Atmospheric Variables

Variable	Release Height (m)	Wind Speed (m/sec)	Atmospheric Stability: Very Stable	Moderately Stable	Neutral	Unstable
a	all		34	97		
b	all		0.025	0.33		
K^2	all		0.0088	0.00025		
n	ground		0.3	0.3	0.25	0.20
n	300 ft.		0.4	0.4	0.25	0.20
C_y	ground	1 - 3	0.18	0.18	0.21	0.35
C_y	ground	4 - 7	0.18	0.18	0.15	0.30
C_y	ground	> 7	0.18	0.18	0.14	0.28
C_y	300 ft.	1 - 3	0.18	0.18	0.15	0.30
C_y	300 ft.	4 - 7	0.18	0.18	0.12	0.26
C_y	300 ft.	> 7	0.18	0.18	0.11	0.24
C_z	ground	1 - 3			0.17	0.35
C_z	ground	4 - 7			0.14	0.30
C_z	ground	> 7			0.13	0.28
C_z	300 ft.	1 - 3			0.15	0.30
C_z	300 ft.	4 - 7			0.12	0.26
C_z	300 ft.	> 7			0.11	0.24

Appendix 2
Distribution of Meteorological Conditions

Wind Speed (m/hr)	Stability Classes: Extremely Unstable (EU)	Moderately Unstable (MU)	Slightly Unstable (SS)	Neutral (N)	Slightly Stable (SS)	Moderately Stable (MS)	Extremely Stable (ES)
< 3	0.00	0.01	0.05	0.95	0.42	0.36	0.26
4 - 7	0.23	0.78	1.73	6.22	1.90	1.31	0.56
8 - 12	0.84	1.59	2.27	15.88	7.05	2.33	0.85
13 - 18	1.44	1.10	1.74	17.13	10.47	3.71	0.62
18 - 24	0.24	0.40	0.65	7.87	4.24	0.91	0.14
> 24	0.00	0.05	0.16	2.87	0.84	0.01	0.00

Appendix 3
Limiting X/Q for Moderately and Very Stable Conditions

The X/Q for neutral conditions can be used conservatively to represent the X/Q for moderately stable and very stable conditions because neutral condition X/Q is larger. This is demonstrated by computing σ_z for the four atmospheric condition classes for a 300 ft. release with a wind speed of 1.01 m/sec. It can be seen from this demonstration that the neutral condition X/Q is also larger for different distances and different wind speeds. σ_z is calculated for the distance of 575 m. listed in Table 3.

Unstable Conditions:
$$\sigma_z = \left(\frac{C_z^2\left(x^{(2-n)}\right)}{2}\right)^{1/2} = \left(\frac{0.3^2(575)^{1.8}}{2}\right)^{1/2} = 64.6$$

Neutral Conditions:
$$= \left(\frac{0.15^2(575)^{1.8}}{2}\right)^{1/2} = 32.3$$

Moderately Stable Conditions:
$$\sigma_z = \left(a\left(1-\exp\left(-K^2t^2\right)\right)+bt\right)^{1/2}$$

$$= \left(97\left(1-\exp\left(-0.00025\left(\frac{575}{1.01}\right)^2\right)+0.33\left(\frac{575}{1.01}\right)\right)^{1/2}\right) = 16.6$$

Very Stable Conditions:
$$= \left(34\left(1-\exp\left(-0.0088\left(\frac{575}{1.01}\right)^2+0.025\left(\frac{575}{1.01}\right)\right)^{1/2}\right)\right) = 6.94$$

X/Q varies with σ_z in the following way:

X/Q is proportional to:
$$\left(\frac{1}{\sigma_z}\right)\left(\exp-\left(\frac{\text{Constant}}{\sigma_z^2}\right)\right)$$

For a decrease in σ_z, X/Q decreases more rapidly.

For increasing X, the σ_z for unstable and neutral conditions increases more rapidly than for moderately and very stable conditions.

For increasing wind speeds, σ_z decreases for moderately and very stable conditions. σ_z for unstable and neutral conditions is unchanged.

APPENDIX B.23
RADWASTE SYSTEM DESCRIPTION

B.23.1 SYSTEM DESCRIPTION

The radwaste system is split into two sub-systems identified as high and low activity. The purpose of this design was to separate highly radioactive basin filter sludge from other plant waste water such as laundry, sump waste and decon solutions which are normally very low in activity.

The high activity system dewaters basin filter spent resins and returns the water to the basin while the low activity system processes waste water through an evaporator. The dewatered filter resins and evaporator bottoms are packaged and shipped as radwaste to a burial or processing site. A description of these systems is as follows:

B.23.2 HIGH ACTIVITY SYSTEM (REFERENCE FIGURE 1-16B)

Spent filter media from the basin filter is backwashed to the filter sludge tank (V-134) approximately every four to six weeks. This backwash consists of approximately 3 ft^3 of filter media mixed with 350 - 400 gallons of water. The slurry is then pumped from V-134 to a High Integrity Container (HIC).

The HIC contains four filter septums at various levels to allow for dewatering and are sized by the manufacturer for the filter medium used at this plant (Powdex Resins). The filter water transfer pump (P-520) is a positive displacement diaphragm pump which takes suction on the HIC and transfers the water through a filter (F-520) to the filter water collection tank (V-508). Most particulate remains in the HIC; however, some fines get through the septum and will be removed by the filter, F-520. F-520 is a backwashable filter that can be cleaned by backwashing to the HIC.

When the filter sludge transfer is completed, or at a later date, the water in V-508 is recirculated utilizing P-520. When samples verify the chemistry and activity of the water is acceptable, the water will be pumped to the basin filter inlet for return to the fuel basin.

During cask flush operations, the flush water may be discharged to either the high or low activity system. The choice of direction will be based on water chemistry and activity.

The HIC is located at the north end of the equipment transfer area (ETA) pit. V-508 is located at the south end of the same pit. P-520 and F-520 are located in the aqueous make-up (AMU) room which is adjacent to the ETA. The ETA pit is covered with 6 in. thick steel plates for shielding. The area directly above the HIC includes an additional 7 in. thick lower plate with a 34 by 34 in. access hole for maintenance, removal, and replacement of the HIC. Connections from the HIC to the system are made via the fill-head which mounts to the top of the HIC. Included are four suction connections for dewatering at various levels in the HIC, a fill

connection, vent connections, a backwash inlet, and instrument hook-ups. Most connections are quick disconnect hoses that attach to the system piping.

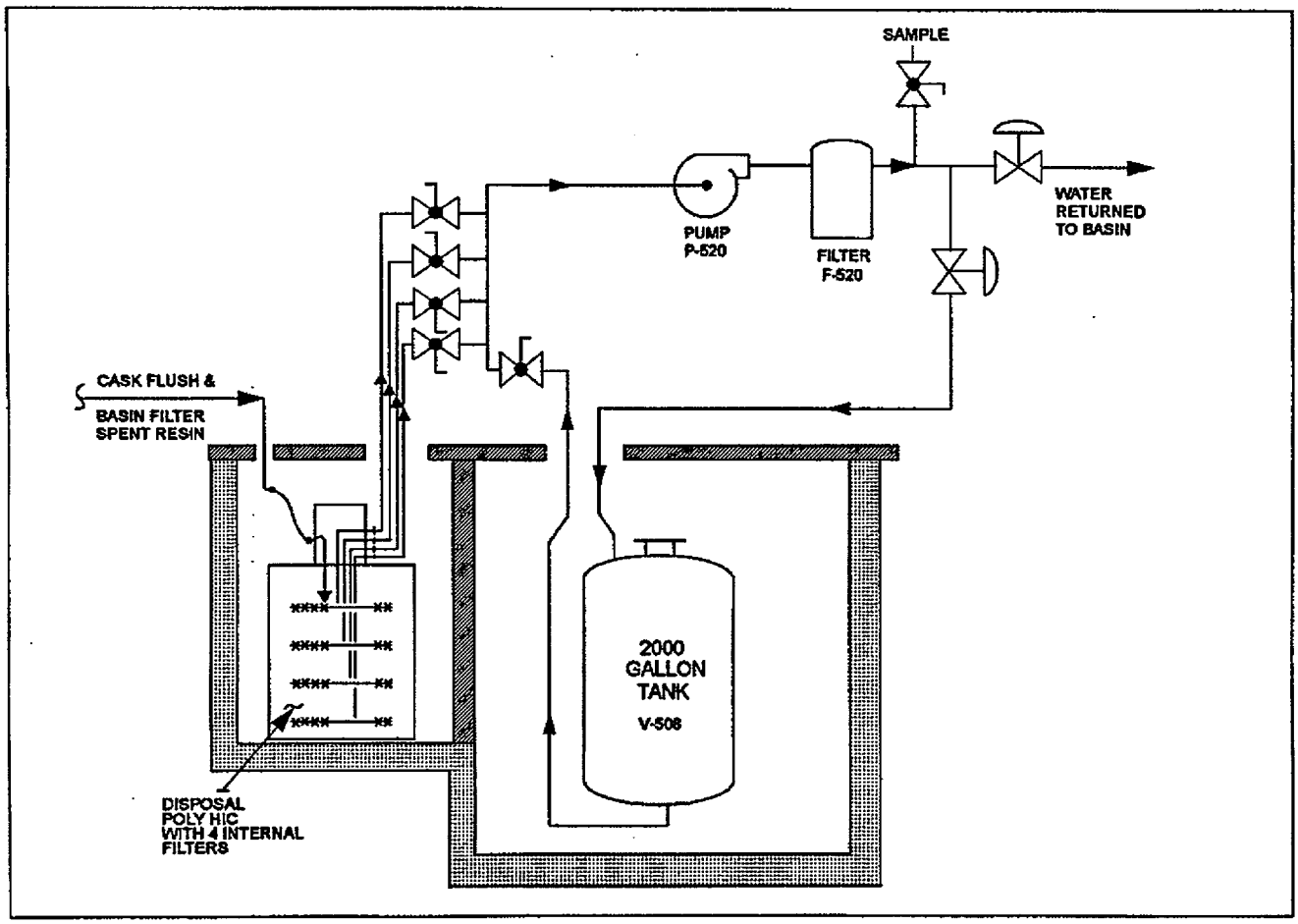

Figure 1-16b. Basin Filter Spent Resin System: Spent resins from the Basin Filter and cask flush solutions are pumped to a shielded Poly High Integrity Container (HIC). Water is removed from the HIC, filtered, and then returned to the Fuel Storage Basin. When filled, HICs are dried and shipped off site for burial.

The HIC is supported by steel plates that position the fill-head assembly at about 10 in. below the steel floor plates in the ETA. Incorporated as part of the support stand is a vertical shielding plate between the HIC and the rest of the pit. This basically provides a 7 ft. by 7 ft. box around the HIC with steel floor and south wall, and concrete on the remaining three sides. Additionally, this area is sealed with a molded plastic liner to contain any spills. Maximum anticipated radiation levels are 40 R/hr at the side of the HIC and 2 mR above the shield plates. HIC capacity is between 122 and 140 ft.^{3} and, at present fill rates, a HIC will be removed every three to four years for disposal.

The HIC is vented to V-508. This also serves as an overflow; however, due to the fill-head connections to the HIC, it is not intended for this purpose. V-508 has overflow connections to the decon cell and is vented to the plant cold vent header.

The HIC manufacturer provided instrumentation for level indication including closed circuit television for visual observation inside of the HIC. These instruments as well as V-508 instrumentation are located in the AMU. The HIC has both local and CAS/SAS high level alarms. The HIC high level alarm also automatically terminates any flow from P-134. V-508 has level indication both locally and in the CAS/SAS with a high level alarm in the CAS/SAS set at approximately 80% of tank capacity.

All system valves, pumps and filters are located in the AMU or above the steel floor plates in the ETA. The pump and filter are in areas where shielding can be easily installed if required.

B.23.3 LOW ACTIVITY SYSTEM (REFERENCE FIGURE 1-16A)

The low activity system consists of discharges from the laundry, and various sources in the basin area. Laundry, dishwasher, decontamination sink, and decontamination shower drains are collected in the laundry waste collection tank (V-509) which has a capacity of 200 gallons. Transfer pump P-509 is a centrifugal pump which takes suction on V-509 and transfers the water through a bag filter (F-509) to the waste water storage and feed tank (V-503). A suction strainer is installed between the tank (V-509) and pump (P-509) to catch foreign objects which could damage the pump. The tank (V-509), pump, and filter are located in room 142 on 37' elevation. This allows for gravity draining of the various effluents to V-509. Inputs to the system are estimated to be 10,000 gallons per year. Of this total 7,800 is from the washing machine, 1,000 from the dishwasher, 1,000 from the decon sink and 200 from the decon shower.

P-509 controls are automatic with pump start at approximately 50% of tank capacity (100 gallons) and shut off at 30 gallons. Tank level indications are both local and in the CAS/SAS with a high and low level alarm in the CAS/SAS. P-509 is a vertical in-line pump which is identical to several other pumps on site. These pumps can be replaced with a spare in minimal time. The tank, pump, and filter are located within a spill containment area that has a liquid sensor which alarms in the control room in the event of a system leak. High filter differential pressure is also alarmed in the CAS/SAS.

Basin pump room, basin pump room addition, expansion gate, basin intrusion, cask service facility, and cask wash area sumps discharge to and are collected in the basin area waste water collection tank (V-104) which has a capacity of 580 gallons. Transfer pump P-104 is a centrifugal pump which takes suction on V-104 and transfers the water through a filter (F-504) to the waste water storage and feed tank (V-503). Inputs to this portion of the system are estimated to be 7,000 gallons per year. Of this total 3,000 is from the cask wash/decon pad sump, 2,000 from the expansion gate, cladding vault, pump rooms and canyon sumps and 2,000 from miscellaneous sources and special projects.

Alternate flow paths have been provided to discharge the cask wash area sump directly into the filter sludge tank (V-134) for treatment in the high activity system. Discharge to V-134 will be

utilized in the event that the cask wash sump contains particulate which is radiologically unacceptable to the low activity system and can be best disposed of in the HIC.

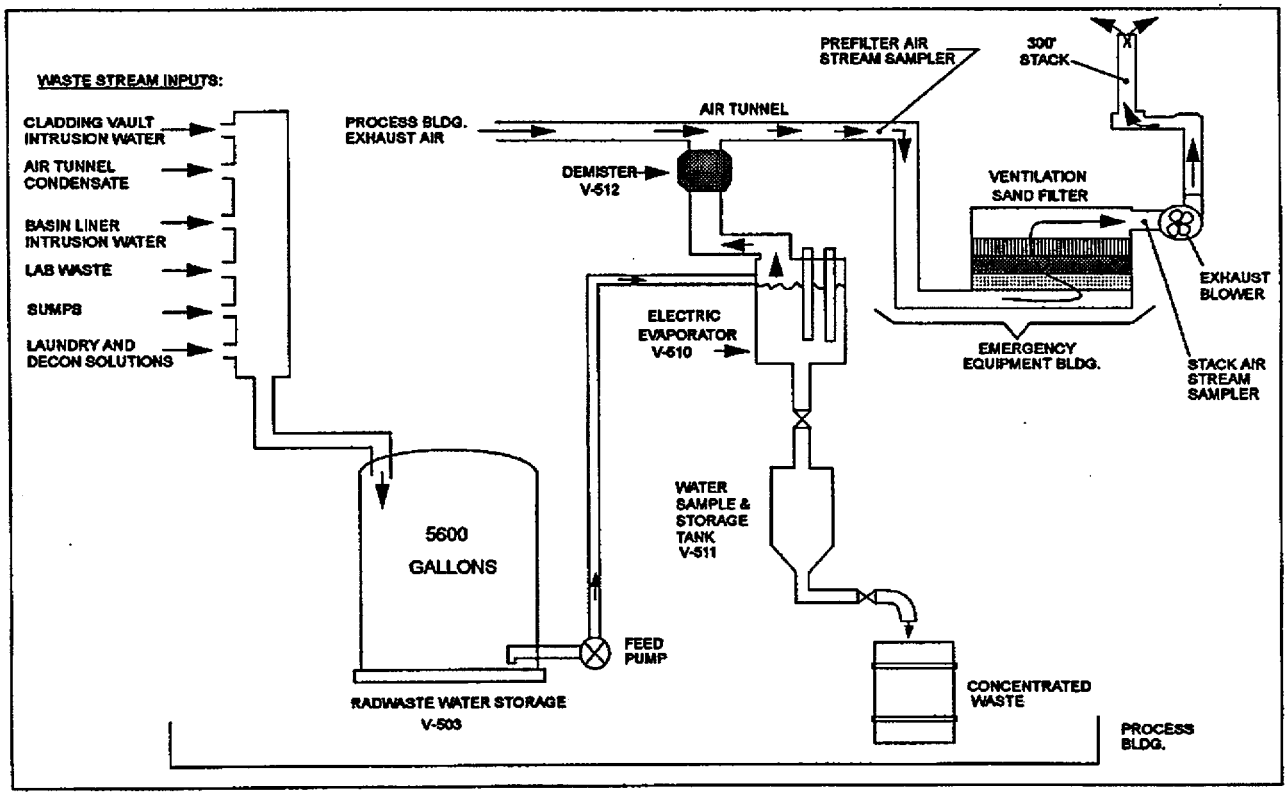

Figure 1-16a. Radwaste System: Low activity radwaste water streams are collected from various sources and piped to the Radwaste Water Storage Tank. Water from this tank is then pumped to an electric evaporator. Evaporator steam is demisted and exhaused via the ventilation system. Evaporator bottoms are put in barrels and shipped off site for processing.

The tank (V-104) and pump (P-104) are located in the basin pump room addition. The tank is vented to the basin filter room which vents to the air tunnel. The various basin area inputs are pumped into a 2 in. drain header which gravity drains to V-104. P-104 pumps the water in V-104 through a filter (F-504) and then to the Radwaste Water Storage Tank V-503. F-504 is a cartridge filter containing six 10 in. cartridges. The filter is located behind a concrete wall which provides shielding for both normal operations and filter replacements.

P-104 controls are manual and require operator action to transfer water to V-503. The manual controls were installed because all the inputs to the system require operator action in the same area. It is anticipated that, unless special work is ongoing, the monthly input to V-104 will be less than 200 gallons.

Level indication for V-104 is by sight glass. The tank level can also be verified by visual examination through the hinged cover on top of the tank. Any leakage of water in this area drains to a sump which has a high level switch that actuates an alarm in the CAS/SAS.

The waste water storage and feed tank (V-503) is in room 133 and is the main storage tank for the low activity system. V-503 has a 5,600 gallon capacity. It is vented and overflows to the canyon decon cell. Level indication is local with a level indicator and both high and low level alarms in the CAS/SAS. This provides for pump protection on the low end and overflow protection on the high end. The total input to V-503 from both sub-systems is estimated to be 20,000 gallons per year.

The evaporator feed pump (P-503) is a centrifugal pump which takes suction on V-503 and supplies water to the evaporator. P-503 operation is controlled by the evaporator level control system. Due to the pump's excessive capacity flow to the evaporator, it is reduced by an orifice installed in the pump discharge piping with bypass back to V-503.

The evaporator is an electric immersion heater unit with a 200 gallon per day capacity and is located on the mezzanine level (elevation 56 ft.) in room 136. The unit holds 70 gallons when full and the water level is controlled by a magnetic float switch. Additional waste water is fed to the evaporator when the level falls about five gallons. This small amount of feed to the unit allows it to return to boiling in approximately five minutes, depending on the concentration at the time.

The evaporator vapor is discharged to the Process Building air tunnel through a moisture separator and demister which drain any liquid back to the evaporator. The vapor passes through the air tunnel sand filter prior to being discharged to the plant ventilation stack. A sight glass is provided in the vapor line to aid in the detection of excessive foaming. The unit has an anti-foam injection system to control foaming. Foaming is also controlled by the use of low foaming detergents in the laundry and for deconning.

Several protective devices and systems are incorporated to automatically shut the evaporator down in the event of an operating problem. These systems include the following:

- High and low level water switches.
- Overflow line to V-511.
- Loss of air tunnel vacuum.
- High temperature.
- High differential pressure across the demister.

Any evaporator shut down will alarm both locally and in the CAS/SAS. Operation of the evaporator is done from a control panel located in a separate area to reduce exposure to personnel.

The evaporator will cycle and continue to concentrate until a 15 to 20 weight % solids is obtained. Liquid solid concentration will be determined by testing. Once this concentration is obtained, the evaporator contents will be drained to the evaporator bottoms tank (V-511). The solution will then be sampled and then drained to a 55-gallon barrel for later disposal.

In the event that the evaporator heaters scale prior to reaching the established concentration the heaters can be removed and cleaned or replaced. As the heaters are removed through the top of the unit this may be performed without draining the evaporator. An alternate method would be to drain the evaporator to the bottoms tank, complete the repairs and then recycle the solution back to the evaporator using pump P-511.

The evaporator bottoms tank (V-511) is located in room 136 on the 48 ft. elevation under the evaporator. This allows for gravity draining of the evaporator through a remotely operated ball valve. V-511 is a 200 gallon tank with an electric blade mixer and immersion heater. The tank is provided with three sample points for analysis of the bottoms in preparation for disposal. The heater and mixer are used to keep the solids in solution. V-511 has a GEMAC LI and density indicator system and overflows to the drum solidification pit. The V-511 high level indication alarms both locally and in the CAS/SAS.

The drum solidification area is a pit located in the bottoms tank room. Normal filling of the drums is accomplished by gravity feed from the bottoms tank (V-511) through system piping and hose connections. Current plans are to ship the wet evaporator bottoms off site for processing. A concrete facility could be added later if cost effective.

Normal radiation levels for various low level system vessels are as follows:

- The Evaporator; 5 to 200 mR at contact.
- The Bottoms Tank V-511 (containing 100 gal. of waste); 10 to 300 mR at contact and 15 mR at 2 ft.
- A 55-gal. waste barrel; to 300 mR at contact, 50 mR at 2 ft.

ESTIMATED VOLUMES OF RADWASTE WATER PER YEAR

*	Water used to dump the basin filter (includes about 1,000 gal cask flush water)	6,000 gal/yr.
*	Decon pad wash water	3,000 gal/yr.
*	Basin exp. gate, Clad vault intrusion, pump room & canyon sumps, etc.	2,000 gal/yr.
*	Laundry, Decon sink and Decon shower (various sources: 7,800 from washing machine, 1,000 from dish washer, 1,000 from decon sink and 200 from decon shower)	10,000 gal/yr.
*	Miscellaneous sources & special projects	2,000 gal/yr.
	Total waste water	**23,000 gal/yr.**

Total volume of waste water that will go to the evaporator.....(23,000 less 6,000) = ... 17,000 gal/yr.