INSTRUCTION SHEET

LICENSE RENEWAL APPLICATION

DEMONSTRATION AND CONDITIONS FOR LICENSE SNM-778

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PART I

LICENSE CONDITIONS

1.0 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

1.1 NAME

Name - McDermott International, Inc.
Babcock & Wilcox
Naval Nuclear Fuel Division
NNFD Research Laboratory

McDermott International Inc. is incorporated under the laws of the Republic of Panama.

Principle Office - 1010 Common Street, New Orleans, Louisiana.

1.2 LOCATION

Address - Babcock & Wilcox
NNFD Research Laboratory
P. O. Box 11165
Lynchburg, Virginia 24506-1165

The NNFD Research Laboratory (site) is located in Campbell County, Virginia, near the James River, approximately four miles East of the city of Lynchburg. Figure 1-1 shows the location of the site with respect to the Commonwealth of Virginia. Figure 1-2 shows the location of the site with respect to a five mile radius. Figure 1-3 shows the location of buildings and facility locations where licensed materials are handled and stored.

1.3 LICENSE NUMBER AND PERIOD

License Number - SNM-778

Docket Number - 70-824

Period of Time - It is requested that this license be renewed for a period of 10 years.

1.4 POSSESSION LIMITS

	Material	Physical Form	Enrichment	Amount
1.	Uranium enriched in U-235	Encapsulated or irradiated	$\geq$ 20 %	3.5 Kg contained U-235
2.	Uranium enriched in U-235	Unencapsulated and unirradiated	$\geq$ 20 %	0.53 Kg contained U-235
3.	Uranium enriched in U-235	Encapsulated or irradiated	5 % to <20%	1.2 Kg contained U-235
4.	Uranium enriched in U-235	Unencapsulated and unirradiated	5 % to <20%	0.5 Kg contained U-235
5.	Uranium enriched in U-235	Encapsulated or irradiated	.711 % to <5%	55 Kg contained U-235
6.	Uranium enriched in U-235	Unencapsulated and unirradiated	.711 % to <5%	11 Kg contained U-235
7.	Plutonium	Unencapsulated and unirradiated		0.05 Kg
8.	Source Material	Any		6000 Kg
9.	Fission Products & Transuranium Elements	Irradiated Fuel		Quantity contained in 4 irradiated fuel assemblies.
10.	Fission Products & Transuranium Elements	Irradiated fuel		5,000,000 Ci.

11. Any byproduct material	Irradiated structural materials & components	50,000 Ci.
12. Byproduct material with at. nos. 3 thru 83	Any	3,000 Ci each total not to exceed 1,000,000 Ci.
13. Transuranium elements	Any	20 milli-Curies each
14. Cf-252	Sealed Sources	4 milligrams
15. Am-241	Sealed Sources	30 Ci
16. H-3	Sealed Sources	100 Ci
17. H-3	Oxide	3 Ci
18. H-3	Ni plated Sc tritide foil	3 Ci

1.5 LOCATION OF POSSESSION AND USE

1.5.1 Licensed material shall be possessed and used at the NNFD Research Laboratory (site).

1.5.2 Byproduct material in the form of sealed sources with activities of up to 500 milliCuries may be possessed and used in locations other than the site for performing instrument calibration, electronic noise analysis, shielding studies, or similar operations.

1.5.3 A restricted zone shall be established on the area north of the Chessie System main line right-of-way with a fence line based on radiation levels not exceeding an exposure dose rate of 500 millirems/year.

1.6 DEFINITIONS

1.6.1 Site means NNFD Research Laboratory.

1.6.2 SRC means Safety Review Committee.

1.6.3 SNM means Special Nuclear Material.

1.6.4 Licensed Material means source, byproduct, or SNM received, possessed, used or transferred under a general or specific license issued by the Nuclear Regulatory Commission.

1.6.5 Research and Development (R&D) means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes, including the experimental production and testing of models, devices, equipment, materials and processes. The administration of licensed material, internally or externally, to human beings is not included in this definition.

1.6.6 Safety Audit Subcommittee (SAS) means the subcommittee established under the SRC to perform audit functions.

1.6.7 Manager, Employee, Community, and Regulatory Relations (Manager, EC&RR) means the position with primary responsibility for the safety of operations at the site.

1.6.8 Authorized User means a person who may work with licensed material unsupervised and may supervise others, not so designated, in the handling of licensed material.

1.6.9 Calibration means a comparison of a measurement standard of known accuracy that is traceable to the NBS with another standard or instrument to detect, correlate or adjust any variation in the accuracy of the item being compared, within the specified range and accuracy of the item. Calibration also includes standardization.

1.6.10 Standardization means, the act of using standards which are traceable to the NBS, a nationally accepted measurement system, or natural phenomena to set up an instrument. Standardization must be performed before and after use.

1.6.11 Unit means (1) a separate laboratory, room, or work area; (2) a transfer cart where SNM is separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center. More than one unit may be on a cart provided the preceding edge-to-edge and center-to-center values are maintained, and (3) a processing bench, glove box, furnace, fume hood, or other similar process equipment or container separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

1.6.12 Standing RWP's are Radiation Work Permits issued for a term not to exceed 6-months, authorizing entry into High Radiation Areas and Airborne Radioactivity Areas to perform routine work.

1.7 AUTHORIZED ACTIVITIES

1.7.1 Licensed material shall be used in the performance of Research and Development (e.g., hot cell examination of irradiated and radioactive components including irradiated fuel; analytical activities for other companies or B&W divisions including laboratory analysis, preparation of and testing of materials and equipment; preparation and modification of radiation sources; and preparation and decontamination of reactor-related hardware for inspecting, evaluating, and measuring reactor components).

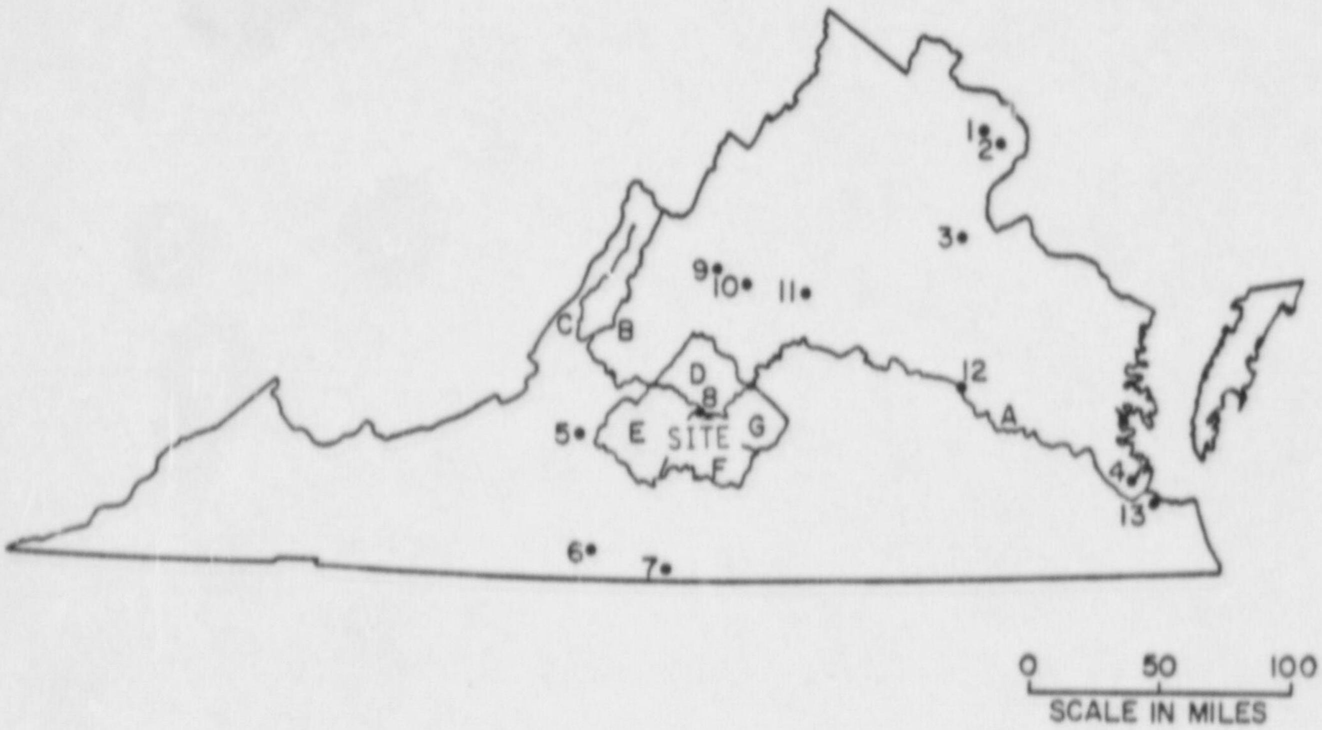
1.7.2 The site may transport and possess licensed material in private carriage between NRC licensed facilities within the United States pursuant to the regulations in 10 CFR 71 and 49 CFR.

1.8 EXEMPTIONS AND SPECIAL AUTHORIZATIONS

1.8.1 The uranium bioassay program sampling frequency shall comply with Tables 2 and 3 of Regulatory Guide 8.11, dated June, 1974, except as follows:

1.8.1.1 When a worker is absent from the site during a period when the bioassay counting service is on site, a special counting shall not be required for those workers for routine exposure control monitoring. The maximum amount of time between in vivo counts shall not exceed 12-months.

FIGURE 1-1



1. ARLINGTON (174, 284)
2. ALEXANDRIA (110, 938)
3. FREDERICKSBURG (14, 450)
4. NEWPORT NEWS (138, 177)
5. ROANOKE (92, 115)
6. MARTINSVILLE (19, 653)
7. DANVILLE (46, 391)
8. LYNCHBURG (54, 083)
9. STAUNTON (25, 504)
10. WAYNESBORO (16, 707)
11. CHARLOTTESVILLE (38, 880)
12. RICHMOND (249, 621)
13. NORFOLK (307, 951)

A. JAMES RIVER
B. COWPASTURE RIVER
C. JACKSON RIVER
D. AMHERST COUNTY (26, 072)
E. BEDFORD COUNTY (26, 728)
F. CAMPBELL COUNTY (43, 319)
G. APPOMATTOX COUNTY (9, 784)

[NUMBERS IN PARENTHESES ( ) ARE 1970 CENSUS DATA]

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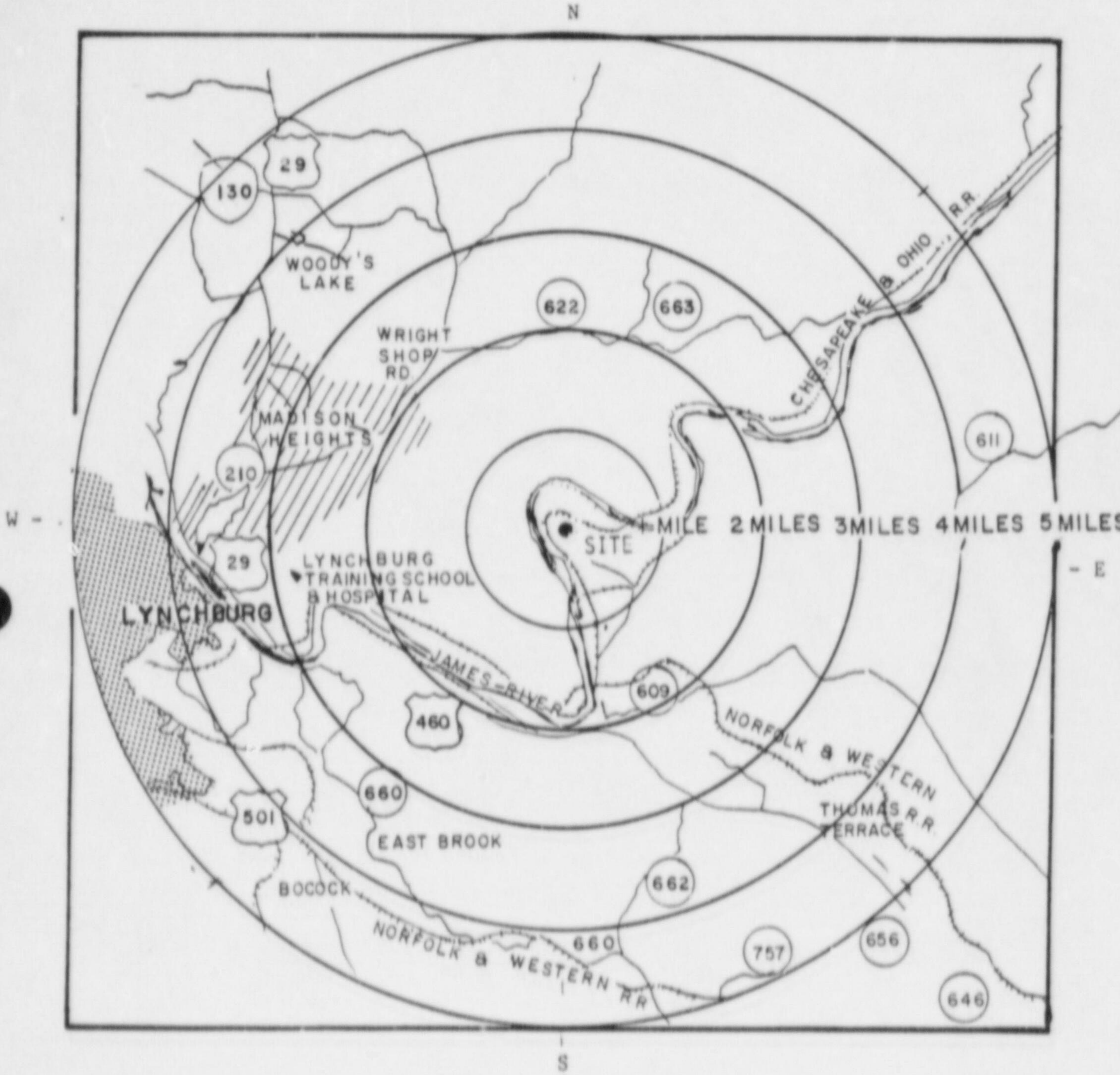
FIGURE 1-2



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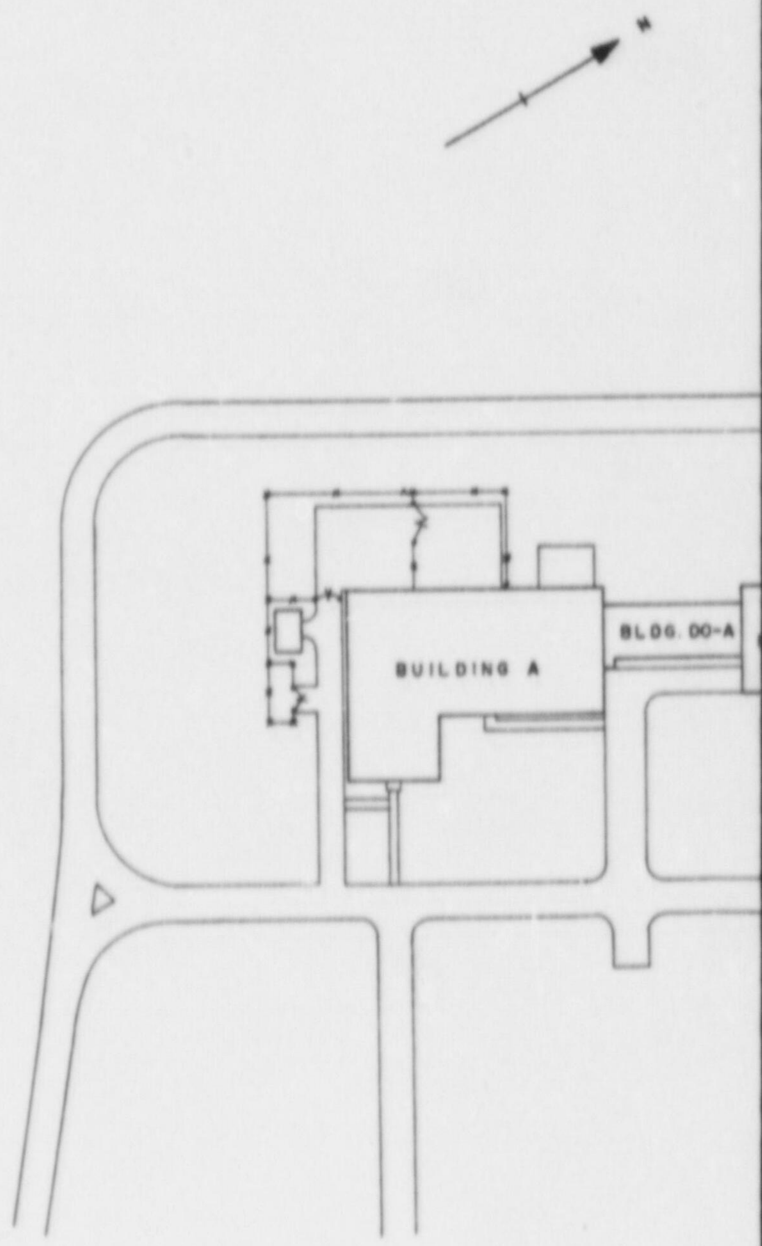
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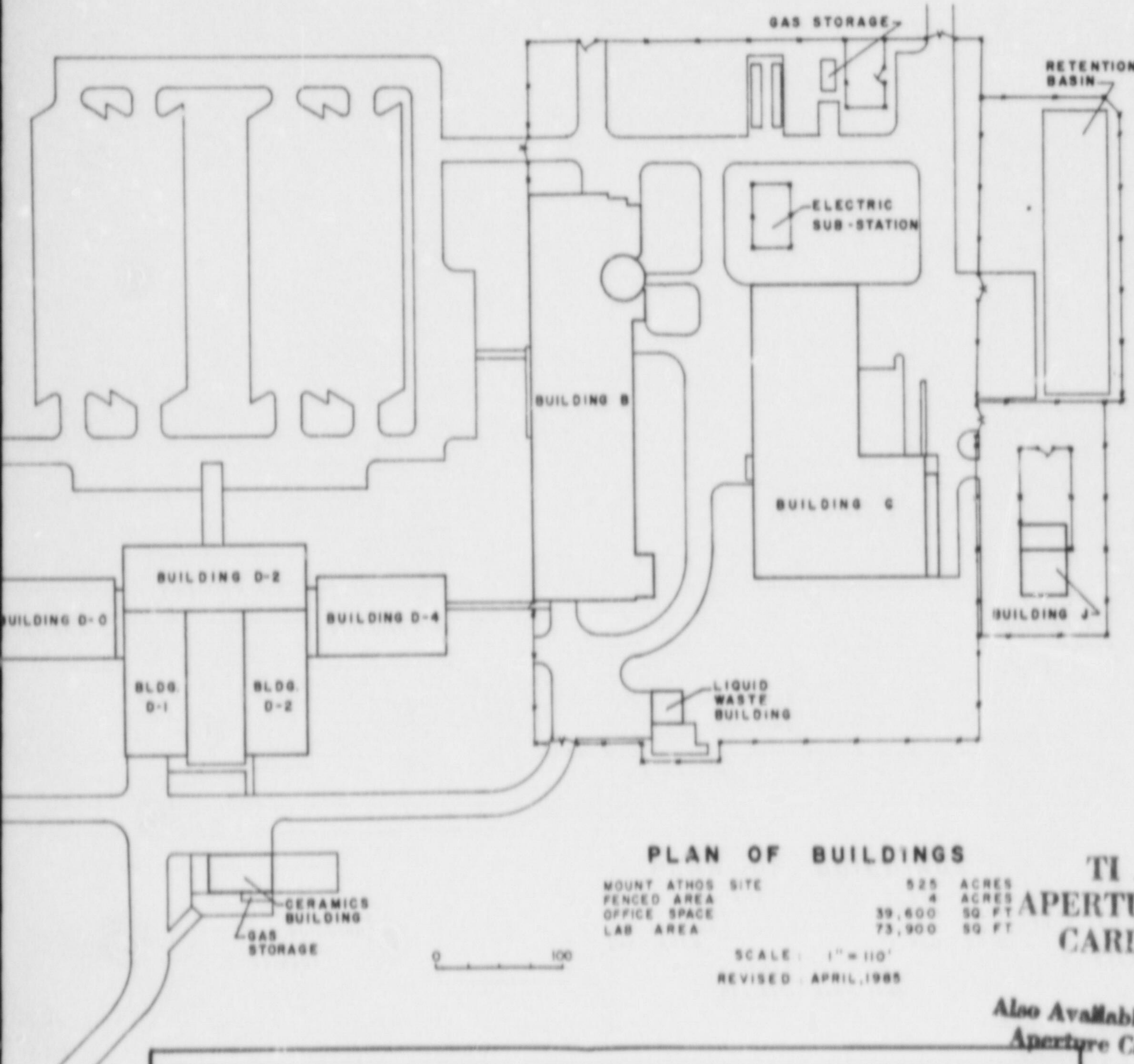
BUILDING A
BLDG. DO-A

FIGURE 1-3



PLAN OF BUILDINGS

MOUNT ATHOS SITE	525	ACRES
FENCED AREA	4	ACRES
OFFICE SPACE	39,600	SQ FT
LAB AREA	73,900	SQ FT

SCALE: 1" = 110'

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TI APERTURE CARD

Also Available On Aperture Card

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Babcock & Wilcox
a McDermott company

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2.0 GENERAL ORGANIZATIONAL AND ADMINISTRATIVE REQUIREMENTS

2.1 POLICY

It shall be the policy of the site to maintain radiation exposures to employees and the general public as low as is reasonably achievable. The facility procedures to ensure the safe handling of licensed material are the Area Operating Procedures.

2.2 ORGANIZATION RESPONSIBILITIES AND AUTHORITIES

2.2.1 Manager, EC&RR - The Manager, EC&RR is ultimately responsible for all safety at the site.

2.2.2 Facility Supervisor - The Facility Supervisor is responsible to the Manager, EC&RR for the safe conduct of all operations at the site and for ensuring that all applicable operations are conducted in compliance with the license and applicable regulations. To fulfill these responsibilities the Facility Super- visor shall have the authority to stop any operation that he feels is unsafe or in violation of license. The Facility Supervisor shall review all new Area Operating Procedures and RWP's, and changes there to, for license and regulatory compliance and for facility safety; and he shall have approval authority for them. He shall submit items for review to the SRC. He shall have approval authority for Area Supervisors.

2.2.3 Area Supervisors - Area Supervisors are recommended by their division management and their appointment shall be jointly approved by the Supervisor, Health and Safety, and the Facility Supervisor. They shall functionally report to the Facility Supervisor. They shall be responsible for the safety and compliance of all operations in their assigned areas. They shall be responsible for the maintaining the exposures of personnel assigned to their area below 300 millirem/week; 1250 millirems/quarter. They shall have approval authority for Radiation Work Permits that apply to their assigned areas. They shall keep the Facility Supervisor advised of all plans for projects and work to be carried out in their areas.

2.2.4 Manager, Safety and Licensing - The Manager, Safety and Licensing reports to the Manager, EC&RR. The Supervisor, Health and Safety, the Accountability Specialist, and the License Administrator report to this manager.

2.2.5 Supervisor, Health and Safety - The Supervisor, Health and Safety is responsible for providing adequate facilities, procedures, and properly trained personnel to implement the Health Physics and Industrial Safety Programs. He is responsible for health physics and industrial safety activities. The Supervisor, Health and Safety reports to the Manager, Safety and Licensing. The Supervisor, Health and Safety has the authority to stop any operation that he believes is contrary to accepted safety practices or license requirements. He shall review all new Area Operating Procedures, Radiation Work Permits and changes thereto, for the radiation safety aspects of the procedure, RWP, or change, and he shall have approval authority for them. He shall conduct training programs for new employees and Authorized Users of Radioactive Material. He shall be responsible for the shipment of licensed material. The Supervisor, Health and Safety shall be a member of the Safety Review Committee but shall not be a member of the Safety Audit Subcommittee. He shall have approval authority for Area Supervisors.

2.2.6 Senior Health Physics Engineer - The Senior Health Physics Engineer shall administer activities of the Health Physics Staff. He shall report to the Supervisor, Health and Safety.

2.2.7 Industrial Safety Officer - The Industrial Safety Officer shall administer the industrial safety program. He shall report to the Supervisor, Health and Safety.

2.2.8 Nuclear Criticality Safety Officer - The Nuclear Criticality Safety Officer shall be responsible for ensuring that no operation at the site results in the inadvertent assembly of a critical mass. He shall review all new Area Operating Procedures and changes thereto, for nuclear criticality safety and shall have approval authority for them. He shall conduct training programs in criticality safety and perform criticality safety calculations. He shall report to the Manager, Safety and Licensing.

2.2.9 License Administrator - The License Administrator shall be responsible for administering the license. He is the primary liaison with the NRC and other federal, state, and local agencies in matters that pertain to nuclear activities. He shall be the coordinator of the SRC and the Safety Audit Subcommittee and shall represent management on both. He shall maintain the permanent records of the SRC and shall be responsible for assuring that appropriate action is taken to correct SAS audit findings that are approved by the Manager, EC&RR. He shall report to the Manager, Safety and Licensing.

2.2.10 Accountability Specialist - The Accountability Specialist shall be responsible for the maintenance and retention of SNM accountability records. The Accountability Specialist shall report to the Manager, Safety and Licensing.

2.3 SAFETY REVIEW COMMITTEE

2.3.1 Function

2.3.1.1 The SRC shall review and approve all new Area Operating Procedures, and shall concur with all changes made to them in the time interval since their last regular meeting.

2.3.1.2 The SRC shall review and approve new projects and major changes to existing projects that utilize licensed materials.

2.3.1.3 The SRC shall review the annual report prepared by the Supervisor, Health and Safety.

2.3.1.4 The SRC shall provide general consulting services in the field of radiation protection and the safe handling of licensed material.

2.3.1.5 The SRC shall review all SAS audit findings, all overexposures and unusual occurrences which must be reported to the NRC. These reviews shall be conducted during the next regularly scheduled meeting following the event and the results of the review shall be documented in the minutes.

2.3.1.6 The SRC Coordinator shall be responsible for resolving comments and recommendations made by the SRC.

2.3.2 Frequency of Meetings

2.3.2.1 The SRC shall meet at least four times annually for the purposes of conducting its business as specified in Section 2.3.1.

2.3.3 Safety Audit Subcommittee

2.3.3.1 The SAS shall perform audits for the Safety Review Committee.

2.3.3.2 The SAS shall audit facilities, procedures, records, and operations for compliance with written requirements and the exercise of acceptable safety practices.

2.3.3.3 The SAS shall perform at least three audits annually, distributed over a 12-month period. Audits shall be made in accordance with written guidance to assure all aspect of 2.3.3.2 are audited.

2.3.3.4 SAS membership shall be appointed by the Manager, EC&RR.

2.3.4 Reporting

2.3.4.1 The SRC shall report to the Manager, EC&RR.

2.3.4.2 The SAS shall report to the Chairman, SRC.

2.3.5 Recordkeeping

2.3.5.1 Minutes of the SRC proceedings shall be prepared by the Chairman, SRC.

2.3.5.2 SRC Minutes shall be forwarded to the Manager, EC&RR by the Chairman, SRC.

2.3.5.3 The permanent records of the SRC shall be kept by the SRC Coordinator.

2.3.5.4 SAS audit reports shall be prepared by the Chairman, SAS.

2.3.5.5 SAS audit reports shall be forwarded to the Chairman, SRC by the Chairman, SAS.

2.3.5.6 SAS audit reports shall be forwarded to the Manager, EC&RR by the Chairman, SRC with comments, as he deems appropriate.

2.4 APPROVAL AUTHORITY FOR PERSONNEL SELECTION

2.4.1 The Manager, EC&RR shall approve the personnel selected for safety-related positions specified in Section 2.2 of this Part and shall appoint the members of the Safety Review Committee in writing. The NNFD shall appoint the Manager, EC&RR.

2.5 PERSONNEL EDUCATION AND EXPERIENCE REQUIREMENTS

2.5.1 Manager, EC&RR - The Manager, EC&RR shall be appointed in accordance with Company policy.

2.5.2 Facility Supervisor - The Facility Supervisor shall have a degree in his related work and three years experience in the use and handling of licensed material, or five years experience in the use and handling of licensed material. He must demonstrate to management proficiency in the application of good principles of radiation protection, industrial safety, and nuclear safety as related to the activities at the site.

2.5.3 Area Supervisors - Area Supervisors shall be Authorized Users and shall have demonstrated sufficient knowledge and experience in the equipment and techniques employed in projects performed in their assigned areas to ensure that all operations are conducted safely and in full compliance with applicable license conditions and area operating procedures.

2.5.4 Manager, Safety and Licensing - The Manager, Safety and Licensing shall have a BS degree in a technical field and five years experience in the nuclear field.

2.5.5 Supervisor, Health and Safety - The Supervisor, Health and Safety shall have a BS degree in a technical field and professional experience in assignments involving radiation protection at the supervisory level. He must have four years experience and demonstrate proficiency in the application of radiation safety principles and be knowledgeable in fields related to radiation protection.

2.5.6 Senior Health Physics Engineer - The Senior Health Physics Engineer shall have a BS degree which shall include at least 20 quarter hours health physics related course work, and two years of radiation control related experience or an MS degree and one year of radiation protection experience.

2.5.7 Industrial Safety Officer - The Industrial Safety Officer shall have at least one year's experience in radiation and industrial safety. He shall be familiar with the codes and requirements of the Occupational Health and Safety Act of 1970 and the National Fire Protection Association.

2.5.8 Nuclear Criticality Safety Officer - The Nuclear Criticality Safety Officer shall have a BS degree in science or engineering. He must have either two years experience with nuclear criticality safety calculations similar to those associated with site activities or he must have one year's experience with nuclear criticality safety calculations similar to those associated with site activities if he has at least an additional two years' experience in nuclear reactor physics calculations.

2.5.9 Accountability Specialist - The Accountability Specialist shall have at least a high school education and three years' experience in the use of licensed material. He must demonstrate to Company management his knowledge of the principles necessary for the accountability and safeguarding of special nuclear materials.

2.5.10 License Administrator - The License Administrator shall have a BS degree in science or engineering and three years experience in nuclear technology or an AS degree in science or nuclear technology with five years experience in nuclear technology.

2.5.11 Safety Review Committee - The SRC membership, as a body, shall have expertise in chemistry, nuclear physics, health physics, and the safe handling of radioactive material. The SRC membership shall have a general understanding of nuclear criticality safety as it pertains to site operation.

2.6 TRAINING

2.6.1 Program 1 - This course is presented to site workers and non-site workers who will be granted access to the restricted area but who will not be granted unescorted access to the controlled areas. The course provides an introduction to radiation and radioactivity (understandable to a non-technical person) and a thorough coverage of safety rules and procedures, including the site emergency procedures. The Supervisor, Health and Safety may modify the course content for those individuals knowledgeable in the basics of radiation and radioactivity. However, safety rules, procedures, and emergency procedures that apply at the site shall be covered.

2.6.2 Program 2 - This course is presented to site workers and non-site workers who will be granted unescorted access to the restricted area and controlled areas but who will not be permitted to work with radioactive materials without supervision. The Supervisor, Health and Safety may modify the course content for those individuals knowledgeable in the basics of radiation and radioactivity. However, safety rules, procedures, and emergency procedures that apply at the site shall be covered. The effectiveness of the training shall be determined by a written examination.

2.6.3 Program 3 - This course shall be presented to site workers and non-site workers who will be granted unescorted access to the restricted area and controlled areas and will be permitted to work with radioactive materials and supervise such work. This course shall meet the requirements for designating a worker as an Authorized User. The Supervisor, Health and Safety may modify the course content for those individuals knowledgeable in the basics of radiation and radioactivity. However, safety rules, procedures and emergency procedures that apply at the site shall be covered. The effectiveness of the training shall be determioned by a written examination.

2.6.4 Retraining - Persons who are designated as Authorized Users shall be retrained annually. Satisfactory completion of the retraining shall be determined by passing a written examination.

2.6.5 Respiratory Protection Training - Training in respiratory protection techniques and equipment shall be required of all workers before the use of such equipment will be permitted. Satisfactory completion of this training shall be determined by passing a written examination.

2.6.6 Respiratory Protection Retraining - Retraining in respiratory protection shall be performed at two year intervals. Satisfactory completion of this retraining shall be determined by passing a written examination.

2.6.7 The training specified in Section 2.6 shall be administered by the Supervisor, Health and Safety, or his designated and qualified alternate.

2.6.8 Nuclear Criticality Safety Training - Nuclear Criticality Safety training provided as a part of the programs specified in Sections 2.6.2, 2.6.3 and 2.6.4, shall be performed by the Nuclear Criticality Safety Officer or his designated alternate. The designated alternate must meet the same minimum qualifications as those specified for the Nuclear Criticality Safety Officer (2.5.8).

2.7 OPERATING PROCEDURES

2.7.1 Area Operating Procedures

2.7.1.1 All operations with licensed material shall be conducted in accordance with Area Operating Procedures or Radiation Work Permits (see 3.1.1).

2.7.1.2 Area Operating Procedures (AOP) - Area Operating Procedures shall be established for all routine operations in which SNM, source and byproduct materials are stored or handled. AOP's shall include those nuclear criticality and radiation safety controls and limits that apply to the operation. Each AOP shall be approved by the Nuclear Criticality Safety Officer or his designated alternate, the Supervisor, Health and Safety or his designated alternate, the Facility Supervisor or his designated alternate, and the Safety Review Committee.

2.7.1.3 AOP's may be revised with the approval of the Nuclear Criticality Safety Officer or his designated alternate, the Supervisor, Health and Safety or his designated alternate, and the Facility Supervisor or his designated alternate. The revised procedure may be used with these approvals until the next scheduled regular meeting of the Safety Review Committee when the revision must be approved by the SRC.

2.7.1.4 AOP's shall be available in each operations area where they apply and shall be followed by site personnel.

2.7.1.5 Distribution of new and revised procedures shall be made in accordance with a document control system which assures that the procedure manuals contain only the most current revision of the procedures.

2.7.1.6 AOP manuals shall be reviewed annually by the Facility Supervisor to assure that the manuals contain the most current revision of the procedures.

2.7.2 Technical Procedures

2.7.2.1 Technical procedures shall be established, reviewed, approved, and followed for Health and Safety or Nuclear Criticality Safety. They shall be reviewed and approved by the Senior Health Physics Engineer or the Nuclear Criticality Safety Officer, respectively, or their designated alternate. The designated alternate for a Senior Health Physics Engineer must meet the minimum qualifications specified in Sections 2.5.6. The designated alternate for the Nuclear Criticality Safety Officer must meet the same minimum qualifications specified in Section 2.5.8. Approval signatures shall appear on the procedure.

2.8 INTERNAL AUDITS AND INSPECTIONS

2.8.1 Nuclear Criticality Safety

2.8.1.1 The Nuclear Criticality Safety Officer or his designated alternate shall conduct internal audits for the purpose of evaluating the nuclear criticality safety aspects of operations. This audit shall be conducted in accordance with written audit guidance. This audit shall be conducted once each calendar quarter. A report of his findings shall be made to the Manager, EC&RR within two weeks of completing the audit. The audit reports shall be forwarded to the Facility Supervisor and the License Administrator. The License Administrator shall be responsible for assuring that the appropriate corrective actions are taken to address the audit findings.

2.8.1.2 The Facility Supervisor shall perform an inspection weekly for compliance with the nuclear criticality safety aspects of the operations. Findings resulting from these inspections shall be reported to the Nuclear Criticality Safety Officer.

2.8.2 Health Physics

2.8.2.1 The Supervisor, Health and Safety or his designated alternate shall conduct internal audits for the purpose of evaluating the health physics aspects of operations. This audit shall be conducted in accordance with written audit guidance. This audit shall be conducted once each month. A report of his findings shall be made to the Manager, EC&RR within two weeks of completing the audit. The audit reports shall be forwarded to the Manager, EC&RR and the License Administrator. The License Administrator shall be responsible for assuring the appropriate corrective actions are taken to address the audit findings.

2.8.3 General Safety and Compliance

2.8.3.1 The SAS performs audits of general safety and compliance. These audits shall be conducted three times annually. The audits shall be distributed over a 12-month period. The SAS shall include an audit of the Health and Safety Group at least once annually. This annual audit shall be performed by a qualified individual who is independent of the Health and Safety Group. Other areas shall be audited for compliance with written requirements and the exercise of acceptable safety practices. Audits shall be made in accordance with written guidance to assure all aspects of Section 2.3.3.2 are audited. The Chairman, SAS shall file a report of the audit findings with the Chairman, SRC, with a copy to the License Administrator and the Facility Supervisor and members of the SRC. The Chairman, SRC shall forward the report to the Manager, EC&RR with comments, as he deems appropriate. The License Administrator shall be responsible for assuring that the appropriate corrective actions are taken to address the audit findings.

2.9 INVESTIGATIONS AND REPORTING OF OFF-NORMAL OCCURRENCES

2.9.1 License Administrator

The License Administrator shall investigate and report, when required, the following types of off-normal occurrences:

2.9.1.1 Excessive levels of radiation from or contamination on packages upon receipt.

2.9.1.2 Thefts, attempted thefts, or losses of licensed material, other than normal operating losses.

2.9.1.3 Incidents as specified in 10 CFR 20.403

2.9.1.4 Overexposure of individuals and excessive levels and concentrations of radioactivity.

2.9.1.5 Failures to comply and defects pursuant to 10 CFR 21.

2.9.1.6 Changes to security, safeguards, or emergency plans made without prior NRC approval, when prior approval is required.

2.9.1.7 Failures to comply with license requirements.

2.9.1.8 Unapproved storage or use of licensed material.

2.9.2 Supervisor, Health and Safety

The Supervisor, Health and Safety shall perform investigations and issue reports of the following:

2.9.2.1 Higher than expected personnel exposures.

2.9.2.2 Higher than expected concentration of airborne activity in the facility.

2.9.2.3 Unauthorized entry into a High Radiation or Airborne Radioactive Material area.

2.9.2.4 Failure of equipment or instrumentation to meet Health and Safety requirements.

2.9.3 Facility Supervisor

The Facility Supervisor shall perform investigations of the following:

2.9.3.1 Any violation of nuclear criticality safety criteria.

2.9.3.2 Any violation of Area Operating Procedures or RWP's.

2.10 RECORDS

The following positions or organizations shall be responsible for maintaining the indicated records, for the period specified. Records may be kept in original form, microfilm or in computer storage. The symbol (*) indicates that the record will be retained until the NRC authorizes its disposition.

2.10.1 Health and Safety Group

Health and Safety Supervisor audits	2 years
Shipping and receiving RM forms	5 years
Waste disposal records	(*)
Personnel dosimetry records	(*)
Results of Bioassays and Whole Body Counting	(*)
Releases to the environment	(*)
Radiation survey data	2 years
Contamination survey data	2 years
Radiation Work Permits (completed)	5 years
Radiation detection instrument calibration	2 years
Leak tests of sealed sources	2 years
Personnel training	(*)
Personnel retraining	(*)
Airborne radioactivity sampling data	(*)
NRC-4 forms	(*)
NRC-5 forms	(*)

2.10.2 Nuclear Criticality Safety Officer

Nuclear criticality safety evaluations and calculations	6 months after termination of the approved process.
Nuclear Criticality Safety Officer Audit Reports.	2 years

2.10.3 License Administrator

Safety Review Committee Minutes	(*)
Safety Audit Subcommittee Audit Reports	2 years
Investigation reports of off-normal occurrences	2 years

2.10.4 Emergency Records

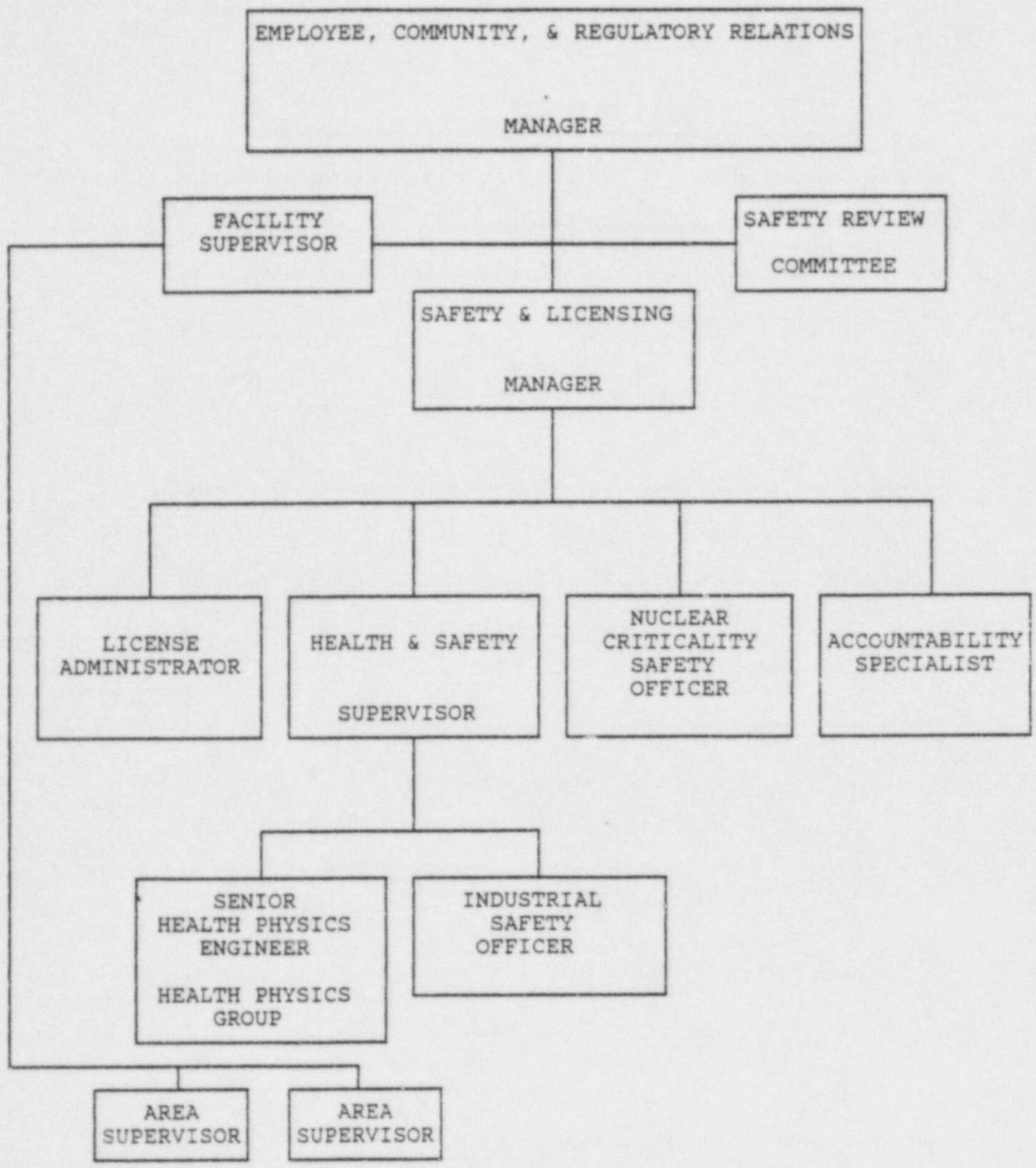
Records pertaining to emergency response and preparedness shall be retained in accordance with Radiological Contingency Plan, Section 8.0.

FIGURE 2-1

SITE ORGANIZATION

NNFD RESEARCH LABORATORY

ORGANIZATION



3.0 RADIATION PROTECTION

3.1 SPECIAL ADMINISTRATIVE REQUIREMENTS

3.1.1 Radiation Work Permits (RWP)

3.1.1.1 RWP's shall be issued whenever the activity is not covered by an Area Operating Procedure and workers are likely to be exposed to levels of radiation or concentrations of radioactive material in excess of those specified in 10 CFR 20.101 & 20.103.

3.1.1.2 RWP's shall be approved by the Area Supervisor, Health Physics Supervisor, and the Facility Supervisor. In the absence of any of the above persons, a designated and qualified alternate may approve RWP's.

3.1.1.3 The RWP shall specify the radiological protection requirements for the operation and specify levels of worker exposure above which a documented ALARA evaluation shall be performed. RWP's that require a documented ALARA evaluation must, in addition to 3.1.1.2, be approved by the Manager, EC&RR.

3.1.1.4 RWP's shall be approved at a meeting of all the signators of the form.

3.1.1.5 The RWP form shall provide space for entering the estimated exposures to the whole body, extremities, and for the job. These are used to identify the areas of exposure concern and do not constitute an exposure goal or limit.

3.1.1.6 The RWP form shall provide space for the Area Supervisor to sign or initial, attesting that the workers have been instructed in the requirements of the RWP.

3.1.1.7 The term of a RWP shall not exceed 30-days, except that Standing RWP's shall have a term not to exceed 6-months.

3.1.2 ALARA Policy

The management of the site is committed to a policy of maintaining exposures as low as is reasonably achievable.

3.1.2.1 Site workers shall be introduced to this policy during their initial training and shall be reinforced during the annual retraining of Authorized Users.

3.1.2.2 The ALARA policy shall be implemented through the Area Operating Procedures and Radiation Work Permits.

3.1.2.3 The ALARA policy shall be enforced by the Facility Supervisor and the Supervisor, Health and Safety in the exercise of their review and approval authority, their authority to terminate operations, and audits.

3.1.2.4 The SRC shall evaluate ALARA performance in exercising their review authority over procedures and proposed new projects and their review of the annual report from the Supervisor, Health and Safety.

3.1.3 Off-Site Possession

Off-site possession and use of licensed material shall be the responsibility of and under the control of site workers specifically approved by the Safety Review Committee.

3.2 TECHNICAL REQUIREMENTS

3.2.1 Access Control

3.2.1.1 High Radiation Areas - Entry into a High Radiation Area or an Airborne Radioactivity Area shall be controlled by an RWP.

3.2.1.2 Contamination Areas - Areas which are determined by the Health and Safety Group to present a risk of spreading radioactive contamination into non-contaminated areas shall be clearly marked at each entrance. Step-off pads shall be provided. Personnel survey instrumentation shall be provided at the step-off pad. The minimum protective clothing required in Contamination Areas shall be shoe covers and lab coats. Exiting such areas shall require personnel to remove their protective clothing and survey themselves with the instrumentation provided. Persons found to be contaminated above background levels must receive the approval of the Supervisor, Health and Safety, prior to leaving the contaminated area.

3.2.2 Ventilation Requirements

3.2.2.1 Air flows within Building B shall be in the direction of highest potential for airborne radioactive material. Air flow directions shall be checked monthly.

3.2.2.2 Potentially contaminated exhaust air from hood, hot cells, and glove boxes shall be discharged through the fifty meter high stack, except as noted in 3.2.2.7.

3.2.2.3 The exhaust stack shall be sampled isokinetically.

3.2.2.4 The stack sampling and monitoring system shall operate continuously except for periods when repair or calibration is required.

3.2.2.5 The following table presents the release limits and action levels associated with the exhaust stack. The Health and Safety Group shall be responsible for responding to releases in excess of these action levels. An operation that results in action levels being exceeded for 4-consecutive time periods, shall be shutdown until the cause is corrected.

STACK RELEASE LIMITS AND ACTION LEVELS

Release Product	Release Limit	Action Level
Beta Particulate	2 mCi/yr	200 uCi/week
Alpha Particulate (long lived)	20 uCi/yr	1 uCi/2 weeks
Kr-85	2500 Ci/yr	70 Ci/week
H-3	130 Ci/yr	3 Ci/week
I-131	6 mCi/yr or 300 uCi/week	200 uCi/week

3.2.2.6 Exhaust systems that cannot be practicably discharged through the 50-meter stack, and where there exists a reasonable probability that the discharges to the atmosphere could exceed 10% of the applicable MPC for an unrestricted area, shall be monitored for gaseous and particulate activity in the effluent.

3.2.2.7 Exhaust air from areas in which there is no airborne radioactive material may be exhausted directly to the roof either with or without continuous sampling, if approved by the Safety Review Committee.

3.2.2.8 Areas equipped with an air monitor may be exhausted to the roof through HEPA filters if the concentration of airborne radioactive material is below the appropriate MPC for an unrestricted area, if approved by the Safety Review Committee.

3.2.2.9 All hoods used for the handling of licensed material shall exhaust through one HEPA filter, except for hoods that are specifically designed and installed for use with perchloric acid.

3.2.2.10 Fume hoods utilized for the handling of unirradiated Pu shall be provided with two HEPA filters in series.

3.2.2.11 Hot cells shall be provided with two stages of HEPA filters.

3.2.2.12 Final HEPA filters which service facilities where licensed material is handled shall be tested, using the cold DOP test, annually or after a final HEPA filter is changed, whichever comes sooner.

3.2.2.13 The acceptance criteria for the testing of final HEPA filters (3.2.2.12) shall be 99.95% of all particles having a light-scattering mean diameter of approximately 0.7 micrometers.

3.2.3 Instrumentation

3.2.3.1 Portable Survey Instruments

3.2.3.1.1 Portable instruments - A relatively large and diverse inventory of portable survey instruments is maintained. These instruments vary in range and sensitivity. The below listing is a representative sampling of the instruments on hand:

Instrument	Sensitivity	Characteristics	Type Radiation
Ionization Chamber	0 - 20K R/hr	6.5Kev - 1.2Mev	Beta & Gamma
Geiger Counter	0 - 1K R/hr	23Kev - 1.2Mev	Beta & Gamma
Proportional Counter (gas flow)	25 - 500K cpm		Alpha & Beta

Scintillation Detector	0 - 50K cpm		Alpha
Geiger Counter	0 - 50K cpm	>40Kev	Beta
Scintillation Detector	0 - 5 mR/hr		Gamma
Neutron Dose	0 - 5K mR/hr	25Kev - 3Mev	Neutron

3.2.3.1.2 Portable survey instruments shall be calibrated semiannually.

3.2.3.2 Air Monitors

3.2.3.2.1 Nuclear Measurements Corp. (NMC) Model AM-2A - This instrument utilizes a gas flow proportional detector with a 1.0 mg/cm^2 thick end window. These instruments are operated as alpha or beta-gamma monitors. They utilize a fixed filter with a nominal air flow of 2.5 to 3 ft^3/min. The alarm setting is set at less than 40 MPC hours above normal background including Radon and Thoron daughters.

3.2.3.2.2 Eberline Model AIM-3S - These monitors are used for alpha monitoring only. They are typically located in areas where Pu or U is being processed. They use a ZnS(Ag) scintillation detector with a fixed filter. The monitor air flow is nominally 20 ft^3/hr. The alarm is set at less than 40 MPC hours above the normal background for Radon and Thoron daughters.

3.2.3.3 Air Samplers

3.2.3.3.1 Mine Safety Appliance (MSA) Model G - These personal air samplers utilize a Millipore field sample cassette. The nominal air flow rate is 2 liters/min. Samples are collected for counting on a low background counter with sufficient sensitivity to detect 25% of the applicable MPC for 8 hour sampling intervals.

3.2.3.3.2 Fixed samplers are located at work stations where the concentration of airborne radioactive material potentially exceeds 25% of the applicable MPC.

3.2.3.4 Criticality Monitors

3.2.3.4.1 Nuclear Measurements Corp. (NMC) Model GA-2TO and GA-2A - These monitors are designed as criticality alarm systems. Detection is by a NaI (Tl) detector operated in the constant current mode. Response is logarithmic and non-saturating. Emergency power is provided. The nominal alarm setpoint is 20 mR/hr. Failure alarm function is provided. Criticality monitors shall be calibrated semiannually.

3.2.3.5 Counting Equipment

3.2.3.5.1 Sharp Low Beta - Air samples and effluent samples may be counted on this instrument. This instrument utilizes a 4.5-inch and a 2.5-inch very thin end window proportional detector. Backgrounds and counter response are tested weekly and the instrument is calibrated annually.

3.2.3.5.2 Beckman Wide Beta - Air samples and effluent samples may be counted on this instrument. It utilizes two 2.5-inch very thin end window proportional detectors. Backgrounds and counter response are tested weekly and the system is calibrated annually. The manual detector is used infrequently and it is tested when used.

3.2.4 Internal and External Exposure

3.2.4.1 Ventilation

3.2.4.1.1 The minimum air velocity across the opening of fume hoods that are used to handle licensed material shall be at least 100 fpm. Hood face velocities shall be measured monthly. Those hoods that do not meet the minimum requirement shall be placed out of service.

3.2.4.1.2 The maximum differential pressure across HEPA filters shall be limited to 4-inches of water, except the hot cell filters which shall be limited to 5-inches of water. HEPA filters shall be changed to prevent exceeding these limits. The differential pressure across HEPA filters shall be checked weekly.

3.2.4.1.3 The minimum differential pressure across the hot cell face shall be 0.25-inches of water. The differential pressure across the hot cell face shall be checked weekly. An additional hot cell fan will be automatically or manually started when the differential pressure reaches 0.25-inches of water.

3.2.4.2 Air Sampling and Analysis

3.2.4.2.1 Continuous air sampling shall be performed in all areas where, in the judgment of the Supervisor, Health and Safety, there exists the potential for exposing personnel to concentrations of airborne radioactive materials in excess of 10% of the applicable MPC.

3.2.4.2.2 Air sampling filters shall be changed daily in areas where sample evaluations indicate concentrations of airborne radioactive materials in excess of 10% of the applicable MPC.

3.2.4.2.3 Air sampling filters shall be changed weekly in areas where sample evaluations indicate concentrations of airborne radioactive materials less than or equal to 10% of the applicable MPC.

3.2.4.2.4 Preliminary evaluation of air sample filters shall be performed within the next working day following their removal. Final evaluation of air sampling filters shall be performed within 7 working days following their removal.

3.2.4.2.5 An investigation by a Senior Health Physics Engineer shall be performed into the cause of unexpected air sampling results that indicate airborne activity at levels between 10% and 25% of the applicable MPC. The Senior Health Physics Engineer shall assign responsibility for completion of any actions that may be indicated by the investigation.

3.2.4.2.6 An investigation by the Supervisor, Health and Safety shall be performed into the cause of unexpected air sampling results that indicate airborne activity at levels exceeding 25% of the applicable MPC. The Supervisor, Health and Safety shall be responsible for specifying corrective actions and assuring that the specified actions are taken.

3.2.4.2.7 If fixed air samplers are used to determine concentrations of airborne radioactivity in the worker's breathing zone, the representativeness of the samplers shall be determined at least once every 12-montns.

3.2.4.3 Bioassay

3.2.4.3.1 Uranium Bioassay Program

1. The uranium bioassay program sampling frequency shall comply with Regulatory Guide 8.11, June, 1974, except as specified in section 1.8 of this application.

2. All workers who routinely work in uranium handling areas shall be subject to the uranium bioassay program. The following are the action criteria for the routine uranium bioassay program:

Analysis	Action Level	Action to be Taken
a. Urinalysis	< 9 ug/l	None
b. Urinalysis	9-16 ug/l	1. Determine if area surveys support the analysis results. 2. If #1 is positive, investigate and correct as needed. 3. Make sure individual is in-vivo counted during the next time that the body counting service is at the B&W site.
c. Urinalysis	> 16 ug/l	1. Restrict the worker from further exposure. Resample the individual within 5 working days. 2. Determine if area surveys support the analysis results. 3. If #2 is positive, investigate the cause and correct as needed. 4. If exposure is confirmed by #2, investigate to determine how exposure was incurred and correct it. If the exposure exceeds 50% of the maximum permissible annual

		dose, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of the restriction.
d. In vivo	< 30 ug U-235	None
e. In vivo	30-120 ug U-235	1. Determine if area surveys support the analysis results. 2. If #1 is positive, investigate and correct as needed.
f. In vivo	> 120 ug U-235	1. Resample the individual within 10 working days. 2. Determine if area surveys support the analysis results. 3. If #2 is positive, investigate the cause and correct as needed. 4. If exposure is confirmed by #1, investigate to determine how exposure was incurred and correct it. If the exposure exceeds 120 ug, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of this restriction.

3.2.4.3.2 Plutonium Bioassay Program

1. All workers who routinely work in Plutonium handling areas shall be subject to the Plutonium bioassay program. The minimum frequency for urine sampling shall be six months. The minimum frequency for in vivo counting shall be annual. Ad-

ditional bioassays shall be performed when, in the judgment of the Supervisor, Health and Safety, conditions during a job and/or other data (air samples, floor smears or clothing contamination) indicate an internal exposure may have occurred.

2. The following are the action criteria for the routine Plutonium bioassay program:

Analysis	Action Level	Action to be Taken
a. Urinalysis	< 0.2 dpm/L	None
b. Urinalysis	> 0.2 dpm/L	1. Resample the individual within 5 working days. 2. The Supervisor, Health and Safety shall consider the need for worker restriction to prevent further exposure until the diagnostic evaluation is complete. Only the Supervisor, Health and Safety may lift any work restriction once it is imposed. 3. Determine if area surveys support the analysis results. 4. If #3 is positive, investigate the cause and correct. 5. If the exposure is confirmed by #1 investigate to determine how exposure was incurred and correct it. If the exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of this restriction.

c. In vivo	< 1.6E-8 Ci Pu-239	None
d. In vivo	≥ 1.6E-8 Ci Pu-239	1. Restrict the worker from further exposure. 2. Resample the individual within 10 working days. 3. Determine if area surveys support the analysis results. 4. If #3 is positive, investigate the cause and correct as needed. 5. If exposure is confirmed by #2, the Supervisor, Health and Safety shall determine the organ dose. If the confirmed exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposures until the Supervisor, Health and Safety authorizes the lifting of this restriction. 6. The restriction in #1 may be lifted by the Supervisor, Health and Safety if the results of the analysis performed under #2 fails to confirm the analysis.

3.2.4.3.3 Fission Product Bioassay Program

1. The fission product bioassay program sampling frequency shall comply with Regulatory Guide 8.26, September, 1980.

2. Additional bioassays shall be performed when in the opinion of the Supervisor, Health and Safety, conditions during the job were such that significant internal exposure may have

occurred. The following are action criteria for additional bioassays.

Analysis	Action Level	Action to be Taken
In vivo	>10% MPOB	Remeasure subject to determine effective half life of the contaminant and plot decay curves. Follow-up program will continue until the contamination present is <5% MPOB or the effective half life has been determined.
Estimation from nasal smears or air sample	>10% MPOB	Submit in vitro sample for analysis within 5 working days.
In vitro	>5% MPOB	Resample excreta to confirm presence of contamination and to establish rate of elimination. Perform isotopic analysis if >10% of MPOB is a possibility.
In vitro	>10% MPOB	In vivo measurement to be made as soon as practicable.

3. The Supervisor, Health and Safety, shall be responsible for evaluations to determine the location and amount of deposition; to provide data necessary for estimating internal dose rates, retention functions, and dose commitments; and to determine if work restrictions or referrals for therapeutic treatment are required for any case where a result indicating a greater than 10%/MPOB deposition of a radionuclide is verified.

3.2.4.4 Protective Clothing

3.2.4.4.1 The use of protective clothing shall be specified in Area Operating Procedures and Radiation Work Permits.

3.2.4.4.2 Protective clothing may also be specified by the Health and Safety Group. In the event of conflicts between the Area Operating Procedure, Radiation Work Permit, and the Health and Safety Group, the decision of the latter shall prevail.

3.2.4.5 Respiratory Protection

3.2.4.5.1 The Respiratory Protection Program shall be conducted in accordance with 10 CFR 20.103, and shall be a responsibility of the Health and Safety Group.

3.2.4.5.2 The Respiratory Protection Program shall be implemented through written and approved procedures.

3.2.4.6 Surface Contamination Monitoring

3.2.4.6.1 The Health and Safety Group shall perform smear surveys in the below listed areas at the indicated minimum frequencies:

Area	Frequency	Action Level (dpm/100cm2)
<---ALPHA--->		
Unirradiated, unencapsulated fuel handling areas	Weekly	5000
Building B Counting Lab.	Monthly	200
Hot Cell Oper. Area	Monthly	200
Scanning Electron Microscopy Lab.	Monthly	200
Exit portals from controlled areas	Biweekly	200
<---BETA + GAMMA--->		
Building B Counting Lab.	Monthly	2000
Scanning Electron Microscopy Lab.	Monthly	2000
Hot Cell Operations Area	Bimonthly	2000

Cask Handling Area	Bimonthly	22000
Radiochemistry Lab.	Bimonthly	22000
Exit portals from controlled areas	Biweekly	2000

3.2.4.6.2 Large area smears are used to survey many square meters of surface area. To determine if these smears indicate that an action level has been exceeded, the assumed area covered shall not exceed 1-square meter.

3.2.4.6.3 Daily surveys shall be performed in the cafeteria, snack bars, and vending machine areas. If contamination is detected in any of these areas, corrective action shall be taken at once.

3.2.4.7 Decontamination

3.2.4.7.1 The Health and Safety Group shall determine and direct the action to be taken to protect personnel and reduce the levels of contamination below those specified in Section 3.2.4.6.

3.2.4.7.2 Decontamination to reduce levels of contamination shall begin within 24-hours of the discovering survey. If the survey is made just prior to the beginning of a holiday or weekend, the contamination shall be marked and labeled, and decontamination shall commence during the first regular workday after the survey.

3.2.4.7.3 Fixed contamination that, in the opinion of the Supervisor, Health and Safety, does not substantially contribute to a worker's exposure, shall be posted and its location and radiation level recorded and its removal shall be scheduled as soon as practicable.

3.2.4.7.4 Fixed contamination that, in the opinion of the Supervisor, Health and Safety, may substantially contribute to workers' exposure shall be posted and removed as soon as practicable.

3.2.4.8 Emergency Evacuation

3.2.4.8.1 Refer to Radiological Contingency Plan, required by Order dated February 11, 1981, as amended.

3.2.4.9 Personnel Monitoring

3.2.4.9.1 Site Workers (Restricted Area) shall be issued a film badge, a SRD, and a TLD.

3.2.4.9.2 Site Workers (Controlled Areas) shall be issued a film badge, a SRD, and a TLD.

3.2.4.9.3 Visitors shall be issued a TLD.

3.2.4.9.4 Non-Site Workers (Restricted Area) shall be issued a film badge, a SRD, and a TLD.

3.2.4.9.5 Non-Site Workers (Controlled Areas) shall be issued a film badge, a SRD, and a TLD.

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4.0 NUCLEAR CRITICALITY SAFETY

4.1 SPECIAL ADMINISTRATIVE REQUIREMENTS

4.1.1 Double Contingency Policy - The Double Contingency Policy as defined in the American National Standard ANSI/ANS-8.1-1983 shall be followed in establishing the basis for nuclear criticality safety of all operations.

4.1.2 Structural Integrity - Where structural integrity is necessary to provide assurance for nuclear criticality safety, the design and construction of those structures will be evaluated with due regard to load capacity and foreseeable abnormal loads, accidents, and deterioration. The Manager, Facilities shall be responsible for determining the structural integrity of equipment when it is necessary to provide assurance of nuclear criticality safety.

4.1.3 Nuclear Criticality Safety Evaluation - All modifications or additions or both to any operation, system or equipment must be approved by the Facility Supervisor. It is the responsibility of the Facility Supervisor, in consultation with the Nuclear Criticality Safety Officer, to determine whether or not a nuclear criticality safety evaluation is required for the proposed modification or addition. The Nuclear Criticality Safety Officer or a person designated by him shall provide any required evaluations, including calculational support. Nuclear criticality safety evaluations shall be reviewed by a second individual, either the Nuclear Criticality Safety Officer or by a person with the same minimum qualifications required for the Nuclear Criticality Safety Officer.

4.1.4 Posting - Each unit shall be posted with the limits of SNM permitted in the unit. The Facility Supervisor shall be responsible for approving the posting of nuclear criticality safety limits.

4.1.5 Labeling - Each container containing greater than 0.5 grams of SNM shall be labeled to show the amount of element, the percent enrichment, when applicable, and the amount of fissile isotope. This condition does not apply to irradiated SNM.

4.1.6 Compliance - Compliance with the nuclear criticality safety requirements shall be in accordance with written area operating procedures, reviewed and approved by the Facility Supervisor, the Supervisor, Health and Safety, the Nuclear Criticality Safety Officer, and the SRC. Area operating procedures shall include all the controls and limits significant to the nuclear criticality safety of the operation. In addition, the Nuclear Criticality Safety Officer shall perform a quarterly audit for compliance with nuclear criticality safety requirements and verifies that process conditions have not been altered that may affect nuclear criticality safety. The results of the audit shall be documented and submitted to the Manager, EC&RR.

4.2 TECHNICAL REQUIREMENTS

4.2.1 Nuclear Isolation - When nuclear isolation is required (the potential neutronic interaction between units is negligible) the unit or units isolated shall be separated from all other SNM by one of the following or equivalent conditions:

1. Twelve inches of water.
2. Twelve inches of concrete with density of at least 140 lb/ft^3 when the unit(s) being nuclearly isolated are one of the units permitted by this license, (i.e., a mass limit specified in Section 4.2.2.2 or an authorized PWR or BWR fuel assembly or portion thereof, pursuant to Section 4.2.3.6.1) provided that the unit or units cannot be representable as a slab which interacts with other SNM primarily through its major face.
3. The edge-to-edge separation of 12-feet, or the greatest distance across an orthographic projection of either accumulation on a plane perpendicular to a line joining their centers, whichever is larger.

4.2.2 Building A

4.2.2.1 General - Building A shall be limited to 40 units as defined in section 1.6 of this Part. Each unit shall be separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

4.2.2.2 Unit Limits - Each unit shall be limited to one of the following:

4.2.2.2.1 Mass limits for mixtures of plutonium and U-235

Pu (wt%)	Limit (total grams fissile)
0	350
1 to 20	313
20 to 40	283
40 to 60	258
60 to 80	237
80 to 100	220

4.2.2.2.2 Mass Limits for Low Enriched Uranium - 850 grams of U-235 as contained in uranium enriched in the isotope U-235 to and including 4 wt%.

1. Whenever uranium enriched to 4 wt% U-235 is being processed under the 850 gram limit, no unit shall be permitted to have fissile material at an enrichment greater than 4 wt% within that laboratory, room, or work area.

2. Whenever an 850 gram, enriched controlled unit is in use in the building, the Facility Supervisor must approve all transfers involving materials with enrichments greater than 4 wt% within the building.

4.2.3 Building B

4.2.3.1 General - Building B shall be limited to 40 units, excluding the hot cells, underwater storage, and the examination of power reactor fuel assemblies. Each unit shall be separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

4.2.3.2 Unit Limits - Each unit shall be limited to the values specified in Section 4.2.2.2 of this Part.

4.2.3.3 Hot Cell - The hot cells, except for examination of power reactor fuel assemblies, shall be limited to the following units:

1. Three units in hot cell no. 1, separated by at least 12-inches edge-to-edge.

2. One unit in each of the other hot cells.

4.2.3.4 Underwater Storage (Transfer Canal & Storage Pool) - SNM in storage under water in the Transfer Canal & Storage Pool shall be in racks or containers limited to the values specified in 4.2.2.2, excluding power reactor fuel assemblies, and separated by 12-inches edge-to-edge.

4.2.3.5 Storage Tubes - SNM in storage tubes shall be limited to the values specified in 4.2.2.2 for each tube. Storage tubes shall be spaced a minimum of 17-inches center-to-center, are approximately 5-inches in diameter, and totally immersed in concrete.

4.2.3.6 Power Reactor Fuel Assemblies - Examination of unirradiated and irradiated power reactor fuel assemblies, including both non-destructive and destructive testing is carried out in Building B subject to existing nuclear criticality safety limits and controls except as modified by the following conditions.

4.2.3.6.1 Fuel assemblies to be studied are identified as:

1. Each assembly shall be of the enriched uranium oxide PWR type with a 15 X 15, or 17 X 17 square pin lattice not greater than 8.6-inches on a side (further identified as a Babcock & Wilcox Mark B or Mark C canless assembly).

2. The maximum initial enrichment in an unirradiated assembly shall not exceed 4.05 wt%.

3. Damaged fuel assemblies may be examined in air. Fuel assemblies which have been damaged can be examined in water if they maintain their 8.6-inches on a side dimensions.

4.2.3.6.1.1 Other PWR or BWR fuel assemblies which do not meet the above may be studied, provided:

1. The unirradiated, fully reflected fuel assembly (fueled with UO_2 only) with all control rods removed is shown by an appropriate nuclear safety evaluation to be subcritical by at least 5 % (K-eff <0.95).

2. The fuel assembly is shown by an appropriate nuclear safety evaluation to be subcritical by at least 5 % (K-eff <0.95) under specific conditions of disassembly.

3. In 4.2.3.6.1.1 Items 1 and 2 above, the primary source for validation data shall be:

o DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," by H. K. Clark of

Savannah River Laboratories, or;

- o Actual critical experiments having approximately the same enrichment and metal-to-water ratio as is present in the fuel assembly to be studied, or;

- o Criticality data supplied by the reactor designer for the actual fuel assemblies.

An appropriate cross-section set will be selected and used to calculate the validation data from one of the above described sources. Any bias between calculational results with that cross-section set and validation data will be included in the results of the safety evaluation called for in #1 and #2 above. A description of and the results of the validation and the nuclear safety analysis will be assembled into a report which shall contain the items specified in Section 4.3.6 of ANSI/ ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

4.2.3.6.1.2 BWR fuel assemblies may be received and studied provided:

1. They are evaluated pursuant to Section 4.2.3.6.1.1 of this Part, or

2. The BWR fuel assemblies have a maximum initial unirradiated enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

4.2.3.6.2 Receipt and Storage

4.2.3.6.2.1 Unirradiated Fuel Assemblies - Unirradiated fuel assemblies will be received at a maximum of two at a time in a shipping container licensed for two assemblies, or one assembly in a shipping container licensed for one assembly. Unirradiated fuel assemblies may be stored in air in the Crane & Cask Handling Area, the Assembly & Machine Shop Area, or the Development Test Area subject to the following conditions:

1. Assemblies may be stored in their shipping container as received.

2. Assemblies may be stored no less than 21-inches apart center-to-center.

3. Assemblies may be stored under water in the hot cell pool, mockup pool, or development test area pool in racks constructed to maintain a 1-foot minimum surface-to-surface separation between assemblies and any other SNM.

4. No more than four unirradiated assemblies may be kept at the site at one time.

4.2.3.6.2.2 Irradiated Fuel Assemblies - Irradiated fuel assemblies shall be received one at a time in a licensed single assembly shipping container or two at a time in a shipping container licensed for two assemblies. Fuel assemblies that have been irradiated will be stored in the hot cell pool which is limited to the following conditions:

1. A maximum of four assemblies or portions thereof may be in the pool at a time.

2. The assemblies shall be stored in a storage rack which is so constructed as to maintain a 1-foot minimum surface-to-surface separation between the stored assemblies and any other fissile material which might be in the pool.

3. Only one assembly may be in a designated work area of the pool at any one time. There shall be at least 1-foot minimum surface-to-surface separation between the assembly in the work area and any other fissile material.

4. Fuel rods which have been removed from an assembly shall be stored in a storage rack providing space in each position for a maximum of 75 rods. All positions shall be spaced from any other fissile material by a minimum surface-to-surface separation of 1-foot.

5. Partially dismantled assemblies will be stored in the assembly storage rack.

6. Each position in the assembly storage rack and in the fuel rod storage rack must limit contained fuel to a square not to exceed the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

4.2.3.6.3 Work Area of Pool Under Hot Cell No. 1 - The work area position under Cell No. 1 is used to load and unload irradiated fuel assemblies into and out of shipping casks and to dismantle both irradiated and unirradiated fuel assemblies. The following conditions govern operations in this work area:

1. Only two assemblies at a time shall be permitted outside of their shipping container provided they are separated by a minimum surface-to-surface separation of 1-foot.

2. An associated storage position shall be permitted for fuel rods or components which have been removed from the assemblies.

3. The assemblies and associated rod storage positions shall be separated from each other and any other fissile material by a minimum surface-to-surface separation of 1-foot.

4. Fissile material and fuel rods or components in the associated storage positions shall each be restricted to a square not exceeding the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

5. Only one fuel rod at a time shall be removed from or inserted into the assembly or the rod storage position. A Maximum of 75 rods shall be permitted in the rod storage position.

6. A fuel assembly may be completely dismantled by withdrawing one fuel rod at a time from the assembly; during all stages of dismantlement, the partially dismantled assembly shall be maintained within the confines of a square not exceeding the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

7. An assembly and its associated rod storage position may be withdrawn from the pool into the cell. Free water drainage from both the assembly and rod storage position as well as 1-foot separation from other fissile materials and each other shall be assured.

4.2.3.6.4 Assembly and Machine Shop and Development Test Area - The work areas on the first floor of Building B may be used to disassemble unirradiated fuel assemblies for testing. The following conditions govern operations in the work area:

1. Only one assembly at a time shall be permitted to be dismantled.

2. An associated storage position will be permitted for fuel rods which have been removed from the assembly, and shall be spaced and stored as stated in items 3 and 4 (4.2.3.6.4) below.

3. The assembly and associated rod storage position shall be separated from each other and any other fissile material by a minimum surface-to-surface separation of 21-inches.

4. The associated rod storage position shall be no larger in any dimension than the fuel assembly. There shall be one such storage position for each assembly to be dismantled. Rods may be stored or handled in a slab up to 4-inches thick provided the slab is separated from other fissile material by a minimum of 12-feet.

5. Only one fuel rod at a time may be removed from or inserted into the assembly or any rod storage position. Only one rod may be in transit to any one location at a time.

6. The fuel assembly may be completely disassembled by withdrawing one fuel rod at a time from the assembly; during all stages of disassembly, the partially disassembled assembly shall be maintained within the confines of the assembly whether damaged or undamaged.

7. Fuel rods may be removed one at a time from this area as required. These rods shall be subject to all fuel handling requirements pertinent to the area they are in.

8. Assemblies may be handled and dismantled under water in these areas (mock-up pool and development test area pool) subject to the same requirements of the hot cell pool.

4.2.3.6.5 Hot Cell Operations - Fuel rods removed from irradiated assemblies may be examined including destructive examination in the hot cells. Operations in the hot cells shall be governed according to the following conditions:

1. An assembly and its associated rod storage position may be withdrawn from the pool into Hot Cell No. 1 pursuant to Item No. 7 of Section 4.2.3.6.3 of this Part.

2. Two units in Hot Cell No. 1 may have a total of 64 fuel rods each, stored, provided that rods shall be confined within a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder, drainage of any free water within the unit shall be assured and the units must be maintained 1-foot from each other and any other SNM in the cell.

3. In addition to the two units of stored rods, another unit limited to the values in 4.2.2.2.1 may be present in Hot Cell No. 1. In this unit under mass control, rods may be destructively examined.

4.2.3.6.6 Fuel Rod Dismantlement - Fuel rods from unirradiated assemblies can be dismantled in any area where the license permits handling of unirradiated fuel. The following conditions must also be met in areas to dismantle fuel rods:

1. The area shall be mass limited to 350 grams of U-235. This area must be separated from the assembly and slab storage area by minimum of 12-feet.

2. Dismantlement must be completed under approved procedures.

4.2.3.6.7 Shipment and Disposal - After examination, assemblies, partially dismantled assemblies, fuel rods, and scrap generated during destructive examination shall be disposed of according to the following conditions.

1. Fuel rods, including fuel rod segments may be placed in any available hole in a fuel assembly, including the instrument and control rod guide tube positions, i.e., 225 and 285 fuel rods in Mark B and Mark C assemblies, respectively. Fuel rod segments shall have their ends sealed, and shall be encapsulated in steel tubing with ends sealed, prior to insertion into an available hole in a fuel assembly.

2. Unirradiated assemblies may be reassembled (one rod at a time) for later use.

3. Assemblies, including partially dismantled assemblies, shall be loaded into shipping casks approved for such assemblies for shipment.

4. Scrap, including rod segments, shall be disposed of according to present LRC procedures and limits.

4.2.4 Building C

4.2.4.1 General - Building C is limited to 90 units. Each unit shall be separated from adjacent units by at least 8-inches edge-to edge and 36-inches center-to-center.

4.2.4.2 Unit Limits - Each unit shall be limited to the values specified in section 4.2.2.2 of this Part.

4.2.5 Outside Storage

4.2.5.1 General - Outside storage consists of underground storage, shipments, and the fenced storage area located adjacent to Building J.

4.2.5.2 Underground Storage - Radioactive materials stored in underground storage tubes shall be limited to one SNM unit per tube, with values as specified in section 4.2.2.2 of this Part. Tubes shall be spaced 20-inches center-to-center and are 5-inches in diameter, and totally immersed in concrete.

4.2.5.3 Shipments - Each shipment of fissile material being stored outside must conform with all license requirements for the type of shipping container. Additionally, each shipment must be nuclearly isolated from all other SNM.

4.2.6 Dry Waste - Dry waste is accumulated in 55-gallon drums, or other suitable containers, with a maximum of 45 grams of SNM per container. These containers may be located throughout the laboratories as required to collect contaminated laboratory waste. Filled containers are transferred, to the radioactive waste storage building after scanning. Dry waste containing 0.5 grams of SNM or less per container may be stored in a fenced, locked and paved outside storage area adjacent to Building J.

5.0 ENVIRONMENTAL PROTECTION

5.1 EFFLUENT CONTROL SYSTEM

5.1.1 Responsibility - The Supervisor, Health and Safety is responsible for the Effluent Control System.

5.1.2 Solid Waste - Solid radioactive waste, including solidified liquid wastes, shall be sent off site to a licensed disposal facility.

5.1.3 Liquid Waste - Low-level, liquid radioactive waste is discharged from process areas to the Liquid Waste Disposal Facility. This facility is comprised of several tanks where the waste is accumulated for eventual transfer to the B & W Naval Nuclear Fuel Division (NNFD) for ultimate release to the James River.

5.1.3.1 Prior to release to NNFD the contents of a tank shall be mixed and sampled.

5.1.3.2 The contents of liquid waste tanks shall not be released to the NNFD unless the concentration of radioactivity is less than 25% of the MPC values of Table I, Col. 2, of 10 CFR 20, Appendix B. The limiting values in water shall be determined in accordance with the note at the end of Appendix B, 10 CFR 20.

5.1.3.3 Process liquid wastes may be collected and stored indoors. These wastes may be solidified and handled as dry waste.

5.1.3.4 Storage tanks in the Liquid Waste Disposal Facility shall be inspected visually upon each tank voiding or annually, whichever is sooner, to assure that there is no unsafe accumulation of Special Nuclear Material. Storage tanks that have not been used during a year will not be inspected.

5.1.3.5 Samples of liquid waste are grab sampled. A small portion of the sample is pipetted into a planchet and brought to dryness. This planchet is counted on a low background counter, either Low Beta or Wide Beta, and the waste activity concentration is calculated. The samples are counted for gross alpha and beta. Waste tanks that may receive Sr-90 waste will have their samples analyzed for Sr-90.

5.1.3.6 Waste tanks that indicate concentrations of activity greater than those specified in section 5.1.3.2 shall be appropriately diluted prior to release.

5.1.3.7 The NNFD must approve the release of liquid waste to their waste treatment facility prior to the release.

5.1.3.8 The 10,000 sq. ft. Storm Drain Collection Pond shall be grab sampled quarterly. The sample shall be analyzed for gross alpha and gross beta.

5.1.4 Gaseous Effluent - Discharge air from process areas is released to the general environment through the 50-meter high stack. The discharge rate of the stack is approximately 20,000 cubic feet per minute. The annual discharge volume is 1.1E10 cubic feet. Planned discharges to the air shall be in compliance with 40 CFR 61. The annual exposure resulting from these planned discharges shall not exceed 25 millirem whole body and 75 millirem to any organ.

5.1.4.1 Action levels - The action levels for releases from the stack are specified in section 3.2.2.5 of this Part I.

5.1.4.2 Analyses - The fixed filter of the stack particulate monitor shall be counted on the Low Beta or Wide Beta counting system, after an appropriate decay period. The results shall be recorded and maintained on file.

5.1.4.3 Sampling - The stack shall be sampled isokinetically on a continuous basis.

5.1.4.4 Monitoring - The stack sample shall be passed through a monitoring system that consists of the following:

1. Particulate Monitor - The stack particulate monitor consists of an alpha and beta channel, with a dual channel ratio detector. This monitor uses a fixed filter and a nominal sampling flow rate of 2 - 3 cubic feet per minute. The detector is a thin window (1.0 mg/cm^2) gas flow proportional detector. Alpha and Beta-gamma radiations are monitored through two single channel analyzers and log rate meters. The ratio between these two channels is also displayed as a logrhythmic ratio. This system effectively compensates for the presence of Radon and Thoron daughters and increases the sensitivity of the system. Alpha and Beta-gamma sensitivities

are similar for both channels. Alarm settings, based on the ratio system, are sufficient to alarm at or below short term stack concentrations that are specified in section 3.2.2.5 of this Part I and that which would result in concentrations in unrestricted areas exceeding 10 times the applicable limits given in 10 CFR 20 for the nuclides in use at the site.

2. Gas Monitor - The stack gas monitoring system consists of a shielded chamber with one or two GM tube detectors (30 mg/cm^2 stainless steel). The Beta-gamma count rate is directly proportional to the stack concentration and system sensitivity is approximately 3E-9 uCi/ml per CPM for Kr-85. A conventional alarming and recording log ratemeter is used to monitor the gas channel. The alarm level is set to activate below the level representing 70 Curies/week of Kr-85.

5.2 ENVIRONMENTAL MONITORING

5.2.1 The environment surrounding the site and the Mount Athos plant site is sampled periodically to determine whether the radiation and radioactive material levels in the area surrounding the site have changed as a result of the operations at this location.

5.2.2 The following types of samples shall be taken at the below indicated frequencies:

1. Site boundary air sample - monthly.
2. Grab sample of the James River above and below the point of discharge - monthly.
3. Continuous sampling of rain water.
4. Grab sample of river silt - quarterly.
5. Direct radiation survey shall be made of the water channel passing through the railroad right-of-way - annually.
6. Direct radiation survey shall be made at the east end of the canal - annually.
7. Vegetation sample - semiannually.

8. Direct radiation monitoring at the site boundary - continuously.

9. Accumulated water from the soil retention basin shall be sampled and if its activity exceeds 10% of the concentration specified in Appendix B, Table 2, column 2, of 10 CFR 20, the collected water shall be disposed of through the liquid waste disposal system - annually.

5.2.3 The evaluation of environmental sampling shall be performed by either the site personnel or a qualified outside concern.

6.0 SPECIAL PROCESS COMMITMENTS

The site is engaged in research and development and for this reason there are a large number of small special processes that are special only because they are outside of the few repetitive operations. Examples of operations performed at the site are given in section 1.7.1. The site relies on established administrative controls to determine what proposals fall outside of the bounds of work that is authorized by the license, in which case amendments are applied for. Those proposals that are authorized by the license but are significantly different from previously reviewed proposals shall be brought before the Safety Review Committee for review and approval.

7.0 DECOMMISSIONING PLAN

The site is committed to decommissioning the facilities which have been used for the use and storage of licensed material at the end of their useful life. At the time of this application for renewal of License No. SNM-778, two programs are underway to decommission Buildings A and C. It is presently estimated that these two facilities will be decontaminated and ready for release for unrestricted use by March, 1987.

7.1 PLANNING CONSIDERATIONS

7.1.1 The history of the facility shall be determined to facilitate the identification of services, equipment, and areas that should be included in the survey plan.

7.1.2 The decontamination of facilities and equipment must meet the levels of contamination specified in Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976. In addition, a reasonable effort will be made to further reduce contamination levels to those which are as low as reasonably achievable.

7.1.3 No covering will be applied to remaining surfaces until it has been determined that contamination levels are below those of Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976, and until it has been determined that a reasonable effort has been made to further reduce contamination below those specified above.

7.1.4 The radioactive contamination of interior surfaces of pipes, ductwork, and other conduits will be determined by taking measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of interior conditions. If such access locations are not likely to be representative, or if interior surfaces are inaccessible, the interior surfaces will be assumed to be contaminated in excess of levels specified in Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and

8.0 RADIOLOGICAL CONTINGENCY PLAN

The site shall maintain and execute the response measures of the Radiological Contingency Plan submitted to the NRC on November 15, 1983, in accordance with provisions of the February 11, 1981 order. The site shall maintain implementing procedures for the Radiological Contingency Plan as necessary to implement the Plan. The site shall make no change in the Radiological Contingency Plan that would decrease the response effectiveness of the Plan without NRC approval. The site may make changes to the Radiological Contingency Plan without prior NRC approval if the changes do not decrease the response effectiveness of the Plan. The site shall maintain records of changes that were made to the Plan without prior approval for a period of two years from the date of change and shall furnish the Chief, Uranium Fuel Licensing Branch, Division of Fuel Cycle and Material Safety, NMSS, USNRC, Washington, DC 20555 and Region II, a report containing a description of each change within six months after the change is made.

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PART II

SAFETY DEMONSTRATION

9.0 OVERVIEW OF OPERATION

9.1 CORPORATE INFORMATION

9.1.1 LICENSEE:

MCDERMOTT INTERNATIONAL, INC.
BABCOCK & WILCOX
NAVAL NUCLEAR FUEL DIVISION
NNFD RESEARCH LABORATORY

9.1.2 Address:

Babcock & Wilcox
Naval Nuclear Fuel Division
NNFD Research Laboratory
P. O. Box 11165
Lynchburg, Virginia 24506-1165

9.1.3 Principal Offices:

McDermott International, Inc.
Babcock & Wilcox
1010 Common Street
P. O. Box 60035
New Orleans, Louisiana 70160

9.1.4 Principal Officers:

J. E. Cunningham
Chairman of the Board &
Chief Executive Officer
1010 Common Street
P. O. Box 60035
New Orleans, Louisiana 70160
U. S. Citizen

Robert E. Howson
President & Chief Operating
Officer
1010 Common Street
P. O. Box 60035
New Orleans, Louisiana 70160
U. S. Citizen

John A. Lynott
Executive Vice President
Chief Financial Officer
1010 Common Street
P. O. Box 60035
New Orleans, Louisiana 70160
U. S. Citizen

Robert E. Howson
President and Chief Operating Officer
Babcock & Wilcox
1010 Common Street
P. O. Box 60035
New Orleans, Louisiana 70160
U. S. Citizen

9.1.5 State of Incorporation:

Delaware

9.1.6 Alien or Foreign Control:

Babcock & Wilcox is incorporated under the laws of the State of Delaware. In 1978 McDermott Inc., a Delaware corporation, acquired Babcock & Wilcox. In 1983 McDermott International, Inc., incorporated under the laws of the Republic of Panama, became the parent company of the McDermott group of companies. This reorganization was reviewed by the NRC prior to its implementation. In a letter dated December 17, 1982, to Mr. J. H. MacMillan, William Dirks summarized the Commission's finding that the change was not inimical to the common defense and security or the health and safety of the public. Based on this conclusion, the change was approved under Section 104(b) of the Atomic Energy Act as it related to the operation of a critical experiment reactor owned by Babcock & Wilcox at the site. Such approval was not required under the Act for material licensees. The organization of McDermott International, Inc. has not changed since that action.

9.2 FINANCIAL QUALIFICATIONS

9.2.1 The financial qualifications of the Corporation to continue operations at the site and to perform the necessary decommissioning at the end of plant life is demonstrated in the latest (1984) copy of the Corporation's Annual Report, which is enclosed with this application.

9.3 SUMMARY OF OPERATING OBJECTIVE AND PROCESS

9.3.1 Research and development activities, utilizing licensed material, are conducted at the site in support of the operating divisions of Babcock & Wilcox and for other companies and government organizations. The broad range of projects that have been conducted pursuant to the license cannot be described in terms of through-put or any single process. Radioactive materials are handled and stored, principally in Building B. That building houses the Hot Cells, Radiochemistry Laboratory, Scanning Electron Microscopy Laboratory, Metallurgy Laboratories, Analytical Chemistry Laboratory, Fatigue and Fracture Laboratory, Failure Analysis Laboratory, Crane and Cask Handling Area, a Hot Machine Shop, the Counting Room, and a Health Physics Laboratory.

Licensed material in the form of liquid waste is collected in tanks that are located in the Liquid Waste Disposal Facility. Solid radioactive waste is stored in Building J, the Annex to Building J, and the storage area adjacent to Building J.

9.4 SITE DESCRIPTION

The site is located on the James River about four miles east of Lynchburg, Virginia. The site, which comprises 525 acres, lies within Campbell County and borders on Amherst County. The site occupies about 13.6 acres of the site. The location of the site within the Commonwealth of Virginia is shown in Figure 9-1.

The irregularly shaped property is bounded on three sides by a large loop of the James River and on the fourth side by State Route 726, which closely follows the base of Mount Athos. This mountain rises rapidly from about 500 feet MSL to 900 feet MSL, making it the dominant feature of the surrounding landscape. The Babcock & Wilcox property consists of large sections of relatively flat floodplain along the James River lying at about 470 feet MSL. The interior of the property is largely composed of rolling hills, one of which rises to almost 700 feet MSL. The property boundary and topography within about two miles of the site are shown in Figure 9-2.

The land in the immediate vicinity of the plant is sparsely inhabited. The severe topography makes it unsuitable for commercial farming. The Lynchburg Foundry, a producer of light metal castings, occupies a parcel of land which abuts the south boundary of the Babcock & Wilcox property. The Foundry is approximately .5 miles from the site.

The site is serviced by a spur of the Chessie System Railroad which runs through the Babcock & Wilcox property. The property is also conveniently located for truck and automobile access. About three miles from the site, State Route 726 connects with U.S. Highway 460, a major link between Roanoke and Richmond. The site is located about 100 feet above the James River and for that reason no dams on the river would threaten the site should they fail.

9.5 LOCATION OF SITE BUILDINGS

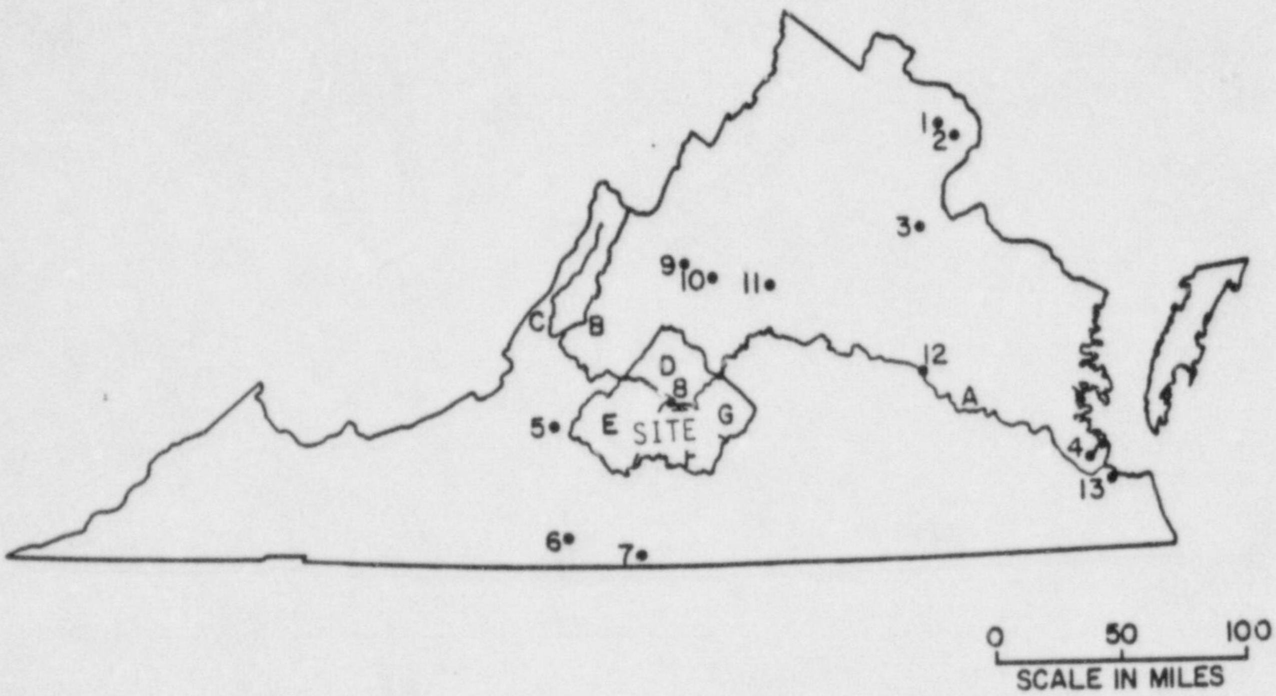
9.5.1 Figure 9-3 shows the layout of buildings at the site. All buildings are of masonry construction.

9.6 LICENSE HISTORY

9.6.1 License SNM-778 was issued on September 16, 1966. Since that time the following renewals and amendments have been approved:

February 15, 1974	First renewal
July 21, 1980	Second renewal
August 28, 1981	Amendment No. 1, approved a change in the organization.
February 25, 1982	Amendment No. 2, approved the Radiological Contingency Plan

FIGURE 9-1



1. ARLINGTON (174, 284)
2. ALEXANDRIA (110, 938)
3. FREDERICKSBURG (14, 450)
4. NEWPORT NEWS (138, 177)
5. ROANOKE (92, 115)
6. MARTINSVILLE (19, 653)
7. DANVILLE (46, 391)
8. LYNCHBURG (54, 083)
9. STAUNTON (25, 504)
10. WAYNESBORO (16, 707)
11. CHARLOTTESVILLE (38, 880)
12. RICHMOND (249, 621)
13. NORFOLK (307, 951)

A. JAMES RIVER
B. COWPASTURE RIVER
C. JACKSON RIVER
D. AMHERST COUNTY (26, 072)
E. BEDFORD COUNTY (26, 728)
F. CAMPBELL COUNTY (43, 319)
G. APPOMATTOX COUNTY (9, 784)

[NUMBERS IN PARENTHESES ( ) ARE 1970 CENSUS DATA]

FIGURE 9-2

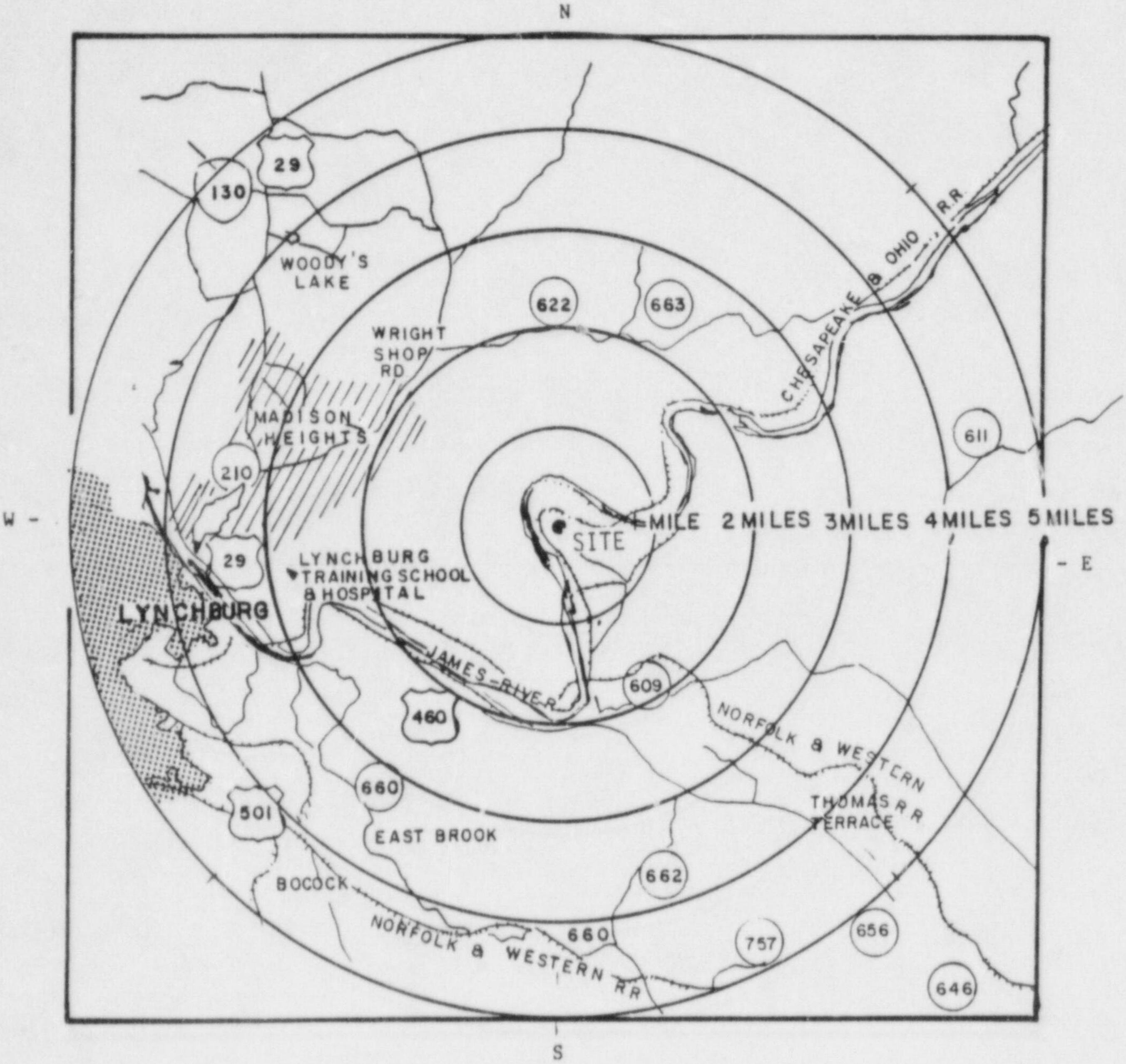


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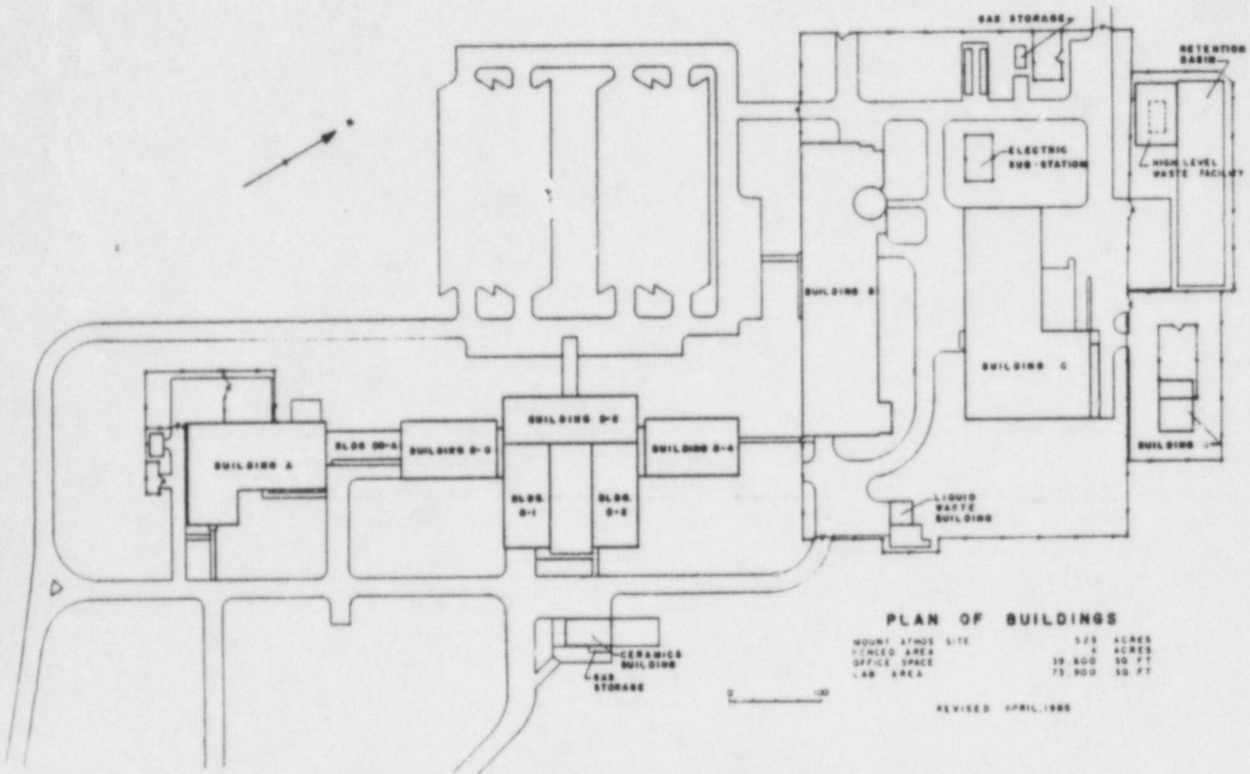
FIGURE 9-3



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10.0 FACILITY DESCRIPTION

10.1 PLANT LAYOUT

Figures 10-1 through 10-3 show the layout of the site buildings.

10.2 UTILITIES INCLUDING EMERGENCY POWER

10.2.1 Potable Water

Potable water is provided to the site by the NNFD. It is pumped from wells. It is stored and treated at the NNFD and is gravity fed to the site.

10.2.2 Process Water

Process water is provided to the site by the NNFD. The source of process water is the James River. It is pumped from the river, filtered by the NNFD and is gravity fed to the site. There is a storage capacity of 6,000,000 gallons on site. Process water is also used for fire fighting.

10.2.3 Gas

Natural gas is supplied at the site via pipeline which enters the B & W property on the western side of the site. Natural gas is used for space heating, fuel for emergency engines, and laboratory uses. This system is provided with a backup source of propane gas which is stored on site.

10.2.4 Fuel Oil

Fuel oil is available for space heating to provide a backup source in the event of curtailed availability of natural gas. Fuel oil is purchased locally and stored on the site site. There is a storage capacity of approximately 24,000 gallons.

10.2.5 Electricity

Electricity is furnished to the site from a substation located on the west side of the site. This source provides the normal source of power for the stack fans, hot cell fans, criticality monitors, emergency evacuation alarm, and lighting. When normal site power is lost, emergency sources are provided for these loads in the following manner.

of being locked and provided with side panels which permit the roof to fit flush with the top of the block walls. Containers are loaded into the Annex from the top. A curbing will be placed on the approach side of the addition to prevent a loading vehicle from accidentally contacting the wall. Two individuals are involved in loading containers into this facility to prevent a container from striking the walls. This facility provides storage of waste that is contaminated with irradiated fuel and is being stored on site until it is accepted by the DOE under the Nuclear Waste Policy Act of 1982. The maximum quantity of SNM per container shall be limited to 45 grams.

10.4.2.5 The Outside Waste Storage Area is located adjacent to Building J. This area is fenced, locked and paved. Waste stored in this area is limited to that contained in closed metal containers. Each container is limited to not more than a Type A quantity (10 CFR 71.4) or 0.5 grams of fissile material or both. Pu shall not be stored in this area. Containment of stored waste is assured by a quarterly visual inspection by the Supervisor, Health and Safety.

10.4.2.6 The High Level Waste Storage Tubes are located adjacent to the south side of the Liquid Waste Disposal Facility. These tubes are constructed of two sections of iron pipe, immersed in concrete, and below ground level. The upper section of pipe (approximately 42-inches long) is 6-inches in diameter. The lower section (approximately 80-inches long) is welded to the upper section and is 5-inches in diameter. Each tube is fitted with a concrete-filled iron plug. These tubes are locked and under the direct control of the Health and Safety Group. Waste stored in these tubes is limited to that which is produced in the Hot Cells and must be in closed metal containers. The quantity of fissile material permitted in each tube is limited to one unit.

10.5 FIRE PROTECTION

10.5.1 Codes and Standards - The development and building construction program of the site has taken place over the period 1955 to the present. For the three main buildings under consideration in this renewal request, the design and construction efforts took place from 1955 to 1969. There have been a number of alterations and use changes over the past ten years, but generally these changes have not significantly altered the structural characteristics of the buildings.

All three buildings were built as staged or "added-on" phased construction. Building A was built in four distinct phases, Building B was built in two stages, and Building C was constructed in five phases. Building A was designed in-house by B&W engineering personnel. Building B was designed by Wiley & Wilson, a Lynchburg consulting engineering firm. Building C was also primarily designed by Wiley & Wilson, with some design by B&W.

The physical layout of all three buildings is highly functional, i.e., based on the specialized requirements of research work related to the nuclear industry. For the most part, the building structure envelopes are quite conventional in nature, both from a design and construction materials standpoint. With the exception of highly specialized portions of these buildings, such as the hot cells, engineering design of the buildings would be considered as state-of-the-art for light industrial/heavy commercial class buildings (for each of the design and construction time periods involved).

The overall quality of the building construction is well above average. Aside from some roof leakage problems and minor settlement cracking in some of the masonry construction, the performance of the building structures and envelopes has been good. There have been no repairs related to significant structural defects in any of the three buildings. As would be anticipated for a complex of this type and importance, maintenance of the buildings has been excellent and contributes to the overall good condition of such a facility.

During the period of design and construction for Buildings A, B, and C, it should be noted that there was very little in the way of code requirements or guidance for construction of such a facility. Virginia did not adopt a state-wide building code until September 1, 1973. Up until that time, various localities in the state had adopted their own local building codes; the Southern Building Code being the one generally used. Many counties, however, had no code at all; Campbell County, in which the site is located, had no building code during this time period. The only state-wide code directly applicable to building construction prior to 1973, was the Virginia Fire Safety Regulations, enacted in 1949.

The lack of a state-wide building code should not be taken as implication that the design and construction of the site facilities were accomplished on an inferior basis. On the contrary, where good accepted engineering and construction practices, coupled with stringent requirements from insuring companies such as Factory Mutual, are used as the main criteria for such facilities, the resulting structures usually far exceed the minimum requirements of various building codes. Such is typically the case for all three buildings under consideration, when examined from the load capacity standpoint as specified in the present Virginia building code, BOCA (Building Officials and Code Administrators) 1978, Seventh Edition.

Conventional construction materials are used throughout all three buildings. Structural steel yield strength varies from 33,000 PSI (ASTM A7 steel) for the 1955 construction to 36,000 PSI (ASTM A36 steel) for the 1969 construction. Concrete strengths vary from 3000 PSI for conventional cast-in-place concrete construction to 5000 PSI for the precast prestressed concrete elements found in Building B. Concrete reinforcing steel is typically ASTM A615, Grade 40 (40,000 PSI yield strength). Working stress design was used as the basis for concrete and steel design for all structures on site. Applicable design criteria used for the facility includes the standards of the American Concrete Institute (ACI), the Prestressed Concrete Institute (PCI), and the American Institute of Steel Construction (AISC). These various standards serve as both design and code basis for the respective types of construction, both at the time of original design as well as the present.

10.5.2 Insurance Inspection Reports - The site is inspected twice annually by the Arkwright-Boston Insurance Company on behalf of the Mutual Atomic Energy Reinsurance Pool (MAERP). The inspection reports list the following items in each report; housekeeping, maintenance & repair, Supervision fire equipment, watchmen, radioisotope handling, areas sprinklered, water supply, all valves found open, criticality control, and until the decommissioning of the last reactor, nuclear reactor operation. These reports have consistently found that the site meets the requirements in each category for a "satisfactory" rating. On a few occasions there have been recommendations that the site add fire protection equipment when the use of an area has been changed. Each such recommendation has been addressed at the site in a manner that has been found acceptable to the inspectors upon their reinspection. The reports on which the above statement is based are dated from 1977 through 1985.

10.5.3 Fire protection equipment is installed in response to recommendations made by the Industrial Safety Officer, the Corporate Fire Protection Engineer, or the insurance underwriters. Installed systems are approved and inspected by Factory Mutual Engineering

Association. Routine inspection and maintenance is described below:

EQUIPMENT	MAINTENANCE	RESPONSIBILITY	REFERENCE
Portable fire extinguishers	Insp./test	Industrial Safety	NFPA 10 FM 4-5
Fire hoses	Insp./test	Industrial Safety	NFPA 10
Sprinklers	Test	Plant Engineering	NFPA 13 FM 4-5
Fire suppres. systems (Halon)	Inspection	Plant Engineering	NFPA 12-A FM 4-8N Mfg.
Smoke det.	Test	Plant Engineering	Mfg.
Heat det.	Test	Plant Engineering	Mfg.
Housekeeping	Inspection	Health and Safety	---
Emer. equip.	Inspection	Health and Safety	Mfg.

10.5.4 Combustible Waste Storage - Combustibles are not routinely stored at the site except; when work requiring such materials is in progress, in containers for shipping and receiving, or in sprinklered areas. Combustible wastes are discarded in metal containers and disposed of by an off site disposal firm. Contaminated combustible waste is discarded in metal containers and shipped to an off-site licensed disposal facility.

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11.0 ORGANIZATION AND PERSONNEL

11.1 SITE ORGANIZATION

11.1.1 The Manager, Employee, Community, and Regulatory Relations (Manager, EC&RR) - The Manager, EC&RR is responsible for all site operations. The Manager, Safety and Licensing, Manager, Facilities, and the Facility Supervisor report to him.

11.1.2 Facility Supervisor - Research and development work at the site will be performed by personnel who do not report to the Manager, EC&RR. Therefore, the positions of Facility Supervisor and Area Supervisor have been established to control the workers and their activities.

The Facility Supervisor shall report to the Manager, EC&RR. He shall be responsible for the safety of all operations performed pursuant to License SNM-778. He shall utilize the expertise of the Supervisor, Health and Safety, the Accountability Specialist, Nuclear Criticality Safety Officer, and the Industrial Safety Officer to ensure the safety of operations.

11.1.3 Area Supervisors - Area Supervisors are selected by their Division Management and shall be jointly approved by the Facility Supervisor and the Supervisor, Health and Safety. Area Supervisors functionally report to the Facility Supervisor and are responsible for the safe performance of all activities in their assigned area and that all activities within their assigned areas are performed in full compliance with the license.

11.1.4 Manager, Safety and Licensing - The Manager of Safety and Licensing is appointed by and reports to the Manager, EC&RR. He is responsible for the proper management of the materials accounting function, licensing function, nuclear criticality safety function, and the Health and Safety Group. He manages the allotment of funds and other resources and assures the proper assignment of personnel priorities. The Supervisor, Health and Safety, Accountability Specialist, Nuclear Criticality Safety Officer, and License Administrator, report to him.

11.1.5 Supervisor, Health and Safety - The Supervisor of Health and Safety is appointed by the Manager, EC&RR and reports to the Manager, Safety and Licensing. The Supervisor directs the overall operation of the Health and Safety Group and the Industrial Safety Officer.

He also serves on the Safety Review Committee. He has the authority to stop any operation that he believes is contrary to accepted safety practices, or license requirements. The Supervisor has overall responsibility for the shipment and receipt of licensed material and exercises signature authority on all Area Operating Procedures. He performs audits of the site for compliance with Health and Safety rules. The Senior Health Physics Engineer and Industrial Safety Officer report to him.

11.1.6 Senior Health Physics Engineer - A Senior Health Physics Engineer reports to the Supervisor, Health and Safety. He administers the activities of the Health Physics staff, which include:

1. Performing area surveys
2. Administering the air sampling program
3. Administering the respiratory protection program
4. Administering the bioassay program
5. Leak testing radioactive sources
6. Supervising shipping and receiving of licensed material
7. Supervising and coordinating the waste disposal program
8. Assisting in personnel, equipment, and facility decontamination
9. Conducting radiation safety training
10. Providing expertise in all aspects of radiation protection
11. Generating, maintaining and distributing records and reports that are required by NRC regulations or Health Physics procedures
12. Providing expertise in health physics to the Facility Supervisor.

11.1.7 Industrial Safety Officer - The Industrial Safety Officer reports to the Supervisor, Health and Safety. His responsibilities include the following:

1. Administering the industrial safety program

2. Reviewing proposed facility changes to ensure fire safety

3. Providing expertise in fire prevention to the Facility Supervisor and the Safety Review Committee

4. Performing tests, maintenance, and inspection of fire protection, control, and extinguishing equipment

5. Providing training for the site Fire and Rescue Team and off site support agencies

6. Inspecting all areas of the site periodically to ensure:

   a. Proper storage and use of flammable solvents

   b. Proper placement of fire extinguishing equipment

   c. Elimination of fire hazards

   d. Reduction, to the extent practicable, of the accumulation of flammable materials

   e. Proper use and maintenance of electrical equipment.

7. Working with Area Supervisors to formulate safety rules and elimination of hazards

8. Investigation of all personnel injuries

9. Keeping management informed concerning industrial safety activities

10. Conducting industrial safety training.

11.1.8 Accountability Specialist - The Accountability Specialist reports to the Manager, Safety and Licensing. He is responsible for the maintenance and retention of SNM accountability records. He prepares and transmits the reports required by regulation to inform regulatory agencies of SNM transactions.

11.1.9 License Administrator - The License Administrator reports to the Manager, Safety and Licensing. The License Administrator is responsible for administering the license. He is the primary liaison between the site and the NRC and other federal, state, and local agencies regarding nuclear matters. He is the coordinator of the Safety Review Committee and Chairman of the Safety Audit Subcommittee and represents site management on both. The License Administrator is responsible for ensuring that corrective action is taken in response to audit findings as they pertain to licensed activities.

11.1.10 Nuclear Criticality Safety Officer - The Nuclear Criticality Safety Officer is appointed by and reports to the Manager, Safety and Licensing. The Nuclear Criticality Safety Officer is responsible for ensuring that no operation at the site can lead to the inadvertent assembly of a critical mass. To help assure this, he has signature authority for all new Area Operating Procedures and changes to these procedures, he observes operations, institutes educational programs if and when he deems them necessary, and carries out confirming nuclear criticality safety calculations.

The Nuclear Criticality Safety Officer will inspect all site operations where special nuclear material is being processed, quarterly. Other areas may be inspected less frequently, but all licensed facilities will be inspected at least twice a year. He will consider area operations when scheduling these inspections and will, if necessary, schedule his inspection at more frequent intervals. His consideration should include inspection of new operations, an audit of nuclear safety records, a check for area posting, a review of current practices and a review of corrective actions recommended during previous audits and the status of the recommended actions. He shall submit a report of his finding to the Manager, EC&RR, with a copy to the License Administrator. Prior to the submission of the report, he will discuss its contents with the Facility Supervisor. The following information is to be included:

1. Areas visited
2. Operations observed
3. Unsafe practices and situations noted
4. Nuclear safety activity of the quarter
5. Recommendations.

11.1.11 Facility Supervisor - The Facility Supervisor is appointed by and reports to the Manager, EC&RR. He shall be responsible to the Manager, EC&RR for the safe conduct of all operations at the site and for ensuring that these operations are conducted in accordance with all license conditions. The Facility Supervisor shall review and have approval authority for Area Operating Procedures. He shall have authority to terminate any operation that he deems contrary to license conditions, Area Operating Procedures, or general safety conditions. The Facility Supervisor shall become familiar with all license conditions and procedures concerned with radiation safety, nuclear safety, industrial safety, and nuclear materials safeguards. He may consult with the following personnel to ensure compliance with all safety regulations and principles:

Supervisor, Health and Safety

Nuclear Safety Officer

Industrial Safety Officer

Accountability Specialist

11.1.12 Safety Review Committee - The Safety Review Committee (SRC) shall be comprised of at least five technically trained and experienced members appointed by the Manager, EC&RR. One member shall be selected by the Manager, EC&RR to be the SRC Chairman. The Chairman shall preside at the meetings and keep the minutes. The Manager, EC&RR shall appoint an Alternate Chairman who shall act for the Chairman during absences. One member shall be appointed by the Manager, EC&RR to be the SRC Coordinator. The Coordinator shall represent site management on the SRC, set the meeting agenda, and maintains the permanent files of the Committee.

The SRC membership shall have expertise in chemistry, nuclear physics, health physics, and the safe handling of radioactive material. The SRC membership shall have a general understanding of nuclear criticality safety as it pertains to site operations. Consultants with special expertise are available to the Committee when needed.

The SRC shall meet at least four times a year. A quorum shall consist of a simple majority of the membership including the Chairman. The SRC shall review and approve all Area Operating Procedures. It shall review and approve new projects that utilize licensed material that are significantly different from previously reviewed and approved projects. The SRC shall review the annual report issued by the Supervisor, Health and Safety which summarizes site workers' exposures, environmental releases, and a summary of the ALARA program accomplishments. The SRC Chairman shall forward the Committee minutes to the Manager, EC&RR, with a copy to the SRC Coordinator.

The Manager, EC&RR shall appoint the members of the Safety Audit Sub-committee (SAS). The SAS shall be comprised of at least two individuals, one of whom shall be designated as Chairman and he shall report to the Chairman, SRC. The SAS shall audit site operations at least three times annually, with successive audits separated by at least two months. Additional audits may be performed at any time. The SAS Chairman shall develop the audit report and submit it to the SRC Chairman. The SRC Chairman shall submit the audit report to the Manager, EC&RR with appropriate comments, with a copy to the License Administrator.

11.2 EDUCATION AND EXPERIENCE OF KEY PERSONNEL

11.2.1 Safety and Licensing Manager - Richard L. Bennett

Education:

B.Ch.E. - Chemical Engineering, University of Delaware, 1958

Experience:

(1985-Present) Babcock & Wilcox, Manager, Safety and Licensing, Lynchburg Research Center, Lynchburg, Virginia.

See Section 11.2.1

(1982-1985) Babcock & Wilcox, Manager, Building C Decommissioning, Lynchburg Research Center, Lynchburg, Virginia

He was responsible for decontaminating facilities that were used for preparation of experimental quantities of nuclear fuels containing plutonium.

(1973-1982) Babcock & Wilcox, Supervisor, Process Technology Group, Lynchburg Research Center, Lynchburg, Virginia

This group was responsible for long-range studies, design assistance, start-up assistance, and preparation of environmental reports and safety analyses related to nuclear fuel conversion. Some of the specific projects performed by the group were preparation of the designs for a low-enriched nuclear fuel conversion plant, preparation of a conceptual design for a spiked nuclear fuel fabrication plant, process engineering assistance to nuclear fuel conversion plants, development of a halide volatility scrap recovery process, development of alternative effluent treatment systems for various nuclear fuel conversion processes, and evaluation of fabrication methods for advanced fuels.

(1971-1973) Babcock & Wilcox, Senior Research Engineer, Lynchburg Research Center, Lynchburg, Virginia

He was responsible for the conceptual design of a facility to treat the effluent from a nuclear fuel plant and developing and evaluating processes for recovering byproducts from B&W wastes.

(1959-1971) American Cyanamid Company, Process Engineer, Piney River, Virginia

He has had broad experience in chemical engineering. This includes research and development, designing equipment and processes, testing and operating new equipment, pilot plant operation, process engineering, and economic evaluation. He has specific knowledge in pigment manufacture, effluent treatment, and byproduct recovery.

Professional Affiliations:

American Institute of Chemical Engineers (Member)
American Nuclear Society (Member)

11.2.2 Supervisor, Health and Safety - Gary S. Hoovler

Education:

B.S. - Nuclear Engineering, University of Virginia
M.S. - Nuclear Engineering, University of Virginia
- 15 hours toward Master of Engineering Administration, George Washington University
- Respiratory Protection for Nuclear Industry

Experience:

(1986-Present) Babcock & Wilcox, Supervisor, Health and Safety, Lynchburg Research Center, Lynchburg, Virginia

Mr. Hoovler is responsible for the Health Physics and Industrial Safety functions at the Lynchburg Research Center; reporting to the Manager, Safety and Licensing.

Mr. Hoovler is responsible for assuring that all radioactive materials at the LRC are properly handled, labeled, and stored. He is responsible for the proper packaging and shipping of radioactive materials, and for radioactive waste disposal. He is responsible for the proper use, storage and disposal of other hazardous materials at the LRC.

Mr. Hoovler is responsible for establishing, maintaining and administering training programs in health physics and industrial safety for employees. He is responsible for reviewing all Area Operating Procedures, Radiation Work Permits, and all Technical Procedures that apply to the Health Physics and Industrial Safety operations. He is a member of the Safety Review Committee.

1984-1986) Babcock & Wilcox, Project Manager, Decommissioning, Lynchburg Research Center, Lynchburg, Virginia

Mr. Hoovler was the Project Manager for the Decommissioning Program reporting the the Director, LRC.

He developed methods and directed activities for decontamination and survey of Building A and the Critical Experiment Facility, which together compromise an area of 14,000 square feet. He was also responsible for the development of methods and directing the activities for the decontamination and survey of the 20,000 square foot Plutonium Development Laboratory. The objective of both projects was for their release for unrestricted use and designation as non-use areas for NRC-licensed materials.

(1981-1983) Babcock & Wilcox, Supervisor, Radiation Experiments Group, Lynchburg Research Center, Lynchburg, Virginia

Mr. Hoovler was Supervisor of the Radiation Experiments Group in the Nuclear Physics Section. He helped to develop the soil assay method used in the plutonium decontamination project, worked on the EPRI Radiation Control Program, and performed experimental work developing radiation gauges for several company applications.

(1976-1981) Babcock & Wilcox, Research Experiments Group, Lynchburg Research Center, Lynchburg, Virginia

Mr. Hoovler worked in the Reactor Experiments Group of the Nuclear Physics Section. He joined B&W in 1976 as a research engineer. In 1978, after receiving his Senior Reactor Operator's License, he was appointed as the CX-10 Operations Supervisor and promoted to Senior Research Engineer in 1980.

He worked on the Department of Energy's Spent Fuel Storage critical experiment program and served on ANS Working Group 15.1.

He worked on the neutron spectrum unfolding code SAND II, eddy current analysis, and on experiments to demonstrate the use of beryllium-gold as a passive technique for measuring fission product distribution in spent fuel.

11.2.3 Senior Health Physics Engineer - Steven W. Schilthelm

Education:

B.S. - Nuclear Engineering, University of Wisconsin, Madison, 1983
M.S. - Health Physics, University of Wisconsin, Madison, 1985
- Domestic & International Shipping of Radioactive Material.

Experience:

(1985-Present) Babcock & Wilcox, Senior Health Physicist, Lynchburg Research Center, Lynchburg, Virginia

Mr. Schilthelm is responsible for administering the Health Physics Program at the Lynchburg Research Center. His duties include external and internal exposure control, shipping and receiving of radioactive material, maintaining the respiratory protection program, preparation and presentation of radiological safety training courses, maintaining the support for licensed activities.

Mr. Schilthelm is the Emergency Radiological Safety Officer and is the designated alternate for the position of Supervisor, Health and Safety.

(1984-1985) Research Specialist, Synchrotron Radiation Center, University of Wisconsin, Madison, Wisconsin

Mr. Schilthelm was responsible for Radiation Surveys and subsequent shielding calculations and design at the 800 Mev electron accelerator/storage ring. He co-authored a shielding upgrade proposal that was presented to the National Science Foundation, and he provided the experimental basis for the proposal. Mr. Schilthelm presented a paper at the 1985 Health Physics Society meeting, entitled "Radiation Survey Measurements at the Aladdin Synchrotron Light Source."

Professional Affiliation:

American Nuclear Society (Member)
Health Physics Society (Member)

11.2.4 Industrial Safety Officer - Reginald R. Spradlin

Education:
- Graduate, Appomattox County High School
- Certified Instructor Trainer, Basic Cardiac Life Support, American Heart Association
- Certified Instructor, First Aid & Advanced First Aid, American Red Cross
- Training in the following areas:
  Industrial Safety
  Fire Fighting
  Rescue
  Extrication
  Fire Protection
  Fire Extinguishing Equipment and Materials
  Arson Investigation.

Experience:

(1972-Present) Babcock & Wilcox, Industrial Safety Officer, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin is the LRC's Industrial Safety Officer. As such he is responsible for compliance with the regulations of the Occupational Health and Safety Administration. He advises the LRC on the standards and requirements of the National Fire Protection Association and performs reviews of equipment and systems for compliance with NFPA standards. He performs inspections of facilities and equipment for fire protection purposes. He reviews facility changes and modifications to ensure fire safety. Mr. Spradlin performs tests, maintenance, and inspection of fire protection, control and extinguishing equipment. He is responsible for investigating all accidents, and keeping his management informed of safety activities. He performs fire and rescue training for the members of the LRC's Fire and Rescue Team, and serves as the Captain of the team. He is a certified Shock Trauma Technician, an Emergency Medical Technician, and certified instructor in CPR and Standard and Advanced First Aid.

(1971-1972) Babcock & Wilcox, Accountability Technician, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin served as the Accountability Technician. In this capacity he was responsible for the recordkeeping system for SNM accountability in the Plutonium Development Laboratory. He recorded all transfers of SNM, performed inventories, and updated the unit log records.

(1969-1971) Babcock & Wilcox, Health Physics Technician, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin was a health physics technician in the Plutonium Development Laboratory. He was responsible for performing contamination surveys of the facility, assisting in the monitoring of bagging operations, and supervising decontamination. He implemented the surveillance program for airborne radioactive material. He performed maintenance, testing, and calibration of alpha particle survey instrumentation and counting equipment. He implemented the respiratory protection program in that laboratory.

(1967-1969) Babcock & Wilcox, Plant Engineering Technician, Lynchburg Research Center, Lynchburg, Virginia

As a plant engineering technician, Mr. Spradlin performed installation, modification, and repair of facilities, equipment, and experimental apparatus at the LRC. He performed these duties on electrical, mechanical and plumbing systems.

(1952-1967) Mead Corporation, Maintenance Superintendent, Mead Paper Company, Lynchburg, Virginia

Mr. Spradlin served in several capacities during this period, including: finishing operation, paper machine operation, Millwright, Maintenance Foreman, Maintenance Superintendent, Safety Inspector and Accident Investigator.

Professional Affiliations:

Concord Rescue Squad - Founding President
American Heart Association - Cardiac Care Committee

11.2.5 Accountability Specialist - Kenneth D. Long

Education:

Graduate - White Sulphur Springs High School, 1958
Certificate - Bookkeeping, Central Virginia Community College, 1983

Experience:

(1974-Present) Babcock & Wilcox, Accountability Specialist
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long, as the Accountability Specialist, is responsible to the Manager of Safety and Licensing for the accurate accounting of all Special Nuclear, Source, and Byproduct material at the LRC. He is responsible for recording all transfers of SNM that are made within the LRC and for preparing the reports and records of off site transfers. He prepares all NRC/DOE 741 Transaction Forms. He is responsible for the timely completion of inventories of licensed material. He initiates the paper work required for all shipments of licensed material.

In addition to his normal duties he is a Document Custodian. In this capacity, he is responsible for the safe storage of all classified DOE and DOD documents at the LRC. He is also an authorized classifier and an authorized courier of classified material.

(1970-1974) Babcock & Wilcox, Shipping & Receiving Clerk
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long was responsible for the shipment and receipt of all materials at the LRC. This assignment included the processing of all the necessary forms and documents used for shipping and receiving licensed materials as well as the many items that are required for operation of a research and development laboratory.

(1967-1970) Babcock & Wilcox, Technician
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long was a technician in the Plutonium Development Laboratory during this period. He performed chemical operations utilizing uranium and plutonium materials and was responsible for the accountability of SNM materials into and out of his area.

Professional Affiliations:

Institute of Nuclear Materials Management (Senior Member)
Nuclear Materials Control Committee, B&W (Secretary)
American Nuclear Society, Virginia Chapter (Member)

11.2.6 License Administrator - Arne F. Olsen
Facility Supervisor - Arne F. Olsen

Education:

AAS - Nuclear Technology, Central Virginia Community College, 1978

Experience:

(1972-Present) Babcock & Wilcox, Senior License Administrator and Facility Supervisor, Lynchburg Research Center, Lynchburg, Virginia

Mr. Olsen is responsible for preparing, amending, and administering the licenses that the LRC possesses with the NRC and the Commonwealth of Virginia. He acts as the primary liaison between the LRC and the NRC and other federal, state, and local agencies regarding nuclear matters. He coordinates the visits made by the NRC's Office of Inspection and Enforcement, and coordinates the LRC's compliance with NRC and state regulations and the licenses. He is the coordinator of the Safety Review Committee and is Chairman of the Safety Audit Subcommittee, and represents LRC management on both. Mr. Olsen is the Facility Supervisor and as such is responsible to the Manager, Lynchburg Technical Operations for the safety of all operations at the LRC.

Mr. Olsen is the Alternate LRC Security Officer, Alternate Emergency Officer and an internal auditor.

(1968-1972) Babcock & Wilcox, Health Physics Technologist, Lynchburg Research Center, Lynchburg, Virginia

In this capacity, Mr. Olsen was responsible to the site Health Physicist (Supervisor, Health and Safety) for the implementation of the Health Physics Program in the Plutonium Development Laboratory. This responsibility included the implementation of the smearing, survey, air sampling, environmental sampling, and waste disposal programs.

(1964-1968) Babcock & Wilcox, Technician and Shift Leader, Babcock & Wilcox Test Reactor, Lynchburg Research Center, Lynchburg, Virginia

Mr. Olsen possessed a Senior Reactor Operator's License for the BAWTR. He was in charge on one of four shifts of reactor operators charged with the proper operation and maintenance of the BAWTR. He supervised the loading and unloading of fuel and experiments in the reactor and kept all required records of operations and maintenance performed on his shift.

(1960-1964) U. S. Navy, Reactor Plant Electrical Supervisor, USS Enterprise CVA(N)-65

Mr. Olsen was an Electrician, First Class and was responsible for the proper operation and maintenance of all electrical equipment serving one of the reactor plants aboard the Enterprise.

Professional Affiliation:

Health Physics Society (Member)
Site Environmental Committee, B&W (Member)

11.2.7 Nuclear Criticality Safety Officer - Francis M. Alcorn

Education:

B.S. - Nuclear Engineering, North Carolina State College, 1957
M.B.A - Business Administration, Lynchburg College, 1974
- Graduate study in Nuclear Engineering, University of Virginia

Experience:

(1971-Present) Babcock & Wilcox, Supervisor, Nuclear Criticality Safety Group, Lynchburg Research Center, Lynchburg, Virginia

This group is the Company's central organization which provides guidance, develops and validates the analytical methods needed for criticality evaluations, does criticality calculations, performs nuclear safety audits, and gives assistance to the various divisions of the Company and the Company's customers in matters related to nuclear criticality safety. In addition to his responsibility as supervisor of this group, he is the Nuclear Safety Officer for the Lynchburg Research Center.

(1969-1971) Babcock & Wilcox, Criticality Specialist, Nuclear Safety Engineer, Lynchburg Research Center, Lynchburg, Virginia

Transferred to the LRC as Nuclear Criticality Safety Specialist for Babcock & Wilcox's Naval Nuclear Fuel Plant, Commercial Nuclear Fuel Plant, and the LRC. He was appointed Nuclear Safety Officer for the LRC.

(1964-1969) Babcock & Wilcox, Power Generation Division, Lynchburg, Virginia

Mr. Alcorn was a physicist in the PWR Development Section and was responsible for determining the most economical method for utilizing plutonium as a recycle fuel in B&W's pressurized water reactor concepts. In addition, he was Nuclear Criticality Safety Advisor to the Company's Naval Nuclear Fuel Division.

(1961-1964) Babcock & Wilcox, Nuclear Power Generation Division Lynchburg, Virginia

He has been concerned with core neutron physics analysis and design of the Consolidated Edison Reactor, the Liquid Metal Fuel Reactor, the Babcock & Wilcox Test Reactor, the Advanced Test Reactor, the Heavy Water-Organic Cooled Reactor Concept, and Babcock & Wilcox Pressurized Water Reactor Concepts. He developed methods for and performed calculations for criticality, fuel depletion, nuclear safety coefficients, power profiles, nuclear fuel costs and critical experiment analysis. He has also worked in the areas of kinetic safety analysis.

(1957-1960) Babcock & Wilcox, Atomic Energy Division Lynchburg, Virginia

He functioned as a nuclear engineer doing both core neutron physics and shielding calculations.

(1960-1961) General Nuclear Engineering Corporation, Staff Physicist

Mr. Alcorn engaged in core neutron physics design and analysis of the Boiling Nuclear Superheat Reactor. He also wrote physics articles for Power Reactor Technology which were published by GNEC for the AEC.

Professional Affiliations:

Sigma Pi Sigma (Member)
Tau Beta Pi (Member)

American Nuclear Society - Past Chairman of ANS Nuclear Criticality Safety Division
- Member Standards Subcommittee ANS-8.

11.3 PROCEDURES

11.3.1 Area Operating Procedures (AOP) - All operations with licensed material shall be conducted in accordance with Area Operating Procedures or a Radiation Work Permit. Area Operating Procedures are prepared by any technically competent person. The proposed procedure is delivered to the Facility Supervisor who ensures that the procedure is in the proper format. The Facility Supervisor routes the procedure to the Nuclear Criticality Safety Officer who reviews it to assure that any nuclear criticality safety issues are properly addressed. If the Nuclear Criticality Safety Officer has additions or corrections, he notes them on the procedure and forwards it to the Supervisor, Health and Safety. If the Nuclear Criticality Safety Officer approves it, he signs the procedure in the space provided and forwards it to the Supervisor, Health and Safety. The Supervisor, Health and Safety reviews it for proper radiological and industrial safety content. If he has additions or corrections, he notes them on the procedure and forwards it to the Facility Supervisor. If the Supervisor, Health and Safety approves the procedure, he signs the procedure in the space provided and forwards it to the Facility Supervisor. The Facility Supervisor reviews it for general safety and determines its impact on other work and facilities. The Facility Supervisor is responsible for resolving all additions or changes recommended by the previous reviewers. When the procedure is approved by the three reviewers, the Facility Supervisor forwards it to the Safety Review Committee. The Safety Review Committee (SRC) may approve the procedure as written, approve the procedure conditionally with specific changes to be made prior to issuance or the SRC can disapprove it. The SRC coordinator signs for the SRC when approval is voted. The procedure may be implemented subsequent to SRC approval.

Revisions to AOP's will follow this same approval route, except that the revised procedure may be implemented after receiving the approval signatures of the Nuclear Criticality Safety Officer, Supervisor, Health and Safety, and the Facility Supervisor. The revised procedure will be placed on the agenda for the next regularly scheduled meeting of the SRC. AOP manuals shall be placed in areas where the procedures apply.

11.3.2 Technical Procedures - Technical procedures provide detailed technical standards and instructions for performing specific tasks. Pursuant to this license application, they are not intended for use by operations personnel and are not distributed in the same manner as AOP's. Neither are they necessarily approved by the Safety Review Committee.

Technical procedures for the Health and Safety Group and the Nuclear Criticality Safety Group are reviewed and approved by the Senior Health Physics Engineer and the Nuclear Criticality Safety Officer, respectively, or by their designated alternates. The distribution list for each procedure is specified in the procedure.

11.4 TRAINING

11.4.1 General Radiation Protection Training

The site provides two training programs covering the nature, use and control of radiation, and radioactivity. These courses are presented to ensure that all site personnel receive training appropriate to their activities and to fulfill obligations under the NRC license to provide such training.

The courses consist of a series of lectures intended to present the proper background and technical base to allow workers to understand the principles of radiation safety. The Supervisor, Health and Safety administers the course and, in general, teaches each course. Where practical, basic general procedures and federal regulations are included and discussed. Training aids, such as motion pictures and self-study materials, are used as appropriate.

Program 1 is intended for site workers and non-site workers who will be authorized access to the restricted area. Program 2 is intended for site and non-site workers who may enter the restricted and controlled areas but who will not be permitted to work with licensed material without supervision. Program 3 is intended for authorized users (those who will be authorized to work with licensed material and to supervise such work).

Training in area operating procedures and special area procedures is the responsibility of the Area Supervisor. This training should be accompanied with appropriate formal and on-the-job training as the job requirements dictate.

11.4.2 Program 1

This course is presented to site workers and non-site workers who will be granted access to the restricted area but who will not be granted unescorted access to the controlled areas. The course provides an introduction to radiation and radioactivity (understandable to a non-technical person) and a thorough coverage of safety rules and procedures, including the site emergency procedures. Subjects include the types of radiation, ALARA, radiation effects on humans, decontamination procedures, radiation exposure to females, warning signs, basic health physics rules, a history of radiation protection, worker's rights and responsibilities, and health physics terms.

11.4.3 Program 2

This course is presented to site workers and non-site workers who will be granted unescorted access to the restricted area and controlled areas but who will not be permitted to work with radioactive materials without supervision. This course is intended to provide the workers with a knowledge of the hazards of working in radiation and controlled areas and ways to minimize their dose. Subjects include types of radiation, radiation exposure limits, ALARA, personnel dosimetry and its use, dose calculation, biological effects, radiation exposure to females, radiation protection measures, warning signs and labels, radiation work permits, emergency procedures, rights and responsibilities of workers, and health physics terms.

11.4.4 Program 3

This course is presented to site workers and non-site workers who will be granted unescorted access to the restricted area and controlled areas and will be permitted to work with radioactive materials and supervise such work. This course is intended for meeting the requirements for designation of a worker as an authorized user. Subjects include fundamentals of radiation, external and internal radiation protection, biological effects, radiation detection, instrumentation, contamination control, license requirements, site organization, rights and responsibilities under 10 CFR 19, ALARA, dose calculation, personnel dosimetry requirements and use, posting and labeling, and health physics terms.

11.4.5 Respiratory Protection Training

Training in respiratory protection techniques will be required of all workers before the use of such equipment will be allowed. This training will be carried out by a qualified individual, as defined in NUREG-0041 (Section 12.1), who will document that such training as been completed. Those persons who direct the work of workers using respiratory protection will be included in the training courses. Biennial retraining will be scheduled, at the discretion of the qualified individual, to ensure that a high degree of proficiency in the use of respiratory protective devices is maintained.

Training in respiratory protection shall include the following subjects:

a. Discussion of the airborne contaminants present in the work environment including their physical properties, physiological actions, toxicity, means of detection, and maximum permissible concentrations (MPC's).

b. Discussion of the importance of selecting the proper respirator based on the hazard and the dangers of using respirators for a purpose other than that intended.

c. Discussion of the construction, operating principles, and limitations of the available respirators.

d. Discussion of the use of engineering controls as a substitute for respiratory protection and the need to make every reasonable effort to reduce or eliminate the need for respiratory protection.

e. Instruction in methods to be used to determine that the respirator is in proper working order.

f. Instruction in fitting the respirator properly, field testing for proper fit, and factors that may influence a proper fit.

g. Instructions in the proper use and maintenance of the respirator.

h. Discussion of the uses of various cartridges and canisters available for air-purifying respirators.

i. Review of radiation and contamination hazards, including a review of other protective equipment that may be used with respirators.

j. Instruction in emergency actions to be taken in the event of respirator malfunction.

k. Classroom instruction to recognize and cope with emergency situations while working with a respirator.

l. Any additional training as needed for special use.

m. The wearer must pass a written examination on the material presented on respiratory protection.

11.5 FACILITY CHANGE

Changes and modifications to buildings, exhaust ventilation systems, gas supply systems, emergency electrical systems, etc. are requested on Form LRC-229, "Facilities Work Order Form" (Figure 9-4). All work orders are forwarded to the maintenance supervisor. The Plant Engineering Supervisor determines if the request involves a facility change. If a facility change is involved, the work order is forwarded to the Facility Supervisor. It is the Facility Supervisor's responsibility to determine that all safety and licensing considerations have been addressed and if the request must be approved by the Safety Review Committee. Space is provided on the form for the approval signatures of the Supervisor, Health and Safety, the Industrial Safety Officer, and the Facility Supervisor.

Completed forms are kept on file by the maintenance supervisor and are audited once a month by the Health Physics Group.

FIGURE 11-1

NNFD RESEARCH LABORATORY

ORGANIZATION

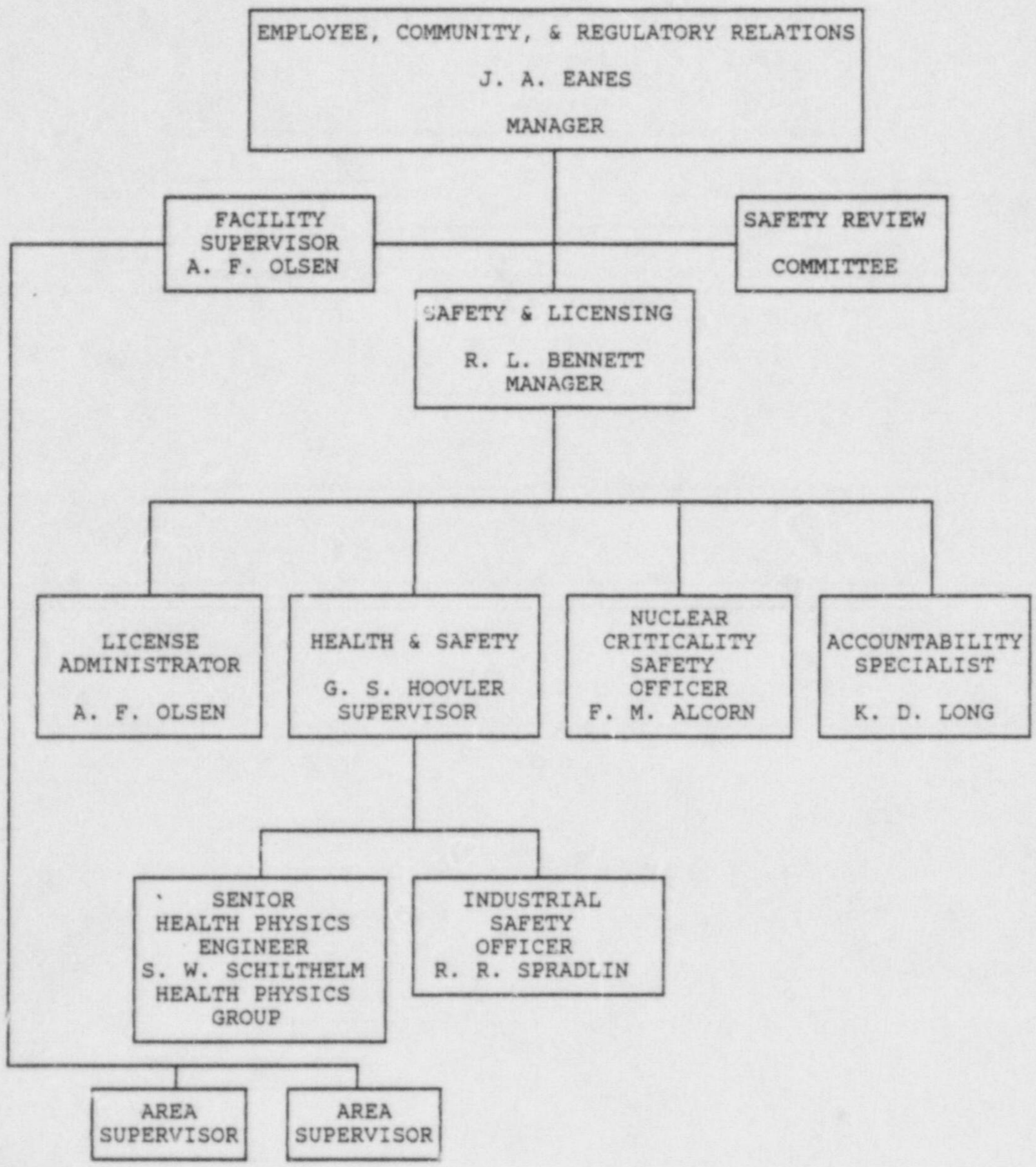

FIGURE 11-2

LRC 229

FACILITIES WORK ORDER FORM

TO: Plant Engineering Date ______

From: ______ Section: ______ Signed: ______

Section Mgr.: ______ Date: ______

Date Required: ______ Charge No.: ______ (Labor) ______ (Material)

DESCRIPTION OF WORK TO BE DONE

______

______

______

______

______

SIGNATURE REQUIRED: Industrial Safety Officer: ______

Health Physics: ______ Facility Supervisor: ______

Space Below This Line For Plant Engineering Use Only

Order Received Date: ______ Signed: ______

Planned Starting Date: ______ Planned Completion Date: ______

Order Completed: Work Order Number

Date: ______ Signature: ______

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12.0 RADIATION PROTECTION

12.1 PROGRAM

The radiation protection program at the site is implemented to protect employees and the general public from the harmful effects of radiation and radioactive material, to comply with NRC regulations, and to maintain personnel exposures as far below the limits established by the NRC as is reasonably achievable.

Implementation of the program requires the active participation of all personnel who work with licensed material or in areas were licensed material is handled. To support the worker, the site has established the Health and Safety organization and vested it with the authority and resources necessary to meet the program goals.

12.2 POSTING AND LABELING

Many areas in the site are required to be posted to indicate the hazard present. This posting is required by the federal regulations and is a fundamental part of an effective radiation protection program. Posting of areas makes the workers aware of the potential hazards in the area and assists workers in keeping their exposures ALARA. Permanent postings are the responsibility of the Health and Safety Group. Temporary postings are the responsibility of Authorized Users. This section discusses the posted areas at the site. Persons not directly familiar with conditions existing in a posted area shall contact the area supervisor prior to entering and shall enter only under his direction.

12.2.1 Radioactive Materials Area - Any area where radioactive materials are stored, handled, or processed in amounts exceeding 10 times the quantities specified in 10 CFR 20, Appendix C is designated a radioactive materials area. Each area is clearly marked at every normal entry with a sign bearing the radiation caution symbol and the words - CAUTION - RADIOACTIVE MATERIAL(S). Monitoring equipment and protective clothing required for use in the area will be specified by the Health and Safety Group.

12.2.2 Contamination Area - This is any area in which loose contamination is present in quantities in excess of those specified in Table 12-24 or an area designated by the Health and Safety Group as one in which there is a risk of contamination. Each contamination area is clearly marked at every normal entry. Work in these areas may require a Radiation Work Permit. Protective clothing, respiratory protection, and personnel monitoring devices required for entry into these areas must be specified by the Health and Safety Group. Entry into the area without the prescribed equipment is prohibited. When exiting a contamination area, workers must remove the protective clothing and monitor himself in accordance with established procedures.

12.2.3 Radiation Area - A Radiation Area is an area in which an individual could receive a radiation exposure to a major portion of the body greater than 5 mRem in 1 hour or 100 mRem in 5 consecutive days. Each radiation area is clearly marked at every normal entry with a sign bearing the radiation caution symbol and the words - CAUTION - RADIATION AREA. Work in these areas may require a Radiation Work Permit. Personnel monitoring devices and protective clothing to be worn in the area will be specified by the Health and Safety Group.

12.2.4 High Radiation Area - Any area in which an individual may receive an exposure to a major portion of the body greater than 100 mRem in 1 hour is a High Radiation Area. High radiation areas are designated by a sign at each normal entrance bearing the radiation caution symbol and the words - CAUTION - HIGH RADIATION AREA. Entry into high radiation areas is limited to qualified persons, or under the direct supervision of a qualified person and, working under an approved radiation work permit. Protective clothing, protective equipment, and personnel monitoring devices appropriate for the area will be specified by the Health and Safety Group and must be worn. When protective clothing is required, each person must remove the protective clothing and monitor himself in accordance with established procedures, when exiting the area.

12.2.5 Airborne Radioactivity Area - This is an area in which airborne radioactivity concentrations could exceed the maximum permissible concentration limits given in 10 CFR 20, Appendix B or in which the concentration of airborne radioactivity averaged over the number of hours individuals are in the area could exceed 25% of the limits given in 10 CFR 20, Appendix B. Each area is clearly designated by a sign at each normal entrance bearing the radiation caution symbol and the words - CAUTION - AIRBORNE RADIOACTIVITY AREA. Entry is limited to those qualified persons classified as radiation workers, working under an approved radiation work permit. No entry is permitted until an appropriate area survey has been made and a member of the Health and Safety Group is present. Protective clothing, protective equipment, and personnel monitoring devices to be worn in the area will be specified by the Health and Safety Group and must be worn. When exiting these areas, each person must remove the protective clothing and monitor himself in accordance with established procedures.

12.3 EXTERNAL RADIATION - PERSONNEL MONITORING

12.3.1 Administrative Exposure Control - Limits for external radiation exposure are set forth in 10 CFR 20.101 and these general limits are used at the site. The applicable exposure limits to be used for operations at the site are:

1. Whole body - 300 mRem/week (with long-term exposure controlled within the 1.25 Rem/quarter limit by the worker's immediate supervisor)

2. Skin of the whole body - 1.5 Rem/week

3. Hands and forearms, feet and ankles - 3.0 Rem/week.

The Manager, Safety and Licensing has the authority to approve whole body exposures up to, but not exceeding, 3.0 Rem/calendar quarter. In emergencies, the Emergency Officer is authorized to allow personnel exposures to the whole body of up to 3.0 Rem/calendar quarter. Higher exposures may be authorized by the Emergency Officer in accordance with the Radiological Contingency Plan.

12.3.2 Personnel Monitoring for Site and Non-site Workers - All site and non-site workers will be issued a film badge, a SRD, and a TLD. This dosimetry will be worn by the workers when they are in the restricted area. When the workers leaves the restricted area they will place their dosimetry on a rack provided for this purpose.

12.3.3 Visitor Monitoring and Escort Requirements - Visitors to the restricted area will be issued a TLD. This dosimetry will be worn by the visitor when they are in the restricted area and will be surrendered to the receptionist when they depart the site. Visitors must be escorted by a site worker when in the restricted area.

12.3.4 Monitoring Devices

The primary device used for monitoring exposure on site is the film badge. The exposure measured by this badge (reported in units of dose equivalent) becomes a part of the workers permanent exposure record. Films are changed monthly and are mailed off-site for evaluation. In some cases, a Health Physics Engineer may choose to base the monthly exposure of an employee on the monthly thermoluminescent dosimeter (TLD). This determination shall be recorded in the employees exposure record.

In general, the worker should wear the dosimeters on the portion of the whole body expected to receive the highest dose (with the exception of extremity dosimetry issued in special cases). The film badge and/or monthly TLD badge should always be worn in the proper orientation to ensure that exposure to non-penetrating radiation (e.g., beta radiation) is recorded. For cases in which the exposure may vary significantly within a small area, several badges may be worn to ensure that the maximum whole body dose is measured. In this context, whole body includes the head, lens of the eyes, the gonads, the upper legs above the knees, and the upper arms above the elbows.

12.3.4.1 Pocket Dosimeters - These dosimeters are small, air-filled ionization chambers used to provide a check of the daily exposure of workers and to ensure that the administrative limit for weekly exposure is not exceeded. Indirect dosimeters are capable of measuring external exposure to gamma radiation in the range 0 to 200 mR (other ranges are also available). These dosimeters are read, recorded, and rezeroed daily. Daily readings are used also as an indication of the need to evaluate the primary dosimeter before the normal exchange period.

Some workers may be issued self-reading pocket dosimeters (SRD). These dosimeters do not require reading and recharging on a daily frequency and the worker may evaluate his accumulated exposure without the need for a special reading device. Workers are encouraged to read their self-reading dosimeters at least on a daily basis. These dosimeters are capable of measuring external exposure to gamma radiation in the range 0 to 200 mR, but other ranges are available.

12.3.4.2 Film Badges - These dosimeters are the primary monitoring device used on site, i.e., the film badge results are entered in the employee's permanent exposure record. Film badges monitor external exposure to beta and gamma radiation typically in the range 15 mRems to 500 Rems. For situations in which neutron exposure is probable, film packets sensitive to neutrons also are used.

Films in use on site are changed monthly and mailed to an off-site dosimetry service for processing (reading, recording, and reporting).

12.3.4.3 Thermoluminescent Dosimeters (TLD) - TLD's are small, solid-state dosimeters capable of measuring external exposure from beta and gamma radiation in the range 10 mRems to 10,000 Rem. The monthly TLD's are used to duplicate the readings of the film badge. These badges are also changed monthly and mailed off-site for processing.

At the discretion of a Senior Health Physics Engineer, persons handling radioactive materials may be issued extremity dosimeters. These dosimeters are small TLD chips attached to a ring and are to be worn on the fingers. TLD "finger rings" are capable of measuring external exposure to beta and gamma radiation in the range 10 mRems to 10,000 Rems. These dosimeters are evaluated on a frequency established by the Health and Safety Group.

12.4 DIRECT RADIATION SURVEYS

Surveys of the direct radiation exposure in areas on site are to be performed on a frequency established by the Health and Safety Group. In general, these surveys require the selection of the appropriate portable survey instruments based upon the anticipated radiation levels, the types of radiation expected, and the nature or type of survey to be performed. Survey maps of the areas to be surveyed may be used to record the measured ambient radiation levels and/or, in some cases, to designate specific areas in which the exposure rates should be measured. The survey should also include a visual examination of the area for any unusual conditions or work habits which could affect the exposures received by personnel working in these areas. Items of this nature should be reported immediately to a Health Physics Engineer or corrected immediately, if practical.

Results of these surveys should be reviewed by a Health Physics Engineer to ensure that the proper posting requirements are in effect for the area and to ensure that appropriate actions are taken to keep all exposures ALARA.

12.5 REPORTS AND RECORDS

The following records will be maintained by the Health and Safety Group for the periods indicated.

Health and Safety Supervisor audits	2 years
Shipping and receiving RM forms	5 years
Waste disposal records	(*)
Personnel dosimetry records	(*)
Results of Bioassays and Whole Body Counting	(*)
Releases to the environment	(*)
Radiation survey data	2 years
Contamination survey data	2 years
Radiation Work Permits (completed)	5 years
Radiation detection instrument calibration	2 years
Leak tests of sealed sources	2 years
Worker training	(*)
Worker retraining	(*)
Airborne radioactivity sampling data	(*)
NRC-4 forms	(*)
NRC-5 forms	(*)

* - indicates that the record will be retained until the NRC authorizes its disposition.

12.6 INSTRUMENTS

12.6.1 Types - The commitment of site management to an effective radiation protection program includes the obligation to provide the adequate equipment and supplies for such a program. It is the responsibility of the Manager, Safety and Licensing and the Supervisor of Health and Safety to ensure the appropriate radiation protection instrumentation is available for use on site. In addition, the Health and Safety Group has the responsibility to ensure that this instrumentation is used properly, and is calibrated, maintained, and repaired as necessary. Minimum instrumentation requirements for maintaining an effective radiation protection program are listed in Tables 12-1 and 12-2. Other specialized instrumentation may not be included in this list. However, the exclusion of these instruments does not imply that their availability does not enhance the effectiveness of the radiation protection program.

TABLE 12-1

PORTABLE RADIATION PROTECTION INSTRUMENTATION

Instrument	Radiation Sensitivity	Range	Window Thickness
Low-range GM	Beta, Gamma	Bkgd. to mR/hr	30 mg/sq. cm.
Intermediate range ion chamber	Beta, Gamma	mR/hr to R/hr	Beta: 1 mg/sq. cm.
High range ion chamber	Gamma	up to 500 R/hr	>100 mg/sq. cm.
Proportional counters	Alpha, Beta	Bkgd. to 500,000 cpm	1 mg/sq. cm.
Proportional counters	Neutron, fast and thermal	Bkgd. to 5000 mRem/hr	N/A
Portable air samplers	Air particulate collection only	N/A	N/A

TABLE 12-2

STATIONARY RADIATION PROTECTION INSTRUMENTATION

Instrument	Radiation Sensitivity	Range	Window Thickness
Laboratory proportional counter	Alpha, Beta	Bkgd. to 100,000 cpm	<1 mg/sq. cm.
Air particulate monitors	Alpha, Beta	Bkgd. up	<1 mg/sq. cm.

Stack particulate monitor	Alpha, Beta	Bkgd. to 1,000,000 cpm	<1 mg/sq. cm.
Stack gas monitor	Beta, gamma	Bkgd. to 100,000 cpm	30 mg/sq. cm.

12.6.2 Calibration - Portable survey instruments shall be calibrated twice annually using approved procedures and sources traceable to the National Bureau of Standards. In addition, frequent operational checks will be performed on survey instruments while in use. For example, Geiger-Mueller survey instruments always indicate the presence of radiation above the ambient background. This provides an indication that the instrument is functioning. Portable alpha survey instruments are equipped with check sources which can be used to ensure that the instruments are operating correctly. Portable ionization chamber survey instruments are not equipped with an internal check source and the user must make sure these instruments are functioning before making a radiation survey.

Fixed and stationary radiation monitoring equipment is calibrated on either a semi-annual or annual basis depending on the applicable manufacturer's recommendations and established health physics procedures. Operational checks are performed routinely by the Health Physics technicians on the laboratory counting equipment and "friskers" located at exits from selected areas on site.

12.7 PROTECTIVE CLOTHING

12.7.1 Clothing - The following is a list of protective clothing that is available for use by personnel during normal and maintenance conditions:

1. Laboratory coats
2. Coveralls
3. Shoe covers, treated fabric (reusable)
4. Shoe covers, plastic
5. Pants, plastic
6. Coats, plastic
7. Hoods, fabric (reusable)
8. Shields, spatter
9. Glasses, plastic
10. Glasses, glass
11. Gloves, plastic

12. Gloves, surgeons
13. Gloves, heat resistant
14. Coats, heat reflective
15. Hard-hats.

12.7.2 Emergency Clothing - In the event of an accident that requires special clothing or personnel protective equipment, the Fire and Rescue Team is provided with the following:

1. Hard-hats, heat resistant with face shields
2. Coats, flame resistant
3. Boots, high top rubber with steel toe shields
4. Gloves, chemical resistant

12.8 ADMINISTRATIVE CONTROL LEVELS

12.8.1 Internal Occupational Exposure

12.8.1.1 Plutonium bioassay action criteria.

TABLE 12-3

PLUTONIUM BIOASSAY ACTION CRITERIA

Bioassay Technique	Action Level	Action To Be Taken
Urinalysis	< 0.2 dpm/L	None
	> 0.2 dpm/L	1. Resample the individual within 5 working days. 2. The Supervisor, Health and Safety shall consider the need for worker restriction to prevent further exposure until the diagnostic evaluation is complete. Only the Supervisor, Health and Safety may lift any work restriction once it is imposed.

		3. If #1 is positive, investigate the cause and correct.
		4. If the exposure is confirmed by #1, investigate to determine how exposure was incurred and correct it. If the exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of their restriction.

TABLE 12-4

PLUTONIUM BIOASSAY ACTION CRITERIA

Bioassay Technique	Action Level	Action To Be Taken
In-vivo	< 1.6E-8 Ci Pu-239	None
	≥ 1.6E-8 Ci Pu-239	1. Restrict worker from further exposure.
		2. Resample the individual within 10 working days.
		3. Determine if area surveys support the analysis results.
		4. If area surveys confirm result, investigate the

cause and take corrective actions.

5. If the resample results do not confirm the exposure, the Supervisor, Health and Safety may lift the work restrictions.

6. If resample results confirm the exposure, the Supervisor, Health and Safety shall determine the organ dose.

7. If the exposure has exceeded 50% of the maximum permissible annual dose, the worker shall remain on a work restriction until the Supervisor, Health and Safety authorizes the removal of the restriction.

12.8.1.2 Uranium bioassay action criteria.

TABLE 12-5

URANIUM BIOASSAY ACTION CRITERIA

Bioassay Technique	Action Level	Action To Be Taken
a. Urinalysis	< 9 ug/L	None
b. Urinalysis	9-16 ug/L	1. Determine if area surveys support the analysis results.

			2. If #1 is positive, investigate and correct as needed.
			3. Make sure individual is in-vivo counted during the next time that the counting service is at the B&W site.
c. Urinalysis	> 16 ug/L		1. Restrict the worker from further exposure. Resample the individual within 5 working days.
			2. Determine if area surveys support the analysis results.
			3. If #2 is positive, investigate the cause and correct as needed.
			4. If exposure is confirmed by #2, investigate to determine how exposure was incurred and correct it. If the exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of this restriction.
d. In-vivo	< 30 ug U-235		1. None
e. In-vivo	30-120 ug		1. Determine if area surveys support the analysis results.

		2. If #1 is positive, investigate and correct as needed.
f. In-vivo	> 120 ug U-235	1. Resample the individual within 10 working days.
		2. Determine if area surveys support the analysis results.
		3. If #2 is positive, investigate the cause and correct as needed.
		4. If exposure is confirmed by #1, investigate to determine how exposure was incurred and correct it. If the exposure exceeds 120 ug, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of this restriction.

12.8.1.3 Beta-gamma activity - Workers who work in areas where beta-gamma internal exposure is likely (Hot Cells, Radiochemistry, Health Physics) shall be in-vivo counted at approximately annual intervals.

TABLE 12-6

FISSION PRODUCT ACTION CRITERIA

Analysis	Action Level	Action to be Taken
In-vivo	>10% MPOB	Remeasure subject to determine effective half life of the contaminant and plot decay curves.

		Followup program will continue until the contamination present is <5% MPOB or the effective half life has been determined.
Estimation from nasal smears or air sample	>10% MPOB	Submit in vitro sample for analysis within 5 working days.
In-vitro	>5% MPOB	Resample excreta to confirm presence of contamination and to establish rate of elimination. Perform isotopic analysis if >10% MPOB is a possibility.
In-vitro	>10% MPOB	In vivo measurement to be made as soon as practicable.

The Supervisor, Health and Safety shall be responsible for evaluations to determine the location and amount of deposition; to provide data necessary for estimating internal dose rates, retention functions, and dose commitments; and to determine whether work restrictions or referrals for therapeutic treatment are required for any case where a result indicating a greater than 10% MPOB deposition of a radionuclide is verified.

12.8.2 External Occupational Exposure - Personnel monitors (film badges, dosimeters, or other suitable devices) are provided to measure the radiation exposure of visitors and workers. Personnel dosimeters issued pursuant to 10 CFR 20.202 shall be read on a monthly basis.

The Area Supervisors are responsible for keeping exposures below 300 millirem per week and 1250 millirem per quarter. The Supervisor, Health and Safety may approve weekly exposures above 300 millirem, but the quarterly limit of 1250 millirem shall not be exceeded without the approval of the Manager, EC&RR. If a worker has received the quarterly limit and the Manager, EC&RR has not authorized exceeding the limit, the worker shall be restricted to prevent further exposure for the remainder of the quarter.

12.8.3 Airborne Activity

12.8.3.1 Air Monitoring Program - Air monitoring in operating areas of the site is accomplished with continuous monitors in predetermined, fixed locations. A monitor is placed in each radioactive materials handling area in which there is a potential for the release of airborne radioactivity. Locations are selected based upon the ability of the monitor to provide a reasonable evaluation of the airborne activity in a particular area and to provide adequate warnings to those in the area of changing conditions. The determinations are made by the Health and Safety Group based upon the operations in the area, the potential for release, the quantity and chemical form of the material.

Alarms are set in accordance with a particular operation, the material being handled, and the potential for release. Actual alarm points are set as low as possible commensurate with the ambient radiation levels in the area. Personnel are instructed through procedures and training to evacuate, up wind, if an air monitor alarms and to notify the Health and Safety Group. Re-entry is controlled by the Health and Safety Group.

12.8.3.2 Effluent Monitors - Potentially contaminated air from chemical hoods, hot cells, and glove boxes is discharged ultimately through the 50-meter stack. Generally, exhaust air containing beta-gamma activity is passed through a single-stage HEPA filter which is sufficient to remove airborne particulates. Air from more hazardous operations, e.g., from glove boxes, is passed through a two-stage HEPA filter.

Discharges through the stack are monitored with a sampling head located in the stack about 25 feet above the base. Air removed by the sampler passes through a fixed filter, into the chamber of the gas monitor, and is returned to the stack. The fixed filter is monitored continuously for alpha and beta activity by a gas-flow proportional counter. The second monitor, the gas monitor operates continuously utilizing a halogen-quenched GM tube. The stack monitor flow rate is maintained at a minimum of 2 cfm. Both monitors are equipped with adjustable alarms. The set points for these alarms are determined by the Health and Safety Group. The alarms are connected to an alarm panel located in the Health Physics Area in Building B. Alarms of the system are responded to by the Health and Safety Group. The alarm condition is first verified by the Health and Safety Group. If the alarm is actual, the exhaust fan is secured, operations personnel are advised to stop all operations with radioactive material, the cause is investigated by the Health and Safety Group, corrected by operations personnel, and the fan restarted.

TABLE 12-7

STACK RELEASE ACTION LEVELS

Release Product	Action Levels
Beta Particulate	200 uCi/week
Alpha Particulate (long lived)	1 uCi/2 weeks
Kr-85	70 Ci/week
H-3	3 Ci/week
I-131	200 uCi/week

12.8.4 Liquid Activity - Liquids containing radioactive material are discharged from the area where they are generated, to the Liquid Waste Disposal Facility. This facility is comprised of a series of tanks. All radioactive liquid waste is held in this facility for sampling prior to release. If the concentration of radioactivity exceeds 25% of the MPC values listed in Table I, Col. 2, of 10 CFR 20, Appendix B, the waste must be diluted to levels that meet this specification. Liquid waste is discharged to the liquid waste processing system at the NNFD. The NNFD must be notified and approve of each discharge from the site prior to discharge. No alarms are associated with this system because its operation is under the positive control of the Health and Safety Group.

12.8.5 Surface Contamination

12.8.5.1 Work Areas - The Health and Safety Group performs smear surveys in the work areas listed in Table 12-8. The frequencies specified in Table 12-8 are minimum frequencies. More frequent surveys are performed based on the level of work performed in the specified areas. Action is taken to protect personnel and reduce the levels of contamination below those specified. The Health and Safety Group will supervise and direct the protection and decontamination activities. Decontamination to reduce levels of contamination will commence within 24 hours of discovery. The Supervisor, Health and Safety shall evaluate and approve any delays on decontamination work that are longer than 24 hours.

TABLE 12-8

SMEAR SURVEYS IN WORK AREAS

Area	Frequency*	Action Level (dpm/100 cm2)
<------------------------ALPHA------------------------>		
Unirradiated, unencapsulated fuel handling areas	Weekly	5000
Building B Counting Lab.	Monthly	200
Hot Cell Oper. Area	Monthly	200
Scanning Electron Microscopy Lab.	Monthly	200
Exit portals from controlled areas	Biweekly	200
<------------------BETA + GAMMA------------------------>		
Building B Counting Lab.	Monthly	2000
Scanning Electron Microscopy Lab.	Monthly	2000
Hot Cell Operations Area	Bimonthly	2000
Cask Handling Area	Bimonthly	22000
Radiochemistry Lab.	Bimonthly	22000
Exit portals from controlled areas	Biweekly	2000

*Minimum frequency specified. More frequent surveys are performed, based on work loads.

Large area smears are used to survey many square meters of surface area. Action levels for large area smears are given below.

TABLE 12-9

ACTION LEVELS FOR LARGE AREA SMEARS

1. Routine Large Area Smears (1000-5000 dpm) - Repeat the large area smear. If results show levels of contamination above 1000 dpm, take smears in smaller areas to locate the source. Decontaminate all areas in which the smear results indicate contamination above 1000 dpm/100 square feet.

2. Routine Large Area Smears (5000-10,000 dpm) - Repeat the large area smear. If results show levels of contamination above 5000 dpm, isolate the contaminated area. Take smears in smaller areas to locate the source. Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm/100 square feet.

3. Routine Large Area Smears (>10,000 dpm) - Isolate the contaminated area. Survey all personnel in the contaminated area. Take smaller smears in the area to locate the source. Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm/100 square feet. Survey all persons leaving the building.

12.8.5.2 Personnel Contamination Surveys - Personnel are required to monitor themselves for activity present on their hands, shoes, clothing and person before exiting a contamination area. Contamination monitors (friskers) are located at all normal exits from contamination areas for this purpose. The detector should be held as close to the surface of the item being monitored as possible, without touching the item, and the probe should be moved at a slow speed over the surface. Allowable levels of contamination on skin surfaces and on items of clothing are given in Tables 12-10. Any contamination in excess of these limits should be reported immediately to the Health and Safety Group. The Health and Safety Group will supervise the decontamination and determine if clothing must be discarded. The approval of the Supervisor, Health and Safety shall be required to allow any individual to leave a contaminated area who is contaminated above background radiation levels.

TABLE 12-10

MAXIMUM PERMISSIBLE CONTAMINATION FOR PROTECTIVE CLOTHING

($dpm/100\ sq.\ cm^2$)

Item	Alpha	Beta + Gamma
Clothing	2,200	22,000
Shoes	22,000	220,000

12.8.5.3 Release of Equipment or Packages - Packages and equipment are surveyed by the Health and Safety Group. The Health and Safety Group has the authority to prohibit the release of items that are found to exceed the limits specified in Annex C to License SNM-778 "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use of Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated July, 1982."

12.9 RESPIRATORY PROTECTION

The primary objective of a respiratory protection program is to limit the inhalation of airborne radioactive materials and other hazardous materials. This objective is normally accomplished through the use of engineering controls, including process, containment, and ventilation equipment. When engineering controls are not feasible or cannot be applied, respiratory protection must be used. The Health and Safety Group is responsible for the implementation of the respiratory protection program. The program is based on the guidance contained in 10 CFR 20, Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," and NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials."

The respiratory protection program will include the following:

1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protection equipment.

2. Written procedures to ensure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to ensure the adequate individual fitting of respirators, as well as procedures to ensure the testing of respiratory protective equipment for operability immediately prior to each use.

4. Written procedures for maintenance to ensure full effectiveness of respiratory protective equipment, including procedures for cleaning and disinfecting, decontaminating, inspecting, repairing, and storing.

5. Written operational and administrative procedures for the control, issuance, proper use, and return of respiratory protective equipment, including provisions for planned limitations on duration of respirator use for any individual as necessitated by operational conditions.

6. Bioassays and other surveys, as appropriate, to evaluate individual exposures and to assess the protection actually provided.

7. Records sufficient to permit periodic evaluation of the adequacy of the respiratory protection program.

8. Determination prior to assignment of any individual to tasks requiring the use of respirators that such an individual is physically able to perform the work and use the respiratory protective equipment. A physician is to determine what health and physical conditions are pertinent. The medical status of each respirator user is to be reviewed at 12-month intervals.

Other details of an effective respiratory protection program can be found in the above mentioned documents and the health physics procedures.

12.10 OCCUPATIONAL EXPOSURE ANALYSIS

12.10.1 External Exposure - The external radiation exposure received by workers is presented in Tables 12-11 through 12-14. Tables 12-11 and 12-12 show the exposures by ranges and the number of workers in each range for calendar years 1984 and 1985 respectively.

TABLE 12-11

1984 EXPOSURES BY RANGE

Annual Whole Body Dose Ranges (Rems)	Number of Individuals In Each Range
No Measurable Exposure	87
Measurable Exposure <0.100	77
0.100 to 0.250	33
0.250 to 0.500	12
0.500 to 0.750	6
0.750 to 1.000	1
1.000 to 2.000	3
2.000 to 3.000	1
3.000 to 4.000	0
4.000 to 5.000	0
5.000 to 6.000	0
6.000 to 7.000	0
7.000 to 8.000	0
8.000 to 9.000	0
9.000 to 10.000	0
10.000 to 11.000	0
11.000 to 12.000	0
>12.000	0
	220

TABLE 12-12

1985 EXPOSURES BY RANGE

Annual Whole Body Dose Ranges (Rems)	Number of Individuals In Each Range
No Measurable Exposure	185
Measurable Exposure <0.100	83
0.100 to 0.250	21
0.250 to 0.500	17
0.500 to 0.750	5
0.750 to 1.000	3
1.000 to 2.000	2
2.000 to 3.000	2
3.000 to 4.000	0
4.000 to 5.000	0
5.000 to 6.000	0
6.000 to 7.000	0
7.000 to 8.000	0
8.000 to 9.000	0
9.000 to 10.000	0
10.000 to 11.000	0
11.000 to 12.000	0
>12.000	0
	318

Table 12-13 presents the exposures received by workers for calendar years 1981 through 1984. The row entitled "Off Site" gives the exposures received by workers at other licensed facilities.

TABLE 12-13

RADIATION EXPOSURE

	1984	1983	1982	1981
Total Person Rems	23.5	18.4	19.4	26.3
Off Site	3.5	2.0	2.5	3.0
LRC	20.0	16.4	16.9	23.3
Average Exposure	0.09	0.088	.105	.137
Number of Workers	220	208	184	192
Highest Exposure	2.25	2.04	1.9	1.7

The exposure received by workers is categorized by group in Table 12-14 for exposures received for calendar years 1983 and 1984.

TABLE 12-14

EXPOSURE BY GROUP (PERSON REMS)

Group	1984	1983
Plant Engineering	5.10	2.25
Project Services	0.07	0.05
Health & Safety	1.85	2.16
Nuclear Materials	11.40	9.60
Chemical & Nuclear Engineering	1.30	1.53
Nondestructive Methods	0.58	0.17
Process Control	0.00	0.49
Systems Design & Engineering	2.12	2.09

Calendar year 1984 brought increased activity in our hot cell facility. This typically results in increased exposures to personnel in the Nuclear Materials, Plant Engineering, and Health and Safety Groups. Table 12-14 reflects this in all categories. Table 12-14 also reflects this increase in two of the three affected groups. Only Health and Safety saw a reduction in the group's exposure. The amount of exposure received from off-site work reversed a three year period of decreases. Table 12-14 reflects this in the increase in the Systems Design & Engineering Group's exposure.

The increases noted in Tables 12-13 and 12-14 do not indicate a decrease in the vigilance given by site management to personnel exposures nor do they suggest a decreased ALARA emphasis. Exposure history at the site shows wide variances because of the variety of work that is performed here. Clear trends have not been evident. If the amount of hot cell work is considered and the fact that objects received for examination exhibit higher levels of radioactivity, the effectiveness of the ALARA program can be appreciated. The preliminary exposure information required on the Radiation Work Permit form was increased in early 1985. This has resulted in many improvements in the manner that cell entries are made.

12.10.2 Internal Exposure - The bioassay sampling, lung counting, and air sampling programs show that the worker is exposed to extremely low levels of respirable activity.

12.10.2.1 Bioassay Results - Urine bioassay samples are taken primarily of workers who perform work with unclad uranium and those involved in any work with plutonium. Table 12-15 below presents the number of urine bioassay samples taken during 1983 and 1984.

TABLE 12-15

NUMBER OF URINE BIOASSAY SAMPLES

Month	1983 U	1983 Pu	1984 U	1984 Pu
January	5	-	20	4
February	-	-	13	6
March	-	-	15	12
April	19	19	18	9
May	-	-	13	5
June	16	16	17	8
July	11	8	15	6
August	10	8	15	7
September	11	14	14	7
October	11	9	16	5
November	3	1	-	-
December	5	5	14	6

In 1983, all samples for uranium were less than 5 micrograms/liter (lower limit of detection), except on four occasions when the analysis indicated the presence of uranium but none met the resample limit of 20 micrograms/liter. All plutonium analyses were below the minimum sensitivity which varied from 0.00 ± 0.1 to 0.3 ± 0.4 dpm per sample.

In 1984, all samples for uranium were less than 5 micrograms/liter (lower limit of detection), except on one occasion 27 micrograms/liter was reported. A resample showed that the level had returned below the lower limit of detection. All plutonium samples indicated 0.0 $\pm$ (0.01 to 0.6) dpm per sample.

12.10.2.2 Air Sampling Results - The air sampling program is the first line of defense for all operations of this type, but the bio-assay program, along with lung counts, is the final step in the estimation of exposure that may occur.

12.10.2.2.1 Table 12-16 presents a summary of the air sampling program for calendar year 1983, for fixed air samplers.

TABLE 12-16

1983 AIR ACTIVITY

(VALUES IN μCi/ml)

Labs	Approximate Average	Maximum Concentration	MPC
15	$3x10^{-15}$	$1.2x10^{-14}$	$1x10^{-10}$
16	$3x10^{-15}$	$1.5x10^{-14}$	$1x10^{-10}$
17	$8.7x10^{-15}$	$1.8x10^{-13}$	$4x10^{-11}$
19	$7x10^{-15}$	$8.7x10^{-14}$	$1x10^{-10}$
27	$2.4x10^{-15}$	$5.7x10^{-15}$	$1x10^{-10}$
44*	$2x10^{-15}$	$6.5x10^{-15}$	$1x10^{-10}$
Cask Handling Area	$1.9x10^{-12}$	$1.27x10^{-10}$	$9x10^{-9}$
	$6.7x10^{-15}$	$4.5x10^{-13}$	$4x10^{-11}$
Hot Cell	$1x10^{-14}$	$1.25x10^{-13}$	$9x10^{-9}$
	$5x10^{-16}$	$1.2x10^{-15}$	$4x10^{-11}$
Recirculated Air "C"	$1.5x10^{-14}$	$3.5x10^{-13}$	$9x10^{-9}$
	$4x10^{-15}$	$1.93x10^{-14}$	$4x10^{-11}$
Waste Storage Area	$1.5x10^{-14}$	$2.6x10^{-14}$	$9x10^{-9}$
	$7x10^{-16}$	$1.7x10^{-15}$	$4x10^{-11}$

Laundry	$3x10^{-14}$	$1.5x10^{-13}$	$9x10^{-9}$
	$3x10^{-15}$	$2.5x10^{-14}$	$4x10^{-11}$
Radio Chem Lab	$7x10^{-14}$	$2.3x10^{-12}$	$9x10^{-9}$
	$1.5x10^{-15}$	$1.5x10^{-14}$	$4x10^{-11}$

*Discontinued in Sept.

12.10.2.2.2 On 338 occasions in 1983, breathing zone air samples were taken to measure the airborne activity to which workers were exposed. In no case was anyone exposed to greater than 2 MPC of airborne activity in any one week. In most cases, respiratory protection was used and exposure levels were at least a factor of 1,000 below the limits.

There are three major operations which require respiratory protection, and several minor ones.

1. Entries into the isolation area behind the hot cell. A supplied air respiratory system was installed in January, 1980, in the hot cell area which has a protection factor of at least 1,000. This system incorporates a double bibb hood which has reduced airborne activity to which a worker is exposed to below detectable levels.

2. Operations outside of the isolation area in the cask handling area using the 3M hood and the supplied air respiratory system. This system incorporates the 3M hard hat which is NIOSH approved with a protection factor of 1,000. Breathing zone samples are taken outside of the hood each time this system is used.

3. Operations in Building C may involve bagging operations with plutonium glove boxes. All operations of this type require respiratory protection. When it is used, a breathing zone sample is taken. Normally, the powered respirator with 1,000 protection factor is used; however, the full face masks with a protection factor of 50 may be used.

4. Other minor operations requiring respiratory protection are: changing HEPA filters, repair work on NPD site

support equipment, and any other operations where Health and Safety believes that there is a potential of airborne activity.

5. It should to be noted that a major operation is occurring in the decommissioning of Building C that is requiring the use of respiratory protection for industrial safety reasons, not for protection from radioactive materials. A number of operations are very dusty (paint chipping, concrete destruction, etc.). A NIOSH approved full flow hard hat system is used. With no protection factor, no one in Building C has been exposed in excess of 2 MPC hr in one week. In most cases, radioactivity above background is undetectable.

12.10.2.2.3 Table 12-17 presents a summary of the air sampling program for calendar year 1984, for fixed air samplers.

TABLE 12-17

1984 AIR ACTIVITY

(VALUES IN μCi/ml)

Labs	Approximate Average	Maximum Concentration	MPC
15*	3E-15	1.6E-14	1E-10
16*	2E-15	5E-15	1E-10
17*	5E-15	3.9E-13	4E-11
19	7E-15	1E-13	1E-10
27**	2.4E-14	7.5E-15	1E-10
Soil Processing***	1E-15	7.4E-15	4E-11
Cask Handling Area	5E-13 5E-15	1.2E-11 5E-13	9E-9 4E-11

Hot Cell	8E-15	6.7E-13	9E-9
	5E-16	1.5E-14	4E-11
Recirculated Air	1.5E-14	1.1E-13	9E-9
Building C	1.5E-15	3.3E-13	4E-11
Waste Storage	1.5E-14	2.9E-14	9E-9
	7E-16	3.3E-15	4E-11
Laundry	3E-14	6.3E-14	9E-9
	2E-15	3.3E-15	4E-11
Radio Chem	3E-14	1.0E-12	9E-9
	1.5E-15	2.4E-15	4E-11

*Discontinued November 1984
**Discontinued June 1984
***Begun May 1984

12.10.2.2.4 On 278 occasions in 1984, breathing zone air samples were taken to measure the airborne activity to which workers were exposed. In no case was anyone exposed to greater than 3 MPC hour of airborne activity in any one week. In most cases, respiratory protection was used and exposure levels were at least a factor of 1000 below the limits.

There are three major operations which require respiratory protection, and several minor ones.

1. Entries into the isolation area behind the hot cell. A supplied air respiratory system was installed in January, 1980 in the hot cell area which has a protection factor of at least 1000. This system incorporates a double bibb hood which has reduced airborne activity to which a worker is exposed to below measurable levels.

2. Operations outside of the isolation area in the cask handling area use the 3M hood and the supplied air respiratory system. This system incorporated the 3M hard hat which is NIOSH approved with a protection factor of 1000. Breathing zone samples are taken outside of the hood each time this system is used.

3. Operations in Building C may involve bagging operations with plutonium glove boxes. All operations of this type require respiratory protection. When it is used, a breathing zone sample is taken. Normally, a 20T air line respirator with a 1000 protection factor is used; however, the full face mask with a protection factor of 50 may be used.

4. Other minor operations requiring respiratory protection are: changing of HEPA filters, repair work on NPD site support equipment, and any other operation where Health and Safety believes that there is a potential of airborne activity.

5. It should to be noted that a major operation is occurring in the decommissioning of Building C that is requiring the use of respiratory protection for industrial safety reasons, not for protection from radioactive materials. A number of operations are very dusty (paint chipping, concrete destruction, etc.). A NIOSH approved full flow hard hat system is used. With no protection factor, no one in Building C has been exposed in excess of 2 MPC hours in one week. In most cases, radioactivity above background is undetectable.

12.10.2.3 In-vivo Results (1983) - Whole body counting was performed by Helgeson Scientific Services, Inc. on 32 workers during 1983. Three had detectable activities, no other workers indicated detectable activity. The results of the three workers with detectable activity is presented in Table 12-18.

TABLE 12-18

WHOLE BODY COUNTS - 1983

(ALL VALUES IN NANOCURIES)

		Worker		
Isotope	MPBB	1	2	3
Cs-137	3E4	8±2	4±2	
Mn-54	3.6E3	5±2		4±1
Co-60	1.1E3		3±1	7±1

In-vivo counting was performed on seven workers during 1983, for plutonium and Americium-241. These results are summarized in Table 12-19.

TABLE 12-19

Am - Pu LUNG COUNTING
1983

(ALL VALUES IN NANOCURIES)

Worker	Pu	Am
1	0	0.00±0.10
2	0	0.00±0.11
3	0	0.13±0.13
4	0	0.00±0.14
5	0	0.00±0.15
6	0	0.00±0.19
7	0	0.00±0.16

In-vivo lung counting was performed on nine workers in 1983, for uranium. The results are listed in Table 12-20. Four of the nine indicated positive results. However, these results were not confirmed in followup urinalyses.

TABLE 12-20

URANIUM LUNG COUNTING

1983

(ALL VALUES IN MICROGRAMS)

Worker	U-235
1	0±30
2	0±43
3	0±39
4	42±37
5	0±41
6	38±33
7	76±45
8	0±39
9	49±44

12.10.2.4 In-vivo results (1984) - Whole body counting was performed by Helgeson Scientific Services, Inc. on 99 workers during 1984. Twelve had positive results but these were very low levels. A summary is presented in Table 12-21.

TABLE 12-21

WHOLE BODY COUNTS

1984

(EXPOSURE VALUES IN NANOCURIES)

Isotope	Number of Workers	Maximum Observed	MPBB
Cs-134	1	3.0	2E4
Cs-137	7	9.0	3E4
Co-60	4	4.0	1.1E3

In-vivo lung counting was performed on 14 workers during 1984 for Plutonium-239 and Americium-241. No plutonium was reported. The presence of Americium-241 was indicated for 5 workers with the highest quantity being 0.26 NanoCuries (±0.14) for one person.

In-vivo lung counting was performed on 20 workers during 1984 for Uranium-235. In 5 instances, the results were positive with the highest result being 48 micrograms (±37) for one person.

12.11 MEASURES TAKEN TO IMPLEMENT ALARA

12.11.1 Irradiated metal specimens had been stored on the roof of the hot cells in an open top cave. This configuration caused this roof to be designated as a high radiation area. Several small totally enclosed caves have been constructed for the storage of these specimens which has eliminated the high radiation area on the cell roof, thus reducing exposures received by workers who periodically enter the area for maintenance on the HEPA filters and to calibrate an area monitor. It also eliminated the radiation area on the roof of Building B which no longer contributes to the exposure of workers who maintain the building ventilation system.

12.11.2 Cleaning of the hot cells contributed significantly to exposure doses of workers. This cleaning operation, which is performed at three or four year intervals, requires the set-up table in the cell to be dismantled. In 1985, this operation was performed remotely with a modified saw so that workers did not enter the cell for this high exposure work.

12.11.3 Trash removal from the hot cell during cell cleaning operations was significant in the past. During the cleaning operation in 1985, trash was remotely loaded into special metal drum liners that were designated to fit into 30-gallon drums and to be handled with long poles. This process modification reduced personnel exposures for this part of the operation considerably.

12.11.4 The site has purchased a TLD reader which provides immediate information on worker exposure. This system is not intended to replace the normal contract service for dose measurement but rather to provide prompt indication of unexpected exposures for non-routine operations. The system makes possible the estimation of exposures to hard to measure areas of the body such as the soles of feet, hands and fingers.

12.11.5 A supplied air respiratory system has been installed to support hot cell work, principally during hot cell entries. This system provides a greater protection factor for workers in addition to providing greater worker comfort while performing the strenuous work.

12.11.6 The Radiation Work Permit (RWP) approval process has been revised. Previously, the worker or his supervisor completed the RWP form and carried it to those personnel who were required to sign it.

This method has been changed such that the workers, Area Supervisors and signators of the RWP gather at a meeting wnere the proposed work scope and methods are discussed in detail. All facets of work are agreed to before any authorization signatures are placed on the RWP. This new approval process requires more time being spent for the planning stage of a task but considerable exposure savings have resulted.

12.12 BIOASSAY PROGRAM

Those workers routinely working in contamination or airborne radioactivity areas will be scheduled for participation in the bioassay program. The Health and Safety Group will select those workers to be sampled in the program. This selection will be based on the probability of exposure, the worker's work habits, the type of work in the area, air sample data, previous bioassay data, etc. Routine bioassay may consist of check or whole-body counting (in-vivo bioassay) or excretion analysis (in-vitro bioassay). In-vivo bioassay is performed routinely by a bioassay service which comes on-site for the evaluations. In-vitro bioassay is performed by a commercial laboratory located off-site.

Bioassay action criteria for plutonium are outlined in Table 12-3 & 12-4. In general, no action is required if the excretion result (i.e., urinalysis) is less than 0.2 dpm/liter or the in-vivo measurement of material in the lung is less than 16 nanoCuries. All compounds of plutonium are considered to be either class W or Y. This classification refers to the most recent evaluation of the ICRP for internal dose calculations. Class W compounds are moderately soluble and clear from the pulmonary region of the lung with half-times in the range 10 to 100 days. Class Y compounds are essentially insoluble and are considered to clear from the pulmonary region with half times of >100 days. No compounds of plutonium are considered by the ICRP to be readily soluble (i.e., class D compounds which clear from the lungs in <10 days).

The bioassay program for uranium generally follows that outlined in Regulatory Guide 8.11, "Application of Bioassay For Uranium," June 1974. There are two exceptions to this general guidance:

1. Workers off-site during the regular visit of the bioassay service will not be scheduled for a special, make-up count, if the count was scheduled only for routine exposure control monitoring.

2. Bioassays of workers working in areas in which both plutonium and uranium may be airborne shall be evaluated for both plutonium and uranium. The Supervisor, Health and Safety may decide to analyze for only one of these elements, if it can be demonstrated that the analysis for a single element is a more sensitive indicator of an uptake.

Bioassay action criteria for uranium are outlined in Table 12-5 & 12-6.

Workers working primarily with beta and gamma emitting radionuclides will also be included in the in-vivo bioassay analysis program. Any worker suspected of an exposure greater than 40 MPC-hours will be scheduled for a bioassay evaluation as soon as practicable after the exposure. Bioassay action criteria for beta-gamma are outlined in Table 12-7.

12.13 AIR SAMPLING AND MONITORING

The presence of airborne radioactive materials in the working areas is determined through the combined use of air samplers and monitors. These programs are discussed below:

12.13.1 Air Sampling Program

The air sampling program can be divided into two categories; fixed and portable. Selection of the sampling category and the frequency of sampling is left to the discretion of the Supervisor, Health and Safety.

12.13.1.1 Fixed Air Samplers - Air samples are obtained at designated points through the use of a central vacuum system. Sampling points are located as close as possible to a permanent operator station to permit continuous sampling of the air near the worker's breathing zone. These samples are usually collected weekly. However, the frequency may vary as the situation dictates.

Normally, these are evaluated within two weeks, after allowing the appropriate decay period for the radon daughter products. However, based on the particular operation, etc., a Health Physics Engineer may determine that it is necessary to evaluate the samples without allowing for the decay period. In these cases, an applicable radon decay correction factor must be applied to the results.

12.13.1.2 Portable Samplers - Air samples in the approximate breathing zone of a worker may be obtained through the use of a lapel sampler. The lapel sampler consists of a small sampling head attached to the worker's lapel (or collar) connected through a small flexible tube to a small air-pump worn at the waist. The flow rates through these samplers are quite low when compared to the fixed system. However, since the sampler is located near the nose and mouth and moves with the worker as he moves about the area, it provides a reasonable estimate of the concentration of airborne radioactivity in the breathing zone of the worker.

Air samples obtained with these samplers are evaluated on a low background, proportional counting system. Factors are applied to the counting results to account for background activity and detector efficiency. All results are reported in units of activity/unit volume of air sampled.

12.13.2 Air Monitoring Program

Air monitoring in operating areas is accomplished with continuous monitors in predetermined, fixed locations. Normally, a monitor is placed in each radioactive materials handling area in which there is a potential for the release of airborne radioactivity. Locations are selected based upon the ability of the monitor to provide a reasonable evaluation of the airborne activity in a particular area and to provide adequate warnings to those in the area of changing conditions. These determinations are made by the Health and Safety Group based upon the operations in the area, the potential for release, and the quantity and chemical form of the material.

Alarms are set in accordance with the particular operation, the material being handled, and the potential for release. Actual alarm points are set as low as possible commensurate with the ambient radiation levels in the area.

12.14 SURFACE CONTAMINATION

12.14.1 Smear Surveying

Smear surveys are performed in all areas specified in the license and which, in the judgment of the Supervisor, Health and Safety, have a potential for surface contamination. The frequency of these surveys will be based upon the potential for contamination in the area, previous experience with contamination in the area, and the need to keep the area free from contamination. Typical areas and survey schedules are listed in Table 12-9, however, both the areas included and the frequencies of surveys are subject to change based upon the current research activities. The frequency of smear surveys in areas not included in the table are generally specified in the procedure covering the particular area.

12.14.1.1 Smear Samples - Smear samples are obtained with small, absorbent filter papers. The smear paper is moved across an area of approximately 100 sq. cm. using about 5 pounds of pressure. The smear may be counted with a portable gas-flow proportional counter capable of detecting alpha or beta radiation. Normally, smear samples are evaluated in a stationary counter located in the Health Physics Laboratory. Appropriate conversion factors are applied to the net counts to express the smear results in units of disintegrations per minute.

12.14.1.2 Large Area Smears - Large area smears are obtained using the dust mop technique in areas around the site, the hot cell operations area, the change room and main hallways in Building B. These smears are intended to indicate the general contamination environment in an area and may lead to a more extensive survey, if unexpected contamination is indicated. Normally, large area smears are evaluated with a hand-held, portable survey instrument (e.g., a gas-flow proportional counter such as the PAC 4G). Actions to be taken in response to the results of large area smears are outlined in Table 12-22.

12.14.1.3 Action Levels - Included in Table 12-24 are the appropriate action levels to be used in designated areas. Decontamination shall be initiated in areas in which the removable surface contamination levels exceed these action levels. The Health and Safety Group shall determine and direct the actions to be taken to protect workers working in these areas and to reduce contamination levels as far below those listed in Table 12-1 as is possible. Normally, decontamination of an identified area shall begin within 24 hours of the discovery.

In some cases, for example, if the contamination is discovered just prior to a weekend or a regularly scheduled holiday, the contaminated area may be marked and posted appropriately. Such a determination shall be made by the Health and Safety Group based upon the severity and extent of the contamination and the potential for further contamination of equipment and/or personnel during the interval. Decontamination of the area shall begin on the first regular work-day after discovery.

TABLE 12-22

ACTION LEVELS FOR LARGE AREA SMEARS

1. Routine Large Area Smears (1000 - 5000 dpm)

   Repeat the large area smear. If results show levels of contamination above 1000 dpm, take smears in smaller areas to locate the source.

   Decontaminate all areas in which the smear results indicate contamination above 1000 dpm per 100 sq. ft.

2. Routine Large Area Smears (5000 - 10,000 dpm)

   Repeat the large area smear. If results show levels of contamination above 5000 dpm, isolate the contaminated area. Take smears in smaller areas to locate the source.

   Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm per 100 sq. ft.

3. Routine Large Area Smears (>10,000 dpm)

   Isolate the contaminated area.

   Survey all personnel in the contaminated area.

   Take smaller smears in the area to locate the source.

Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm per 100 sq. ft.

Survey all persons leaving the building.

NOTE:

Routine large area smears are normally taken in the early afternoon to facilitate clean-up of areas found to be contaminated before the end of the normal work-day.

TABLE 12-23

SMEAR SURVEY FREQUENCIES AND ACTION LEVELS

Alpha Radiation Smear Survey

Area	Frequency	Action Level (dpm/100 sq. cm.)
Unirradiated, unencapsulated fuel handling areas	weekly	5,000
Building B counting laboratory	monthly	200
Hot cell operations area	monthly	200
Scanning electron microscopy laboratory	monthly	200
Exit portals from controlled	twice weekly	200

Beta Radiation Smear Survey

Area	Frequency	Action Level (dpm/100 sq. cm.)
Building B Counting Laboratory	monthly	2,000
Scanning Electron Microscopy Laboratory	monthly	2,000

Hot Cell Operations Area	twice monthly	2,000
Cask Handling Area	twice monthly	22,000
Radiochemistry Laboratory	twice monthly	22,000
Exit Portals From Controlled Areas	twice monthly	2,000

12.14.2 Direct Radiation Surveys

Surveys of the direct radiation exposure are to be performed on a frequency established by a Health Physics engineer. In general, these surveys require the selection of the appropriate portable survey instruments based upon the anticipated radiation levels, the types of radiation expected, and the nature or type of survey to be performed. General maps of the areas to be surveyed may be used to record the measured ambient radiation levels and/or, in some cases, to designate specific areas in which the exposure rates should be measured. The survey should also include a visual examination of the area for any unusual conditions or work habits which could affect the exposures received by personnel working in these areas. Items of this nature should be reported immediately to the Supervisor, Health and Safety, or corrected immediately, if practical.

Results of these surveys should be reviewed by a Health Physics Engineer to ensure that the proper posting requirements are in effect for the area and to ensure that appropriate actions are taken to keep all exposures ALARA.

Action levels for direct radiation surveys are presented in Table 12-24.

TABLE 12-24

CONTAMINATION ACTION LEVELS

Area	Type of Radiation	Fixed Surface Reading	Transferable Surface Contamination (dpm/100 sq. cm.)
Uncontrolled	Alpha	300 dpm/100 sq. cm.	30
	Beta-Gamma	0.1 mRad/h	220
Contamination*	Alpha	3000 dpm/100 sq. cm.	2,200
	Beta-Gamma	1.0 mrad/h**	22,000

* The Supervisor, Health and Safety may raise these action levels. Justification for this action must be documented and forwarded to the Safety Review Committee for their review and approval.

** This action limit applies to contamination areas which are normally radiation areas. This level of contamination will not cause a significant increase in radiation exposure.

NOTE:

This table provides limits above which decontamination must be initiated. These action levels pertain to areas normally accessible to personnel performing normal work functions. The levels do not apply to areas requiring extraordinary precautions for entry, e.g., the Isolation Area, waste water tanks, etc. In these cases, direct health physics coverage is the primary control mechanism.

12.14.3 Personnel Contamination Surveys

Workers are required to monitor themselves for activity present on their hands, shoes, clothing, and person before exiting a contamination area. Contamination monitors (friskers) are located at all exits from contamination areas for this purpose. The detector (probe) should be held as close to the surface of the item being monitored as possible (without touching the item) and the probe should be moved at a speed of about 0.5 inch/second. Allowable levels of contamination on skin surfaces and personal clothing must not exceed background. Permissible levels of contamination on protective clothing are given in Table 12-10. Any contamination in excess of these limits should be reported immediately to the Health and Safety Group.

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13.0 ENVIRONMENTAL SAFETY

13.1 ENVIRONMENTAL MONITORING

Environmental sampling of the area surrounding the site is performed on a regular basis to evaluate changes in the levels of radioactivity in air, water, and vegetation. The minimum environmental program consists of the following.

- one continuous on-site site boundary air sample (Figure 13-1)
- monthly water samples from the James River collected above and below the liquid discharge point (Figure 13-2)
- continuous sampling of rain water on-site (Figure 13-1)
- quarterly samples of river silt and near-river vegetation (Figure 13-2).

Normally, site personnel are responsible for collecting the environmental samples. Analysis of these samples may be performed on-site or the samples may be analyzed by a commercial laboratory.

Environmental sampling data for the calendar years 1982, 1983, and 1984 is given in "Lynchburg Research Center, Environmental Report, October, 1985," Tables 2.2 through 2.6.

13.2 EFFLUENT AIR MONITORING

Planned discharges of air to the environment shall be in compliance with the limits specified in 40 CFR 61. Compliance with 40 CFR 61 demonstrated by calculation that the total annual release limits permitted by the license will not exceed the specified 25 millirem whole body and 75 millirem organ dose limits for persons located at the point of maximum ground level concentration.

Potentially contaminated air from chemical hoods, hot cells, and glove boxes is discharged ultimately through the 50-meter stack. Generally, exhaust air containing beta-gamma activity is passed through a single-stage HEPA filter which is sufficient to remove airborne particulates. Air from more hazardous operations, e.g., from glove boxes, is routed through a two-stage HEPA filter.

Discharge through the stack is accomplished with a large blower, powered normally by a large electric motor operated on off-site power. Emergency power is supplied by an internal combustion engine coupled to the blower shaft through a centrifugal clutch. On loss of off-site power, the engine starts automatically and takes over the load upon reaching the proper speed.

Discharges through the stack are monitored with a sampling head located in the stack about 25 feet above the base. Air removed by the sampler passes through a fixed filter, into the chamber of the gas monitor, and is returned to the stack. The fixed filter is monitored continuously for alpha and beta activity by a gas-flow proportional counter. The second monitor, the gas monitor, operates continuously utilizing a halogen-quenched GM tube. The stack monitor flow rate is maintained at a minimum of 2 cfm. Both monitors are equipped with adjustable alarms. Set points for these alarms are determined by the Health and Safety Group. These alarms are connected to an alarm panel located in the Health Physics Laboratory in Building B.

Air from areas equipped with continuous air monitors (and which is below the applicable MPC for an unrestricted area) may be exhausted, through HEPA filters, directly to the roof of the building. Air from areas which have a low potential for airborne activity may be exhausted directly to the roof of the building.

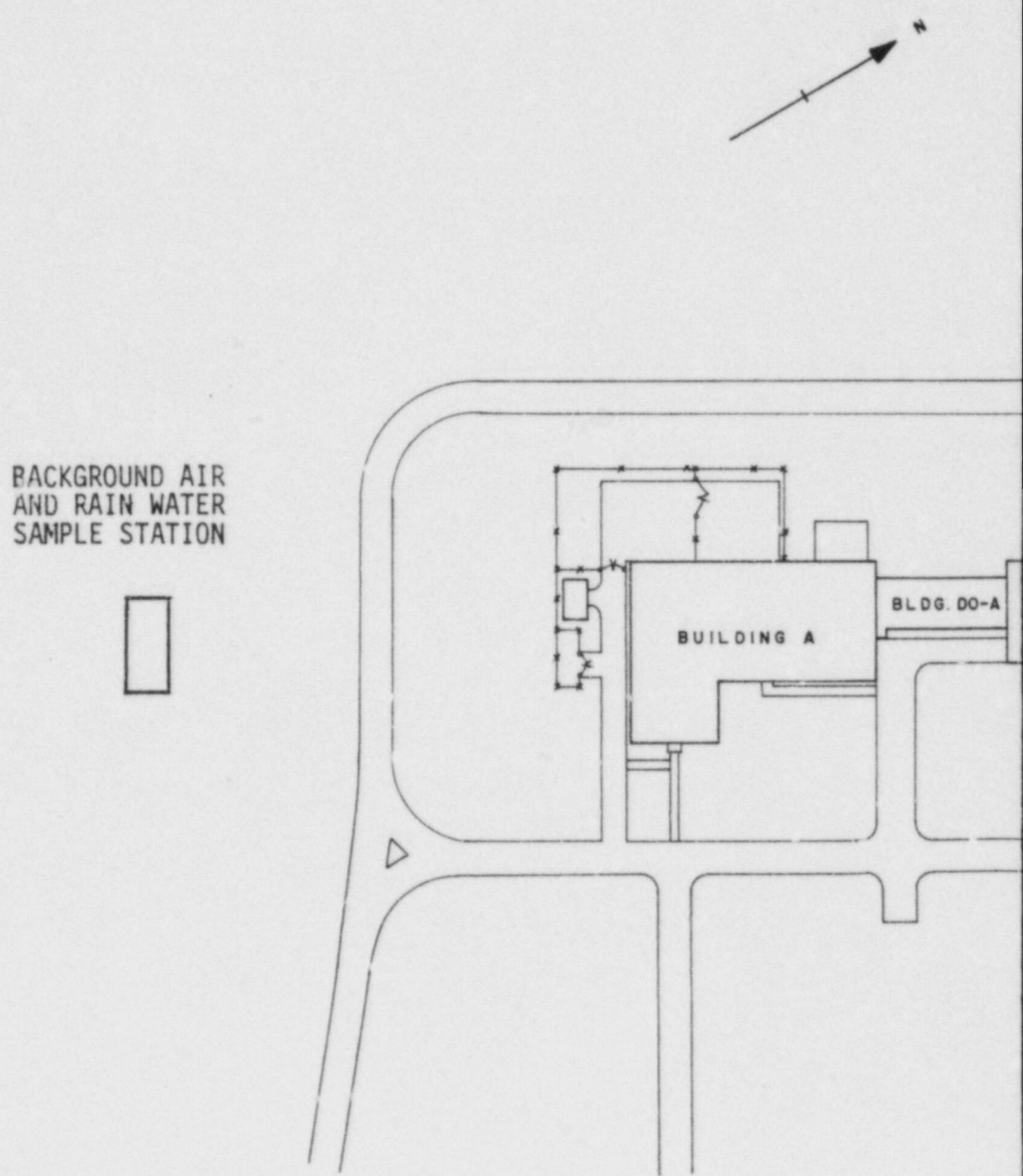
13.3 LIQUID EFFLUENT MONITORING

All potentially radioactive liquids are collected in tanks located in the Liquid Waste Disposal Facility. The contents of each tank are mixed, samples are obtained and are analyzed for radioactivity before the liquids are released to the waste treatment plant at the Naval Nuclear Fuel Division (NNFD).

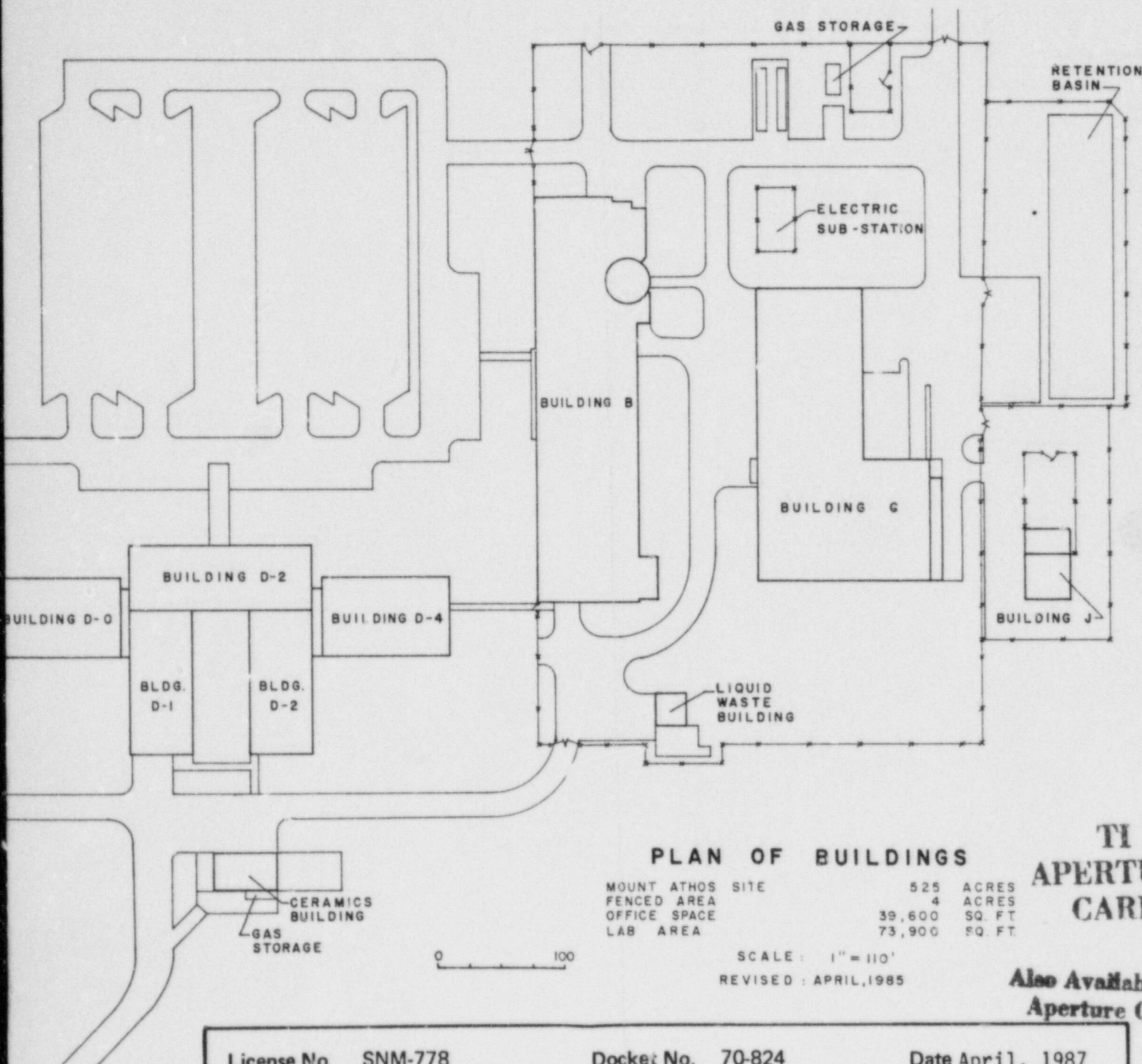
Liquid waste tanks are sampled on a quarterly frequency, before release to the NNFD or at other times determined by the Health and Safety Group. Results of all analyses are reported in units of activity per unit volume and records of these evaluations are retained by the Health and Safety Group.

Water samples are also obtained on a quarterly basis from the retention basin located behind Building C and the holding pond located near Building J.

FIGURE 1


N
BACKGROUND AIR
AND RAIN WATER
SAMPLE STATION
BUILDING A
BLDG. DO-A

3-1 NNFD RESEARCH LABORATORY



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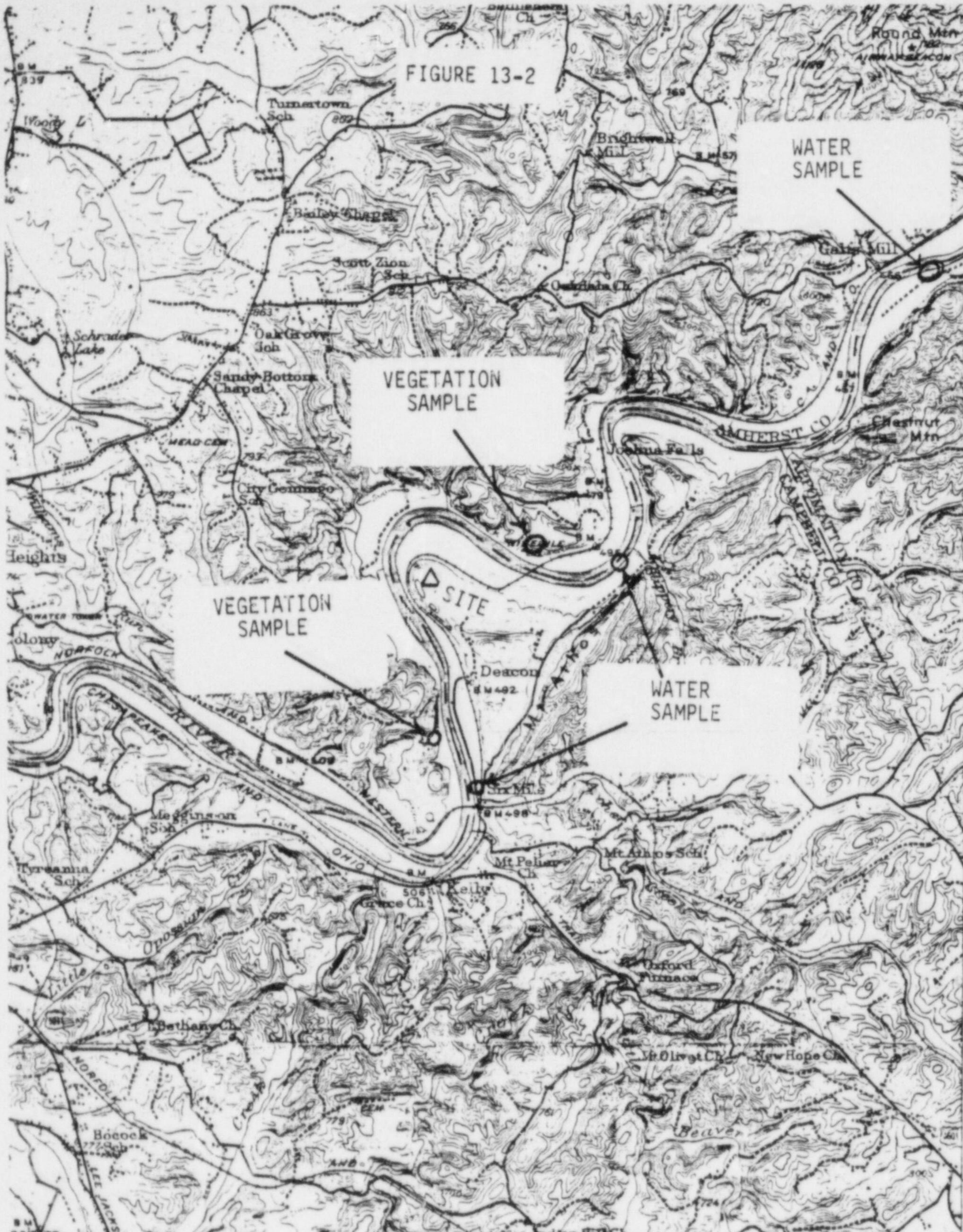

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14.0 NUCLEAR CRITICALITY SAFETY

14.1 ADMINISTRATIVE AND TECHNICAL PROCEDURES

The ultimate responsibility for nuclear criticality safety rests with the Manager, EC&RR. However, first-line responsibility is with the Facility Supervisor supported by the Nuclear Criticality Safety Officer.

The Nuclear Criticality Safety Officer is generally responsible for establishing nuclear safety limits and nuclear safety considerations in operating procedures, processes, and the like. His duties are shown more specifically in the following statement.

The position of Nuclear Criticality Safety Officer has been established at the site. It will be this officer's responsibility to ensure, as far as possible, that no operations on site can lead to the inadvertent assembly of a critical mass. To this end, he will review all new procedures which involve the handling of special nuclear materials as well as changes in old procedures, observe operations, inaugurate educational programs if and when he deems them necessary, and carry out confirming criticality calculations.

This appointment does not in any way relieve the Facility Supervisor of his responsibilities for ensuring the safety of operations, nor will it eliminate the necessity for the reviews by the Safety Review Committee required by the license.

Once a quarter the Nuclear Criticality Safety Officer or qualified person designated by him will inspect all site operations where special nuclear materials are being processed. Other areas shall be inspected less frequently; however, all areas shall be inspected at least once a year. He shall consider area operations when scheduling these inspections and shall, if necessary, schedule his inspection at more frequent intervals. His consideration should include inspection of new facilities, inspection of hazardous non-routine operations, an audit of nuclear criticality safety records, a check for area posting and a review of current practices.

A written report is to be filed with the Manager, EC&RR quarterly with a copy to the License Administrator. Prior to submission of the report, he shall discuss any findings with the Facility Supervisor. The report shall be brief, concerning itself with inspections made during the quarter and with the nuclear criticality safety activity of the quarter.

The following information is to be included:

- o Areas visited
- o Operations observed
- o Unsafe practices or situations noted
- o Nuclear safety activity of the quarter (brief summary)
- o Recommendations
- o Resolution of previous recommendations.

14.2 PREFERRED APPROACH TO DESIGN

Research and Development activities are performed at the site. While the use of safe geometry is the preferred approach in a production facility, it is not appropriate nor practical at a research laboratory. Since most projects require only small amounts of SNM on laboratory benches and in hoods, the preferred approach at the is through safe masses in simple arrays; the lattice density model or arrays found in TID-7016, Rev. 1 is the adopted model. The one exception to use of safe masses is when examining and testing reactor fuel assemblies. The approach to such uses is to accept only a limited number of fuel assemblies and then to maintain the fuel of an assembly within the dimensional envelope of the original assembly's dimensions. Where this is not possible, the fuel of an assembly is handled within the dimensions of safe geometry or as a safe mass.

14.3 BASIC ASSUMPTIONS

This section describes basic assumptions and evaluations that have been made to demonstrate nuclear criticality safety for the specifications of Section 4.2 (Technical Requirements for Nuclear Criticality Safety).

14.3.1 Nuclear Isolation - Special nuclear material is isolated from all other special nuclear material for nuclear criticality safety purposes if any of the three conditions (or equivalent) listed in 4.2.1 are met. These three isolation criteria are accepted industrial practice for maintaining nuclear criticality safety. It is recognized that 12 inches of high density concrete may not be adequate as isolation between two large parallel slabs of SNM; this does not describe any SNM configuration and will not be permitted without additional evaluation and NRC approval.

14.3.2 Building A

14.3.2.1 General - From Figure 22, TID-7016, Revision 1, 74 units is read as the maximum allowable number of units in a cubic array on 24 inch centers (with at least 8 inches edge-to-edge between units), assuming full reflection on the array. The number of allowable units has been reduced from 74 to 40 units to permit use of the 850 grams of U-235 at low enrichment as a unit. The subdivisions defining a unit are for clarification of the general definition of a unit as any physically identifiable accumulation of SNM. The terminology of TID-7016, Revision 1, applies.

14.3.2.2 Mass Limits

1. The mass limits for plutonium, U-233, and U-235 are based on the recommended limits in Table I, TID-7016 (Rev. 1). The values for Pu-235-U mixtures in 4.2.2.2.1 were derived to satisfy the following relationship:

$$\frac{\text{grams Pu fissile}}{220} + \frac{\text{grams U-235}}{350} \leq 1$$

2. The values for U-233 - Pu and U-233 - U-235 mixtures were found by taking the lowest limit of any isotope in the mixture.

3. From DP-1014, uranium metal-water lattices which have the minimum U-235 mass at critical are 2.36 kg for 3.0 wt% U-235 and 1.47 kg for 5.0 wt% U-235. A conservative interpolation between these two points gives 1.9 kg at 4.0 wt% U-235; 45% of this is 850 g U-235. The present array control is based on the lattice density model using Figure 22 and Table IV (modified) in TID-7016 (Rev. 1). Our calculations demonstrate that the 850 gram unit is an allowable unit if the number of units permitted in TID-7016 (Rev. 1) is set at 40.

All computer calculations were made using either the NULIF code for fully reflected spheres or with the Monte Carlo code KENO. Four series of computer calculations were made. Tables 14-1 and 14-3 summarize the results.

14-1 Table for determination of K_{eff} for the mass limits listed in TID-7016 for lattice density model at the upper H/X limit (made with NULIF).

14-2 Table for determination of K_{eff} Vs H/X for 349 grams of U-235 contained in fully enriched uranium metal (made with NULIF).

The K_{eff} values for the lattice density limits ranged from 0.800 to 0.854 and are tabulated in the following table.

TABLE 14-1

K_{eff} TID-7016

Mass, Kg U-235	Sphere Radius, cm	H/U-235	H/Total U	K_∞	K_{eff}
10.0	6.828	2.0	1.87	1.86	0.800
9.0	7.182	3.0	2.81	1.84	0.804
7.3	7.562	5.0	4.68	1.82	0.803
5.2	8.178	10.0	9.36	1.81	0.815
3.6	8.922	20.0	18.71	1.84	0.854

The K_{eff} values for the present limit (349 was used instead of 350) vs H/X are given in the following table.

TABLE 14-2

K_{eff} FOR PRESENT LIMIT

Mass g U-235	Sphere Radius, cm	H/U-235	H/Total U	K_∞	K_{eff}
349	13.32	736.8	689	1.49	0.780
349	9.82	293.8	275	1.76	0.753
349	7.79	146.2	137	1.86	0.671

By analogy, values for Pu would be similar.

The effect of interspersed water moderation in a concrete reflected finite array of 850 gram U-235 units is shown in Table 14-3. These data show that maximum array multiplication occurs with almost no interspersed water.

TABLE 14-3

K_{eff} FOR 6x6x6 ARRAY OF 850 g U-235 UNITS ON 30-INCH CENTERS

(γ = 18.14 cm, = 0.85 g U/cc)

Volume Fraction H2O	$K_{eff} \pm 2\sigma$
1.00	0.833 ± 0.010
0.15	0.826 ± 0.010
0.10	0.851 ± 0.010
0.07	0.868 ± 0.009
0.05	0.907 ± 0.011
0.03	0.924 ± 0.009
0.02	0.931 ± 0.009
0.01	0.937 ± 0.009
0.001	0.930 ± 0.009

The effect of varying the number of 850 U-235 units in a concrete reflected array while maintaining a constant center-to-center spacing with void between them is shown in Table 14-4. Interpolating between the heterogeneous values for the 24 inch center-to-center system predicts a $K_{eff} \pm 2$ for 40 units of 0.936 $\pm$ 0.12 whereas the 36 inch spacing system has about 512 units for the same K_{eff}.

TABLE 14-4

K_{eff} FOR ARRAYS OF 850 GRAM U-235 UNITS ON 24 AND 36 INCH CENTERS

Array Size	Number of Units	$K_{eff} \pm 2\sigma$	Center-to-Center Spacing, in.
4x3x3	36	0.910 $\pm$ 0.010	24
4x3x3	36	0.929 $\pm$ 0.013*	24
4x4x3	48	0.932 $\pm$ 0.010	24
4x4x3	48	0.947 $\pm$ 0.011*	24
4x4x4	64	0.949 $\pm$ 0.011	24
4x3x3	36	0.807 $\pm$ 0.011	36
4x4x4	64	0.845 $\pm$ 0.011	36
5x5x5	125	0.860 $\pm$ 0.010	36
6x6x6	216	0.881 $\pm$ 0.009	36
7x7x7	343	0.912 $\pm$ 0.009	36
8x8x8	512	0.938 $\pm$ 0.008	36
9x9x9	729	0.949 $\pm$ 0.009	36

*Assumes heterogeneous UO_2-water mixture.

From these data it is concluded that for 850 grams U-235 per unit an array of 24-inch centers should be safe for 40 units or less and on 36-inch centers, an array would be safe with 90 units or less. A slight increase in the array multiplication, on the order of 1%, may occur for low levels of interspersed water moderation. However, the safety of these arrays would still be maintained.

To avoid confusion and possible mistakes, additional procedural controls are applied when low-enrichment limits are used.

These preclude enrichment combinations of below and above 4.0 wt% U-235. (These are not necessarily unsafe - no calculations were made and no such combinations are desired.)

4. The unit and its limit (laboratory, furnace, transfer cart, etc.) are established by the Facility Supervisor, who authorizes posting the limit showing the maximum quantity of plutonium, U-233, and U-235 allowed. The fissile material content of the material transferred to or from a unit is established from process records, analyses, or previous analytical data. Only authorized users of SNM may transfer SNM between units and must do so only according to approved procedures. A board, sign, or other acceptable device is used to record the new balance and compares to balance with the unit limit.

14.3.3 <u>Building B</u>

14.3.3.1 General - The demonstration for units and the array is identical to that of Building A (14.3.2.1 & 14.3.2.2).

14.3.3.2 Hot Cell - The demonstration for the units and array is identical to that of Building A. The individual hot cells are isolated from all other arrays by a minimum of 2 feet of high density concrete.

14.3.3.3 Underwater Storage - Transfer Canal - Underwater aluminum or stainless steel storage racks are constructed to ensure 12-inch edge-to-edge spacing of each unit. Units are limited to those in 4.2.2.2.1 & 4.2.2.2.2 excluding PWR fuel assemblies and, since they are separated by 12 inches of water, units are considered isolated. Therefore, any number of these units may be used.

Racks and fixtures are constructed with sufficient integrity and strength to withstand reasonable structural deformity, thereby providing the spacing previously outlined. Supervisory approval is required for removing or inserting any subcritical unit out of or into its storage rack.

There is no credible way in which water can be lost from the storage pool and transfer canal. However, assuming loss of water, stored units would drain and be unmoderated and subcritical.

14.3.3.4 Underground Storage Tubes - Underground storage tubes are 5 inches in diameter, approximately 10 feet long, and on 17 inch centers (minimum) in a straight line. Material stored is first placed in a storage can with an inside diameter of 4-1/2 inches. Maximum units demonstrated safe in Section 14.3.3.2 are stored one per tube. These are nuclearly isolated from each other by 12 inches of concrete (minimum). The average edge-to-edge separation approximates 13 inches of concrete.

14.3.3.5 Power Reactor Fuel Assemblies

14.3.3.5.1 General - The site will receive and examine PWR fuel assemblies for both nondestructive and destructive examination. Irradiated assemblies will have been subjected to a reactor environment. From a nuclear criticality safety viewpoint, these assemblies are in their most reactive state when fresh or unirradiated. Therefore, nuclear safety is demonstrated by appropriate evaluation of the unirradiated assembly. The current plans call for examination of B&W-manufactured fuel assemblies from B&W power reactors. The current models of interest are designated as the Mark B and Mark C canless assembly. The Mark B assembly is described in the SNM license for B&W's Commercial Nuclear Fuel Plant (SNM License No. 1168, Docket 70-1201). In 7.10 of Section III in SNM License 1168, the K_{eff} of the unrodded and fully moderated and reflected assembly is shown to be 0.92 at maximum enrichment. Maximum enrichment is defined as 4.0 percent nominal which could go to 4.05 percent in manufacturing. Table 14-5 shows a comparison of the Mark C and Mark B assemblies. The K_{eff} of the Mark C assembly under the same conditions listed above has a value of 0.9[illegible]. The reactivity as well as the spectral and physics kinetics of these assemblies are essentially the same. All of the nuclear safety calculations shown in this section were made with the Mark B assembly model (except Tables 14-5 and 14-6). Results were obtained for a fully reflected infinite array 12 inch edge-to-edge of maximumly enriched assemblies that were fully moderated, i.e., under water. The Mark B and C assemblies are to be disassembled in air only in an unirradiated state. The similarity in nuclear characteristics and the large decrease in reactivity in air-moderated assemblies ensure nuclear safety during the disassembly operations. Conditions given in 4.2.3.6.1 are sufficient to ensure that these two assembly types are indeed those to be examined. A damaged assembly which is restrained to 8.6 inches on a side will be no more reactive in air or water even if part of the fuel is missing; this will be demonstrated in Section 14.3.3.5.3.

TABLE 14-5

COMPARISON OF THE MARK B AND MARK C FUEL ASSEMBLIES

	Mark B	Mark C
Fuel assembly array	15 x 15	17 x 17
Fuel assembly dimensions, in.	8.45 x 8.45	8.536 x 8.536
Control rod tubes per assembly	16	24
Instrument tube per assembly	1	1
Fuel rods per assembly	208	264
Fuel rod pitch, in.	0.568	0.501
Fuel active height, in.	144	143
Pellet OD, in.	0.370	0.324
Theoretical density, %	92.5	94.0
Enrichment, %	4.0	4.0
Fuel rod clad ID, in.	0.377	0.332
Fuel rod clad OD, in.	0.430	0.379
Fuel rod clad material	Zr-4	Zr-4
V_{water}/V_{fuel} in fuel rod cell	1.65	1.68

V_{water}/V_{fuel} in assembly with water completely filling control rod and instrument cells	1.90	1.98
K_{eff} of one assembly in H_2O	0.92	0.92

Since The Babcock & Wilcox Company is continuing to improve its assemblies and will supply reload fuel to reactors initially fueled by other reactor manufacturers, the site may destructively examine other types of assemblies. The conditions given in 4.2.3.6.1.1 for additional evaluation are adequate to ensure nuclear safety for different assemblies.

Acceptance of BWR fuel assemblies for study is acceptable if the assemblies have a maximum enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm diameter cylinder. By reference to DP-1014 this is 90% of the minimum critical cylinder diameter for an infinitely long, water reflected, optimally moderated cylinder with four wt% enriched heterogeneous UO_2. This value is further supported by Figure 2 (page 10) in ANSI/ANS 8.1-1983 and Figure 2.15 (page 44) in TID-7016, Rev. 2.

14.3.3.5.2 Receipt and Storage

A. Unirradiated Assemblies - Unirradiated fuel assemblies may be stored in their shipping containers since their nuclear safety has been proven prior to their licensing. Assemblies that are unirradiated may also be stored in air if the distance between assemblies is no less than 21 by 38 inches. (Refer to SNM-1168, Docket 70-1201, Section 3, page 173, dated 2/27/81). This distance assures criticality safety for less than 100 assemblies of either the Mark B and/or Mark C assembly types. This ensures the safety of the maximum of four assemblies stored on site. Unirradiated assemblies may also be stored under water (hot cell pool, mock-up pool, or development test area pool).

Assemblies stored in air will be stored either:

1. Horizontally - on the floor or on tables constructed with sufficient integrity and strength to withstand reasonable structural deformity thus assuring the above mentioned spacing.

2. Vertically - in racks and fixtures constructed with sufficient integrity and strength to withstand reasonable structural deformity and assuring the above mentioned spacing.

   Supervisory approval is required to move any other fissile material into the area where the assemblies are stored. No more than four unirradiated assemblies may be stored at once. The limit of four assemblies is an arbitrary limit which the site imposes upon itself and does not affect nuclear safety.

Partially disassembled unirradiated Mark B or Mark C assemblies may also be stored in air. This is safe due to the lower moderation characteristics of air compared to water. Air moderated values of K_{eff} will be less than those shown in Table 14-6.

Fuel rods from unirradiated, disassembled Mark B or Mark C assemblies will be stored in air in slabs not to exceed 4 inches in height (see Section 14.3.3.5.4, statement 2).

B. Irradiated Fuel Assemblies - Assemblies which have been irradiated may also be stored in their shipping containers or in the hot cell pool. Storage in the hot cell pool is limited to four irradiated assemblies. The limit of four assemblies in the pool is an arbitrary limit which the site imposes upon itself and does not affect nuclear safety since each fuel assembly or rod storage position is neutronically isolated from any other fissile material by a minimum of 1 foot of water.

   Racks and fixtures in the pool are constructed with sufficient integrity and strength to withstand reasonable structural deformity, thereby providing the spacing previously outlined. The racks are also constructed to preclude inadvertently placing other fissile material closer than the 1-foot minimum spacing. Supervisor approval is required for removing or inserting fissile material into or out of any of the racks or fixtures. Storage of Mark B and Mark C fuel rods and partially dismantled assemblies into storage racks which restrain the size of each position to a square not exceeding the dimensions of a fresh fuel assembly, i.e., 8.6 inches, is safe based upon the analysis demonstrating safety of as assembly during dismantlement. Fuel rods may also be stored in an ever safe cross sectional area fixture, i.e., a cross sectional area not exceeding that of a 22.5 cm diameter cylinder.

14.3.3.5.3 Work Area Of Pool Under Hot Cell No. 1 - This area will be used to dismantle irradiated and unirradiated assemblies. Nuclear criticality safety for Mark B and Mark C assemblies under varying stages of dismantlement has been demonstrated via use of NULIF and PDQ-07 physics computer codes.

Reactivity was calculated by PDQ (coefficients having been generated by NULIF) for a fully reflected and flooded, unrodded fresh assembly and for the same assembly under five conditions of dismantlement. The cases run with the number of rods removed in each case and the resulting K_{eff} is given in Table 14-6.

TABLE 14-6

REACTIVITY FOR MARK B AND MARK C FUEL ASSEMBLIES UNDER DISMANTLEMENT

Case No.	No. of Removed Rods	Calculated K_{eff} Mark B	Calculated K_{eff} Mark C
1	0	0.891	0.921
2	4	0.894	0.920
3	12	0.897	0.919
4	24	0.898	0.922
5	36	0.896	0.917
6	8	0.888	0.918

Reactivity was also calculated by KENO-IV using the 123-group XSDRN cross section set for a fresh assembly fully submerged in water under conditions of 24 rods removed and with all instrument and control rod guide tube positions loaded with fuel rods. The cases run with the number of rods removed or added and the resulting K-effectives are given in Table 14-7.

TABLE 14-7

K-EFFECTIVE OF INDIVIDUAL MARK B AND MARK C FUEL ASSEMBLIES

Change in No. Of Fuel Rods From Normal	K-effective $\pm$ 2σ Mark B	Mark C
0	.895 $\pm$ .010	.900 $\pm$ .013
-24*	.900 $\mp$ .015	.906 $\mp$ .014
+17	.890 $\mp$ .015	
+25		.876 $\pm$ .016

*Same configurations as case 4 in Table 14-5.

Cases 2 through 5 represent removal of rods "uniformly" through the assembly while Case 6 represents the removal of eight rods clustered about the center. Rods removed are shown schematically in Figures 14-1 and 14-2 for Mark B and C assemblies, respectively. The calculations reported above demonstrate the nuclear safety of an assembly under various conditions of disassembly and reloading. If any of the fuel rods inserted into the fuel assembly are further encased in metal tubing, the assembly would still be safe due to the tubing displacing moderator with absorber. A grouping of 75 fuel rods confined within a 8.6-inch square merely describes a dismantled assembl and is also safe. Fuel rods inserted into instrument and control rod guide tubes shall be held in place with a flat metal plate which shall be bolted to the top of the assembly.

The safety of withdrawing an assembly and its associated rod storage position partially into the cell is demonstrated safe by comparison to a series of KENO runs made for pool storage at a reactor site. To demonstrate the safety of flooding a reactor site storage pool filled with fresh Mark B fuel assemblies, an array of fuel assemblies 14 units wide, infinitely long, and reflected on the sides and bottom by concrete was calculated by KENO. Each assembly was spaced 1 foot from the other on the concrete reflector, as appropriate. Four cases at different degrees of pool flooding were evaluated and are described in Table 14-8.

TABLE 14-8

REACTIVITY FOR AN INFINITE BY 14-UNIT ARRAY OF FUEL ASSEMBLIES

Water Height	Calculated K_{eff}
Fully Flooded	0.951 ± 0.006
3/4	0.946 ± 0.007
1/2	0.928 ± 0.007
0 (dry)	0.506 ± 0.004

The confidence levels quoted above are one standard deviation. K_{eff} for the fully flooded condition is higher than that calculated by PDQ because of simplifications made in running the cases. The series of runs were to demonstrate safety of a partially flooded pool, a much more restrictive condition than partial withdrawal into one cell. The similarity in the Mark B and Mark C nuclear characteristics and the simplifying assumptions assure these calculations are also valid for the Mark C assembly type.

14.3.3.5.4 Assembly and Machine Shop and Development Test Areas - Assemblies of either Mark B or C disassembled in air are far less reactive than the cases listed in Table 14-6. Either assembly type may be disassembled in air. A safe reactivity level (<<0.95) is assured provided the handling in Section 4.2.3.6.4 is followed. The conditions stated in Section 4.2.3.6.4 are based on KENO calculations that show:

1. Two assemblies in air 21 inches or more apart are nuclearly safe.

2. Fuel pins at a maximum enrichment when optimumly moderated are fully reflected in an infinite slab have a K_{eff} = 0.95 if the slab is no more than 4 inches thick.

3. Fuel rods in any configuration or number, up to the number in the assembly, when limited to the confines of the assembly size are no more reactive than the intact assembly (Ref. Table 14-6).

14.3.3.5.5 Hot Cell Operations - Work within the hot cell will, by and large, follow existing controls. Three units in addition to an assembly and its associated rod storage position are permitted within Cell No. 1. Two of the three units are restricted to rods confined within an ever safe cross sectional area, i.e., a cross sectional area not exceeding that of a 22.5 cm diameter cylinder; in addition these two units must be free draining of any water. The third unit of Cell No. 1 under mass control is permitted. All other Hot Cells are limited to one unit each.

14.3.3.5.6 Fuel Rod Dismantlement - Fuel rods of either assembly type may be dismantled. Dismantlement can be performed in any area which present licensing conditions permit fuel handling. In addition, mass control must be limited to 350 grams of U-235, proper spacing must be maintained, and approved procedures must be followed.

14.3.3.5.7 Shipment and Disposal - The conditions of 4.2.3.6.7 are consistent with the above demonstration and/or current limits.

14.3.4 Building C

The demonstration for units and the array is similar to that of Building A (14.3.2.1 and 14.3.2.2). The values of all units in Building C are less than or equal to the value of the maximum storage unit defined in Table IV, TID-7016, Revision 1 (as amended), or they have been evaluated above in Section 14.3.2.2.

The allowable number of units on 36-inch centers is 90 units with at least 8 inches edge-to-edge between units. The allowable number of units according to Figure 22 of TID-7016, Revision 1, is about 190. The number of units has been reduced to 90 to permit the low enriched units. Administrative procedures for posting and controlling transfers of SNM to and from units are those described in 14.3.2.2.4.

14.3.5 Outside Storage

14.3.5.1 General - The underground storage and shipments are nuclearly isolated by distance or matter.

14.3.5.2 Underground Storage - The underground storage tubes are 5 inches in diameter, approximately 20 feet long and 20-inch centers. Maximum units demonstrated safe in Section 14.3.2.2 are stored, one per tube. These are neutronically isolated from each other by 15 inches of concrete.

14.3.6 Dry Waste

Nuclear criticality safety of dry waste is ensured by maintaining the concentration of SNM to a value much less than an ever safe concentration. Forty-five grams of SNM in a 55-gallon drum yields a concentration of less than 0.25 g/liter. These low concentrations are guaranteed by the nature of the material being stored which is contaminated laboratory waste. The nature of the waste as borne out by more than 20 years of experience will maintain an approximate uniform dispersion within the container. Dry waste containers are stored in the radioactive waste building after gamma scanning to ensure that the maximum SNM is not exceeded. There is therefore no requirement in the number or arrangement of containers within the radioactive waste building. One dimensional transport calculation shows that, at a U-235 concentration of 0.25 g/liter with optimum water moderation, a fully concrete reflected sphere having the same volume as 8×10^5 55-gallon drums has a neutron multiplication of ≤ 0.95. Therefore, the 45 grams of U-235 per drum limit is safe in that the maximum number of drums on site cannot credibly exceed 8×10^5.

14.4 ANALYTICAL METHODS AND VALIDATION REFERENCES

Nuclear criticality safety computer calculations presented in this chapter have used the computer codes NULIF, PDQ-07 and/or KENO. The physics codes NULIF and PDQ-07 are not only routinely used in nuclear criticality safety to evaluate highly moderated low-enriched systems but also are the standard codes used by the reactor design group of the Babcock & Wilcox Company (both codes have been certified by the Company's Quality Assurance Program for reactor calculations). The Monte Carlo code KENO is state-of-the-art in industry for nuclear criticality safety evaluations. These three computer codes are described in Appendix A, Section III, pages 3 through 12 of SNM License No. 1168 (Docket 70-1201); validation for these codes are given in Appendix A, Section III, of the same document on pages 19 through 21. Future calculations for nuclear criticality safety will make use of these codes and the Nuclear Criticality Safety Codes in SCALE 3 (NITAWL-S, XSDRNPM-S, KENO-IVS and KENO-Va). SCALE 3 is described in NUREG/CR-0200. Before use of the SCALE 3 package, the proper wording of the various codes will be assured and appropriate benchmarking activity will be carried out.

14.5 DATA SOURCES

Data and Guidance for Nuclear Criticality Safety is taken from one or more of the sources specified below.

1. Calculations using methods described in Section 14.4.

2. "Nuclear Safety Guide, TID-7016, Revision 2," NUREG/CR-0095 (ORNL/NUREG/CSO-6), (June, 1978.)

3. "Nuclear Safety Guide, TID-7016, Revision 1," (1961). TID-7016, Revision 1 is used only for application of the lattices density method which is Table IV and Figure 22 on page 26. Table IV has been modified according to information published in the Federal Register, March 5, 1963 on page 2130.

4. American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Oxide Reactors, ANSI/ANS-8.1-1983).

5. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water" DP-1014, Savannah River Laboratory (1966).

6. H. C. Paxton, "Criticality Control in Operations with Fissile Material," LA-3366(Rev), Los Alamos Scientific Laboratory, (1972).

7. R. D. Carter, et al, "Criticality Handbook," ARH-600 Revised last November 6, 1973, Atlantic Richfield Hanford Company.

14.6 FIXED POISONS

The site does not now use Fixed Poisons to maintain nuclear criticality safety.

14.7 STRUCTURAL INTEGRITY

Where structural integrity is necessary to provide assurance for nuclear criticality safety in any operation, the design and construction of those structures will be evaluated with due regard to load capacity and foreseeable abnormal loads, accidents and deterioration. This engineering activity is the responsibility of the Manager, Facilities with review and approval by a qualified person.

14.8 SPECIAL CONTROLS

There are no special controls for nuclear criticality safety at the site.

FIGURE 14-1

FUEL ROD REMOVAL SCHEMATIC - MARK B

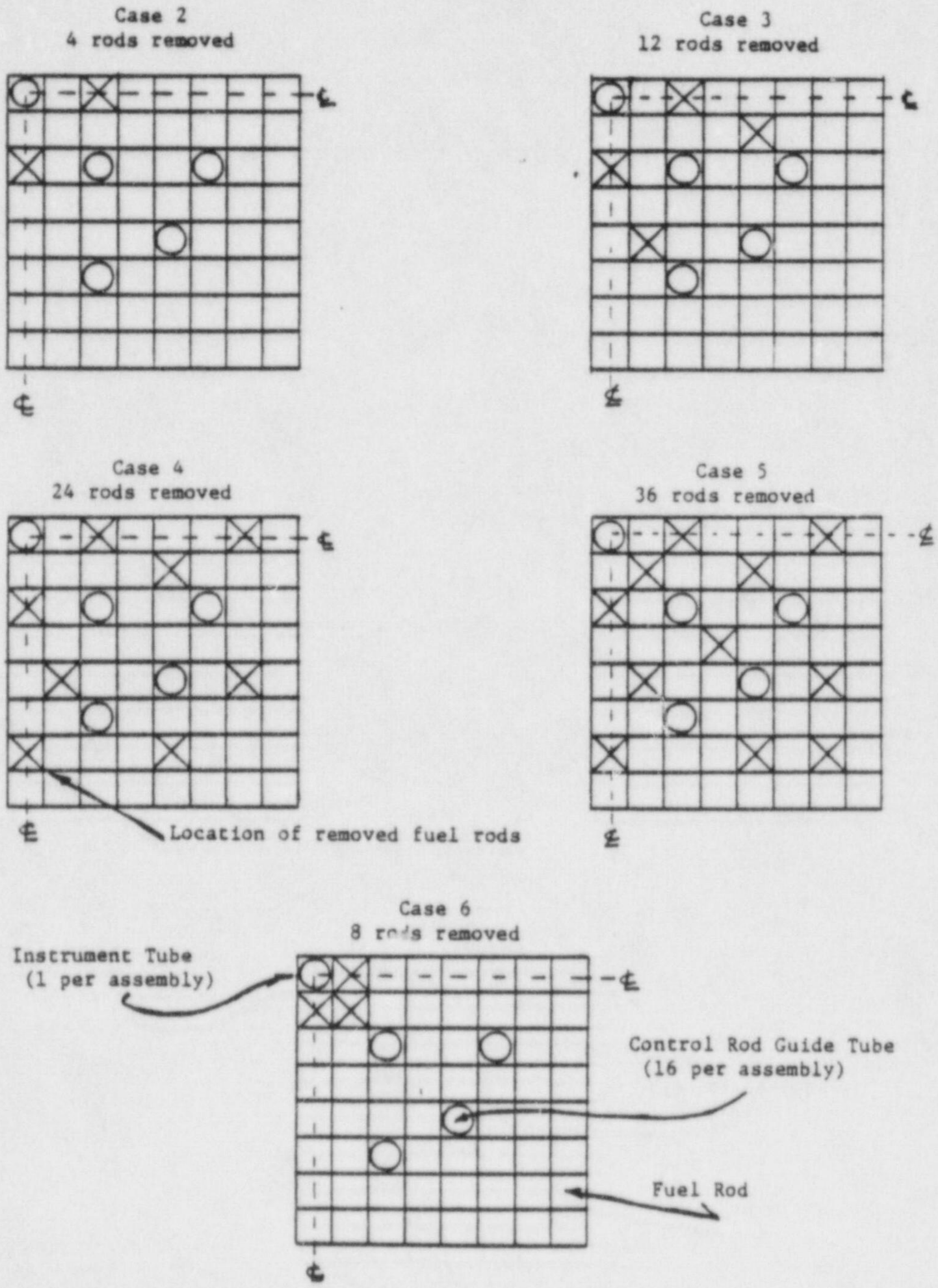
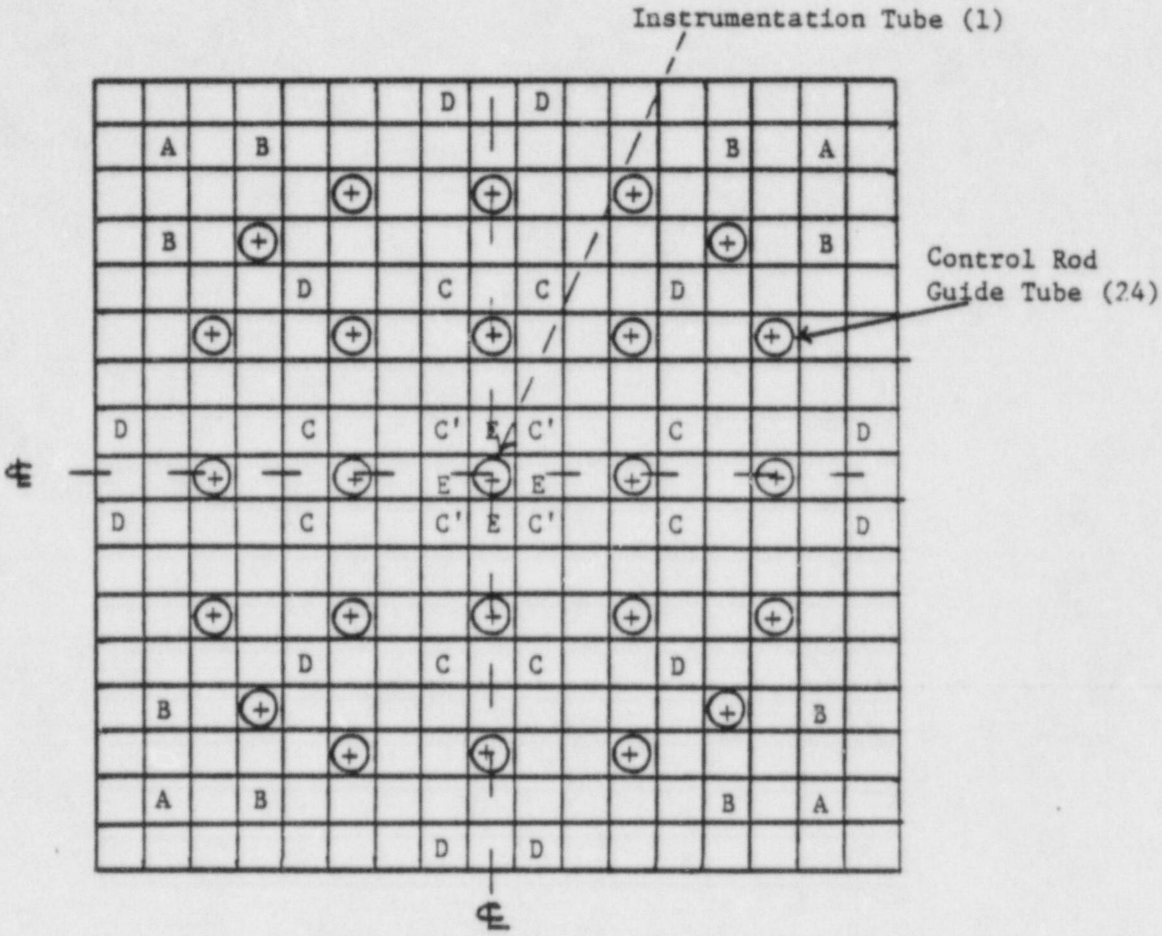


FIGURE 14-2

FUEL ROD REMOVAL SCHEMATIC - MARK C



MARK C FUEL ASSEMBLY (17 × 17)

Fuel Rods Removed for Reactivity Under Dismantlement (Section 4.4.7.3)

No. of Rods Removed	Pins Shown By Letter
4	A
12	A, B
24	A, B, C, C'
36	A, B, C, C', D
8	C', E

15.0 PROCESS DESCRIPTION AND SAFETY ANALYSES

The operations and projects at the site do not lend themselves to flow sheets. There is no operation presently in progress that has a regular measured feed material or a regular measured product output.