70-754

GE Nuclear Energy

POL/LPOR

General Electric Company
Vallecitos Nuclear Center
P.O. Box 460, Vallecitos Road
Pleasanton, CA 94566

December 9, 1992

Jerry J. Swift, Section Leader
Advanced Fuel and Special Facilities Section
Fuel Cycle Safety Branch
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: 1) License SNM-960, Docket 70-754.
2) Application for Renewal; April 21, 1989.

Dear Mr. Swift:

Enclosed is a revision of the application for renewal of License SNM-960 (Ref. 2). Changes were made as a result of discussions with your staff.

One major change from the previous license was the elimination of Table 1 from Appendix A. At your staff's request, Table 1 has been retained in a modified form. Justifications for the changes are given below. Please note that some of the "design criteria" were those used to design the Building 102 hot cell complex in the late 1950's and were never intended to be inflexible limits on operations. As no new hot cells are planned, these criteria have been removed.

Changes to Table 1

Facility		Criteria/Guidelines
A. Hot Cells	1.	Fire extinguishers. Fire extinguishers are not used in cells. A combination of limiting combustibles and ignition sources along with the ventilation system fire suppression system controls in-cell fire hazards.
	2.	Continuous air monitors are used. However, the 10 min. at 10 x MPC has no real meaning as it will vary with the projects conducted in any given cell.
	3.	Dose rate limits were original design criteria (actually based on material now licensed by the State of California) for normal operation. As VNC is a research and development facility, it was never intended that these numbers could not be exceeded for individual projects.
	4.	The liquid waste system is covered in Section 8.9.

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Facility		Criteria/Guidelines
B.	Storage Pools	1. Dose rate limits - see A.3. above.
		2. HEPA filters are addressed in other parts of Appendix A.
C.	Hoods	1. Uncontained Pu is no longer handled at VNC.
		2. Fire extinguishers. Site buildings are now sprinkled.
		3. HEPA filters - see B.2. above.
		4. Liquid waste - see A.4. above.
D.	Glove Boxes	Glove boxes no longer are used at VNC for License SNM-960 material. Three sealed glove boxes on site will be removed when disposal problems are resolved.

If you or your staff have any questions concerning these changes, please contact me at (510) 862-4330. Thank you.

Sincerely,

G. E. Cunningham
Senior Licensing Engineer

/ca

Enclosures

LICENSE CONDITIONS
FOR THE
VALLECITOS NUCLEAR CENTER

APPENDIX A

LICENSE SNM-960

DOCKET 70-754

GENERAL ELECTRIC COMPANY
VALLECITOS NUCLEAR CENTER
P.O. BOX 460
PLEASANTON, CA 94566

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LICENSE CONDITIONS FOR THE VALLECITOS NUCLEAR CENTER

Set forth herein are the technical and administrative requirements proposed for governing the receipt, possession and use of the special nuclear material and associated by-product material (as defined in Section 2.0 of this Appendix), subject to licensing and regulation by the Nuclear Regulatory Commission, in activities other than reactors at General Electric's Vallecitos Nuclear Center near Pleasanton, California.

1.0 AUTHORIZED ACTIVITIES

The following types of activities are authorized.

1.1 PRODUCT PROCESSING OPERATIONS

Assembly, modification, cleaning and repair of unirradiated encapsulated experimental assemblies.

1.2 LABORATORY OPERATIONS

1.2.1 Chemical

Analysis of the chemical and isotopic composition, concentration and behavior of special nuclear materials by wet chemistry and physical measurement techniques.

1.2.2 Metallurgical

Physical analyses and testing of physical and metallurgical properties of special nuclear materials.

1.2.3 Physics and Health Physics

Measurements of radiation and its effects on instruments and on the structure and composition of materials.

1.2.4 Hot Laboratories

Post-irradiation examination, testing, and analysis of fuel elements and materials in shielded enclosures by remote manipulative techniques; research and development and/or pilot plant activities.

1.2.5 Research and Development

Including but not limited to the above.

1.3 GENERAL SERVICES OPERATIONS

1.3.1 Equipment Maintenance and Engineering

Design, fabrication, and testing of equipment containing special nuclear materials and maintenance of such equipment.

1.3.2 Storage

Storage of special nuclear materials other than wastes in shielded containers and/or at locations as applicable and in designated general purpose storage areas.

1.3.3 Transportation and Transfer

Inspection of packaging and preparation for shipment and/or transfer of special nuclear materials.

1.3.4 Decontamination

Decontamination of equipment and facilities.

1.4 WASTE TREATMENT

1.4.1 Liquids

Concentration of the radioactive constituents of liquid wastes by evaporation, chemical treatment, sedimentation, filtration, and ion exchange; solidification and packaging of concentrates for disposal.

1.4.2 Solids

Packaging and storage of wastes contaminated with or containing nonreclaimable special nuclear materials, excluding direct burial in soil.

2.0 POSSESSION LIMITS

The following quantities of special nuclear materials are authorized at the Vallecitos Nuclear Center.

2.1 U-235: 50 kg enriched to less than or equal to 10% for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant by-product material and reactor-produced transuranics.

2.2 U-235: 4 kg enriched to more than 10% for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant by-product and reactor-produced transuranics.

2.3 Plutonium: 100 grams in contained or sealed form in addition to the irradiated quantities as referenced in Subsections 2.1 and 2.2 above.

2.4 U-233: 100 grams in any form.

3.0 DEFINITIONS

As used in the proposed license conditions set forth herein:

3.1 Area Manager means the lowest supervisory position fully responsible for the specific activity or function with which the term is associated. The generic term "Area Manager" does not necessarily refer to the title of any specific position in General Electric's system of organization and position nomenclature.

3.2 Array means two or more interacting fissile material accumulations.

3.3 Criticality Area means any physically identified area or location within which fissile materials are handled under the direction of a single area manager. A criticality area may include more than one criticality limit area.

3.4 Criticality Control means the administrative and technical requirements established to minimize the possibility of achieving inadvertent criticality in the environment analyzed.

3.5 Criticality Limit Area means a designated and physically identifiable locality within which a specific set of criticality control limits governs the use of fissile materials.

3.6 Fissile Material Accumulation means any single accumulation of fissile material; for example, one can of UO_2 powder, one can of fuel pellets, or one fuel element.

3.7 Full Reflection means that degree of reflection equivalent to a tight-fitting shell of greater than 12 inches of water.

3.8 Homogeneous System means a system containing compounds of fissile materials in the form of uniformly distributed particles and fissile compounds in solution.

3.9 Heterogeneous System means a system containing compounds of fissile materials in the form of clumped fissile compounds (e.g., fuel rods in water).

3.10 Normally Subcritical Values means those maximum values which provide safety under normal conditions of operation. Further adjustment of such values may be necessary to incorporate safety margins for the activity as it is conducted in the plant allowing for credible mishaps that could occur in the actual plant situation.

3.11 Nuclear Energy Operations means those components of the General Electric Company engaged in the various aspects of nuclear energy and does not refer to a specific component by title.

3.12 Nuclear Safety means that field of safety comprised of criticality safety and radiation safety.

3.13 Safe Batch means an accumulation of fissile material which is 45% of the critical accumulation considering enrichment, full reflection, and optimum water moderation consistent with the form of the material.

3.14 Site Safety Manager means an individual designated by upper management as having the responsibility for nuclear safety at the entire VNC site.

3.15 Special Nuclear Material means plutonium, Uranium-233, Uranium-235, or any material artificially enriched by any of the foregoing.

3.16 Subcritical Area means a physically identified area or location involving special nuclear materials in quantities of less than 500 grams of U-235, 300 grams of U-233, or 300 grams of plutonium or a prorated combination of such materials under the direction of a single SNM custodian and which is unrelated to any other area where special nuclear materials are handled (a subcritical area is considered unrelated when it meets the isolation requirements of Section 6.10 of this Appendix and is not located in the same room); or an unrelated building or structure under the direction of a single SNM custodian which meets the criteria of 10CFR70.24(a).

4.0 GENERAL ADMINISTRATIVE REQUIREMENTS

4.1 AREA MANAGERS

Operations and activities in a specific criticality area or radiation area shall be directed by the designated area manager. The responsibility for safe operation and control of activities in the area and for the safety of the environs as influenced by the activities conducted therein shall be vested in this position. An area manager shall be proficient in the application of the VNC radiation protection program as it relates to limitations and radiological controls on work activities in his assigned radiation or radioactive materials area. Additionally, each area manager of a criticality area shall be proficient in the application of criticality control procedures and be knowledgeable in the procedures applicable to the criticality area under his management.

4.2 CRITICALITY SAFETY COMPONENT

The Criticality Safety component is defined as that component with designated responsibility to provide authoritative professional advice and counsel to area managers on matters of control against accidental criticality and to measure the effectiveness of the criticality control program.

4.2.1 The functions of the Criticality Safety component shall exclude direct responsibility for operations involving the use of fissile materials, and the Criticality Safety component shall not report to an area manager responsible for an area where fissile material is handled other than a subcritical area.

4.2.2 The Criticality Safety component shall include at least one technically trained person with a bachelor's degree in science or engineering and three years experience in the nuclear field, including one year of directly relevant criticality safety

experience. Should the Criticality Safety component lack the one year of experience, all analyses for the interim period until the experience is achieved shall be verified by two of the individuals described in 5.5 or by the physics advice function in 5.4 of this Appendix.

4.2.3 The manager of the Criticality Safety component shall hold a bachelor's degree in science or engineering and have at least five years experience in a responsible position in a nuclear field such as engineering, physics, or chemistry.

4.3 RADIATION SAFETY COMPONENT

The Radiation Safety component is defined as that component with designated responsibility to provide authoritative professional advice and counsel to area managers at the Vallecitos Nuclear Center on matters of radiation protection and ALARA and to measure the effectiveness of the radiation protection program.

4.3.1 The functions of the Radiation Safety component shall exclude direct responsibility for operations involving the manufacture of nuclear products or processing of nuclear materials. The Radiation Safety component shall be responsible to establish and maintain the radiation safety program to ensure the protection of employees at the Vallecitos Nuclear Center and of the community. The radiation safety program shall include as a minimum: the evaluation of release of radioactive effluents and materials from the site, establishment of procedures and training programs to control contamination and exposure to individuals, the review of calibration and maintenance activities for radiation detection instruments, maintenance of appropriate records and reports, review of radioactive material handling practices, and review of change procedures (see Section 4.6), including ALARA considerations.

4.3.2 The minimum qualifications of personnel assigned functional responsibilities in the Radiation Safety component shall be as follows:

a. Manager: B.S. degree in science or engineering with five years experience in assignments involving radiation protection.

b. Specialists: B.S. degree in science or engineering with two years experience in assignments involving radiation protection or eight years of experience in health physics or radiation protection.

c. Monitors: High school with two years experience in handling radioactive materials or two years of college and four months experience; also, successful completion of a General Electric certification program which includes written and oral examinations covering radiation protection procedures and/or criticality procedures.

4.4 VALLECITOS TECHNOLOGICAL SAFETY COUNCIL

The functions of the Vallecitos Technological Safety Council (VTSC) shall include responsibility for review of reportable incidents and the nuclear safety program, and for contributing professional advice and counsel on criticality and radiation safety policy. The VTSC shall review annually the site safety and compliance program performance to include effluent releases and occupational exposures in terms of ALARA and focus on trends for corrective action as necessary.

Its deliberations in any calendar quarter shall be reported in writing to the Site Safety Manager (SSM) (see Section 3.14) and retained two years. Should the VTSC choose not to meet in any calendar quarter, the SSM shall be notified in writing. The Vallecitos Technological Safety Council is an independent review body and shall consist of at least five senior members of General Electric's technical and/or management personnel appointed by the SSM and shall include competence in the physics, chemistry, and engineering disciplines.

4.5 COMPLIANCE POLICY

General Electric shall establish and maintain a policy statement in writing expressing requirements for compliance with the terms and conditions of special nuclear materials licenses and applicable NRC regulations. Each area manager shall know and understand the provisions of this policy.

4.6 CHANGE PROCEDURES

Activities which do not involve a change in license conditions but which require procedures, facilities, or equipment substantially different from those previously used shall not be initiated until the Radiation Safety component has completed a review and technical evaluation to assure adequacy of health and safety features and compliance with license conditions and ALARA policy where applicable.

In a criticality limit area, such activities shall not be initiated until the Criticality Safety component has completed a criticality analysis or a review of the existing criticality analysis for the proposed activity. Such changed activities shall be initiated in accordance with written procedures issued by the area manager.

5.0 CRITICALITY CONTROL ADMINISTRATIVE REQUIREMENTS

5.1 PROCEDURES

Area managers shall develop and maintain written criticality control procedures incorporating limitations established by the Criticality Safety component and shall assure that these procedures are made readily available to concerned personnel through posting of limits or other appropriate notifications.

5.2 ANALYSIS REQUEST

A request for criticality analysis or review of existing analyses shall be prepared in writing by, or under the direction of, an area manager for any activity or change in activity requiring a change in criticality controls or changes in environment of possible consequence to criticality safety. The activity or change shall not be initiated until receipt of an approved criticality analysis or documentation of a review.

5.3 CRITICALITY CONTROL ANALYSIS

Qualified individuals of the Criticality Safety component shall be responsible for the establishment of safe batches, safe geometries, safe concentrations, safe spacing of special nuclear materials, and limitations of other nuclear parameters utilizing information set forth in the request for criticality analysis and verification and assessment of normal and potential environmental conditions of significance to criticality control.

5.4 NEUTRON PHYSICS ADVICE

An organizational component competent in neutron physics and the technical methods for criticality computations shall verify the use of any new computational methods for criticality control. This component also shall provide technical review and verification of criticality safety analyses as requested by the manager of the Criticality Safety component pursuant to 5.5.

5.5 VERIFICATION OF CRITICALITY SAFETY ANALYSIS RESULTS

The manager of the Criticality Safety component shall assure that the results of criticality analyses are reviewed. This review may be performed by the manager of the Criticality Safety component provided he has acquired a minimum of two years experience in the methods used or by a member of the Criticality Safety component other than the individual who performed the analysis or an independent reviewer of equivalent education and experience. The neutron physics advice function (5.4) may provide technical review of computation methods. The review need not involve a recalculation of the results.

5.6 RECORDS

Records of criticality control analyses shall be maintained in sufficient detail and form to permit independent review of the method of calculation and results. Such records shall be retained for a period of at least six months following the cessation of activities to which they apply.

5.7 CRITICALITY CONTROL INSPECTIONS

5.7.1 The Radiation Safety component shall be responsible for inspecting each criticality area on an ongoing basis to assure compliance with criticality control procedures. Items to be inspected shall include, but not be limited to, the posting of criticality control limits, the labeling of SNM containers with information needed for criticality

control purposes, the movement of SNM, and general criticality safety work practices.

5.7.2 A program of inspection shall be performed by the Criticality Safety component to determine that actual operations conform to the physical situations on which the calculations of criticality limits have been based. Inspection reports shall be furnished to area managers. Where situations are identified which require corrective action, such reports shall so indicate. Corrective or follow-up action shall be taken in accordance with VNC Safety Standards and/or site Nuclear Safety Procedures.

Operations which handle routinely in process more than one safe batch (as defined in Section 3.13) of fissile material shall be inspected on a quarterly basis. All operations shall be inspected on an annual basis.

5.8 TRAINING PROGRAM

Area managers shall assure that new employees receive instruction in criticality safety and plant operating and emergency procedures prior to their working with special nuclear materials in excess of one safe batch (see Section 3.13). A criticality control training program, approved by the Criticality Safety component, shall be maintained to emphasize the need for following criticality control procedures and to aid personnel in understanding the various parameters which are essential to the maintenance of subcritical conditions. The program may be conducted by the Criticality Safety component, some other portion of the Nuclear Safety component, or combined with training performed by operating components. It may be combined with radiation safety training. A written test shall be completed by each employee taking the course. The test shall be evaluated and the results forwarded to the appropriate area manager.

Employees requiring criticality safety training shall receive refresher training at least biennially.

5.9 MONITOR ALARM SYSTEM

5.9.1 No transfers of fissile materials between criticality limit areas shall be permitted in criticality areas required to have a monitor alarm system unless the system is operable. Transfers may resume following repair and verification of the monitoring system's operability.

5.9.2 Exemption from the monitor alarm requirement of Section 70.24(a)(2) of 10CFR70 is granted for the Remote Handling Operation (RHO) pool and hot cells. At the high-level solid waste storage facility and the storage pit in the RHO pool area, for the purpose of compliance with 70.24(a)(2) of 10CFR70, the source of a possible accidental condition of criticality may be considered as the accessible surface of the earth or concrete shielding.

5.9.3 Specific exemption from the prescribed preset alarm points (5 to 20 mRem/h) is granted for those monitoring devices where the routine movement of by-product materials could result in gamma fields at the monitoring devices in excess of those that would result in a dose rate of 20 mR/h. Further, the alarm setting for these monitoring devices under the above conditions shall not exceed 500 mRem/h.

5.9.4 Exemption from the requirements of Section 70.24 of 10CFR70 is granted for each area in which there is not more than one shipment of packages containing special nuclear materials licensed pursuant to 10 CFR Part 71 for transport outside the confines of the Vallecitos Nuclear Center (packages in any shipment which depend on special arrangement for nuclear safety shall be retained in that same arrangement during such storage); or one safe batch of finished reactor fuel rods or assemblies, providing no activities could cause rearrangement of fuel-bearing portions into more reactive configurations; or which meets the requirements of a subcritical area as defined in Section 3.15 of this Appendix.

6.0 CRITICALITY CONTROL CONDITIONS - TECHNICAL AND ANALYTICAL REQUIREMENTS

6.1 GENERAL REQUIREMENTS

Criticality controls shall be based on the normal environmental conditions in the criticality area and also on all credible abnormal conditions that could affect the criticality safety in that area. These factors shall include but not necessarily be limited to moderation, reflection, neutronic interaction, fissile material accumulation size, physical form, chemical and isotopic composition, nuclear poisons, density, concentration, reliability of equipment, type of operation, and training and supervision of operating personnel.

In evaluating the subcriticality of individual accumulations, process configurations, or arrays of fissile material, the two-contingency criterion shall be applied; that is, two unlikely, independent and concurrent accidents or changes in process conditions must occur before criticality is credible.

6.2 CALCULATIVE METHODS

Criticality analyses shall utilize calculative methods that have been validated by the calculation of models of critical experiments of similar geometrical configuration and material content; the maximum negative bias in the multiplication factor, defined as $(-\Delta k_{bias}) = k_{calculation} - k_{experiment}$, determined in the validation shall be applied to the analysis as given in the relationship below. The effective multiplication factor of the accumulation or array of accumulations shall satisfy the relationship:

$$k_{cal} + \Delta k_{uncertainty} + (-\Delta k_{bias}) \leq 0.95$$

where: k_{cal} is the calculated effective multiplication factor by the calculative method, and $\Delta k_{uncertainty}$ is the statistical uncertainty (2σ) in the mean value of the calculated effective multiplication factor, and

($-\Delta k_{bias}$) is the maximum negative bias determined from the validation of the calculative method. If the bias is positive, the ($-\Delta k_{bias}$) term shall not be used, so that overstatements in the effective multiplication factor shall not be corrected.

Calculative methods shall be subject to independent review and verification by persons with demonstrated competence in neutron physics computations before use.

6.3 NORMALLY SUBCRITICAL VALUES - INDIVIDUAL ACCUMULATIONS

Facilities and processes involving the accumulations listed below shall adhere to the two-contingency principle (Section 6.1). The two-contingency principle may be achieved through physical constraints and inventory controls or a combination thereof.

Normally subcritical values of nuclear parameters for individual accumulations under normal operating conditions shall be as shown below except that the effective multiplication factor shall not exceed 0.95 in any case.

6.3.1 For accumulations limited by mass, the normally subcritical mass shall not exceed 74% of the critical mass.

6.3.2 Not more than 45% of the critical mass shall be contained in any accumulation in a criticality limit area in which double batching is credible.

6.3.3 For accumulations limited by volume, the normally subcritical volume shall not exceed 76% of the critical volume.

6.3.4 For accumulations limited by dimension, the normally subcritical cylinder diameter or slab thickness shall not exceed 90% or 88%, respectively, of the critical dimension where U-235 is the fissile constituent, nor 85% in all other cases.

6.3.5 For accumulations limited by concentration, the normally subcritical concentration shall be less than that concentration which is equivalent to the following atomic ratios of hydrogen to fissile material: H/U-235, 5200; H/U-233, 7600; and H/Pu (fissile), 7600.

6.3.6 Accumulations limited by the effective multiplication factor shall be limited by Section 6.2.

6.4 NORMALLY SUBCRITICAL VALUES - INTERACTING ACCUMULATIONS

Facilities and processes involving arrays of fissile materials shall adhere to the two-contingency principle (Section 6.1). The two-contingency principle may be achieved through physical constraints and inventory controls or a combination thereof.

Normally subcritical (as defined in Section 3.10) numbers of interacting accumulations shall be as listed below except that the effective multiplication factor for the array shall not exceed 0.95.

6.4.1 Not more than 69% of the critical number of accumulations whenever the critical number has been determined by experiment; or

6.4.2 Not more than 50% of the critical number of accumulations in any array calculated by the density analog method (Ref.: ARH-600); or

6.4.3 The number of accumulations determined by the solid angle method as described in TID-7016, Rev. 2, paragraphs 4.29 through 4.34. The method may be applied only to accumulations that are well moderated (e.g., aqueous solutions) at the point of the maximum multiplication factor, and array reflection must be no more effective than a thick water reflector spaced at distances from the accumulations comparable to the spacing between accumulations.

6.4.4 Interacting accumulations limited by the multiplication factor shall be limited by Section 6.2.

6.5 INTEGRITY OF STRUCTURES

6.5.1 Whenever criticality control is directly dependent on the integrity of a structure used to retain a special nuclear material storage array, the structure will be designed with an adequate strength factor to assure against failure under foreseeable loads or accident conditions. Materials of construction will be selected to resist fire, and the degree to which any corrosive environment might affect nuclear safety will be considered and corrosion-resistant materials or coatings applied as necessary. The design shall be reviewed by a qualified engineer.

6.5.2 Whenever criticality control is directly dependent on the integrity of a neutron isolating structure, the structure will be designed to assure against loss of integrity through foreseeable accident conditions such as fire, impact, melting, corrosion or leakage of materials. The design shall be reviewed by a qualified engineer.

6.6 OPTIMUM CONDITIONS OF MODERATION AND REFLECTION

Unless moderation or reflection is controlled by physical and administrative means in accordance with Sections 6.7 or 6.8 below, calculations of criticality control limits shall be made assuming that optimum light-water moderation and full reflection by light water exist, consistent with the form of the fissile material and its container. When mass limitation is to be employed for criticality control of an accumulation of fissile material, optimum moderation is that degree of moderation which minimizes the mass of material that is required to achieve criticality. When a geometric limitation is to be employed for criticality control of an accumulation of fissile material, optimum moderation is that degree of moderation which maximizes the material buckling.

6.7 CONDITIONS OF MODERATION OTHER THAN OPTIMUM

Notwithstanding Section 6.6 above, criticality control limits may be based on conditions of moderation other than optimum when all of the applicable following conditions exist:

6.7.1 The fissile material specifications require the state of moderation be greater or less than optimum.

6.7.2 No single credible mishap will result in a more effective degree of moderation than assumed in the specific criticality analysis.

6.7.3 The undermoderated material accumulation is either contained within a noncombustible water barrier, or of such physical form and natural arrangement that the application of water cannot in any credible mishap result in a more effective degree of moderation than that assumed in the specific criticality analysis.

6.7.4 The overmoderated fissile material accumulation is assayed and periodically inspected or sampled to assure the proper chemical concentration and composition are maintained.

6.8 CONDITIONS OF REFLECTION OTHER THAN FULL REFLECTION

Notwithstanding Section 6.6 above, criticality control limits may be based on less than full reflection when all of the following conditions are satisfied:

6.8.1 The fissile material environment provides the unreflected condition or a specific degree less than fully reflected.

6.8.2 The environment of the fissile material is such that more effective reflection cannot occur through a single credible mishap.

6.8.3 The fissile material accumulation is within a structure that will preclude more effective reflection by personnel or other objects in the environment.

6.9 MODERATORS

If special nuclear materials are to be stored, handled or used in the presence of heavy water, beryllium or graphite, possible moderating and reflecting effects of those materials shall be considered.

6.10 NUCLEAR ISOLATION

Two accumulations or arrays of accumulations may be considered as being nuclearly isolated from each other only if an edge-to-edge separation exists which is not less than one of the following or its nuclear equivalent:

6.10.1 Twelve inches of water.

6.10.2 The larger of 12 feet or the greatest distance across an orthographic projection of either accumulation or array on a plane perpendicular to a line joining their centers.

7.0 RADIATION CONTROL ADMINISTRATIVE REQUIREMENTS

7.1 RADIATION STANDARDS

General Electric shall establish and maintain a comprehensive set of standards for operational health and safety, including ALARA considerations. Such standards shall be reviewed by the Radiation Safety component and the appropriate area managers prior to issuance. The Radiation Safety component shall review the standards biennially thereafter.

7.2 RADIATION CONTROL INSPECTION

Activities involving special nuclear material shall be inspected by the Radiation Safety component on a quarterly basis. Conditions of an unusual or uncertain nature that could lead to radiological health and safety problems shall be referred to the area manager immediately for correction.

7.3 RADIATION SAFETY TRAINING

A training program in radiation protection shall be in effect and shall include training in the requirements of 10CFR19 and 10CFR20, methods of controlling radiation exposure, license requirements, protective methods, and basics of radiation effects. All personnel working in radioactive materials areas or in radiation areas shall receive an indoctrination lecture prior to starting work followed by additional training commensurate with the work environment as determined by the area manager.

Personnel working with radioactive materials or assigned to work in radiation areas shall receive refresher training at least biennially.

8.0 RADIATION CONTROL CONDITIONS - TECHNICAL AND OPERATING REQUIREMENTS

8.1 AIR CONTAMINATION CONTROL

The following design criteria for ventilation systems shall be used to provide air contamination control.

8.1.1 Airflow shall be from areas of lesser contamination to areas of higher contamination. Potential accident conditions shall be considered.

8.1.2 Duct flow velocities and design shall be such to minimize possible accumulation of contamination.

8.1.3 Air shall be sampled continuously in normally occupied areas in which dispersible SNM is handled. Samples are analyzed for gross alpha and gross beta-gamma.

8.1.4 In general, potential airborne radioactivity-producing operations shall utilize close-capture ventilation devices, e.g., hoods and high-velocity local exhaust.

8.1.5 All ventilation exhaust systems for facilities that routinely handle dispersible radioactive materials in quantities in excess of the limits specified in Appendix C of 10CFR20 shall be sampled continuously as proof of filter performance.

8.1.6 At least one filter in each such stream (8.1.5) shall be equipped with a device for measuring differential pressure that shall be read monthly. Filters shall be changed when readings deviate from specification values or following evidence of damage.

8.2 INSTRUMENT CAPABILITY

Dose rate range capabilities of portable instrumentation shall include the following:

gamma	-	1 to 50,000 mR/h (10-5,000 μR/hr for environmental monitoring)
beta	-	4 to 200,000 mrad/h
neutrons	-	1 to 5,000 mRem/h

8.3 CONTAMINATION DETECTION

8.3.1 Contamination detection capabilities of portable instrumentation shall be:

alpha	-	200 to 300,000 disintegrations per minute
beta-gamma	-	twice background to 80,000 counts per minute on a Geiger-Mueller counter

8.3.2 Contamination detection capabilities of portable instrumentation when used in the smear survey technique shall be 200 disintegrations per minute for alpha and 500 disintegrations per minute beta-gamma.

8.3.3 Detection equipment shall be maintained on site to meet the requirements of Section 8.11 of this Appendix for contamination-free items.

8.4 SAMPLE DETECTION

Sample detection capabilities for laboratory analysis of effluents shall be ≤0.1 times the concentrations specified in 10CFR20, Appendix B, Tables I and II.

8.5 EXPOSURE DETECTION

Personnel dosimeters shall be capable of detecting gamma, beta, and x-ray radiation. Additional neutron detection capability shall be available as appropriate. Primary dosimeters (film badges, TLD dosimeters) are worn in a manner as to record the most representative exposure.

8.6 INSTRUMENT CALIBRATION

Portable monitoring instruments shall be calibrated upon initial acquisition, after major maintenance, and at least annually.

Fixed gamma area monitors shall be source checked at least annually.

8.7 LABORATORY COUNTING INSTRUMENTS

Control charts indicating system performance shall be maintained on all laboratory instruments used for counting health physics samples.

8.8 INSTRUMENT CHECK SOURCES

Field check sources shall be available for use in functional response checks of portable radiation-measuring instrumentation.

8.9 LIQUID WASTE DISPOSAL

Potentially contaminated liquid wastes shall be collected, solidified, and disposed of as solid waste.

8.10 AIRBORNE EFFLUENT CONTROL

Potentially contaminated airborne effluents shall be released through HEPA filter systems which shall be at least 99.95% efficient for 0.3-micrometer particles. Such effluents shall be limited at the point of release to the atmosphere so that the annual average concentration at the site boundary does not exceed the concentrations specified in 10CFR20, Appendix B, Table II, Column 1. The limits at the points of release shall be based on a χ/Q calculation, reconcentration factors, and the effects of other site stacks. If the environmental sampling program indicates a reconcentration of these materials, the release levels shall be lowered such that the reconcentration trend is reversed.

These represent maximum release limits. Actual normal releases shall be controlled to meet ALARA objectives (see Section 10.0).

8.11 CONTAMINATION-FREE ARTICLES

Articles which have been handled, used or stored in areas with a potential for contamination with radioactive materials may be disposed of or transferred to persons not licensed to possess radioactive materials when all of the following conditions are satisfied:

8.11.1 Either all surfaces are accessible for survey or it is reasonable to assume from the design and usage that no radioactive materials could have contaminated inaccessible surfaces without having contaminated the accessible surfaces as well.

8.11.2 Articles are considered contamination free which meet the requirements of Annex C to this license.

8.12 BIOASSAY

Persons who work routinely in areas where there is potential for internal deposition of radioactive materials shall be subject to determination of the extent of radioactive materials intake by techniques such as analysis of urine or whole body counting.

Bioassay data shall be evaluated by the radiation safety function.

8.12.1 Whole Body Counting: VNC employees shall receive a whole body count at least annually. An investigation shall be initiated if a whole body count result exceeds 10% of a Maximum Permissible Body Burden or 100% of an ICRP Investigation Level (Ref.: ICRP Publication No. 10).

8.12.2 Urinalysis: Urinalysis may be performed on a routine or on a job-by-job basis when radioactive materials which cannot be directly detected by the whole body counter and which are not tagged with isotopes detectable by the whole body counter are handled. All samples above background are investigated.

8.13 HIGH RADIATION ALARM EXEMPTION

Exemption from the provisions of 20.203(c)(2) of the Commission's regulations is granted for an area or location which is located within the 94-acre inner fenced exclusion area of the Vallecitos site provided such area or location is bounded by a distinctly colored rope or chain providing a barrier where dose rate values are not higher than 100 mR/h and is posted as a High Radiation Area in accordance with 10CFR20.203(c) and is:

8.13.1 used for the temporary storage of strong, weatherproof containers not exceeding 1,000 mR/h at any point of readily accessible surface; or

8.13.2 an area established for any continuous period of 30 days or less in which there are contained radioactive materials, the readily accessible surfaces of which do not exceed 1,000 mR/h at any time and which do not exceed 100 mR/h at the barrier.

TABLE 8.1. EQUIPMENT AND FACILITY DESIGN CRITERIA AND GUIDELINES

Facility	Alarms, Interlocks and Safety Features	Basis
Hot Cells	Door interlock (operationally checked twice per year).	Prevents both doors from opening simultaneously during normal operation. Interlock can be nullified during cell cleanout.
	Stepped plugs in cell penetrations.	Prevents beam-type exposure to personnel at cell face.
	Check valves on all inlet lines for liquids and gases.	Prevents release of contaminants to work area through backflow.
	Effluent air double filtration - HEPA filters.	Provides contingency in case of cell fire and loss of first filter.
	> 0.02 in. of H_2O ΔP (instrument checked for accuracy annually).	Prevents blow-back to working area; positive flow through openings around plugs, etc., > 125 ft/min.
Underwater Storage Pools	Anti-syphon drain.	Prevents water draining out.
	Automatic filling.	Prevents loss of shielding by evaporation, etc.
Hoods	Airflow designed to 125 lineal ft/min (flow checked quarterly).	Prevents carryout of radioactive materials.

9.0 ON-SITE TRANSFERS

9.1 TYPE A AND UNIRRADIATED SNM

Transfers of Type A quantities of radioactive materials or transfers of unirradiated SNM between buildings on the VNC site shall be made in such a manner as to limit the probability of any release of radioactive material to the environs of the VNC site.

9.2 TYPE B

Transfers of Type B irradiated special nuclear materials with associated by-product materials may be made in containers approved by the NRC pursuant to 10CFR71 but without the crash/fire overpacks or in containers for which written and approved procedures have been generated in accordance with Section 4.6 of this Appendix.

10.0 EFFLUENT AND ENVIRONMENTAL SAMPLING

10.1 EFFLUENT SAMPLING

Gaseous and liquid effluents are sampled for compliance with 10CFR20 at the site boundary. Action levels have been assigned to various parameters. When an action level is reached, the cause must be investigated. (Note: Corrective action may or may not be required.)

10.1.1 Gaseous Effluent Sampling (Continuous)

Table 10.1 Stack Action Levels

Parameter	Concentration ($\mu Ci/cm^3$)
Gross Alpha	3.0×10^{-14}
Gross Beta-Gamma	1.4×10^{-11}
I-131	4.6×10^{-11}
Noble Gases	6.4×10^{-6}

The values in Table 10.1 are based on the Building 102 stack. Not all stacks are sampled for all parameters.

10.1.2 Liquid Effluent Sampling

Each site retention basin is sampled and the sample counted prior to release to the environment.

Table 10.2 Liquid Effluent Action Levels

Parameter	Action Level ($\mu Ci/cm^3$)
Gross Alpha	3.0×10^{-8}
Gross Beta-Gamma	5.0×10^{-8}

10.2 ENVIRONMENTAL SAMPLING

10.2.1 Groundwater Sampling

Two groundwater sampling stations are maintained on site. One is located south of the main active operations areas in the main groundwater flow. The second is located upstream. They are sampled on a quarterly basis.

Table 10.3 Groundwater Action Levels

Parameter	Action Level ($\mu Ci/cm^3$)
Gross Alpha	3.0×10^{-8}
Gross Beta-Gamma	5.0×10^{-8}

Additional wells are located around Building 102 to monitor potential contamination from a leak in the building storage pool (repaired in 1975). No action levels are assigned to these wells.

10.2.2 Stream Bottom

The bottom soil at the point where the natural drainage ditches exit the site is sampled annually.

Table 10.4 Stream Bottom Action Levels

Parameter	Action Level
Gross Alpha	3.0×10^{-5} $\mu Ci/gm$
Co-60	8.0×10^{-6} $\mu Ci/gm$
Cs-137	1.5×10^{-5} pCi/gm

10.2.3 Vegetation

Vegetation is sampled annually.

Table 10.5 Vegetation Action Levels

Gross Alpha	1.0×10^{-5} pCi/gm
Gross Beta-Gamma (after adjustment for K-40)	5.0×10^{-5} pCi/gm

SPECIAL NUCLEAR MATERIAL
LICENSE RENEWAL APPLICATION
FOR THE
VALLECITOS NUCLEAR CENTER

MARCH 27, 1989

REVISED DECEMBER 1, 1992

LICENSE SNM-960

DOCKET 70-754

GENERAL ELECTRIC COMPANY
VALLECITOS NUCLEAR CENTER
P.O. BOX 460
PLEASANTON, CA 94566

SPECIAL NUCLEAR MATERIAL
LICENSE RENEWAL APPLICATION
FOR THE
VALLECITOS NUCLEAR CENTER

(REVISED 12/1/92)

INTRODUCTION

The General Electric Company has been engaged in nuclear energy work since 1955. One of the major installations at which the Company conducts this work is the Vallecitos Nuclear Center located near Pleasanton, California.

OBJECTIVES OF THIS APPLICATION

The basic objective of this application is to obtain a renewal of License SNM-960, Docket 70-754.

APPLICATION

General Electric hereby submits the enclosed information describing all activities at the Vallecitos Nuclear Center in which special nuclear materials are utilized, except those activities subject to licensing pursuant to 10 CFR Part 50, in support of our application for renewal of License SNM-960. Several sections contained in the previous submittal have been eliminated due to reduced activities. The facilities previously described in the eliminated and/or replaced sections have been discontinued. On the basis of this information, General Electric requests that License SNM-960 be renewed.

By_____

G. E. Cunningham
Senior Licensing Engineer
Irradiation Processing

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1.0 GENERAL INFORMATION

1.1 CORPORATE AND FINANCIAL

This application is filed by General Electric Company, a New York corporation with a principal place of business at One River Road, Schenectady, New York. A list of the names of the directors and officers appears in General Electric's latest Annual Report, a copy of which is forwarded to the Commission each year for inclusion in Docket 70-754. All correspondence may be addressed to 3135 Easton Turnpike, Fairfield, Connecticut 06431.

General Electric is not controlled by any alien, foreign corporation, or foreign government; it is controlled by its Board of Directors and the Officers elected by the Board.

General Electric is a publicly held corporation whose stock is traded on the principal security exchanges. The applicant has no knowledge or any information indicating any appreciable ownership of General Electric stock by an alien, foreign corporation, or foreign government. No person owns of record or is known by General Electric to own beneficially one percent or more of the outstanding shares of its capital stock.

Financial information required by 10CFR70.22 appears in the aforementioned Annual Report.

1.2 LOCATION AND GENERAL DESCRIPTION OF VALLECITOS NUCLEAR CENTER

The General Electric Vallecitos Nuclear Center (VNC) is operated in conjunction with the operation of a 100-kW nuclear research reactor, for the development and examination of reactor fuels, and to provide nuclear products and services. It is located near the center of the Pleasanton quadrangle of Alameda County, California. The laboratory is east of San Francisco Bay, approximately 35 air miles east-southeast of San Francisco and 20 air miles

north of San Jose. The site is indicated on the Bay Area map, Figure 1.1. The nearest sizeable towns are Pleasanton with a population of approximately 51,000 located 4.1 air miles to the north-northwest, and Livermore with a population of approximately 57,000 located 6.2 air miles to the northeast. A United States Veterans Administration Hospital with a population of approximately 500 is located about four miles to the east.

The site is on the north side of Vallecitos Road (State Route 84), which is a two-lane paved highway. The Union Pacific railroad lies about two miles west of the site. The laboratory site consists of approximately 1,600 acres, about one-third of which is relatively flat. Approximately 1,400 acres of the site are leased for raising feed crops and cattle grazing.

There is very little industrial activity within a 10-mile radius of the plant. A small amount of light industry is located at Pleasanton, Fremont and Livermore, but these towns are not industrial centers. The city of San Jose to the south, 20 miles distant, and Oakland and San Francisco, 30 and 35 miles, respectively, to the northwest are the major industrial centers in the vicinity. In the southeast quadrant, there are no industries and very sparse population for 20 miles and beyond.

The property on which the laboratory buildings are located is drained by ditches leading to Vallecitos Creek. This creek discharges to Arroyo de la Laguna near the north end of Sunol Valley, two to three miles southwest of the property.

Water is supplied from the Hetch-Hetchy aqueduct by means of a 14-inch line capable of supplying over 3,000,000 gallons per day. A 500,000-gallon storage tank is provided on the laboratory site. One hundred thousand gallons are reserved for fire protection.

Electrical power is supplied by Pacific Gas & Electric to the main laboratory substation from whence it is distributed to each building on the site.

A sewage treatment system is provided in the southwest corner of the site. Effluent from this system is disposed to site land.

1.3 GENERAL PLANS AND USES OF SPECIAL NUCLEAR MATERIALS

This application requests authorization under Title 10, Code of Federal Regulations, Part 70, to receive and possess the special nuclear material designated in Section 1.4 herein; to receive and possess the special nuclear material and associated by-product material produced by the irradiation thereof; and to use said special nuclear materials in research and development activities as defined in Section 70.4(j), in chemical and physical analysis, and examination and investigation of nuclear fuels, associated materials and devices at the Vallecitos Nuclear Center.

1.4 SPECIAL NUCLEAR MATERIAL POSSESSION LIMITS

1.4.1 Vallecitos Nuclear Center

The special nuclear materials used in connection with activities authorized by License SNM-960 at the Vallecitos Nuclear Center will not at any time exceed those limits listed in Section 2.1 of Appendix A to License SNM-960.

1.4.2 Form and Enrichment Specifications

The majority of the Vallecitos Nuclear Center activities are conducted in facilities and under procedural controls which accommodate any chemical or physical form and any U-235 isotopic content. If specific limitations are placed on these parameters in connection with an individual activity in order to assure the radiation or nuclear safety of that work, the limit is described in the appropriate section of this application entitled, "General Plans and Uses of Materials".

1.5 PRINCIPAL VALLECITOS FACILITIES

Descriptions of the principal buildings and laboratories in which special nuclear materials are used at the Vallecitos Nuclear Center site are set forth in this section with the primary objective of general orientation. The locations of these facilities are shown in Figure 1.2. The specific activities conducted in each of these facilities and their safeguards equipment and procedures are discussed in later sections.

1.5.1 Radioactive Materials Laboratory

The Radioactive Materials Laboratory (RML) is located in Building 102. This laboratory is a shielded facility equipped with remote manipulators to conduct experiments and analyses with irradiated reactor fuels and other radioactive materials. The facility also includes a storage pool and dry pit storage.

1.5.2 Radiochemistry Laboratory

Adjacent to RML, on the main floor of Building 102 and providing analytical support to it, is a radiochemistry laboratory equipped with standard chemical and radiochemical apparatus. This laboratory primarily is used to analyze samples of materials prepared in the RML.

1.5.3 Metallurgy, Chemistry, and Ceramics Building

A second major laboratory building in the 100 Area is the Metallurgy, Chemistry, and Ceramics Laboratory (Building 103). This two-story building consists of laboratories, variously equipped with laboratory apparatus designed to handle moderate quantities of radioactive materials, and offices. The functions served by this facility are research, development, and analytical chemistry services.

1.5.4 Building 105

Just north of Building 102 is Building 105. The principal facilities located in this building are an experimental reactor (the Nuclear Test Reactor) and laboratories. The laboratories in Building 105 use only minute quantities of special nuclear materials.

1.5.5 Engineering Shop

Building 106 contains various maintenance shops (e.g., machine, carpentry, electric, and instrument calibration facilities).

1.5.6 Solid Radioactive Waste Storage Facility

Solid radioactive wastes generated at the various laboratory and facility locations are stored in the waste storage facility located approximately midway between the deactivated Vallecitos Boiling Water Reactor (VBWR) and General Electric Test Reactor (GETR) areas. This storage area includes shielded horizontal tubes for storing 5-inch and 7-1/2-inch-diameter waste liners.

1.5.7 Waste Evaporator Plant

The Waste Evaporator Plant is located adjacent to the deactivated VBWR site. This plant is used to concentrate and solidify liquid radioactive wastes generated at the Vallecitos Nuclear Center or other licensed facilities prior to transfer to authorized waste disposal firms or waste burial sites. Such wastes may contain minute quantities of special nuclear material.

1.5.8 Reactors and Auxiliary Facilities

The ESADA-Vallecitos Experimental Superheat Reactor (EVESR), the VBWR, and the GETR are deactivated.

1.5.9 400 Area

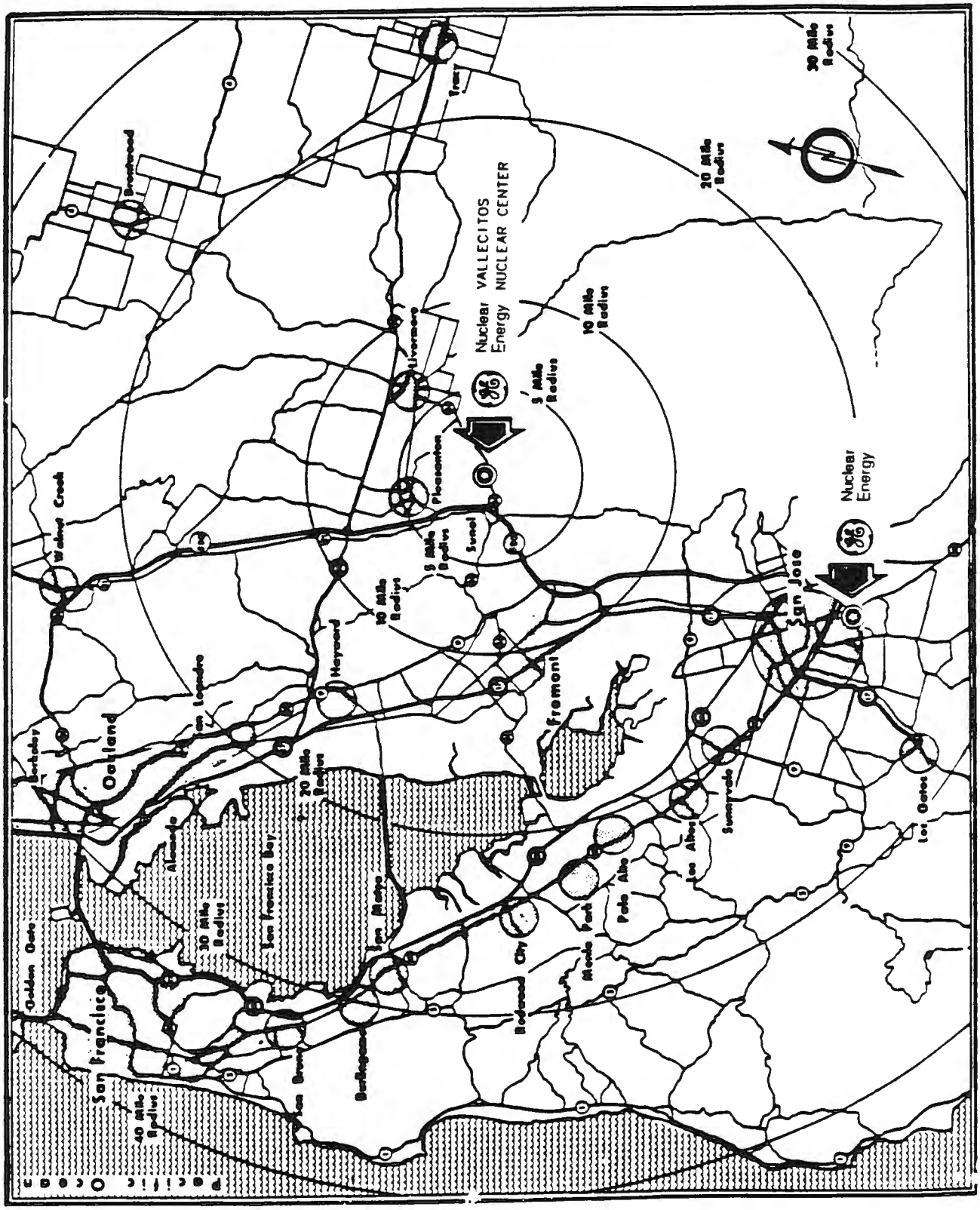
The 400 Area consists of two buildings, 400 and 401. Building 401 is devoted chiefly to offices, while Building 400 currently is used for storage of nonradioactive materials.

1.5.10 Building 104

This building is used for warehousing.

1.5.11 Building 300

This building contains a radiochemistry training laboratory and a reactor fuel cladding examination facility.



BAY AREA MAP

Figure 1.1

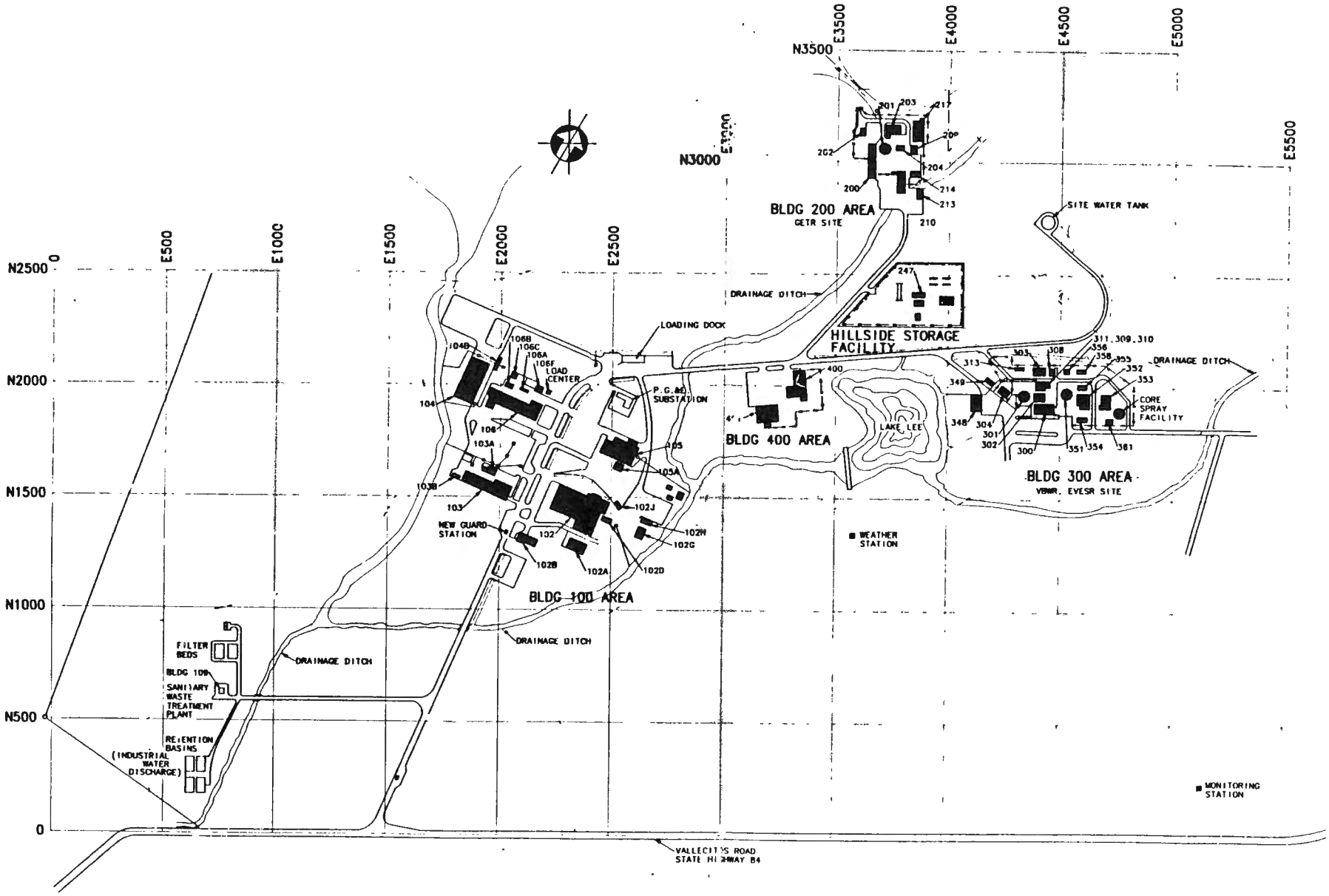


FIGURE 1.2. VALLECITOS NUCLEAR CENTER BUILDING LOCATIONS

2.0 ORGANIZATION AND ADMINISTRATION

All Vallecitos Nuclear Center (VNC) activities are conducted under the management of General Electric Nuclear Energy (GE-NE).

Several components of GE-NE perform activities at VNC. However, the Manager, Irradiation Processing Operation (IPO), who is responsible for regulatory compliance at VNC, reports to the Vice President & General Manager, GE-NE.

2.1 ORGANIZATION

An organization chart for VNC is included at the end of this section as Figure 2.1. An organization chart for the Nuclear Safety function is included as Figure 2.2.

2.2 DELEGATION OF RESPONSIBILITY

The management of the components at VNC has established a policy of protection of employees, the public, and the environs from potential industrial, radiation, and nuclear hazards that could occur through activities conducted in each component's facilities. He has delegated the responsibility for implementing this basic policy through line managers to the manager and supervisor of each activity in which radioactive materials are handled, used or stored. Additionally, IPO has experienced and competent staff personnel to provide expert advice and guidance to all components in matters of radiation and criticality safety. The Manager, IPO, has been delegated responsibility to act as the chief compliance safety officer for all VNC operations. Industrial safety is provided for VNC by the Industrial Safety & Hygiene function located in San Jose.

With respect to the radiation and criticality safety programs, the basic line and staff relationships are illustrated in Figures 2.3 and 2.4, respectively.

2.3 SITE MANAGEMENT REVIEW FUNCTION

The Vallecitos Technological Safety Council (VTSC) is composed of senior site management and/or technical personnel and has been established as a site-wide evaluation function with responsibility for management evaluations and reviews of VNC activities, including the Nuclear Safety function. Assigned functions include review of VNC activities to assure that major risks are identified, considered and resolved and that the application of policies and procedures is consistent between the site components.

2.4 RESPONSIBILITIES OF THE NUCLEAR SAFETY COMPONENT

a. Radiation Safety. Establish and administer a radiation safety program to ensure the protection of employees and the general public. Provide monitoring support, a dosimetry program, an environmental monitoring program, and employee training in the methods for minimizing exposure through proper use of survey instruments and protective clothing and devices. Maintain all radiation exposure records required by regulatory agencies.

b. Nuclear Safety Compliance and Review. Provide review of reportable incidents, new facilities or major changes to facilities and operations control standards, and professional advice and counsel on nuclear and radiation safety policy.

c. Licensing. Represent VNC components in regulatory activities concerning radiation protection and licensing. Procure, administer and interpret NRC licenses and regulations, state and local government licenses and regulations, and provide advice and counsel to VNC components and their customers on regulatory matters.

d. Radiological Engineering. Assist in improved operation of VNC facilities and the overall radiation protection program through the analyses of existing systems, equipment and operation thereof and the recommendation of improved systems, equipment or methods. Generally, these studies will be safety oriented.

e. Training. Establish and administer a site-wide training program at VNC to ensure adequate knowledge of radiation control procedures.

f. Criticality Safety. Perform criticality control analyses to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment. Audit the criticality control environment and conduct educational programs in criticality matters.

g. Emergency Preparedness and Response. Maintain the site-wide emergency procedures; coordinate and audit training.

2.5 OTHER COMPONENTS CONTRIBUTING TO OVERALL SAFETY

In addition to the principal radiation and nuclear safety functions of the staff components previously described, the following components and their functions make significant contributions to the overall safety program.

a. Nuclear Materials Safeguards. Establish and administer the basic system of special nuclear material control. Assure that SNM control policies and practices of all components are coordinated with the requirements of nuclear safety, licensing and shipping groups. Perform reviews of compliance with internal control procedures and regulatory requirements.

b. Physical Security. Develop and operate security programs to safeguard special nuclear materials and corporate property.

c. Transportation and Materials Distribution. Assure all radioactive materials shipments are in compliance with applicable Federal and State regulations.

d. Industrial Safety and Hygiene Function. Develop programs to protect the employees from industrial hazards, including operation of medical and safety education programs.

e. Environmental Safety Function. Develop programs to protect the employees and the general public from exposure to hazardous materials and assure proper disposal of hazardous wastes.

2.6 TECHNICAL PERSONNEL CAPABILITIES

The primary responsibility for operational radiation safety for the operations conducted in the various Vallecitos facilities involving special nuclear material rests with the supervisor or manager of each facility. Equally important are the knowledge and experience of personnel in the Nuclear Safety function. Since the issuance of License SNM-960 in 1966, the Commission's Region V office has inspected VNC to assure that adequate levels of technical expertise are maintained in all positions. Resumés for key personnel are included as Addendum A to this section.

2.7 IMPLEMENTATION OF CRITICALITY CONTROL PROGRAM

The program for protection against accidental conditions of criticality is implemented by means of functional responsibility assignments. Managers whose operations require the use of quantities of special nuclear materials approaching a theoretical minimum critical mass, or greater, are responsible for integrating and measuring the efforts of line and staff participants in this program. The principal participants and their responsibilities are outlined below.

2.7.1 Responsibility of the Manager

It is the responsibility of any manager of an activity involving special nuclear materials to establish and maintain a records system to accurately account for the quantities of such materials used and transferred, thereby providing a basis for controlling against excessive accumulation or loss.

It is the responsibility of any manager of an activity involving more than 500 grams of U-235, through the designated supervisor or engineer with specific responsibility for a given detailed activity area, to:

a. Know and understand the specific limits of all nuclear parameters applicable to his area of responsibility.

b. Determine the need for a criticality analysis. This need arises whenever there is: (1) a change in the parameters of special nuclear material such as enrichment, chemical and physical form, density, etc.; (2) a change in the parameters such as quantity or dimensions of unit accumulations; (3) a change in the spacing or dimensions of processing, handling or storage equipment; or (4) a change in environment such as the addition of fire sprinkler systems of possible consequence to criticality safety.

c. Prepare in writing a request for criticality analysis stating the activity or area involved, including a description of the special nuclear material, process description, and equipment description with appropriate charts, diagrams, and drawings. All special nuclear material shall be included whether license or license-exempt and whether as usable forms or as scrap and waste.

d. Upon receipt of results of a criticality analysis in writing from the Criticality Safety component, take all necessary steps to implement the appropriate controls such as:

(1) Posting of limitations at strategic locations at the criticality areas where they apply.

(2) Issue appropriate procedures for establishing and maintaining equipment and materials in the arrangement and quantities specified by the criticality analysis.

(3) Assure boundaries identifying each separate criticality area and position are marked conspicuously.

(4) Assure all operating personnel are advised of any special limitations or controls other than as outlined above and are trained to follow the necessary precautionary procedures.

(5) Establish procedures for handling off-standard processing conditions.

2.7.2 Responsibility of Criticality Safety Component

It is the responsibility of the Criticality Safety component to:

a. Review requests for criticality analysis to verify completeness of submitted information and, as appropriate, personally inspect the equipment or location involved to obtain a first-hand determination of the possibility of interactions with other special nuclear material in the environment.

b. Make criticality control calculations to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment in accordance with authorized computational methods.

c. Furnish a written criticality analysis to the requesting operation for each request received. The person in charge of the Criticality Safety component shall identify the analyses which involve systems in which the direct achievement of criticality may be the consequence of error in analysis and shall assure that such analyses are reviewed and verified.

d. Perform an inspection at least annually of the degree to which actual operations conform to physical situations on which the criticality calculations were based.

e. Establish a criticality control educational program to emphasize the need for following precise instructions.

2.7.3 Responsibility of Radiation Safety Component

It is the responsibility of the Radiation Safety component through its staff of qualified and experienced radiation monitors to:

a. In their normal rounds of duty be continuously alert to possible criticality safety practices inconsistent with the appropriate criticality analyses and/or procedures.

b. Be familiar with applicable parameters for criticality control and use these parameters as standards for inspection.

c. Verify that physical conditions and activities comply with the conditions specified in the criticality analysis.

d. Initiate immediate action to discontinue any SNM movement and shut down any operation which in its judgment does not fully comply with the criticality analysis.

2.8 IMPLEMENTATION OF RADIATION SAFETY PROGRAM

The program for protection against radiation hazards is implemented by means of functional responsibility assignments. Managers are responsible for integrating and measuring the efforts of line and staff participants in the program.

2.8.1 Responsibility of the Manager

It is the responsibility of the manager of an activity or area involving radioactive materials to:

a. Take all necessary steps to plan and organize the work of his component in accordance with approved radiation safety standards and operational procedures.

b. Identify needs for operational procedure revisions when there is a planned change in conditions such as types or quantities of radioactive materials or equipment modifications.

c. Integrate the results of reviews, inspections, engineering assessments and investigations made by the Radiation Safety component to correct or improve operational procedures, controls and performance.

2.8.2 Responsibility of Radiation Safety Component

It is the responsibility of the Radiation Safety component to:

a. Establish and administer through a system of instructions a radiation control program and maintain all radiation exposure records required by regulatory agencies.

b. Review and approve new operations in the design stage, identifying potential hazards and providing recommendations for their avoidance through education, training methods, and equipment application.

c. Inspect each radiation area to assure compliance with radiation safety procedures and limits.

d. Provide management with reports of inspection results showing trends in activities involving radioactive materials.

2.9 TRAINING PROGRAMS

2.9.1 Radiation Safety

Every new employee at VNC who routinely handles radioactive materials normally receives a radiation safety orientation, "New Employees Radiological Safety Orientation", within thirty days of reporting to the site. If an employee's area manager determines that the employee will be exposed regularly or frequently to radiation and/or radioactive materials, the employee is instructed in radiation protection such that he is able to protect himself and is made aware of the degree of hazard involved. A training program entitled, "Radiological Safety At Vallecitos Nuclear Center", is completed by each such employee normally within one year of his starting date at VNC. This course is scheduled and conducted by the Radiation Safety component.

The training course includes the following elements:

a. basic principles of radiation safety,
b. Company policies and operating procedures,
c. radioactive materials handling methods and shielding requirements,
d. emergency procedures,
e. requirements of NRC regulations, and
f. NRC license requirements.

Follow-up training commensurate with the work environment and the employee's work performance is determined by employee supervision. The employee also receives on-going training in the form of on-the-job demonstrations, periodic safety meetings, etc.

Employees whose work assignments may include the need for the use of respiratory protection equipment also receive the "Respiratory Protection Training Course" (RPTC). The course provides both instruction and hands-on experience in the proper use and fitting of the respiratory protective equipment. NOTE: Respiratory protection equipment is worn only as a precaution. VNC does not assign protection factors or take credit for

respiratory protection for routine work. Should respiratory protection be mandatory for a given job, the Change Authorization will assure compliance with 10CFR20.

2.9.2 Criticality

Every new employee who will work regularly with special nuclear material in areas in which quantities of special nuclear material capable of forming a critical mass (considering such criteria as existing moderation, reflection, enrichment, etc.) are present is instructed in the principles of criticality safeguards and is made aware of the degree of hazard involved. This instruction is completed within one year after the employee's starting date at that facility.

Area supervision is responsible for informing all personnel at work or otherwise present in their area of specific procedures for criticality control and for appropriate administrative action to assure compliance with these procedures.

2.9.3 Other Programs

Radiation monitors receive the "Radiation Monitoring Technicians Certification Course". This is an intensive training program designed to qualify participants as VNC-certified Radiation Monitoring Technicians.

Other courses are available from the Radiation Safety component on an as-needed basis.

2.9.4 Records

Records of individuals receiving training courses are kept in a central computerized record file.

2.10 INDEPENDENT REVIEW AND CHANGE AUTHORIZATION PROCEDURE

2.10.1 Definitions

a. Change Authorization. A Change Authorization is the VNC mechanism for independently reviewing and documenting changes. Any addition, alteration, deletion, modification or substitution which results in a different position, course or direction not previously analyzed, adds a new capability, performs a different function, modifies performance characteristics, or introduces a hazard not previously analyzed will constitute a change. Descriptions of existing facilities and standard methods of operation normally are contained in documents supporting Federal and State licenses or previously approved operating instructions and may be used to assist in determining if the proposed activity constitutes a change.

b. Facility. A facility consists of any permanent structure and that apparatus which, wholly or in part, is integrated with the structure, i.e., shielding, piping, ventilation, reactor core, etc.

c. Equipment. Equipment is defined as that apparatus which operates independently or in conjunction with the facilities to perform the various VNC activities.

d. Experiment or Test. An experiment or test consists of any device or combination of devices (not to include the facility or equipment wherein the experiment or test is housed) designed and operated for the purpose of obtaining a predetermined objective, or any nonroutine operation or manipulation of objects, systems or process streams for the purpose of gaining information.

e. Procedure. Procedures are the written guidelines established by facility management to describe and define the methods and instructions for operations.

2.10.2 Purpose

Any addition, alteration, deletion or substitution which adds a new capability, performs a different function, modifies performance characteristics, or introduces a hazard not previously analyzed requires an independent review by use of the Change Authorization procedure. A Change Authorization is prepared whenever the work involves changes to:

a. Facilities, equipment, or processes so that safety or regulatory compliance considerations differ from those previously analyzed.

b. Radioactive material limits.

c. Hazardous or potentially hazardous industrial materials where such change is significant in terms of quantities or use.

The independent review of items is conducted so that the hazards (both direct and indirect) of the proposal are recognized and appropriate safeguards are provided to eliminate or reduce the probability and severity of potential accidents. While procurement, fabrication, selective installation or testing, etc., may proceed prior to the final CA review and approval, actual implementation of the proposed change should not proceed until this review and approval are received.

The Change Authorization is processed in accordance with a written procedure and reviewed by the Nuclear Safety component and by the Industrial Safety component as appropriate.

2.10.3 Procedures

a. Preparation of Change Authorization. Knowledgeable individuals prepare Change Authorizations which describe in sufficient detail the nature of the changes and the effect on safety, including applicable drawings and specifications, acceptance test procedures (ATP's), quality control requirements (if applicable), means for assuring personnel indoctrination for initiating method changes and specifying the responsible supervision.

Certain minor changes and changes previously evaluated and documented need only internal review and documentation. The Nuclear Safety component will assist the initiating component in making this determination.

b. Review of Change Authorization. All Change Authorizations (CA's) are reviewed by the initiating component, by the Nuclear Safety component and, as appropriate, by Industrial Safety and are approved by the appropriate area or facility manager or his designated alternate. Recommendations for the change needed for safety may be added by the reviewers following discussion with facility personnel as appropriate. The Nuclear Safety component has the responsibility for determining whether a proposed change constitutes an unreviewed safety question or other license or technical specification violation. No CA should be implemented until review is completed and the CA is signed by the manager of the Nuclear Safety component or his designated alternate.

c. Distribution. A Change Authorization file will be maintained by each facility. A copy of the Change Authorization form must be filed with the Nuclear Safety component, with other copies distributed as appropriate.

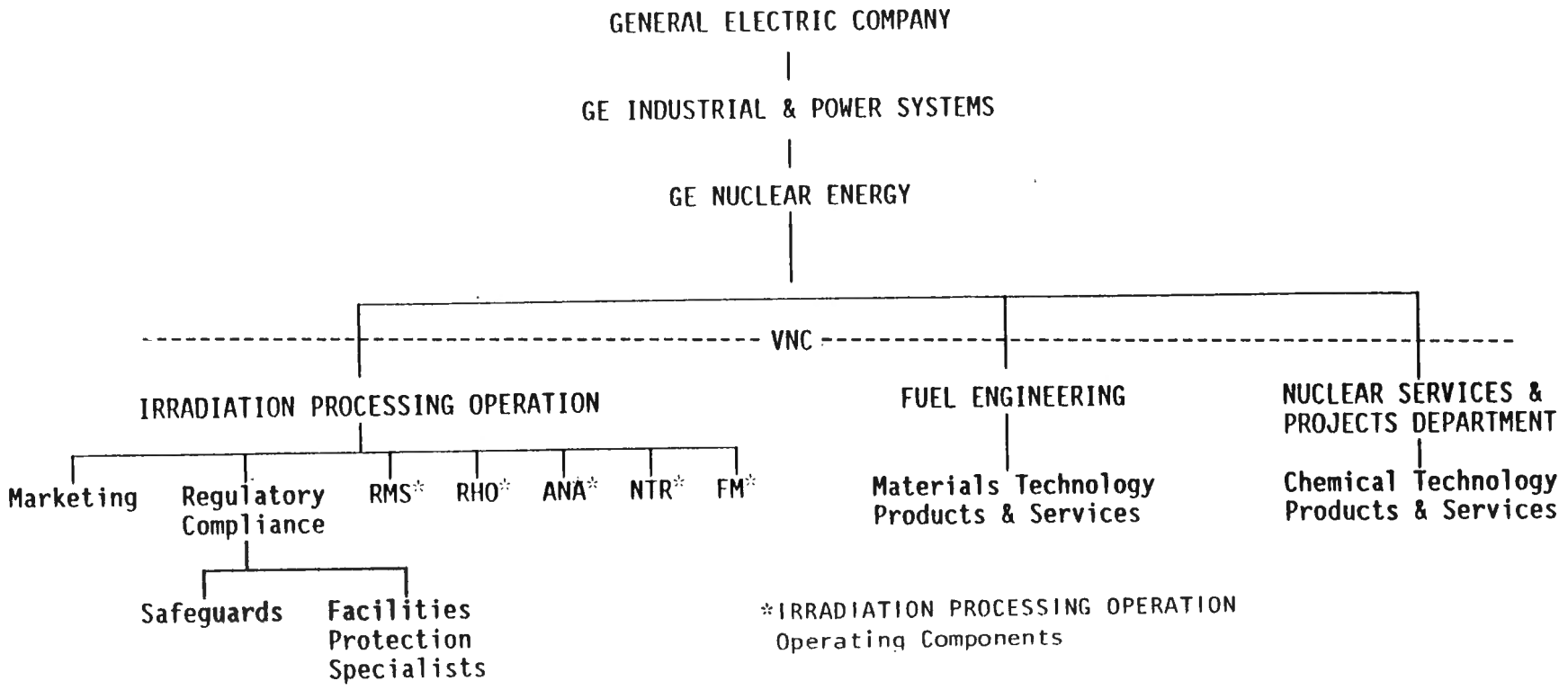


FIGURE 2.1. VALLECITOS NUCLEAR CENTER ADMINISTRATIVE ORGANIZATION CHART
(Components shown below the dotted line are located at VNC.)

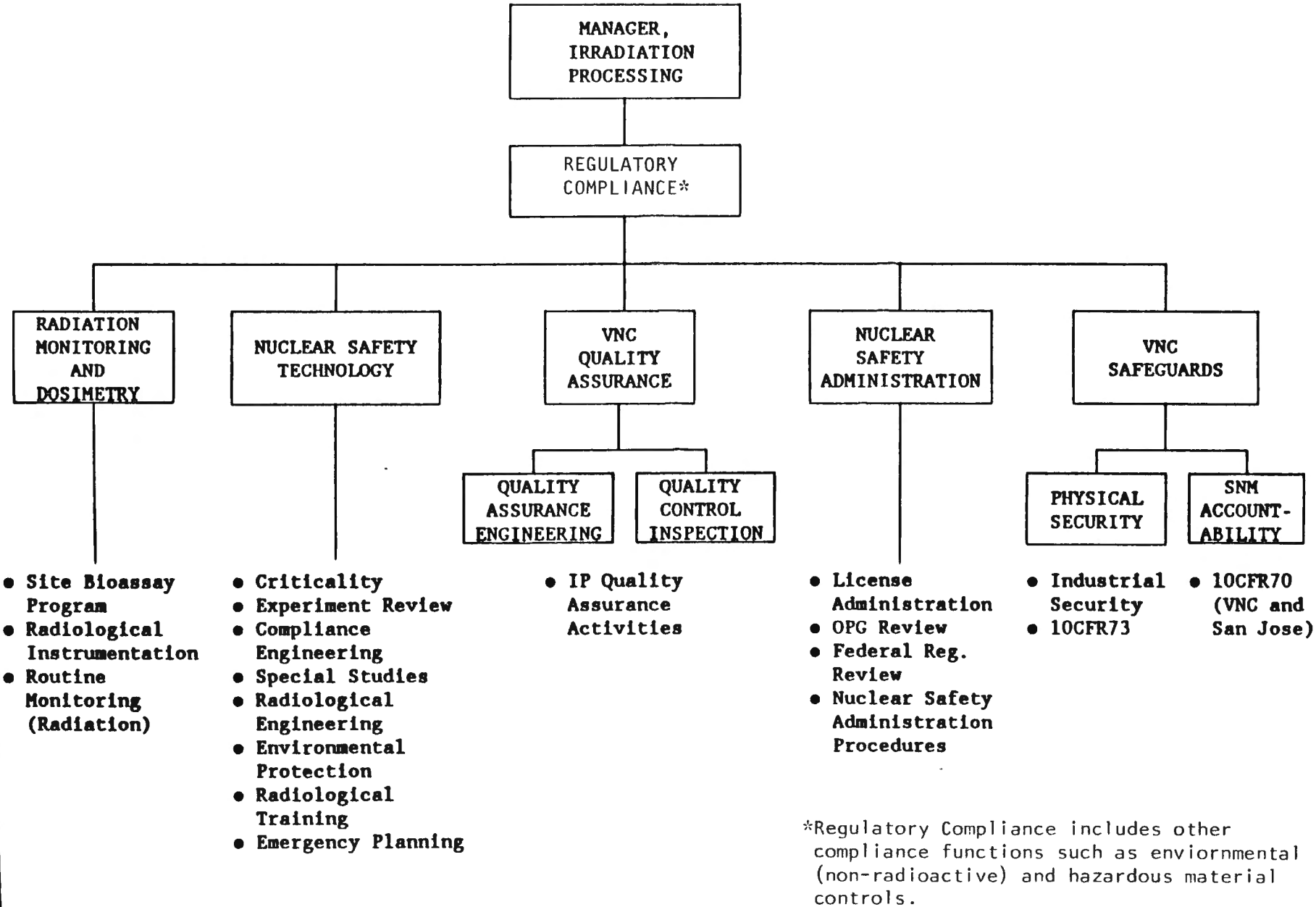


FIGURE 2.2. NUCLEAR SAFETY ORGANIZATION

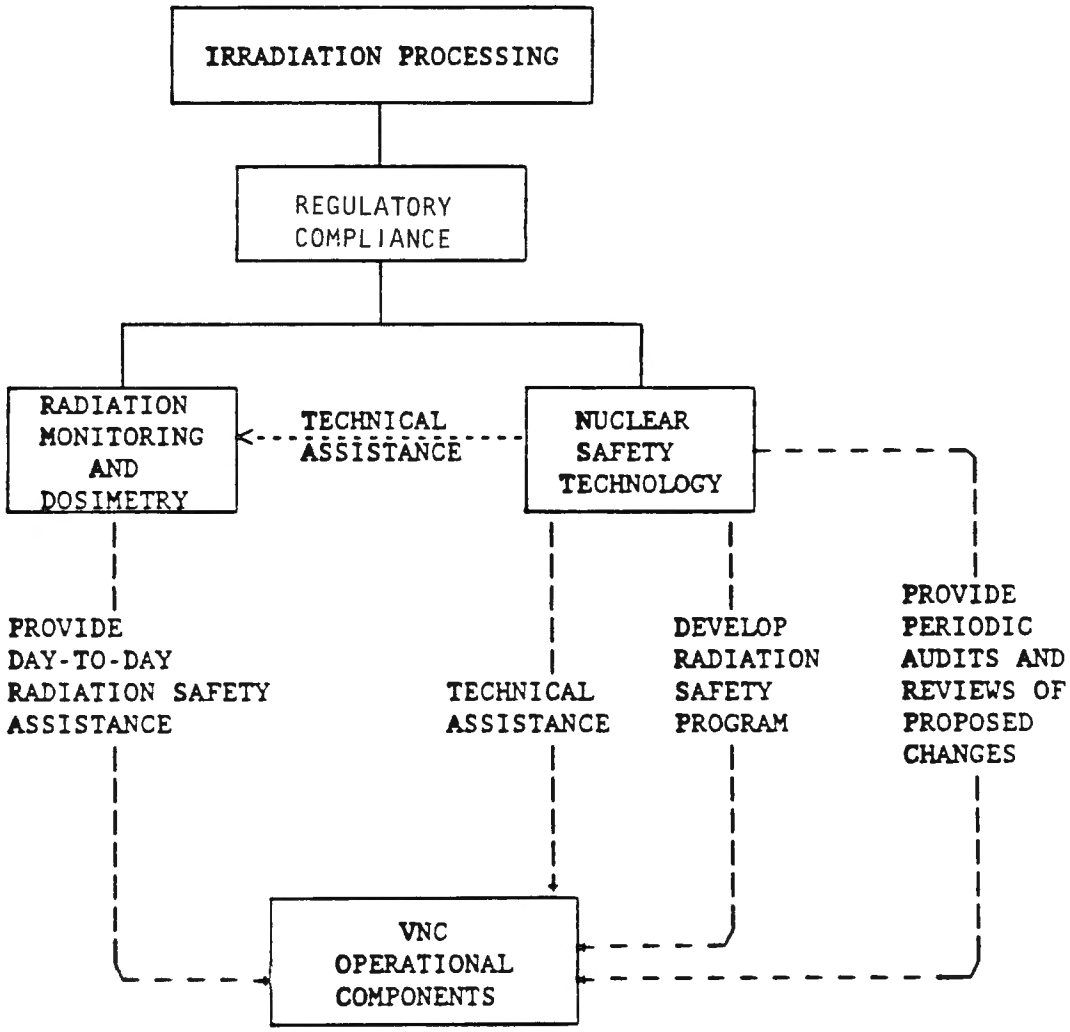


FIGURE 2.3. RADIATION SAFETY STRUCTURE

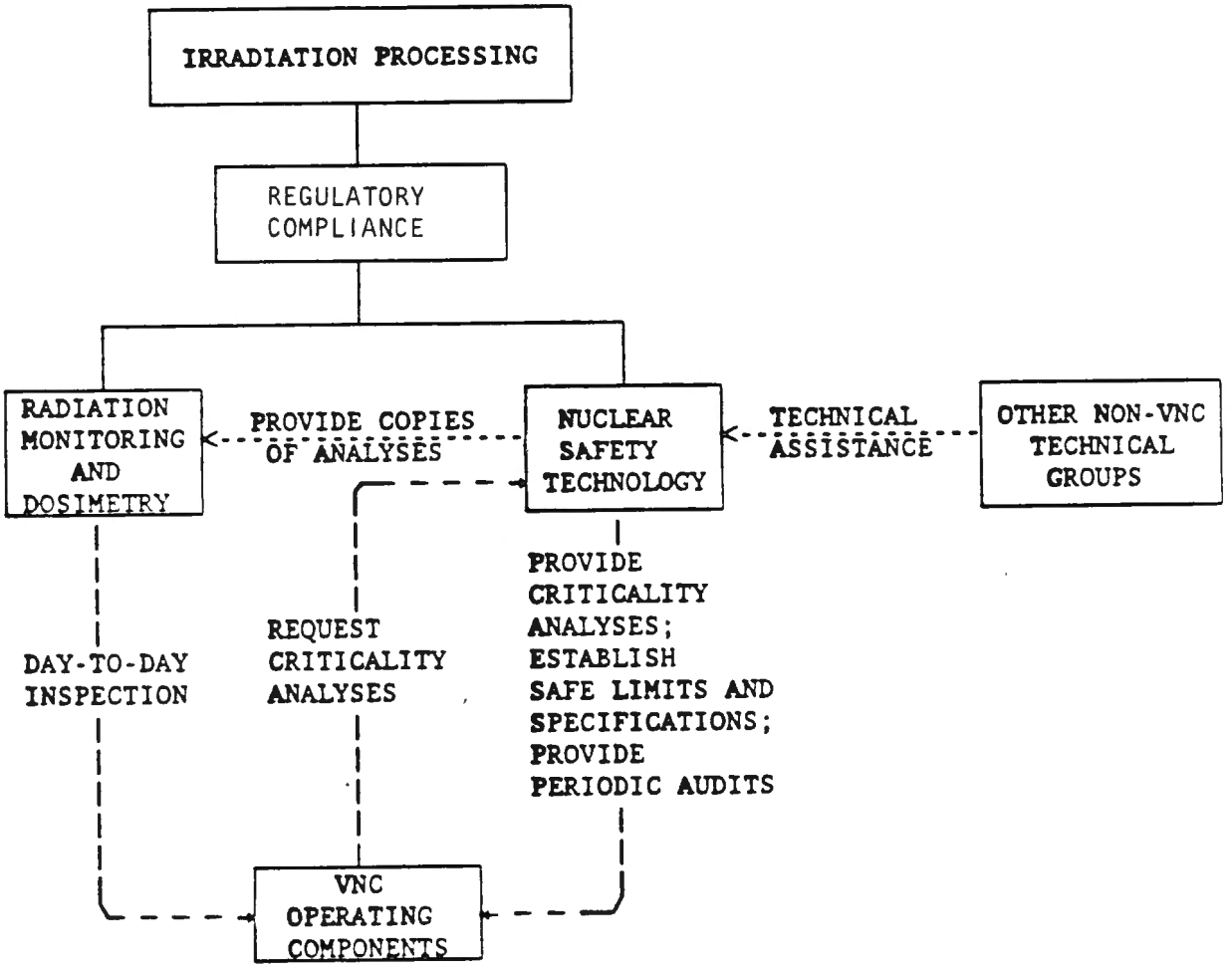


FIGURE 2.4. CRITICALITY SAFETY STRUCTURE

ADDENDUM A TO SECTION 2.0

RESUMES OF KEY MANAGERIAL AND SAFETY PERSONNEL

A.1 G. L. Stimmell; Manager, Irradiation Processing Operation;
B.S., Mechanical Engineering, California State University, Fresno;
M.B.A., University of Santa Clara.

Mr. Stimmell joined the General Electric Company, Vallecitos Nuclear Center (VNC) in 1964 on the Vallecitos Training Program. He had assignments in Nuclear Safety, Test Engineering, the Radioactive Materials Laboratory, and Design Engineering. In 1971, he became a Project Engineer responsible for irradiation programs conducted in the GE Test Reactor (GETR). In 1974, he became Manager, Design and Test Engineering, where he managed a multi-disciplined engineering unit responsible for the design and analysis of experiments for irradiation in the GETR, and design of equipment for several VNC facilities, including the GETR and RML. During this period, the engineering unit also conducted analyses in support of license renewal for the GETR and the Nuclear Test Reactor (NTR).

From 1984 to 1986, Mr. Stimmell became a Senior Program Manager, Neutron Sources. In this position he was responsible for establishing the Californium-252 neutron source production facilities and the development of neutron source markets.

From 1986 to 1988, Mr. Stimmell was the Manager of Project Engineering for the Nuclear Waste Repository Project where he was responsible for the development and design of waste packages and repository equipment for a high-level waste repository in salt.

From 1988 to 1992, he became Manager of Systems Engineering for the Advanced Liquid Metal Reactor project. In this capacity he was responsible for the coordination and specification of overall plant design requirements and the integration of the systems level plant design. He was also responsible for

the refueling system concept development, plant operations and maintenance assessments and staffing estimates, and fuel cycle facility concept development. He was also responsible for coordination and support of international collaboration by arranging economics and safety technical exchanges with the European Fast Reactor Associates.

In 1992, Mr. Stimmell returned to VNC as the Manager of Irradiation Processing. He is responsible for the operation of the radioisotope production and post-irradiation examination facilities, the operation of the Nuclear Test Reactor and its support functions, and three shut down reactors maintained under possess-only licenses. Mr. Stimmell has been designated by the General Managers of the General Electric divisions represented at VNC as the manager responsible for the nuclear safety, industrial safety, and environmental compliance for the entire site.

Mr. Stimmell is also a Licensed Professional (Mechanical) Engineer in the State of California.

A.2 J. H. Cherb; Manager, Regulatory Compliance; B.S., Mechanical Engineering, Sacramento State College, 1967; Licensed Professional Engineer, State of California (No. 15034).

Mr. Cherb joined the General Electric Company (GE), Vallecitos Nuclear Center (VNC) in 1965. After completing a training assignment in the radiation safety organization he worked for most of his career in a variety of engineering and managerial positions related to activities at VNC.

Engineering positions held by Mr. Cherb include Plant Engineer, Test Engineer, Process Engineer, Project Engineer, and Senior Quality Assurance Engineer. Responsibilities included maintaining and modifying equipment and systems at the Vallecitos Experimental Superheat Reactor and providing technical direction to operating units in the conduct of experimental programs at the General Electric Test Reactor (GETR). He was a major contributor to the establishment of the quality system at VNC which includes a Nuclear Regulatory Commission-approved Quality Assurance Program for Shipping Packages for Radioactive Material in accordance with 10CFR71. One year he worked as a Field Engineer (Nuclear Systems) where he served as a resident QC representative during power reactor pressure vessel fabrications and had similar assignments at several power reactor construction sites.

Mr. Cherb has 11 years managerial experience in quality assurance at VNC, seven of those years in the Nuclear Safety organization. In 1986 he was appointed Manager, Nuclear Safety (now Regulatory Compliance). His primary responsibilities include developing the radiological safety criteria at VNC and monitoring the radiological conditions to assure safety of workers and the public.

A.3 G. E. Cunningham, Senior Licensing Engineer; B.S., Physics, Louisiana State University, 1958.

In 1958, Mr. Cunningham joined the General Electric Company in Richland, Washington, under the Technical Graduate Rotational Training Program. From 1960 to 1966 while with General Electric in Richland, he was an engineer in the health physics group becoming responsible for all technical aspects of health physics for a chemical reprocessing facility. His responsibilities included exposure records, development of administrative and technical procedures, developing bioassay schedules, liaison work with other site components and with the Atomic Energy Commission, dosimetry studies, and environmental surveillance.

From 1966 to 1967, Mr. Cunningham was employed by Isochem, Incorporated, Richland, Washington. His work chiefly involved an investigation into the merits of Cobalt-60 and Cesium-137 for commercial irradiation application.

In 1967, Mr. Cunningham joined General Electric at VNC as a Criticality Specialist, providing criticality work and consultation for other General Electric sites and backup capabilities for site health physics.

In 1971, Mr. Cunningham became responsible for licensing and liaison activities between VNC and the various regulatory agencies.

From March, 1975, to October, 1975, Mr. Cunningham assumed the additional responsibility for the establishment of the VNC Physical Security and SNM Accountability Program. Since October, 1975, he has reassumed his duties as the site licensing administrator.

Mr. Cunningham is a past member of the Board of Examiners of the American Board of Health Physics.

A.4 B. M. Murray, Radiological Engineer; B.S., Mechanical Engineering, Chico State College, 1963.

Mr. Murray has had 30 years of experience in nuclear safety and operations activities at the Vallecitos Nuclear Center. He has expert capabilities in radiological engineering and risk evaluations, and his most recent assignment includes radiological engineering responsibilities for both the VNC and San Jose sites and responsibility for criticality safety.

From 1963 to 1965, Mr. Murray's assignments on a training program included Nuclear Safety operations and technical analysis, BWR operations, and test reactor operations. He received a General Electric Test Reactor Operator's License in 1964.

From 1965 to 1974, Mr. Murray held the positions of Facility Engineer, Product Manager, and Isotope Production Supervisor in the Remote Handling Operation. His assignments included equipment design, project managing, sealed source fabrication, supervision of a processed isotope production crew, and facility planning and modification.

Mr. Murray's assignments as Nuclear Safety Engineer during the period 1974 to the present time include: technical risk evaluations, computer-aided shielding calculations, internal and external dosimetry measurement and evaluation, radioactive effluent system evaluations and release limits, hypothetical accident evaluations, facility operating license application input preparation, stack monitoring system evaluations and recommendations, operating standards and procedures preparation and review, health physics instruments evaluation and specification, and review of proposed site activities. He became a California Registered Professional Mechanical Engineer in 1974. He attended and completed course work for the Health Physics Society School on Internal Radiation Dosimetry, Principles and Practices in June, 1983, and the University of New Mexico-sponsored Nuclear Criticality Short Course in May, 1985.

A.5 A. F. Mindt; Specialist, Radiation Monitoring; B.S., Education, University of Minnesota, 1959.

Mr. Mindt joined the General Electric Company, Vallecitos Nuclear Center (VNC) in 1967 as a Chemical Laboratory Technician performing tests and reporting results for radioisotopes analysis, burnup fuel analysis, plutonium fuel analysis, and environmental studies.

In 1970, Mr. Mindt became Supervisor, Isotope Production. In this capacity he was primary licensee for pharmaceutical manufacturing, directing the operations of four remote handling facilities. These facilities were designed for the chemical, cryogenic and physical separation and purification of radioactive nuclides. In addition to the direction of personnel, this position required preparation of procedures insuring compliance to good manufacturing practices, State and Federal license requirements, and nuclear safety regulations.

From 1973 to 1979, Mr. Mindt was Specialist, Isotope Production, responsible for quality, production schedules, cost controls and compliance with regulatory requirements for four product lines containing 30 different radioactive line items.

From 1979 to 1987, Mr. Mindt held several administrative positions including Production Programs Manager; Manager, Electronics Manufacturing; Manager, Materials Planning; and Specialist, Site Services.

In 1988, Mr. Mindt joined the Nuclear Safety group where he became certified as a Radiation Monitoring Technician. He served in that capacity, evaluating radiological conditions, specifying appropriate radiation measures, and overseeing the work environment. He conducted surveys and inspections recommending appropriate radiation techniques for a variety of activities.

He took the position of Specialist, Facilities Protection, in 1990 where he served as senior GE representative on site during the back shifts, weekends and holidays, providing nuclear safety services as well for the reactor areas, facilities functions and reservation-wide surveillance.

Having assumed the position of Specialist, Radiation Monitoring, in 1992 he now coordinates the work of the Radiation Monitoring Technicians and the Specialists, Facilities Protection. In addition, he serves as an alternate to both positions as needed for emergency or high work load periods.

3.0 RADIATION PROTECTION FACILITIES AND EQUIPMENT

Permanent laboratory buildings and other nuclear facilities make up the Vallecitos Nuclear Center complex. The buildings and facilities are designed, equipped and maintained to perform work with radioactive materials in a manner providing a high degree of safety for persons employed at the laboratory and for residents of surrounding areas. Typical major equipment and facilities available for the protection of health, life, and property are set forth in this section.

3.1 PORTABLE MONITORING INSTRUMENTS

Monitoring instruments from the following list are available in adequate supply to provide for essential monitoring and for scheduled calibration and maintenance.

PORTABLE MONITORING INSTRUMENTATION

	Instrument Type	Range
1.	GM Detector	0-500 k cpm, beta-gamma
2.	Ionization Chamber (low energy)	0-300 mrad/h, beta-gamma
3.	Ionization Chamber (CP)	1-250 k mR/h, gamma 4-1,000 k mrad/h, beta
4.	Ionization Chamber (gas multiplication)	1-1,000 k mR/h, gamma 20-20,000 k mrad/h, beta
5.	Geiger Tube (telescopic)	1-1,000 k mR/h, gamma
6.	Micro-R Meter	0-5,000 μR/h, gamma
7.	Scintillation Counter - Sodium Iodide (Tl)	0-500 k cpm, gamma
8.	Neutron Rem Meter (BF_3)	0.5-5,000 mRem/h, neutron
9.	Alpha Survey Probes (gas proportional and ZnS)	200-1,000,000 dpm; alpha
10.	Portable Air Samplers	0-8 cfm

3.2 FIXED MONITORING EQUIPMENT

Listed below are types of equipment installed for monitoring quantities or concentrations of radioactivity.

- Air samplers and monitors utilizing GM, proportional, scintillation, and semiconductor detection modes with moving and fixed filtering units which are capable of alarming at air concentrations equivalent to MPC's in less than four hours for most of the commonly encountered radioisotopes. Fixed filter units consist of 47- or 50-mm-diameter filters and constant flow control regulators. Stack sampling and monitoring units include isokinetic probes with GM, proportional, scintillation, semiconductor and/or flow-through ion chambers and appropriate filter media.

- Fixed gamma monitors with ranges from 0.1 mR/h to 10^6 R/h are located in areas with potentially hazardous gamma fields.

- Hand-and-shoe counters and/or hand-held probes are provided at principal exit points for beta-gamma and alpha as required.

- Environmental surveillance is provided by a number of TLD dosimeters located on the VNC site and at its perimeter. Four permanent environmental air sample stations also are located on site.

3.3 PROTECTIVE CLOTHING

Protective clothing is provided to assure the necessary protection of personnel. The amount and type of protective clothing required for a specific activity or area are determined on the basis of contamination potential. Available protective clothing includes caps, laboratory coats, coveralls, boots, overshoes, shoe covers, gloves, and respiratory protection, either filtered or independent air-supplied types. Other special protective equipment is available for use from time to time.

3.4 FILM BADGES AND POCKET DOSIMETERS

Film badges or TLD's are worn where there is potential for radiation exposure. Albedo neutron dosimeters are used where appropriate. Pocket ionization chamber dosimeters may be used. If pocket dosimeter results indicate an off-scale or unexpectedly high reading, the badge is processed; and if these results are confirmed, the circumstances are investigated and the individual is removed from radiation work if appropriate. TLD extremity dosimeters are worn when radiation exposure to the hands is expected to exceed total body exposure to a significant degree. Film and TLD dosimeters are serviced by a commercial vendor.

3.5 RADIOACTIVE WASTE FACILITIES

The wastes containing the most significant quantities of special nuclear material are items such as irradiated fuel specimens which have been examined and analyzed at the Building 102 hot cell complex. Some of the wastes are delivered to a licensed waste contractor while others are stored at the Hillside Storage Facility (HSF) complex. Other radioactive wastes ordinarily contain only gram or milligram quantities of special nuclear materials. Facilities used for waste storage and handling at the site are specified in the sections which follow.

3.5.1 Dry Wastes

Dry contaminated wastes are placed in sealable drums, tubes, boxes, or casks available at each laboratory or facility where such wastes may be generated. Each laboratory or facility maintains a designated area for temporary waste storage. Dry wastes are transferred to Building 102 for final inspection and any necessary repackaging. Waste packages then are transferred to the site radioactive material storage facility, HSF.

The facility has a horizontal tube facility for storage of high-level radioactive material contained in sealed encapsulations called "waste liners". The horizontal tube facility is made of two rows of 40-foot-long concrete-lined steel pipes mounted horizontally and covered with earth. Eleven of the tubes have a 6-inch inside diameter, and seven have a 10-inch inside diameter. The tubes in either row are spaced on 3-foot centers, and the rows are spaced 3 feet apart with the tubes in the bottom row offset halfway between the tubes in the upper row. Shielding is provided on the top and sides of the facility by a minimum of 6 feet of compacted earth. Shielding at the front and back ends consists of 3 feet of concrete in which the pipes have been anchored, plus concrete-filled step plugs with a minimum of 3 feet of concrete shielding in the plug. Additional above-ground space for lower level waste or other materials is available within this fenced and posted facility.

3.5.2 Liquid Wastes

Liquid wastes are routed from laboratory sinks and gravity drains leading from sources known to be or potentially contaminated through regulated pipe lines to retention tanks located in each building where such wastes are generated. Such wastes are transferred periodically to a waste treatment plant for concentrating and solidifying the liquid wastes which are described in Section 13 of this application.

Other liquid waste (excluding sanitary waste) flows through a separate piping system into any three of four 60,000-gallon retention basins. After sampling and determining that radioactivity, if any, is within permissible discharge levels (3.0×10^{-8} gross alpha and 5.0×10^{-8} gross beta-gamma), the water in the basin is released.

Sanitary wastes are treated, and the waste waters are sprinklered on site.

3.6 EMERGENCY EQUIPMENT

A vehicle is available to Radiation Safety and can be equipped quickly with a supply of protective clothing, first aid equipment, respiratory protection equipment, and portable instrumentation and sampling equipment for use during emergencies. Emergency equipment also is stored in selected areas on site.

3.7 INDUSTRIAL SAFETY EQUIPMENT

In conjunction with the radiation safety program at VNC, industrial health and safety of VNC personnel also are emphasized. Some of the major protection facilities and equipment which are available include portable extinguishers, sprinkler systems, a dispensary attended part time by a registered nurse, and a wide range of typical industrial safety equipment.

3.8 CRITICALITY ALARM SYSTEMS

In any Vallecitos Nuclear Center area in which special nuclear material containing more than 500 grams of U-235 is used or stored and does not otherwise qualify as a "subcritical area" as defined in Section 3.16 of Appendix A, a monitoring system, including gamma or neutron-sensing devices which will energize an audible alarm in the event of criticality, is installed and maintained. The system in use on site is described in the following paragraphs.

3.8.1 LiI Crystal System

This monitoring system consists of a lithium-iodide crystal coupled to a photomultiplier tube. Failure of any detection circuit component which would prevent criticality detection activates a warning light on the unit. Failure of any signal-producing component is detected during the monthly test.

The alarm level of the system is pre-set at not less than 5 mRem/h nor more than 500 mRem/h (this setting is an exemption from 10CFR70.24 contained in License SNM-960). The system is capable of energizing the alarm when the radiation level at a distance of 1 foot from the special nuclear material is 300 R/h or 2.1×10^6 neutrons per square centimeter per second.

Sensing devices are positioned within 120 feet, air equivalent, of every required location where special nuclear material is handled, used or stored. The system is tested by exposing the detectors to appropriate sources and sounding the alarm monthly. The alarm system is designed so that the alarm continues to sound until reset. The alarm is clearly audible in all locations where radiation exposure may result from an accidental criticality incident. If a facility does not have emergency backup power, all movements of SNM are suspended during a power failure.

3.8.2 Excepted Areas

Criticality sensors are not provided below the surface of the water in the RML storage pool or within the RML cells, nor in the horizontal tube solid waste facility. Shielding surrounding the special nuclear material is as follows:

RML Storage Pool	16 feet of water
RML Hot Cells	1.5 to 3.0 feet of high-density concrete
Horizontal Waste Storage	5.5 feet of compacted earth

3.9 ANALYTICAL LABORATORY COUNTING EQUIPMENT AND CAPABILITIES

The following is a summary of the normal capabilities of the analytical laboratory counting room for radiation safety samples.

Sample Type	Instrument	Minimum Detection Limit*
Air and Exhaust Stack Samples	Alpha Proportional	3×10^{-15} μCi/cc
	Beta Proportional	7×10^{-15} μCi/cc
Charcoal Cartridges (I-131)	NaI(Tl)	2×10^{-13} μCi/cc
Smears	Alpha Proportional	7×10^{-8} μCi
	Beta Proportional	3×10^{-6} μCi
Water (Retention Basin)	Alpha Proportional	3×10^{-8} μCi/cc
	Beta Proportional	5×10^{-8} μCi/cc

*Based on standard sample size and counting times.

4.0 RADIATION PROTECTION PROCEDURES

A system of Vallecitos Safety Standards establishes the site radiation and criticality protection and regulatory compliance programs. The manager of the Nuclear Safety component issues the standards with review and comment of the managers of the major organizational components located on the site.

Currently, there are about 40 standards dealing with radiation protection matters. The principal features of these are summarized below.

4.1 PERSONNEL WORK RULES

Requirements are established to prevent or minimize the hazards of radioactivity and radioactive materials. Eating, storing or preparing food, smoking, or storing tobacco are not normally permitted in areas where there is a potential for contamination with radioactive materials. Food containers may not be used for storing or handling radioactive materials.

General Electric furnishes protective clothing for service in areas where contamination is likely to contact personal clothing. Protective clothing standards are set by the site Radiation Safety function to assure effective quality, positive identification, and to avoid use for other than its intended purpose. The amount and type of clothing for any specific activity are assessed on the basis of potential for personnel contamination.

4.2 LIMITS OF RADIATION IN CONTROLLED WORK AREAS

All Vallecitos locations where there is a potential for radiation exposure are classified (radiation area, high radiation area, etc.) in accordance with the definitions of 10CFR20, Sections 20.202 and 20.203. General Electric's philosophy of protection is to keep radiation exposure at the lowest reasonably achievable level in all cases.

With respect to operations which could produce airborne radioactive contamination, managers of facilities are responsible for providing ventilation equipment to meet the concentration limits of 10CFR20.103 without the necessity, or credit, for personal respiratory devices during routine operations. In certain nonroutine situations where adequate ventilation equipment could not ensure control of airborne material, respiratory protection of demonstrated integrity is utilized. In such cases, respiratory protection equipment will be used in accordance with NUREG 0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials".

The respiratory equipment currently in use at VNC is approved by the National Institute of Occupational Safety & Health (NIOSH) and, as such, achieves compliance with 10CFR20.

U.S. Divers Company's Survivair self-contained breathing apparatus (SCBA) or other NIOSH-approved SCBA having a backpack air supply, hose, harness, pressure-demand regulator, and a facepiece may be worn in emergency situations.

VNC has adopted only equipment that is approved by NIOSH.

No individual will be permitted to work more than 20 hours per week under conditions requiring masks. No individual will be permitted to work in a mask until he has received a medical clearance for respirator use and has been thoroughly instructed in methods of proper use, fitting, and field testing of respirators.

4.3 PERSONNEL MONITORING

Instructions for the use of film and TLD badges, finger TLD dosimeters, and pocket dosimeters include the proper part of the body on which the device is to be worn and procedures to prevent spurious readings. In addition, personnel are instructed to use monitoring instruments upon leaving a radioactive materials area. Hand-and-shoe counters are also provided at some locations.

In order to determine the extent to which individuals may have internally accumulated any radioactive materials, two bioassay procedures are established. Special analyses for specific radioactive materials are performed whenever a need is indicated.

Persons who work routinely with radioactive materials which are readily detectable gamma emitters may be tested additionally, or alternatively, by the whole body counting technique. A whole body counter is operated by the Nuclear Safety function. These tests are performed at least annually. More frequent schedules are maintained for personnel who routinely handle radioactive materials. The whole body counter technique provides the advantage of producing results more accurately and more promptly than urinalysis and, therefore, is used wherever practicable.

The Nuclear Safety function is advised of all additions to the work force, changes in individual assignments, termination of employees, and any occurrence which may have resulted in internal deposition. This procedure assures that appropriate schedules are maintained for biological assay.

4.4 SURVEYS

Surveys to assure radiation safety are made routinely in order to detect any unfavorable trends or conditions. Special surveys also are conducted as warranted by the suspected or potential presence of radiation or radioactive material. Routine survey schedules are established whenever work programs are initiated and are changed when the work is discontinued or the scope of work is revised. Survey methods are formulated to meet the needs of the particular type of radioactive materials used and in the light of the equipment capability. Original survey reports are retained by the Radiation Safety function with copies to the supervisor of the surveyed area for information and action as appropriate.

Environmental surveys are made both within the VNC site and at off-site locations. These surveys involve samples of ground water, soil, vegetation, and air.

The allowable stack discharge dilution factor, i.e., the ratio of the effluent concentration in a stack to the concentration at the site boundary, for any given discharge stack on site is a function of the actual annual meteorological conditions, the stack flow rate, and the distances from the stack to the site boundaries. Technically, there could be an allowable dilution factor for each radioactive effluent stack on site; but for simplicity and conservatism, the Building 102A stack, which has the largest potential source term and the highest flow rate, is used to determine a single allowable dilution factor for all SNM-960 licensed stacks on site.

To calculate the worst-case airborne effluent concentrations at the edge of the site in the event of a release, it is assumed that all releases are from ground level during moderately stable (Type F) atmospheric conditions with a one-meter-per-second wind speed. No effective stack height is assumed because of limited stack heights and their proximity to buildings. The wind is assumed to blow toward the point of interest during the release. Using calculational methods and data outlined in Application Amendment No. 48 to Docket 50-18, HW-SA-2809, and Regulatory Guide 1.3, a concentration of 3.6×10^{-9} $\mu Ci/cm^3$ will result from an average emission of 1 $\mu Ci/sec$ from a 360-meter distance source.

Notwithstanding the above calculations, all effluent air from operations involving potential particulate activity is filtered; and, therefore, special nuclear material concentrations have been and are expected to continue to be routinely well below permissible levels for unrestricted locations at the points of discharge from the stacks.

Routine calibrations of survey instruments are performed as follows:

a. Alpha calibrations by use of sources which are traceable to plutonium and uranium standards prepared by the National Bureau of Standards.

b. Beta calibrations by use of a slab of natural uranium in radioactive equilibrium with Th-234 and Pa-234.

c. Neutron instrument calibrations are performed by measuring a response to a pulse generator and to an Am-Be neutron source.

d. Gamma calibrations using Cobalt-60 sources standardized with a meter which, in turn, was calibrated with traceability to the National Bureau of Standards.

All radiation monitoring instruments are calibrated as frequently as deemed necessary to assure reliability during use. Portable radiation monitoring instruments are calibrated on an annual basis, before initial use, and after repair.

Stack particulate monitor systems are calibrated by placing a uniformly distributed radioactive source in the same geometry as the filter paper used for collecting particulates. Stack gas monitor systems are checked routinely by placing a reference gamma source on the side of the Kanne chamber and observing whether the response falls within prescribed limits. Calibration with a known radiogas standard has been performed to verify this procedure. Iodine monitors also are source checked routinely.

4.5 POSTING AND LABELING

Instructions are established implementing the posting and labeling requirements of 10 CFR Part 20. Additional precautionary signs may be utilized to meet special requirements and detailed procedures. The area supervisor is responsible for maintaining the proper posting and labeling.

However, in view of the degree of control inherent in the security of the site and the required prerequisite training program for employees, there is an exemption from the provisions of 10CFR20.203 for high radiation area alarms which General Electric deems should be continued.

An exemption from the alarm provisions of 20.203(c)(2) is requested for any area or location which is:

a. (1) Used for the temporary placement of strong, weatherproof containers such as shipping casks not exceeding the limits of 49CFR73.393, i.e., 200 mR/h at any point of readily accessible surface; or

(2) An established area in which there may be radioactive materials in containers, the readily accessible surfaces of which do not exceed dose rates of 1,000 mR/h at any time and which do not exceed 100 mR/h at the barrier for any continuous period greater than 30 days, such as radioactive waste pickup areas from which wastes are collected on a weekly or monthly basis; and is

b. Located within the 94-acre inner fenced exclusion area of the site; and is

c. Bounded by a distinctively colored rope or chain providing a barrier at dose rate values not greater than 100 mR/h and posted as a High Radiation Area in accordance with 10CFR20.203(c).

4.6 WASTE DISPOSAL

Detailed procedures for the packaging, storage and removal of contaminated material which no longer is useful are established by operating components and reviewed by the Nuclear Safety function. The procedures define low, high, and intermediate levels of solid waste on the basis of a contact dose rate at the surface of outer packaging and provide specifications for container packaging to prevent loss of contents, repackaging of damaged units, labeling of contents and similar requirements. Procedures for handling or disposal of liquid wastes in the various waste treatment facilities described in Section 3 are detailed in similar instructions. Area supervision is responsible for adherence to proper handling procedures, for obtaining approved containers and for arranging for transportation of wastes to the appropriate site storage or treatment facility.

4.7 REPORTS AND RECORDS

Records of surveys, personnel exposure records, and other records indicating the degree or nature of individuals' exposure are maintained by the Radiation Safety function. These records currently are retained for an indefinite period.

4.8 INDUSTRIAL SAFETY PROGRAM

The effectiveness of the radiation and criticality safety program is in large measure influenced by a sound industrial safety system, and vice versa. Good housekeeping practices, modern fire prevention equipment, and training in handling toxic or explosive chemicals are but a few benefits of industrial health and safety which carry into the nuclear areas. For that reason, compatible program philosophies and policies are maintained. General Electric places direct responsibility for safety on the individual employee and for directing safe practices on his area manager.

Other important elements include pre-employment medical examinations; medical surveillance of personnel; control of flammables, explosives, and toxic materials; and administration of fire prevention equipment requirements in accordance with applicable codes and regulations.

4.9 VALLECITOS EMERGENCY PROCEDURES

Plans for the prompt and rapid response to emergency situations are set forth in the emergency procedures. The procedures are revised from time to time to reflect changes in equipment and organization at the site. The procedures address emergencies which could arise from accidental criticality or release of radioactive material.

4.9.1 Nuclear Emergencies

The accidental criticality alarm is a distinct sound which is activated automatically when criticality detection instrumentation in that local area reaches a pre-set limit. Personnel in the affected area immediately evacuate.

At the same time, the origin of the signal is designated at the main security building. The site-wide public address system is used to instruct all personnel at the site concerning further action. Upon the sounding of any alarm, an assessment is made of the indicated or reported situation, and whatever action is deemed necessary to minimize personnel injury and property damage is taken. A radio communication system and a dedicated telephone network can be used for conference contact with VNC site managers representing most fields of technical competence at the site, with the person reporting the emergency, and with the security building.

If the nature and severity of the emergency requires evacuation, either complete or partial, instructions are broadcast.

Personnel who have been trained in the use of survey instruments and protective apparel are available. Guidelines to assist management in formulating dose limits for emergency exposures for the protection of human life, recovery of victims, and protection of health and property are available. A system which has undergone a criticality accident will be left undisturbed until competent review has produced a plan to cope with the situation. Calibrated instruments are available at strategic locations.

Injured persons, if any, may be taken to the hospital in Livermore. Approximate travel time to the hospital is 15 minutes. Arrangements have been made with the hospital to receive and care for injured persons who are contaminated, with supervision by competent Nuclear Safety personnel. In the event of spurious alarms, the all-clear will be announced on the public address system.

4.9.2 Non-nuclear Emergencies

Other alarms (automatically and manually operated) are used for fire or other non-nuclear emergency. The senior supervising person at a facility will decide when an emergency exists and will take appropriate action for the safety of people and equipment.

4.10 RADIATION SAFETY REVIEW

The design of new VNC facilities, or major changes in existing facilities in which radioactive materials are to be handled, must be reviewed by the Radiation Safety function to insure that adequate health protection facilities are provided. Policy design and operating elements are considered in this review. Materials containment, exposure reduction (both operating and maintenance), design life, and assessments of potential hazards are used in the final design phase whereby such hazards are engineered out.

The Area Manager concerned, or his designated representative, is responsible for initiating requests for radiation safety review of proposed facilities and radiation safeguards.

5.0 CRITICALITY SAFETY PROGRAM TECHNOLOGY

5.1 INTRODUCTION

Administrative responsibilities for implementing the criticality safety program at the Vallecitos Nuclear Center (VNC) are given in Section 2.0 of this application. The responsibility of qualified personnel to maintain activities at all times within specified limits is emphasized.

In making a criticality analysis, the two-contingency criterion is used: the occurrence of at least two unlikely, independent, and concurrent changes in one or more of the conditions originally specified as essential to nuclear safety is required before a nuclear accident is possible. All analyses receive an independent review.

An organizational component competent in neutron physics and the technical methods for criticality computations shall verify the use of any new computational methods for criticality control.

In the discussion below, information about computational tools useful for performing criticality analyses at VNC is presented. Sources of nuclear parameter data useful in performing hand calculations are discussed. Critical experiments useful in evaluating the computational tools for criticality analyses are identified. Any negative biases that must be applied are identified. Finally, the adjustment factors for normal operating conditions are presented.

5.2 MINIMUM CRITICAL QUANTITIES AT ISOLATED LOCATIONS

When only small quantities of special nuclear materials are required at any one time within an isolated facility, accidental criticality hazards are easily and definitely avoided by control of inventory to quantities substantially lower than the minimum quantities of such materials that have been experimentally found to be capable of criticality under optimum laboratory conditions.

As used in this section, the term "isolated" means physically separated from other areas where special nuclear materials are used. The procedures for accounting for inventories of SNM normally will be under specifically designated custodians. The minimum neutronic isolation barrier is taken as the equivalent of 12 inches of water or a distance of 12 feet, or the greatest distance across an orthographic projection of either accumulation or array on a plane perpendicular to a line joining their centers.

5.3 DETERMINATION OF CRITICAL PARAMETERS OF ACCUMULATIONS AND ARRAYS OF FISSILE MATERIAL

5.3.1 Experimental Critical Data

Experimental critical data are available in the literature from which critical parameters for fissile materials such as mass, sphere volume, cylinder diameter, slab thickness, and concentration may be determined. Data are often presented so that values of critical parameters may be read directly from curves or tables. Some documents from which critical parameters may be obtained are:

a. Karlsruhe Symposium papers, 1961.

b. A.H.S.B. Handbook 1, "Handbook of Criticality Data".

c. TID-7028, "Critical Dimensions of Systems Containing U-235, Pu-239 and U-233".

d. TID-7016, Rev. 1 and Rev. 2, "Nuclear Safety Guide".

e. LAMS-3067, "Los Alamos Critical Mass Data".

f. LAMS-2537, "Correlations of Experimental and Theoretical Critical Data".

g. Y-1272, "Y-12 Plant Nuclear Safety Handbook".

h. ARH-600, "Criticality Handbook".

i. Oak Ridge Criticality Data Center Report, ORNL-CDC-5.

j. DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods In Water".

When the geometry of the accumulation is other than a simple one, data obtained from this literature may be used to determine critical parameters by using simple buckling conversion methods or by using approximate, but conservative, values for the accumulation dimensions.

5.4 ADJUSTMENT FACTORS FOR NORMAL OPERATING CONDITIONS

Criticality analyses shall utilize calculative methods that have been evaluated by the calculation of models of critical experiments of similar geometrical configuration and material content; the maximum negative bias in the effective multiplication factor, defined as $(-\Delta k_{bias}) = k_{calculation} - k_{experiment}$, determined in the evaluation shall be applied to the analysis as given in the relationship below. The effective multiplication factor of the accumulation or array of accumulations shall satisfy the relationship:

$$k_{cal} + \Delta k_{uncertainty} + (-\Delta k_{bias}) \leq 0.95$$

where: k_{cal} is the calculated effective multiplication factor by the calculative method, $\Delta k_{uncertainty}$ is the statistical uncertainty (2σ) in the mean value of the calculated effective multiplication factor, and ($-\Delta k_{bias}$) is the maximum negative bias determined from the validation of the calculative method. If the bias is positive, the ($-\Delta k_{bias}$) term shall not be used, so that overstatements in the effective multiplication factor shall not be corrected.

Normally, subcritical values of nuclear parameters for individual accumulations under normal operating conditions shall be as shown below. However, in no case shall the effective multiplication factor exceed 0.95.

a. For accumulations limited by mass, the normally subcritical mass shall not exceed 74% of the critical mass.

b. For accumulations limited by volume, the normally subcritical volume shall not exceed 76% of the critical volume.

c. For accumulations limited by dimension, the normally subcritical cylinder diameter or slab thickness shall not exceed 90% or 88%, respectively, of the critical dimension where U-235 is the fissile constituent, nor 85% in all other cases.

d. For accumulations limited by concentration, the normally subcritical concentration shall be less than that concentration which is equivalent to the following atomic ratios of hydrogen to fissile material: H/U-235, 5200; H/U-233, 7600; and H/Pu (fissile), 7600.

e. Not more than 45% of the critical mass shall be contained in any accumulation in a criticality limit area in which double batching is credible.

Normally, subcritical values of nuclear parameters of interacting accumulations shall be:

a. Not more than 69% of the critical number of accumulations whenever the critical number has been determined by experiment; or

b. Not more than 50% of the critical number of accumulations in any array calculated by the density analog method (Ref.: ARH-600); or

c. The number of accumulations determined by the solid angle method as described in TID-7016, Rev. 2, paragraphs 4.29 through 4.34. The method may be applied only to accumulations that are well moderated (e.g., aqueous solutions) at the point of the maximum effective multiplication factor, and array reflection must be no more effective than a thick water reflector spaced at distances from the accumulations comparable to the spacing between accumulations.

6.0 LABORATORY BUILDING 102

6.1 LOCATION AND GENERAL DESCRIPTION

Building 102 is a single-story with basement, concrete and steel structure located as shown in Figure 6.1. The predominant feature of the building is the multikilocurie facility known as the Radioactive Materials Laboratory (RML). General support laboratories for these facilities also are located in the building. The current layout of the building is shown in Figure 6.2. The laboratory areas of the main floor are separated from general office areas by a fire wall. Inherent in the building design are general service facilities such as the main ventilation system, decontamination rooms, and other areas which are described later in this section.

6.2 VENTILATION SYSTEM

6.2.1 Air Supply

Inlet air to Laboratory Building 102 is provided by 30 air conditioning units furnishing modulated, filtered and tempered outside air to the operation and office areas. The capacity of the inlet air system is a nominal 65,000 cubic feet per minute.

6.2.2 Direction of Flow

Airflows in the laboratory are from areas of low radioactivity toward and through areas of higher radioactivity. The arrows in Figure 6.2 indicate general direction of flow. The system of directional airflow minimizes the possibility of accidental contamination of nonradioactive areas. Airflows are all single pass and continuous except a portion of the RML operating gallery air which is recirculated through absolute filters.

The operating areas are supplied at rates of 6 to 40 changes per hour. A minimum of six air changes per hour is being exhausted from the RML cell spaces. This amount of airflow to the cells is adequate for control of contamination.

6.2.3 Pressure and Flow Rates

Small static pressures (-0.01 to -0.03 inch of water) are used in the isolation of thin-walled general work areas, while static pressures from 0.02 to 0.20 inch of water are used to isolate the RML cells. The flow diagram as shown in Figure 6.2 indicates the airflow. Glove box contamination control is effected by maintaining from 0.5 to 1.0 inch of water negative static pressure. At this static pressure, air velocities of 125 linear feet per minute or greater are achieved through openings, an adequate rate to provide the necessary control.

6.2.4 Filtration and Exhaust

The main exhaust equipment is located in Building 102A (Figure 6.1). During regular working hours and any other times radioactive material is handled, two of six exhaust fans are used to provide a normal exhaust flow of approximately 43,000 cfm with the other fans in a standby status. Exhaust air from Building 102 is routed to Building 102A via overhead ducts. At Building 102A, this exhaust air is passed through 90 HEPA filters and then discharged to the atmosphere through a 66-inch-diameter, 75-foot-high stack.

Exhaust air from the RML Hot Cells is prefiltered at the outlet of each cell and is routed to a filter bank of 10 HEPA filters in the Building 102 basement before it joins the main exhaust stream at Building 102A. Two activated charcoal filters are available as needed. Two booster fans are available as required to aid in the transfer of the RML Hot Cell effluent from Building 102 to Building 102A.

The filter banks and the exhaust connections, at their point of origin, utilize high-efficiency filters. The filter system is 99.95% efficient for 0.3-μm-diameter homogeneous particles of dioctyl sebacate. The filters are constructed of fire-resistant materials and are housed in noncombustible duct work.

The exhaust connections have adjustable dampers where necessary to provide the capability for balancing flows throughout the system.

Figure 6.3 is a schematic diagram of the ventilation exhaust system.

Fire protection is provided to the exhaust system by: (1) a CO_2 suppression system and a water fog system for the Hot Cell basement exhaust filter bank; (2) a water spray system for the Hot Cell charcoal filters; and (3) a water fog spray for the main exhaust ducting from Building 102. None of the water suppression systems can discharge into facilities (e.g., Hot Cells, glove boxes) containing SNM.

6.2.5 Emergency Exhaust

In the event of the loss of utility power, emergency power is provided by a 335-kW diesel-driven electric generator. The generator is activated automatically by power loss and is designed to reach full capacity within one minute. All critical equipment normally supplied with normal power through the main building switchboard will be supplied with emergency power. This equipment includes two exhaust fans, fans supplying air to areas where radioactive materials are handled, and the main stack monitoring equipment.

The generator is supplied with fuel from a 1,000-gallon storage tank. At an estimated fuel consumption rate of 20 to 25 gallons per hour, this supply should be sufficient for at least 40 hours of continuous, full-load operation.

The building alarm panel and exhaust fans are supplied with normal power through the Motor Control Center (MCC). The solid-state logic portion of the MCC is a solid-state, battery-powered computer that provides for automatic switching of equipment during transitions between normal and emergency conditions. During normal and emergency power operations, it performs the following functions automatically:

a. If one of the operating fans should fail, it will provide an audible and visual alarm; identify and shut down the failed fan; and start a standby fan. The control panel will indicate which fan has failed.

b. In case a fire is identified by a detector in the exhaust ducting, the computer operates the exhaust system on reduced flow until the "fire" condition is cleared manually.

6.2.6 Stack Monitoring

The Building 102 stack monitoring system was designed for the monitoring of the primary isotopes released during normal operations in Building 102 and 102A. The system has been designed to detect beta-gamma particulate and gaseous activities. Each detector output signal is transmitted to a processor/recorder which processes the data into a desired format and displays information on a recorder printout or trace. The flow schematic of this system is shown in Figure 6.4.

6.2.6.1 Noble Gas Monitor

The flow-through ion chamber is a detector to measure activity in gases. The test gas flows between the outer pair of electrodes, called the "scrubber", to which a potential is applied. Here, ionization from previous decays is cleared from the gas. The uncharged test gas then passes into the 16.3-liter active volume. The ions produced by any radioactive decay occurring in the active region drift to the collecting electrode. The resulting current is measured by the electrometer. The sample flow rate is 3 cfm.

The response for Xe-133 is 1.45×10^{-7} amp/μCi/cc.

6.2.6.2 Beta-Gamma Monitor

The beta-gamma particulate monitor is utilized to evaluate the total amount of beta particulate activity and the total amount of a pre-selected gamma-emitting isotope in a sample airstream withdrawn from the stack. The sampled air (1.5 cfm) passes through a 47-mm filter which will remove approximately 99.9% of all particulates greater than 0.3 μm in diameter.

The beta-gamma activity collected on the particulate filter is monitored by a 2-inch-diameter pancake-type GM detector having a window thickness of 7 to 8 mg/cm^2 which is located approximately 1/2-inch from the face of the filter.

The GM detector portion is calibrated with a Cl-36 beta source for which the efficiency is approximately 35% (4π).

6.2.6.3 Permanent Records

The 47-mm filter and a charcoal cartridge are sent to the site counting room for analysis (alpha, beta-gamma and iodine). These analyses provide the basis for the permanent stack effluent emission records.

6.3 DECONTAMINATION ROOM

A decontamination room is available for cleaning tools and equipment. Equipment in the decontamination room includes standard filtered airflow-type hoods connected to the building ventilation system, a large stainless steel sink draining to the building radioactive waste system, and miscellaneous other items of equipment typical of that used for decontamination work. The room is located in a high radiation area.

6.4 AIR SAMPLERS

The air in various parts of laboratory rooms, offices and other occupied areas is sampled on a continuous basis. Approximately 80 sample stations in the RML and radiochemistry areas have been provided. Air is drawn through each sample station filter paper. Flow rates routinely are determined using standard equipment for such determinations. Filter papers are removed from the stations on a routine schedule and counted for alpha and/or beta-gamma activity. Continuously monitored, alarming air sample stations also are provided in the RML.

In addition, spot check air sampling is done whenever a new operation is introduced to assure absence of significant contamination in the breathing air; and several continuous air monitors are in use for alpha and beta-gamma determination. Alarms are sounded if concentrations increase significantly.

6.5 LIQUID WASTE HANDLING

Sinks and gravity drains in Building 102 radioactive materials areas are routed to a filter which removes particulates. The liquid then is pumped to a 3,000-gallon tank. The tank periodically is emptied, and the waste liquid is sent to the liquid waste evaporator facility to be concentrated, solidified, and transferred to a commercially licensed waste disposal contractor. This liquid waste material contains only minute quantities of SNM.

Highly radioactive liquid wastes are solidified at the hot cell work stations for transfer to the waste contractor. Such wastes contain only gram quantities of SNM and have no potential for criticality.

6.6 EMERGENCY POWER

In addition to the exhaust system as described in Section 6.2.5, the Building 102 emergency generator also provides emergency power to the criticality alarms, the breathing air alarm (loss of pressure alarm), the fire alarms, the telephone equipment, and an annunciator panel which transmits these alarms to the security building.

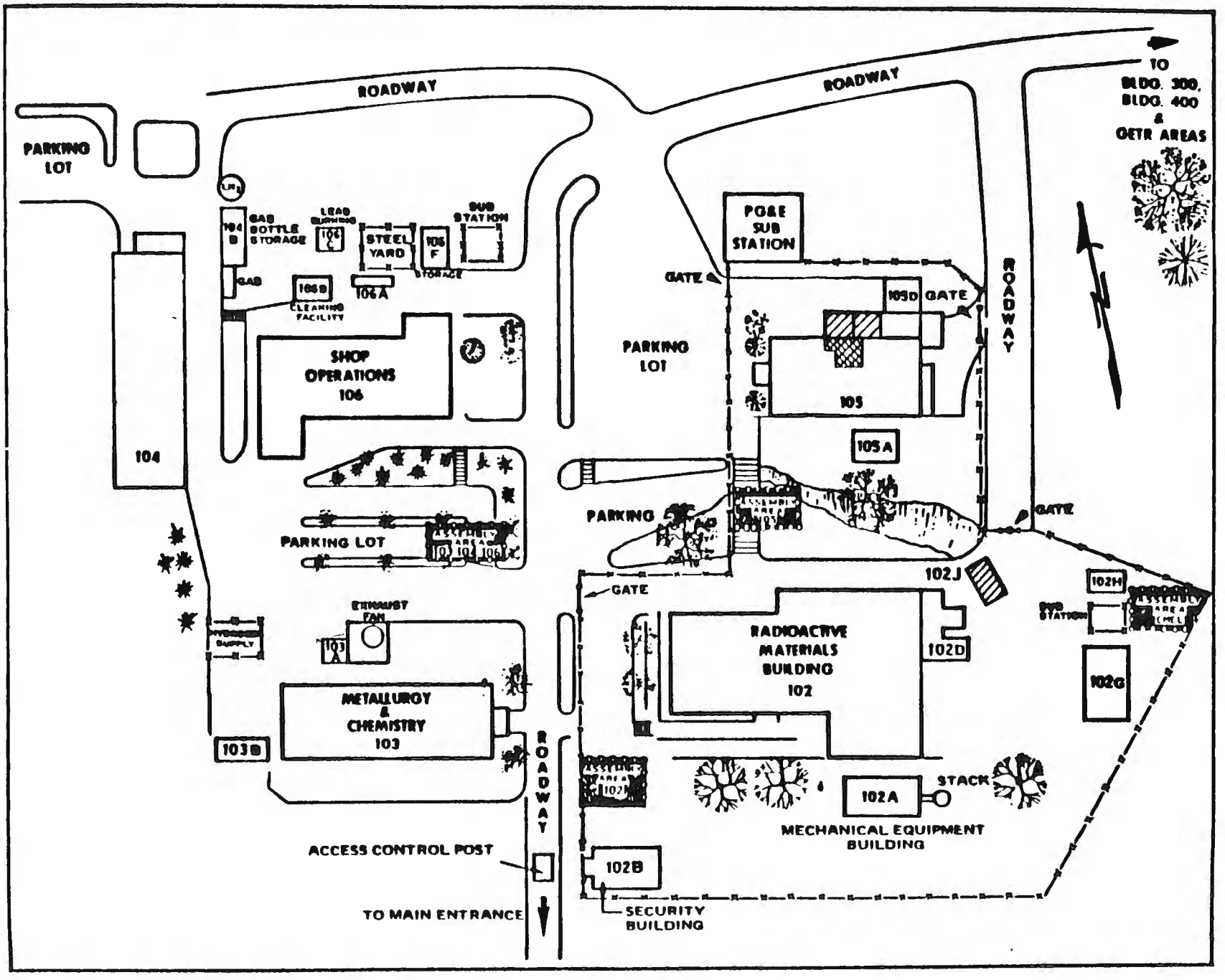


FIGURE 6.1.1 100 AREA AND SURROUNDINGS

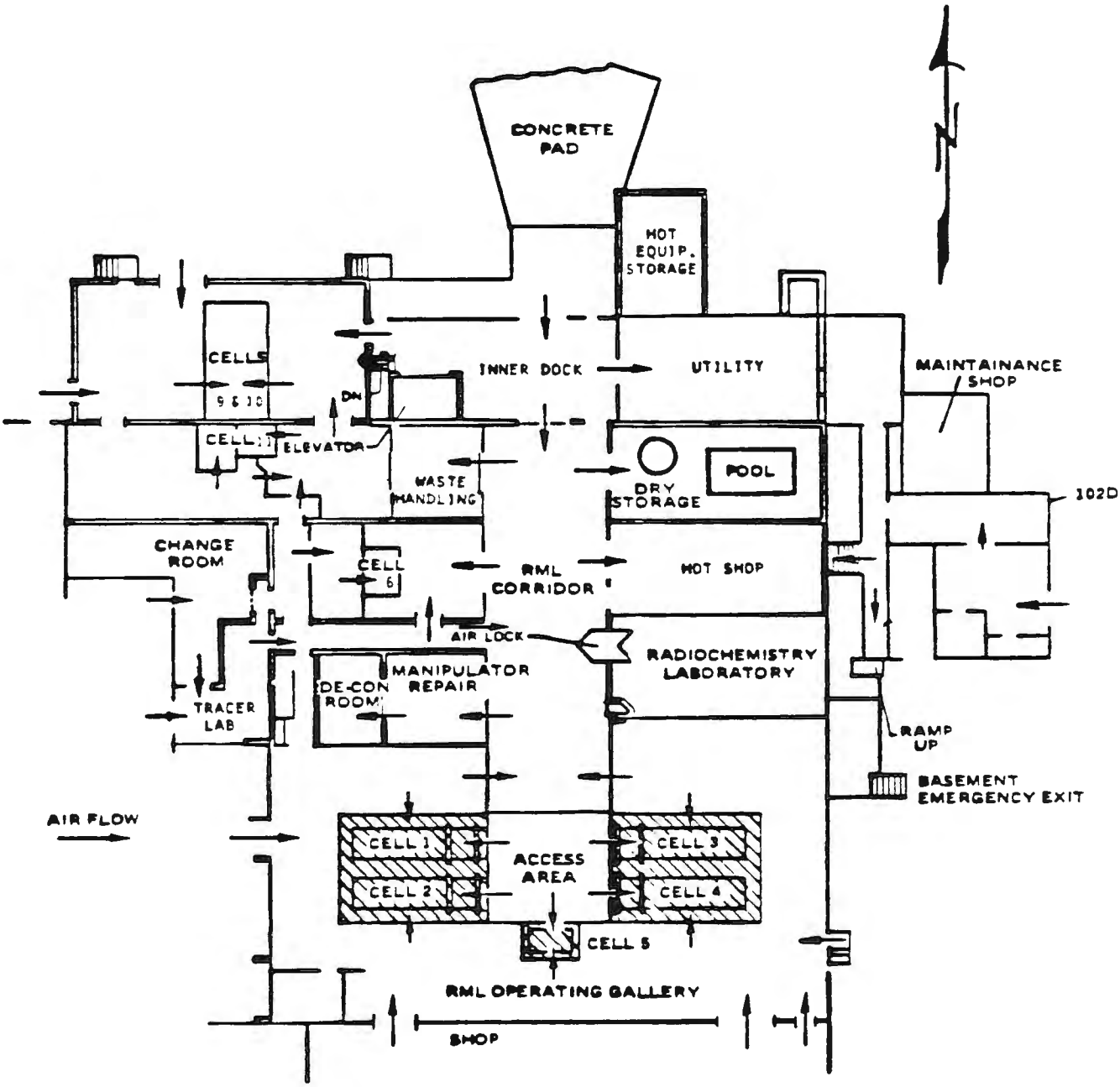
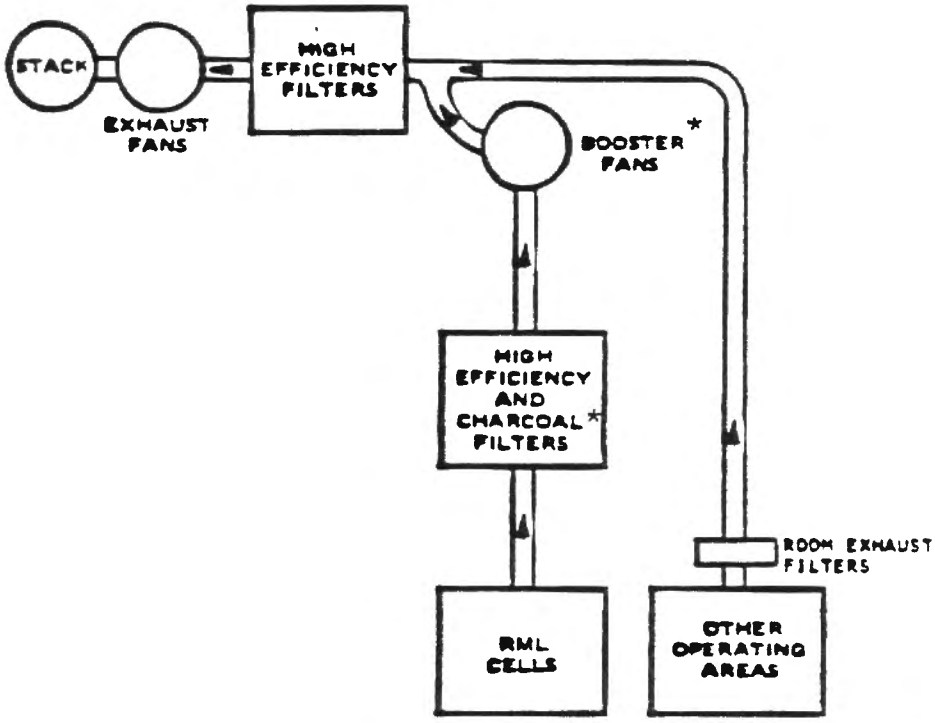


FIGURE 6.2. BUILDING 102 MAIN FLOOR - RADIOACTIVE MATERIALS AREAS



*Normally bypassed

FIGURE 6.3. BUILDING 102 EXHAUST SYSTEM SCHEMATIC DIAGRAM

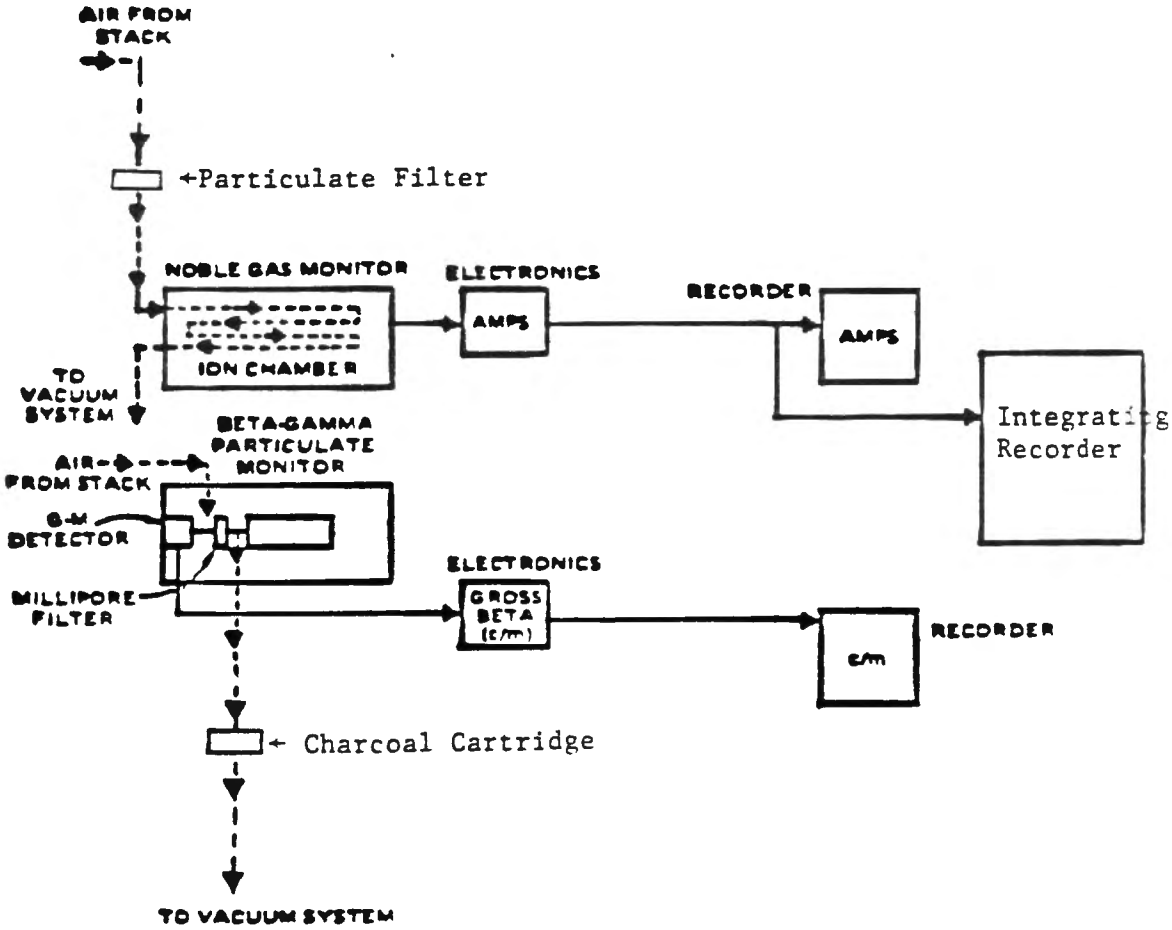


FIGURE 6.4. BASIC STACK SAMPLING SYSTEM FOR BUILDING 102

7.0 RADIOACTIVE MATERIALS LABORATORY

7.1 LOCATION AND GENERAL DESCRIPTION

The Radioactive Materials Laboratory (RML) is the term applied to a series of shielded facilities located in Building 102 used for the examination of irradiated reactor fuels and irradiated hardware and radioisotope production, including support and service areas. A plan of the RML area is shown in Figure 6.2. Although there is a number of hot cells in the RML complex, special nuclear material licensed under License SNM-960 is handled predominantly in Hot Cells 1, 2, 3, 4 and 5; and, accordingly, only those cells will be addressed in detail in this section. A sketch of these RML cells is included as Figure 7.1.

7.2 GENERAL PLANS AND USES OF MATERIALS

RML provides a high-level shielded facility in which safe and efficient non-nuclear testing and examination of irradiated reactor fuels, materials and components may be conducted. The facility also is equipped for the handling of large quantities of by-product materials.

7.3 RML FACILITIES

7.3.1 General Description

Equipment and facilities of the Radioactive Materials Laboratory are classified into several major areas: the cell operating area, the general laboratory and cell service corridor, and the cell area. All areas are on the main floor of Building 102 as shown in Figure 6.2. These areas are described more fully in the sections immediately following.

7.3.2 High-Level Cells (Hot Cells 1, 2, 3 and 4)

The cell area is the heart of the Radioactive Materials Laboratory as it is here that work on irradiated material is performed. A double or twin cell block for high-level work is located on either side of the south end of the service corridor, constituting a total of four high-level cells. Each of these cells can contain safely in excess of one million curies of 1-MeV gammas. The high-level cell walls are 36 inches thick up to a height of 12 feet, where they are reduced to 24 inches to take advantage of reduced shielding requirements and to provide a set-back for overhead manipulator rails. The shielding material is a high-density concrete made with ferro-phosphorous aggregate with a poured weight of 300 pounds per cubic foot. Shielding in the vertical direction is provided by the 3-foot-thick concrete roof above the cells.

Each cell can be equipped with an overhead bridge-mounted manipulator and a 3-ton bridge crane running on the same rails. The units are designed to traverse the entire length of the cell and the radiation lock and can be moved out of the cells on removable extension rails. This equipment provides handling capability for transfer of irradiated materials in heavy shielded casks.

The viewing window at each operating station is a 3-foot-thick, lead glass window with glass of sufficient density (at least 6.2 gm/cc) to give shielding equivalent to the cell wall.

Each of the high-level cells as shown by Figure 7.1 has a 6-foot-long radiation lock for entry from the access corridor. These radiation locks are formed by hydraulically operated bi-parting steel shielding doors. The outer door is 18 inches thick, and the inner door is 15 inches thick. A 3.5-foot-diameter by 9-foot-deep pit is provided in the radiation lock floor to accommodate very large casks and permit long irradiation assemblies to be withdrawn. The access corridor width and floor loading specifications were based on the use of a 15-ton-capacity fork lift truck in this area. An intercell transfer system is provided in the common wall between each pair of cells.

Each high-level cell has an operating area approximately 17 feet long by 6.5 feet wide and is 14 feet high inside. There are four operating stations in each cell. Three of the stations are located along the side wall of the cell and the fourth at the end wall.

There is a total of 16 operating stations available. Typically, the elements of each cell operating station are a pair of through-wall master-slave manipulators, a viewing window, and the "in-cell" work or experiment apparatus required for carrying out one or more functional tests or operations. Six to seven feet of width are available at each station internally and externally. This permits adequate space for one or two operators and the miscellaneous control and operating equipment which is required on the cold side.

Water is piped to the hot cells by an independent supply system. A 25-gallon vented storage tank located on the top of Cell 1 is filled by a water line not directly connected to the tank. This tank feeds an adjacent 42-gallon pressure tank through a connecting check-valved pump line which, in turn, supplies pressurized water at a maximum of 60 psi to a process water header with branches to each hot cell. This independent system provides positive assurance against feedback of cell water to the potable water system as well as preventing large quantities of water (more than 42 gallons) to enter the cells in the event of piping failure. This system currently serves only Cells 1 and 5.

7.3.3 Cell Door Interlock System

The high-level cells are equipped with inner and outer steel shielding doors which form a radiation lock. The cell doors are hydraulically operated and controlled by a panel at the cell operating face. The operating controls are interlocked such that the outer door normally cannot be opened when the inner door is open. A key-locked override switch is provided for unusual circumstances such as cell decontamination activities.

7.3.4 Cell 5

A smaller cell designed for conducting metallographic work occupies the south end of the main access corridor. This cell is used primarily for the preparation of samples for metallographic examination and microhardness testing. Remotely operated equipment for this cell includes sample mounting, polishing, cleaning, and etching equipment as well as a remotely operated microhardness tester. Metallographic samples after polishing and etching can be checked with the optics of the hardness tester before being transferred remotely out of the cell to a modified research metallograph contained in a shielded enclosure. The working area of the metallography cell is approximately 5.5 feet deep, 8 feet wide, and 8 feet high. The front wall contains two 18-inch by 13-inch lead glass windows for direct observation of the working area. The cell walls are 18-inch-thick magnetite concrete. The back of the cell, which opens onto the access corridor, contains a safe-type door for equipment and personnel access. Metallographic or microhardness test samples are introduced to this sample entry port by means of a special transfer cask.

7.3.5 Pool Facility

A 16-foot by 8-foot by 16-foot-deep, stainless-steel-lined water pool is available for underwater transfer, examination, repair, assembly or disassembly, and storage of irradiated materials. Overhead crane facilities are available for cask handling in the pool or on the floor of the room. A filtered exhaust port is located adjacent to the pool. Dose rates of the surface of the pool are of the order of 20 mR/h at maximum storage capacity.

The pool is equipped with portable types of tools including tongs, hooks, tables, racks, and lights. The water is circulated at about 30 gpm through strainers and a tank containing 6 cubic feet of nuclear-grade ion exchange resin, thereby maintaining excellent water clarity and decontamination. Resin is replenished as necessary. Spent resins are removed to waste drums for disposal.

7.3.6 In-Floor Dry Storage Pit

Located immediately adjacent to the RML storage pool, a dry storage pit is available for temporary storage of irradiated fuel assemblies, rods or other rod-shaped materials. This storage device consists of 19 recessed pipes fabricated of 6-inch Schedule 40 steel pipe 46.5 inches long attached to two horizontal circular steel plates 91.25 inches in diameter and 6 inches thick. The device rests on a ledge cast in the concrete floor so that the top surface is flush with the floor with the tubes extending downward below the floor surface. The pipes are arranged to provide a minimum center-to-center spacing of 18 inches. Twelve-inch-thick stepped plugs provide shielding of sources up to 500 R/h at 1 MeV to dose rates of 2.5 mR/h.

Fuel materials are placed in sealed tubes called waste liners prior to insertion into the storage pipes. Movement of the material is accomplished by means of shielded transfer casks which have top and bottom entry ports. Handling tools which are attached remotely to the waste liner can permit insertion and withdrawal of stored objects without encountering direct radiation from the top of the open cask. Each tube has two 1-inch-diameter holes located 3 inches above the bottom and one 1/4-inch hole in the bottom plate to prevent accumulation of water.

7.3.7 Other Cells

Cells 6, 9, 10 and 11 (see Figure 6.2), are used in support of research and development activities authorized under California License No. 0017-60.

7.3.8 Radiochemistry Laboratory

The radiochemistry laboratory, used principally for the analysis of samples of irradiated fuel materials from the RML, is located immediately adjacent to the RML area. Hoods or glove boxes are connected to the previously described 102 ventilation system and are designed to provide a minimum face velocity of 125 linear feet per minute.

7.3.9 Servicing Areas and Equipment

The Radioactive Materials Laboratory section of Building 102 also contains several shielded cells used in non-SNM work, a waste analyzing area, an equipment decontamination room, a machine shop, equipment storage areas, and a manipulator repair room.

A hot shop facility now occupies the former Plutonium Analytical Laboratory (PAL) next to the radiochemistry laboratory. The PAL was surveyed completely (except for the area behind a metal security wall) and decontaminated following the discontinuance of plutonium operations. The area now is used for the repair of contaminated equipment. Only traces of SNM are present in the contamination.

7.3.10 Criticality Alarm Sensors

The Radioactive Materials Laboratory is monitored for criticality accidents by a detector located at the storage pool area entrance (Figure 6.2).

7.4 RML CRITICALITY CONTROL SYSTEM

Special nuclear materials used in connection with RML operations are principally in the form of oxides in irradiated fuel elements and experimental capsules. The spectrum of material and activity types may be quite broad. This required flexibility has been taken into account in the establishment of criticality controls and is reflected in considerably larger safety margins than might be appropriate to more routine or repetitive situations. For example, each CLA in each fuel examination cell is limited to 45 percent of a critical number of units (fuel rods, assemblies, etc.); and each fuel examination cell is limited further so that criticality is not possible if all of the fissile material in the cell comes together simultaneously under conditions of optimum water moderation and full water reflection; but normal activities preclude moderation to any degree, and

sources of water to the cells are limited and virtually accident proof. Criticality controls governing RML and supporting activities are described in the following sections.

7.4.1 Receiving

RML work is predominately on the post-irradiation fuel examination work but also involves work with various radioisotopes, including encapsulations, and the examination of experimental fuel capsules. Shielded casks used for SNM shipments or on-site transfers have received criticality analyses demonstrating their safety either to receive NRC certification or to meet the requirements of License SNM-960.

Quarantine zones in the radioactive materials storage area (Hillside Storage) and the north side of Building 102 are used for incoming shipments from off site awaiting evaluation of the contents. Containers are spaced according to DOT and NRC requirements.

Casks of irradiated material also may be received on the Building 102 dock.

7.4.2 Storage

Irradiated special nuclear materials may be stored in shielded shipping or transfer casks (temporary), in the RML pool, or in the dry pit storage facility. When material is stored in casks, the spacing between casks and the number accumulated in a given area will be consistent with the DOT requirements for safe shipments.

Pool storage is limited to locations which have been analyzed both for the critical parameters of the accumulation of material in each storage location and the interaction between storage of uranium or mixed oxide irradiated fuel rods or fueled experimental capsules. All storage locations are isolated from one another by a minimum of 12 inches of water, and the locations have been analyzed for all varying degrees of interspersed moderation. The fuel rods/capsules stored in each rack are limited to 45% of the smallest estimate

of the initial number of pins with consideration given to the fissile composition, pellet diameter, and enrichment. When two or more types of rods/capsules are stored in the same location, the limit for the most restrictive type is used. The pool is further limited so that criticality is not possible even if all the SNM in the pool came together simultaneously.

The dry pit limit is currently 300 grams fissile for each of the 19 positions. Each position has a minimum edge-to-edge spacing of 12 inches. Fuel rods or assemblies are sealed in aluminum waste liner cans similar in design to DOT Specification 2R containers which exclude water in the "as-stored" situation.

There are no credible means for rearranging the tube spacing. If the pit were flooded accidentally, even with optimum moderation within the sealed liner units, the fully loaded facility would be safe since each unit would be isolated by 12 inches of water.

7.4.3 Operations

Criticality controls for the hot cells and for cold limit areas are based on the principle of limiting the quantity of fissile materials in terms of 45% of the minimum critical mass or number of units for each limit area. In determining the minimum critical mass or number of units (e.g., irradiated fuel rods), such factors as the form (e.g., fuel rod, powder, solution), enrichment, and composition of the special nuclear material and the available moderation and reflection are considered.

The limit area control is implemented by an electronic data processing system. An inventory log is kept for each limit area showing the quantity of fissile material in each unit transferred to and from the limit area.

Each cell is isolated by 36 inches of concrete and from its interlock by a 15-inch-thick steel door.

The analytical services provided to RML by the Radiochemistry Laboratory necessitate only small samples of special nuclear material in the lab area. Simple mass control therefore is established. Not more than 300 grams U-235 are permitted within the confines of the Radiochemistry Laboratory at any time. If mixtures of these fissile isotopes are present, the lowest limit applicable is the limit for the entire laboratory.

7.4.4 Waste Disposal

Upon completion of hot cell work, fuel-bearing materials and components are sealed in prenumbered waste liners and removed to one of the storage locations described above and in accordance with the criticality control limits of the storage location or transferred to the Hillside Storage location in a shielded cask.

Other forms of waste materials are handled in previously described Building 102 liquid waste and ventilation systems.

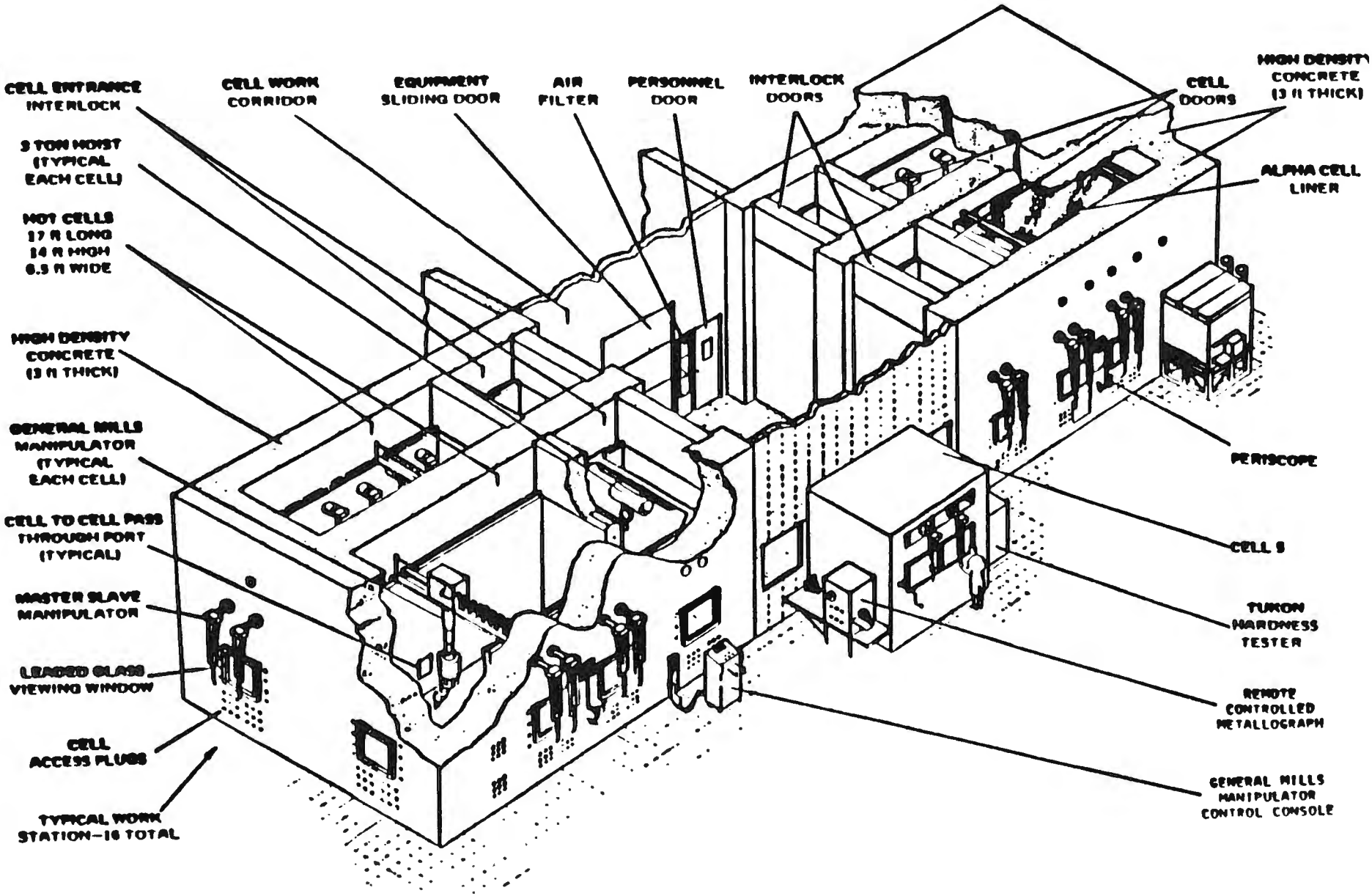


FIGURE 7.1. RADIOACTIVE MATERIALS LABORATORY HOT CELLS

10.0 CHEMISTRY, METALLURGY AND CERAMICS LABORATORY - BUILDING 103

10.1 LOCATION AND GENERAL DESCRIPTION

The Chemistry, Metallurgy and Ceramics Laboratory, Building 103, is a two-story building with a partial basement located directly across the access road from Building 102 (Figure 6.1). The building has a total floor area of approximately 22,000 square feet, including approximately 11,000 square feet of laboratory space. A corridor runs the length of the building on each floor. The laboratories are located on one side of the corridor and offices on the other as shown in Figures 10.1 and 10.2.

10.2 GENERAL PLANS AND USES OF MATERIAL

Special nuclear material is used in research and development activities including analytical, test specimen fabrication, maintenance, and calibration work and for sources of radiation necessary to support research and development programs. At the conclusion of experiments, radioactive materials usually are re-worked, reused, stored, transferred off site to persons authorized to receive, or discarded as wastes.

10.3 LABORATORY FACILITIES AND EQUIPMENT

10.3.1 General

Facilities located on the first floor include electron microscopes, x-ray equipment, microscopes, machine shop equipment of various kinds, ultrasonic equipment, electrochemical equipment, mounting presses, gas purification trains, hoods, and similar experimental equipment deemed necessary.

The chemistry areas on the second floor consist of typical chemical laboratories, a counting room, and an instrument room. Equipment in the chemistry laboratories includes the following: various types of spectrophotometers, fluorimeters, gas chromatographs and a plasma emission spectrometer; other miscellaneous laboratory equipment; lead caves and glove boxes; vacuum systems, including necessary instrumentation; hoods designed for handling radioactive materials; counting instrumentation; and mass spectrometers of various types.

A concrete storage vault for special nuclear material is provided on the ground floor. The vault has walls and ceiling of 8-inch minimum thickness, without penetrations, and a single locked door.

A waste storage building of corrugated steel and aluminum on a concrete pad is provided for temporary storage of packaged wastes and scrap materials which result from the licensed chemistry, metallurgy, and ceramics activities. They consist of paper, glassware, plates, rods, wire, samples, and other waste residue materials which result from such research, development, and analytical activities. These wastes normally will contain small amounts of by-product, source, and special nuclear materials. They are transferred to the site waste handling component (Building 102) for final packaging and disposal.

10.3.2 Ventilation System

Inlet air supply for Building 103 is provided by air conditioning units furnishing filtered and tempered outside air to the building. Air passes from the office areas through 2-inch fiberglass roughing filters in the laboratory door grills, thereby minimizing the passage of lint and dust into the laboratories. These filters also minimize backflow of potentially contaminated material in the unlikely event of complete exhaust system failure. Laboratory doors are equipped with automatic closers.

Air is withdrawn through the hoods and glove boxes passing first through individual high-efficiency filters at each hood or glove box. Only filters having a minimum efficiency of 99.97% for 0.3-micron-diameter homogeneous

particles of dioctyl sebacate, DOS, are used in the laboratory effluent ventilation system. From the individual filters, the air is conducted through a second filtration in one of two parallel banks of high-efficiency (99.95% for the system) filters. Thus filtered, it is discharged through a 48-foot-high, 5-foot-diameter stack. The high-efficiency filters are fabricated of fiberglass to provide high resistance to fire. Filter frames are metal or chemically impregnated for resistance to fire, and permanent duct work is metal or polyvinyl chloride. Each laboratory room used to conduct activities with radioactive materials is equipped with air sampling devices. However, airflow rates are adequate to permit routine operations with nuclear materials without the use of personnel respiratory protection. The main exhaust blower operates at approximately 36,000 cubic feet per minute. If complete ventilation failure occurs, an evacuation alarm is sounded automatically.

From 9 to 12 air changes per hour are provided for most laboratory rooms. However, in some rooms the airflow rate may be as high as 15 air changes per hour. Hood exhausts are dampered individually to maintain minimum face velocities on the order of 125 linear feet per minute across the openings. Glove boxes are operated at approximately -0.5 inches of water with respect to the room. Appropriate instrumentation indicating airflow and/or differential pressure is available.

The efficiency of the Building 103 ventilation filter system has been demonstrated by years of exhaust stack sampling data.

10.3.3 Radioactive Waste Facilities

Dry contaminated waste materials generated in Building 103 are packaged as indicated in Section 3.5.1. Waste containers are transferred directly to the site radioactive material storage facility or may be placed in a waste storage building approximately 11 feet by 12 feet constructed on a concrete pad adjacent to Building 103 (Figure 10.1). This building is of corrugated steel and aluminum construction. The waste storage building is conspicuously posted in accordance with 10CFR20.203. Waste packages stored in this area

are removed periodically to the site waste storage facility for delivery to a licensed waste disposal contractor.

Liquid contaminated wastes originating in Building 103 are routed from laboratory sinks and gravity drains through regulated pipes to waste retention tanks. Two tanks, each of approximately 5,000-gallon capacity, are provided. The tanks are equipped for representative sampling and for draining to drums or tank trucks. All tank wastes are sent to the site waste evaporator for concentration and solidification.

10.3.4 Criticality Alarm

Building 103 currently is monitored for a criticality accident by three detectors as described in Section 3.8. Approximate locations of the sensors are shown in Figures 10.1 and 10.2.

10.3.5 Fire Protection

Building 103 is provided with an automatic sprinkler system. Fire extinguishers also are located strategically throughout the laboratory areas. Accidental fires in the hood enclosures are of low potential. Three types of fuel are available for such accidents: general paper and plastic combustibles, flammable solvents, and pyrophoric materials. VNC fire prevention procedures minimize the fire potential; however, extinguishing equipment and materials are provided at strategic locations in the building. Additionally, smothering agents such as Metl-1-X, asbestos cloth, etc., are available in each enclosure where sufficient fire probability exists. Special precautions are taken when quantities of special nuclear materials are handled in hoods or glove boxes to minimize fire hazards in these enclosures. For example, metal containers will be used for pyrophoric materials.

10.4 BUILDING 103 PROCEDURES

In addition to the VNC general procedures described elsewhere herein, other procedures specific to the health and safety of experimental and supporting operations conducted in the chemistry, metallurgy and ceramics laboratories, Building 103, are employed. The essence of these procedures is as follows.

10.4.1 Survey and Contamination Control Procedures

Monitoring and step-off procedures are observed at points of transition from controlled areas to uncontrolled areas. In addition, hand-and-shoe monitoring stations are provided at the building exits. Surveys at the beginning of irradiated experiments assure the adequacy of local shielding.

10.4.2 Hood Station Procedures

Basic operations to be performed at hood stations normally involve gram or milligram quantities of uranium and TRU materials. Blending, sample and specimen preparation, and analytical operations are the most frequent activities utilizing such material. Standard procedures for these activities are used.

10.4.3 Ventilation System Maintenance

The filtered and monitored ventilation system for Building 103 is described in Section 10.3.2. The high degree of efficiency of this system is due in part to careful maintenance and operating procedures. The main filter bank in the system is dual so that the system can be run on one bank while the other bank is being changed. The duct velocities are low enough to allow the main flow control and balancing to be done at each of the primary filter box connections. The main system basically runs as a large manifold of relatively constant suction so that primary connections can be made to the system as needed allowing the overall blower capacity of the main system to be maintained. Damper locations are restricted to dampers at the exit of

each primary absolute filter to control the flow of that primary branch and dampers that normally are open at the main filter bank to permit running the system on one bank while the filters in the other bank are changed.

All filter boxes are provided with differential pressure gages to measure pressure drop across the filters. The flows, correlated with the pressure changes, give a good indication of the state of the filters. Any filters found to be below standard are changed. Filter boxes are surveyed by Radiation Safety for contamination prior to changeout. This survey on the second bank of filters also is a measure of the effectiveness of the first filters.

Exhaust air from the Building 103 laboratories is sampled continuously. The filtered air that leaves the building is exhausted up a 48-foot stack which extends 20 feet above the building. The exhaust air is sampled continuously for iodine and particulate activity at a point 24 feet below the top. The particulates are collected on a high-efficiency, 47-mm membrane filter; and iodine is collected on a charcoal cartridge. The filter and cartridge are evaluated on a weekly basis.

10.4.4 Waste Disposal

Normally, dry contaminated wastes occur in the form of paper towels, glassware, and similar nonsalvageable laboratory apparatus as well as small quantities of uranium as, for example, test specimens and analytical sample residues. These materials are packaged, sealed, recorded as to quantity and type of nuclear material, placed in the designated waste storage building (Figure 10.1), and transferred to the site radioactive materials storage facility until final disposal from the site. Storage of dry wastes in Building 103 is limited to packages having surface dose rates of 100 mR per hour or less.

The two 5,000-gallon-capacity radioactive liquid waste tanks are monitored routinely and, when approximately filled, are sampled and their contents routed to the site waste evaporator.

10.5 BUILDING 103 CRITICALITY CONTROL

The activities and work programs conducted in Building 103 are experimental and analytical rather than industrial. Experimental operations rarely require large quantities of fissile special nuclear materials at a time but do require a wide variety of forms, compounds, and concentrations.

Criticality control of these materials is relatively simple for it is based on the issuance of a safe quantity by a materials custodian for use in any single laboratory room at any one time for the majority of the activities conducted. Stock supplies of special nuclear materials normally are stored in a vault when not in laboratory use. Directions for storage arrays in the vault limit the amount and spacing of the material to safe systems assuming optimum moderation and reflection. Full or partial flooding of the vault is unlikely; however, since there is no water service in the vault and, except for the steel vault door and electric lighting in the ceiling, there are no other penetrations.

The storage vault is isolated by concrete walls, ceiling and floor at least 8 inches thick. It is situated between a rest room and a lunch room. Areas immediately above the vault on the second floor do not contain special nuclear material. No special nuclear materials are permitted in any of the adjacent rooms. Within the storage vault, special nuclear materials are stored in specified locations. Figure 10.3 shows the current configuration.

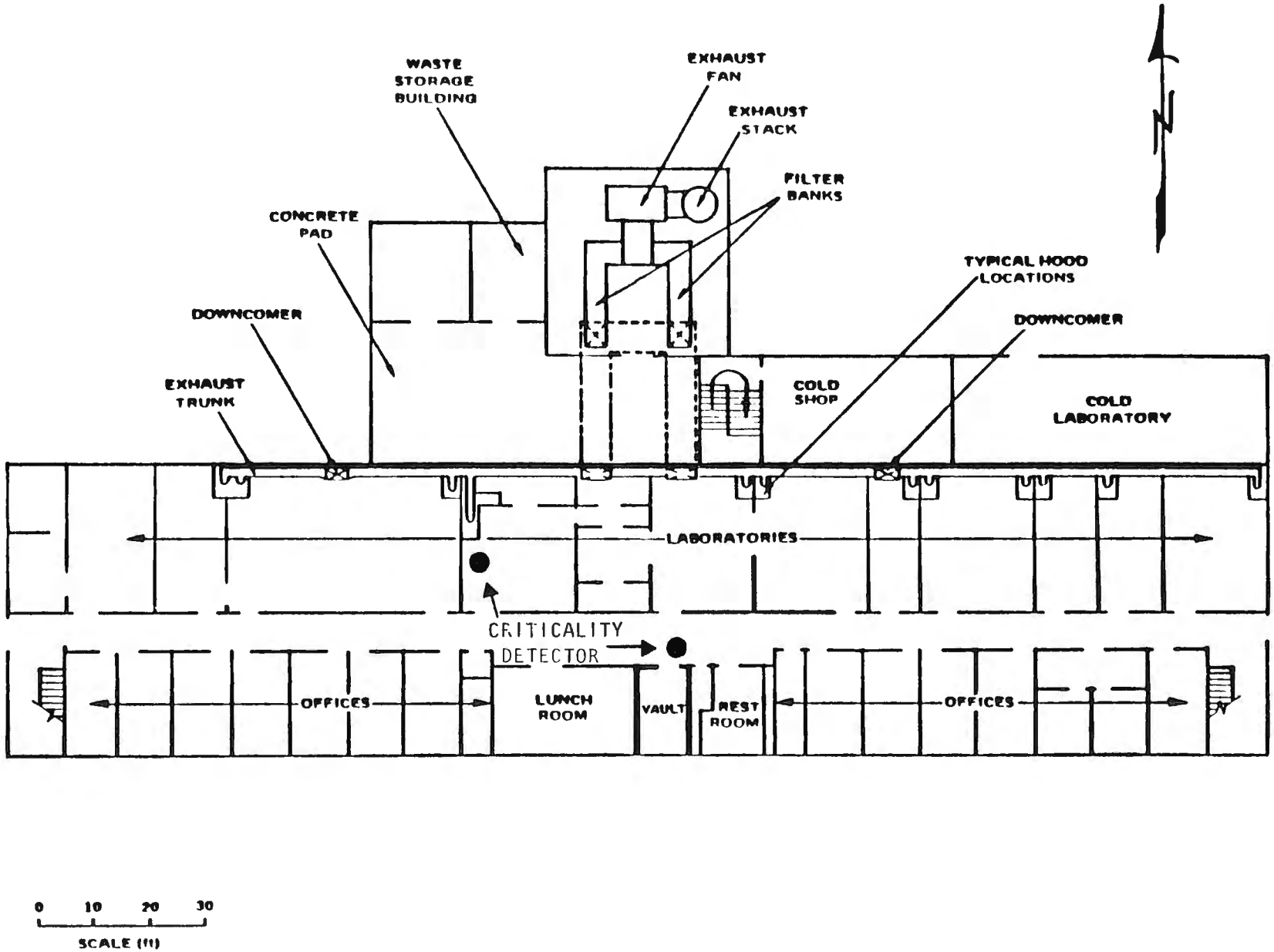


Figure 10.1. Building 103 Floor Plan (Ground Level)

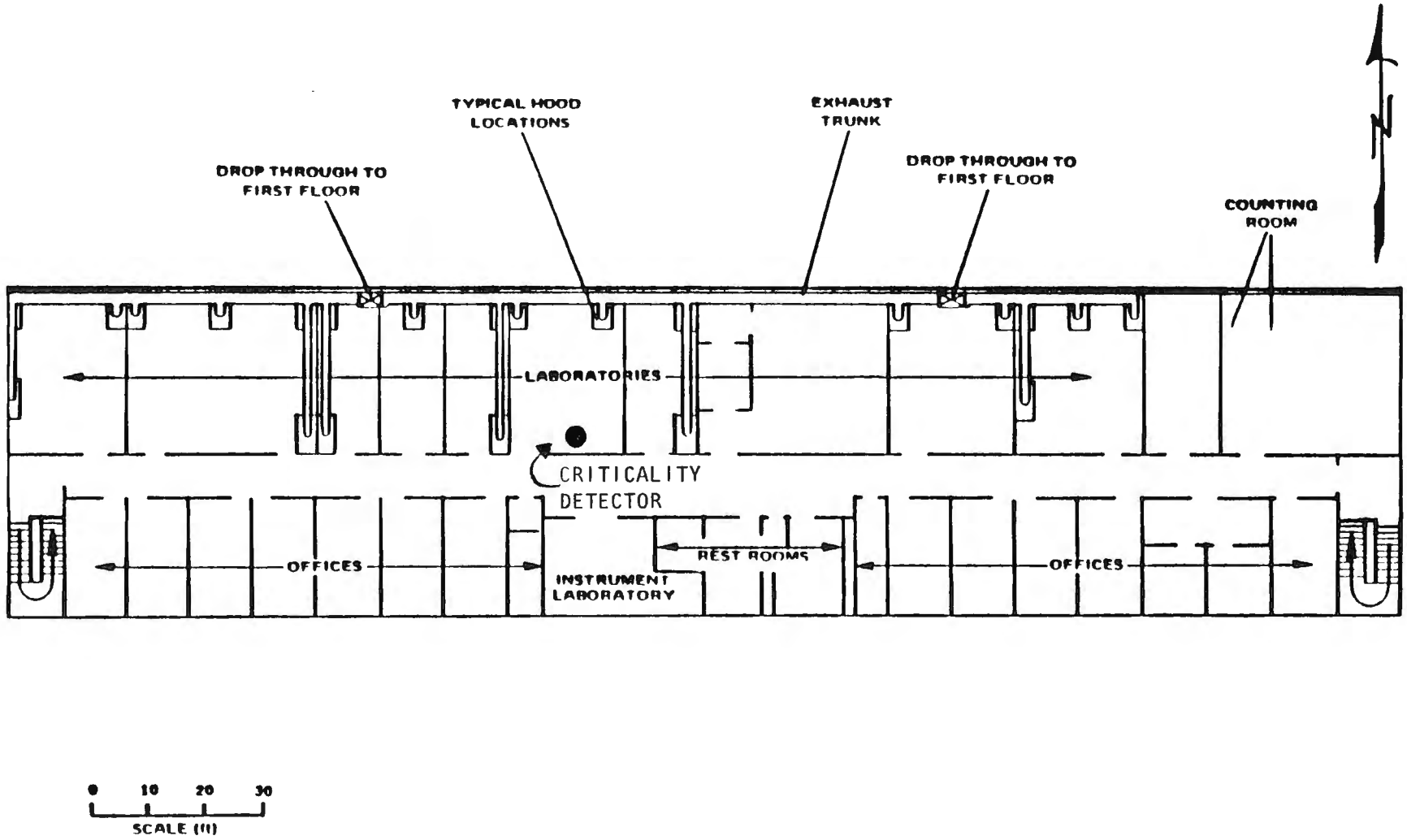


Figure 10.2. Building 103 Floor Plan (Second Floor)

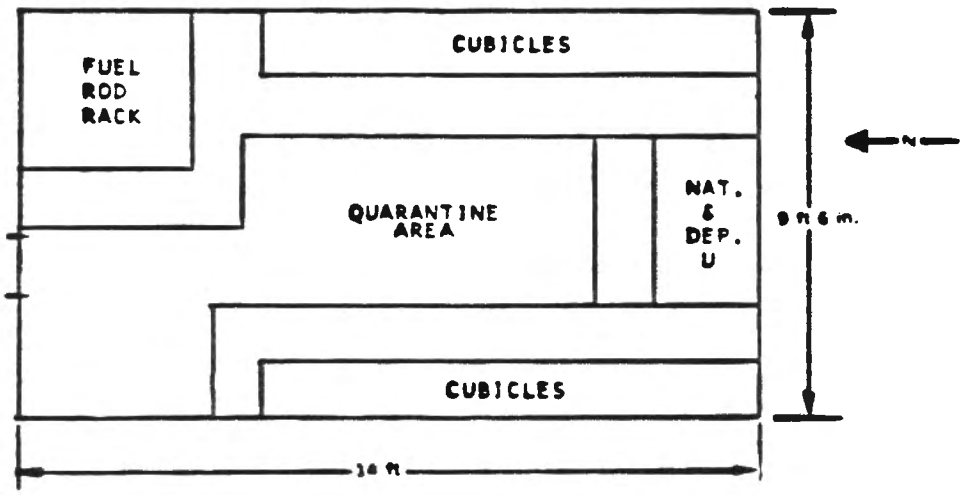


FIGURE 10.3. BUILDING 103 VAULT LAYOUT

12.0 BUILDING 105

12.1 LOCATION AND GENERAL DESCRIPTION

Building 105 is located immediately north of Building 102 in the 100 Area (Figure 6.1). The building houses the Nuclear Test Reactor (NTR) and the Advanced Nuclear Applications Laboratory (Figure 12.1).

12.2 NTR FACILITIES

The Nuclear Test Reactor (NTR) is licensed pursuant to 10CFR50 (License R-33). SNM licensed pursuant to License SNM-960 is taken to the NTR North Room or the NTR South Cell (Figure 12.1) primarily for purposes of neutrographic examination. Such material is in the form of sealed units, and only one safe batch (<45% of the minimum critical accumulation) is permitted in the North Room or South Cell at any one time. The materials are handled pursuant to the safety provisions and requirements of License R-33.

12.3 OTHER LABORATORY AREAS (ADVANCED NUCLEAR APPLICATIONS)

Other laboratory areas in Building 105 use SNM only as sealed sources, standards, foils, or as electronic components under general license.

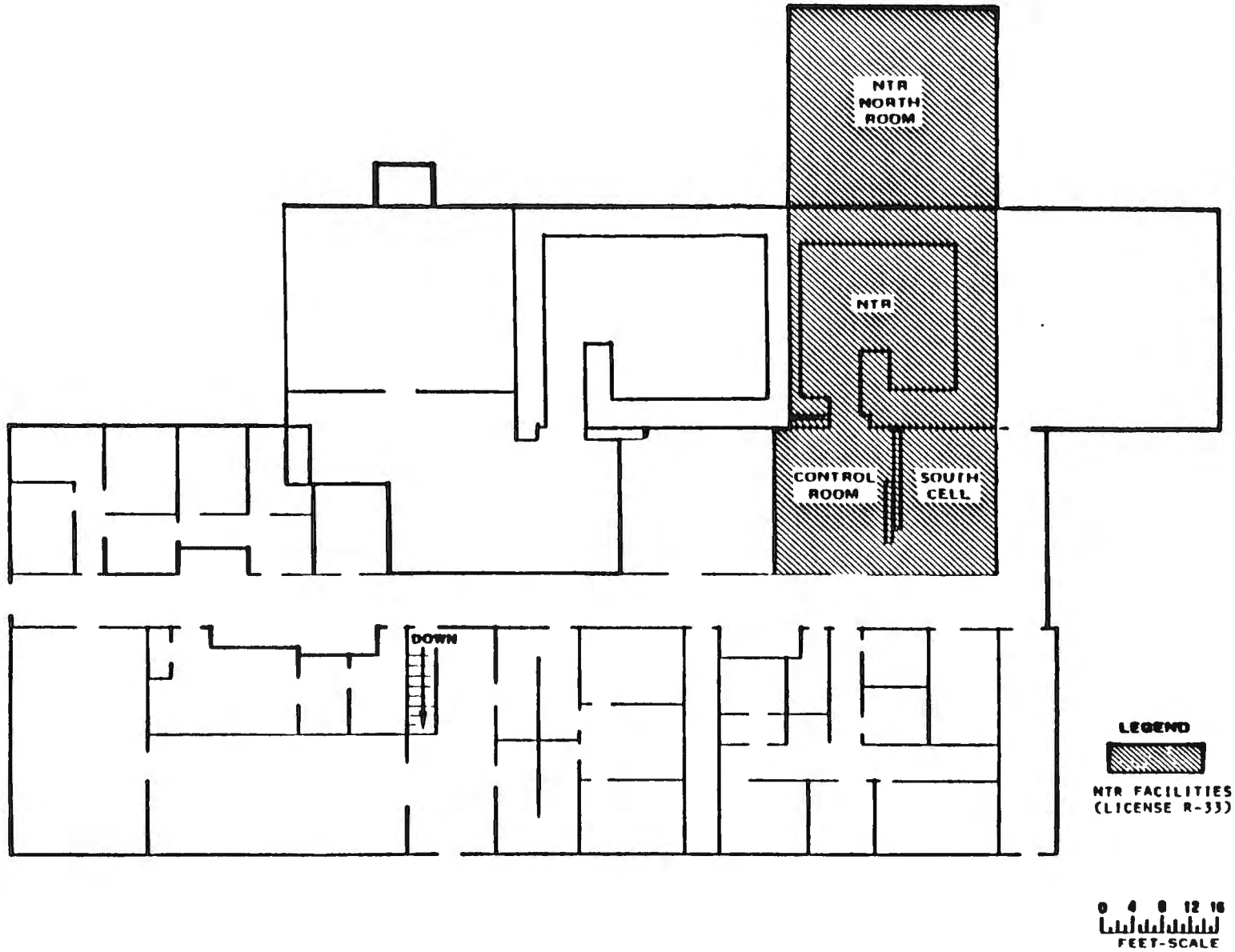


Figure 12.1. Building 105 Floor Plan

13.0 WASTE HANDLING

13.1 RADIOACTIVE LIQUID WASTE EVAPORATOR PLANT (BUILDING 349)

Activities described in this section involve the transfer of liquid radioactive wastes including those containing small quantities of special nuclear material generated at various facility and laboratory installations at the Vallecitos Nuclear Center site to the Radioactive Liquid Waste Evaporator Plant, Building 349; the concentration of these wastes in Building 349; the discharge of processed effluents; and the transfer of concentrated waste materials to a licensed waste disposal contractor. The estimated annual plant throughput is 100,000 gallons of liquid waste.

Each facility operated under License SNM-960 which generates liquid radioactive waste (i.e., Buildings 102 and 103) are equipped with liquid radioactive waste retention tanks. All lines in each building which potentially could carry radioactive materials are connected to the retention tanks. The tanks are emptied as necessary.

Prior to transferring liquid wastes to the Waste Evaporator Plant, samples are taken for gross alpha and beta, total uranium and plutonium, and U-235 analyses. During the first half of 1988, uranium levels ranged from <0.02 to 24.0 ppm with an average of 5.7 ppm. A log of the total quantity of U-235 in the plant at any one time is maintained. The entries to this log are based on the analyses of samples of the liquid waste, and the U-235 inventory has been typically between 50 to 70 grams.

The Waste Evaporator Plant is not a production or utilization facility within the meaning of 10 CFR Part 50, Section 50.2(a)(3) ["a facility designed or used for the processing of irradiated materials containing special nuclear materials..."]. None of the components of the plant make possible the separation or purification of isotopes from each other.

13.1.1 Location and General Description

The Waste Evaporator Plant is housed in a metal building constructed on a poured concrete pad with integral concrete shielding walls around the processing vessels at high activity points. The shielding is designed to provide radiation levels in operating areas from normal wastes to less than 5 mR/h.

The building entrance opens into a change room and operations control room. General radiation levels in this area normally are less than 2 mR/h. Figure 13.1 shows the floor plan for the building.

The Waste Evaporator Plant is located in an area to the northwest of the deactivated VBWR facility (Figure 1.2).

13.1.2 Liquid Waste Processing

Liquid wastes are collected from the various site accumulation tanks and transported to the Waste Evaporator Plant by fork lift truck in a specially designed 1,500-gallon stainless steel tank. The waste transfer tank is equipped with internal vertical baffles to prevent cyclic shifting of the liquid material during transport. All external protuberances such as pipes, valves, gages, etc., are arranged or guarded in such a manner that they cannot come in contact with other vehicles or objects upon the roadway yet are readily accessible for manual operation for loading and unloading. Experience in transporting liquid wastes in this way shows dose rates to the fork lift truck operator generally have been less than 5 mR/h.

In the direct evaporation process, the liquids are pumped continuously from the feed tank or chemical treatment tanks directly into the evaporator. Here, the wastes are concentrated through a vertical-tube, natural convection evaporator. The vapor is treated in associated equipment, including a high-efficiency demister entrainment separator and a condenser. Effluent waters from the evaporator are collected in the monitoring tanks for analysis

and disposal by evaporation. If further decontamination is necessary, the water can be re-routed to the feed storage tank for reprocessing or ion exchange treatment.

13.1.3 Concrete Mixing Station

Liquid waste concentrates from the evaporator are collected in a receiver and discharged into DOT Specification 17-H 55-gallon drums or other approved containers. The concentrates then are mixed with a cement-diatomaceous earth mixture or equivalent for solidification. After solidification, the drums are sealed and prepared for disposal as dry solid waste. The drums as loaded will provide compliance with appropriate DOT regulations. Solidified waste contained in drums is monitored; tagged with the radionuclides present, curie content, and radiation level; and stored awaiting removal from the Vallecitos site by a licensed waste disposal contractor or other approved methods. Waste drums are not buried at the VNC site.

Small amounts of liquid wastes incompatible with evaporation may be solidified directly in 55-gallon drums.

Storage of dry wastes at the Evaporator Plant is limited to packages having surface dose rates of 100 mRem per hour or less. Waste drums having radiation readings in excess of 100 mRem per hour are transferred promptly to the high-level waste storage area. The building waste storage area is classified as a Radiation Area and so posted.

13.1.4 Ventilation

Effluent air from equipment and locations such as tank vents, hood vents, and sample points not requiring pre-filtering is exhausted directly to the main building filter exhaust system. Effluent air from the vented side of the condenser along with other points of suspected higher activity is manifolded into pre-filters before being exhausted into the main building exhaust ducts.

The main exhaust then is directed through a HEPA filter system (99.95% efficient for 0.3-μm particulate). Filters appropriate for high relative humidity service are used.

The filtered air then is discharged to the atmosphere through a continuously sampled stack at a point approximately 7 feet above the roof of the building. Approximately 18 air changes per hour maintain proper contamination control. Air discharged from the stack at the rate of 3,000 cfm is driven by an electrically powered blower mounted at the base of the stack.

13.1.5 Monitoring Procedures

Monitoring and/or step-off procedures are observed at points where each potentially contaminated regulated area exits into the clean or nonregulated areas. In addition, survey instruments are provided at convenient locations for final surveying. The doors to the process equipment area are alarmed or padlocked in accordance with 10CFR20.203(c)(2).

13.1.6 Criticality Control

The facility meets the requirements of a subcriticality area as defined in Section 3.16 of Appendix A to License SNM-960.

13.2 SOLID WASTE HANDLING

A single organization has been designated by VNC management as responsible to assure that all solid wastes leaving the site meet the appropriate regulatory requirements. For the purposes of this application, it will be referred to as the Waste Handling Function (WHF).

13.2.1 Solid Waste Accumulation

Solid wastes are accumulated at each location where radioactive materials are handled. The majority of wastes fits the Low Specific Activity (LSA) category as defined in the Department of Transportation regulations.

For each waste accumulation, the generating component is responsible for maintaining a listing of all material in the accumulation. Each accumulation with its listing is forwarded to the WHF for final inspection and/or repackaging.

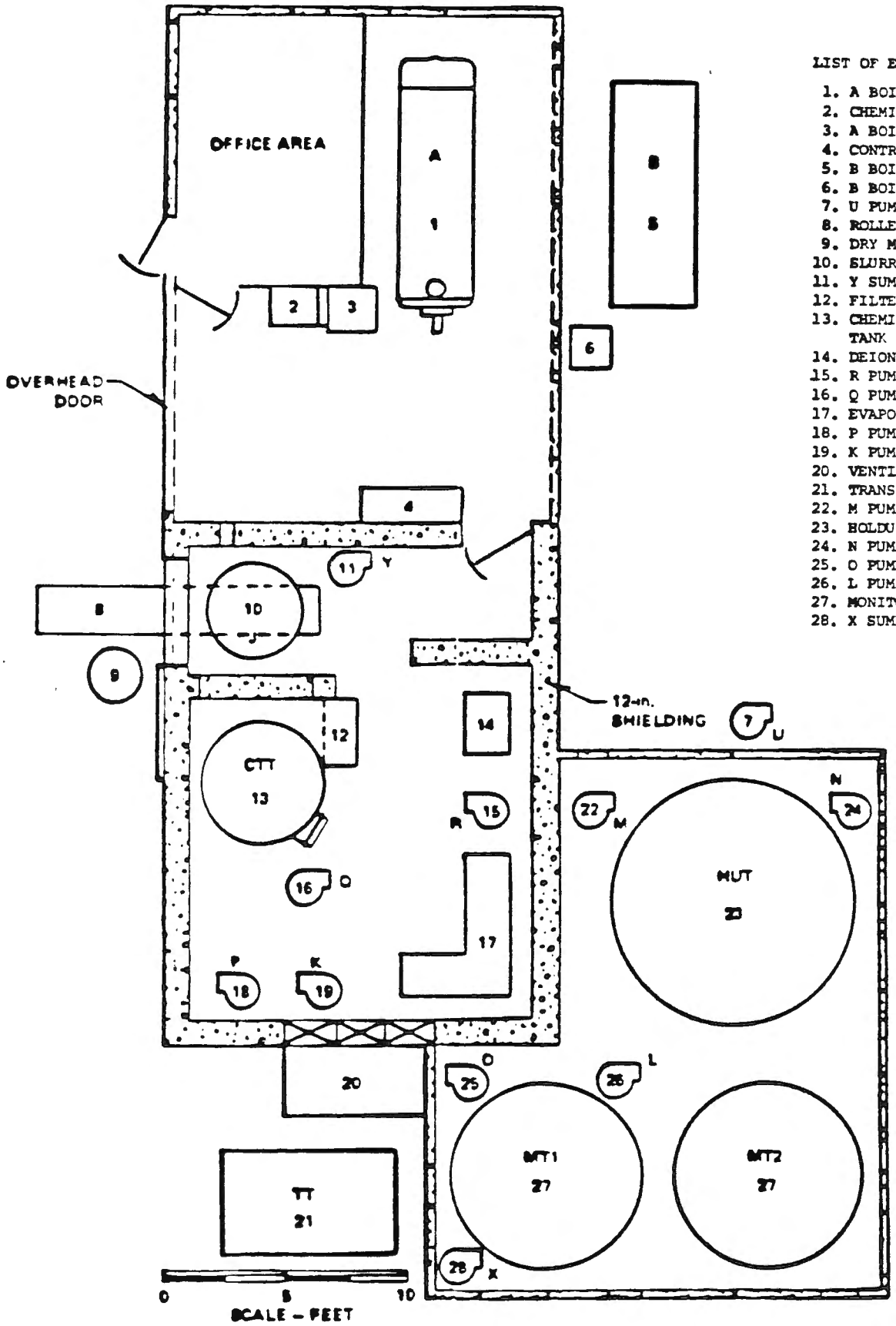
The WHF is responsible for implementing waste volume reduction methods, as appropriate.

13.2.2 Solid Waste Storage

Solid waste materials are stored in the site radioactive materials storage facility described in Section 3.5.1 and shown on Figure 13.2.

Waste materials that include or are associated with significant quantities of special nuclear materials are placed in containers called waste liners. Waste liners are stored in the horizontal tube facility bunker. Fifty-five-gallon drums and boxes containing lower level wastes are stored in a covered facility.

Limits on the maximum quantity of special nuclear material that may be loaded into any containers for purposes of waste storage have been established by Nuclear Safety. For 55-gallon drum storage, calculations were made assuming optimum water moderation and spherical geometry for the individual masses within each drum. No credit was taken for neutron absorption by the materials between the individual units. On this basis, limits were established which provide criticality safety for an essentially infinite array of these drums. This limit is set in this fashion since it is possible to store drums above grade in almost any geometry. However, normal storage arrangement would be a planar array one drum high.



LIST OF EQUIPMENT

1. A BOILER
2. CHEMICAL FEEDER
3. A BOILER FEED
4. CONTROL PANEL
5. B BOILER
6. B BOILER HOT WELL
7. U PUMP
8. ROLLER CONVEYOR
9. DRY MIXER
10. SLURRY TANK (J)
11. Y SUMP PUMP
12. FILTER
13. CHEMICAL TREATMENT TANK (CTT)
14. DEIONIZERS
15. R PUMP
16. Q PUMP
17. EVAPORATOR SYSTEM
18. P PUMP
19. K PUMP
20. VENTILATION SYSTEM
21. TRANSPORT TANK (TT)
22. M PUMP
23. HOLDUP TANK (HUT)
24. N PUMP
25. O PUMP
26. L PUMP
27. MONITOR TANKS
28. X SUMP PUMP

FIGURE 13.1. WASTE TREATMENT PLANT

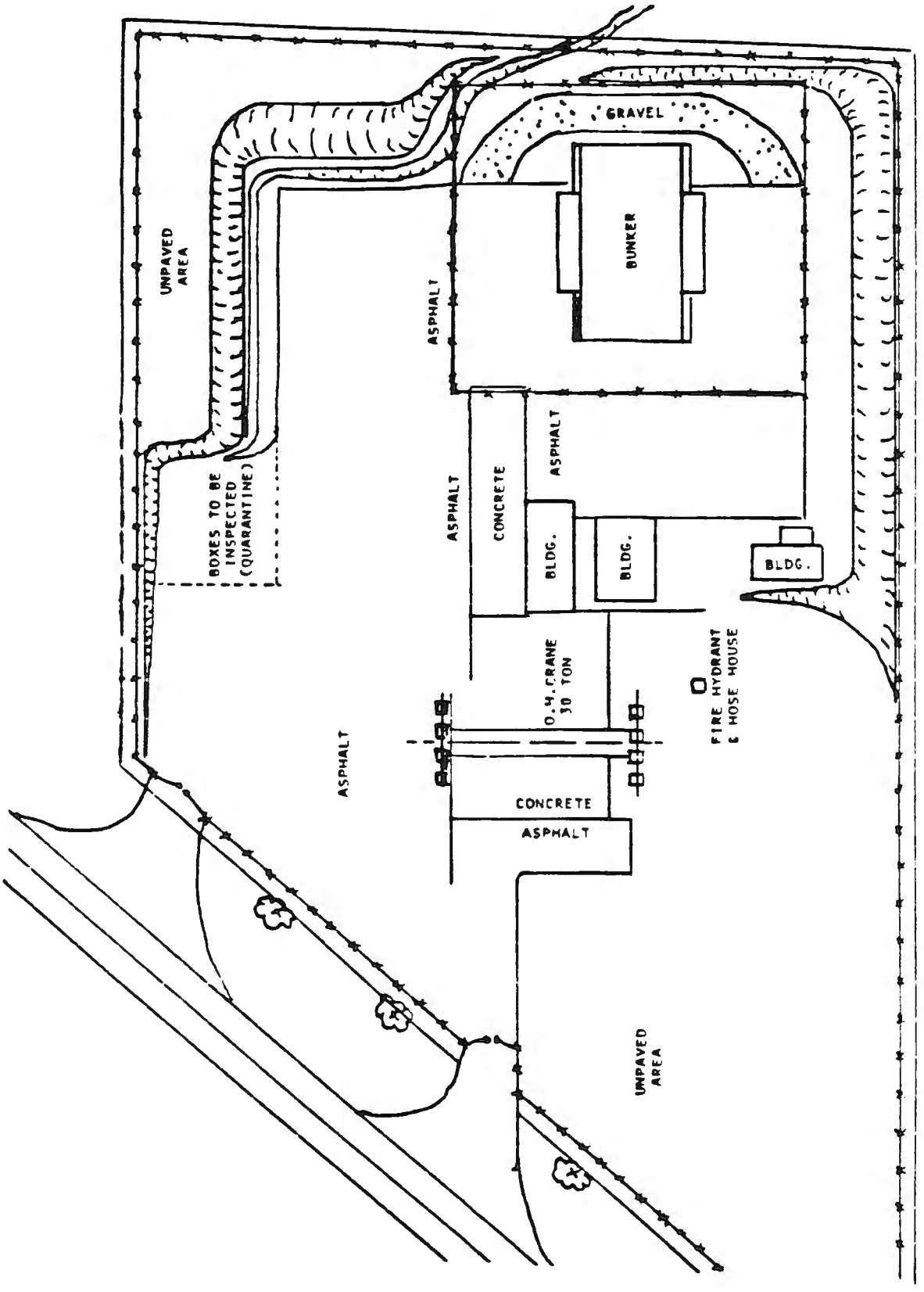


Figure 13.2. Hillside Waste Storage