HITACHI

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attn: Document Control Desk

September 30th, 2010

Subject: Renewal of Materials License SNM-960

Attachment: License Application

Reference: 1) Renewal Letter, Feb 25, 2010 from GEH (Turner) to NRC
2) Non-acceptance letter, Apr 16, 2010 NRC (Dorman) to GEH (Turner)

GE Hitachi (GEH) hereby makes an application pursuant to 10 CFR 70.33 to renew Special Nuclear Materials License SNM-960 for a term of 10 years. The current license expired June 30, 2010. However, based on a previous license application (References 1 & 2) GEH is considered in 'timely renewal.' On the basis of this application GEH further requests that the authorized activities in the existing license continue in effect until the application has been finally determined as provided in 10 CFR 2.109 (a).

This submission revises and supersedes the application submitted in Reference 1, addressing the concerns enumerated in the third paragraph of Reference 2.

Please contact me on (925) 862-4344 if you have any questions or would like to discuss this matter further.

Sincerely,

Digitally signed by
David W. Turner
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David W. Turner
Manager, Vallecitos Nuclear Center

CHAPTER 1.0
GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

The primary purpose of the GE Hitachi (GEH) Vallecitos Nuclear Center (VNC) facility in Alameda County, California is the measurement, engineering evaluation and support of light water reactor fuels, fuel materials, components and inspection tools. The SNM-960 authorized activities are co-located with the licensed activities for DPR-1 Vallecitos Boiling Water Reactor [VBWR], TR-1 GE Test Reactor [GETR], DR-10 Empire State Atomic Development Agency Vallecitos Experimental Superheat Reactor [EVESR] (all in safe store), and R-33 Nuclear Test Reactor [NTR]. Also conducted on site are by-product activities covered under the State of California license CA-0017-01 including sealed source manufacture and research and development. The safety, environmental, quality assurance and emergency preparedness aspects of the research and development, measurement and evaluation operations are managed and controlled as described in this license.

1.1.1 SITE DESCRIPTION AND LOCATION

VNC 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 area map, Figure 1.1. The properties surrounding the site are primarily used for agriculture and cattle raising, with some residences, which are mostly to the west of the property. The nearest sizeable towns are Pleasanton, with a population of approximately 67,000 located 4.1 miles to the north-northwest and Livermore with a population of approxiamtely 81,000 located 6.2 miles to the northeast. A United States Veterans Administration hospital with a population of approximately 500 is located about 4 miles to the east.

The site is on the north side of Vallecitos Road (State Route 84), which is a two and four-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 light industrial activity within a 10-mile radius of the plant. San Jose (20 miles south), Oakland (30 miles northwest) and San Francisco (35 miles northwest) are major industrial centers.

The site is located in the Livermore Upland physiograpic area. The majority of the site is undeveloped with hills ranging in elevation from the 1,000 to 1,300 feet above mean sea level. Approximately 135 acres located in the southwest corner of the property situated between the 400 and 600-foot topographic contours are developed.

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 or 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.

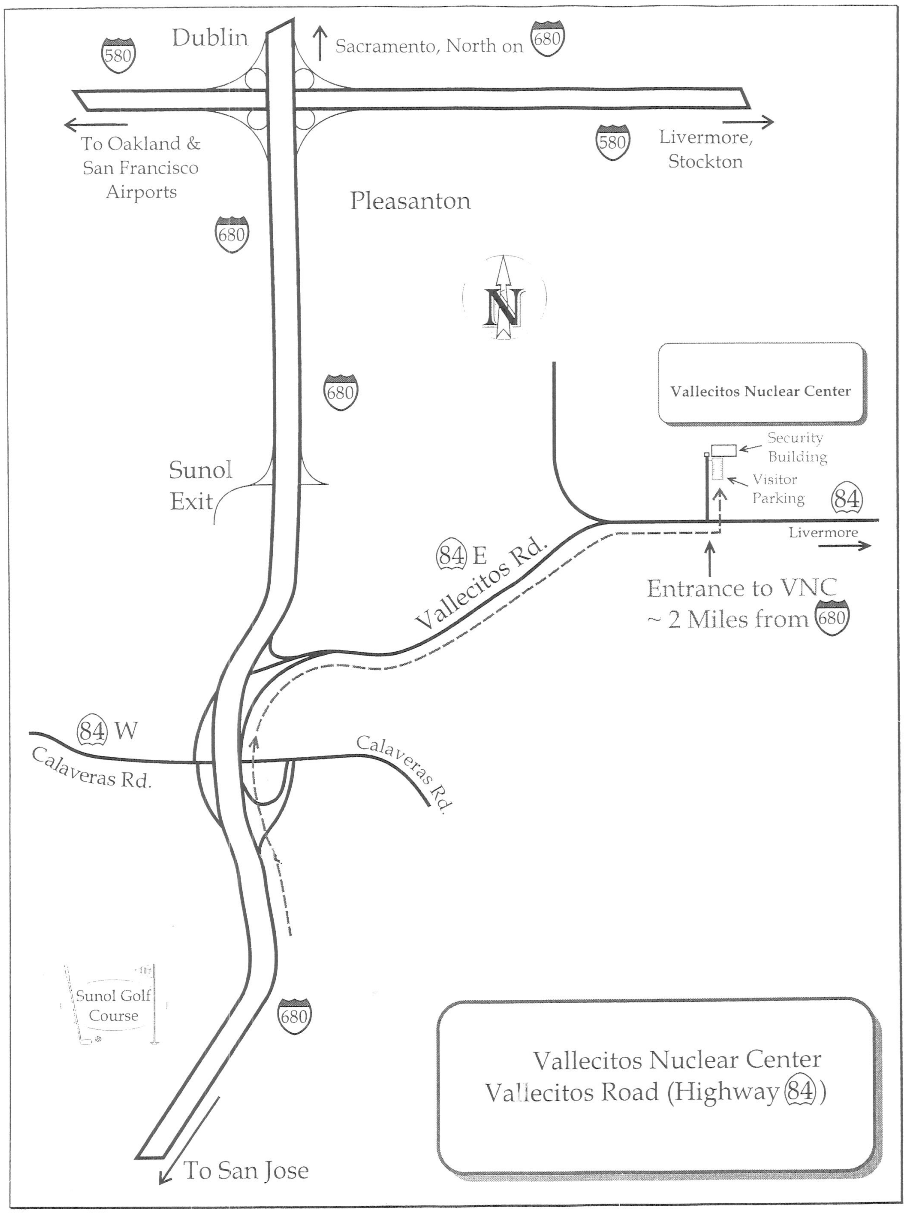
Electrical power is supplied by Pacific Gas and Electric to the main laboratory substation from whence it is distributed to each building on the site.

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

Industrial wastewater is processed in a system of surface impoundments and used as irrigation water for grasslands on site.

FIGURE 1.1

Area Map



1.1.2 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 current 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.

1.1.2.1 Radioactive Materials Laboratory

The Radioactive Materials Laboratory (RML) is located in Building 102. This laboratory is a shielded (hot cell) facility equipped with remote manipulators to conduct experiments and analyses with irradiated reactor fuels and other radioactive materials. The facility also includes the Building 102 storage pool and dry pit storage. The RML is the principal location for operations conducted under this license. Addition description of the RML is provided in section 1.1.5.

1.1.2.2 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. The hot cells in the radiochemistry laboratory are not as heavily shielded as the RML hot cells and typically only small quantities of irradiated fuel or hardware are examined in this area.

1.1.2.3 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 offices and laboratories, variously equipped with laboratory apparatus designed to handle quantities of radioactive materials that are typically less than 1gm U235 and are limited to less than a critical mass. The functions served by this facility are research, development, and analytical chemistry services.

1.1.2.4 Building 104

Building 104 is used for warehousing and training.

1.1.2.5 Building 105

Just north of Building 102 is Building 105. The principal facilities located in this building are an experimental Nuclear Test Reactor (NRC License R-33) and laboratories leased to another company. The laboratories leased to another company are primarily engaged in trace element analysis. The laboratories in Building 105 (operated by the contractor) use only minute quantities (less than accountable amounts) of special nuclear materials under this license.

1.1.2.6 Building 106

Building 106 contains Engineering Shop Operations which includes various maintenance shops (e.g., machine, carpentry, electric, maintenance support), and instrument calibration facilities. Calibration sources associated with these activities are maintained in Building 106.

1.1.2.7 Building 107

Building 107 is the Hazardous Waste and Chemical Storage Building. No radiactive materials or SNM-960 licensed activites are allowed in this building. Secondary containment is provided in case of spills.

1.1.2.8 Solid Radioactive Material and Waste Storage Facility

Solid radioactive wastes and material samples generated at the various laboratory and facility locations are stored in the Hillside Storage Facility located approximately midway between the deactivated Vallecitos Boiling Water Reactor (VBWR) and General Electric Test Reactor (GETR) areas. The hillside storage area includes shielded horizontal tubes for storing liners containing samples and waste materials. The GETR storage area primarily consists of material stored in drums or shipping containers prior to shipment.

1.1.2.9 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.

1.1.2.10 Reactors and Auxiliary Facilities

The ESADA-Vallecitos Experimental Superheat Reactor (EVESR), the VBWR, and the GETR are deactivated. The NTR is operational and is engaged primarily in neutron radiography under a separate NRC license.

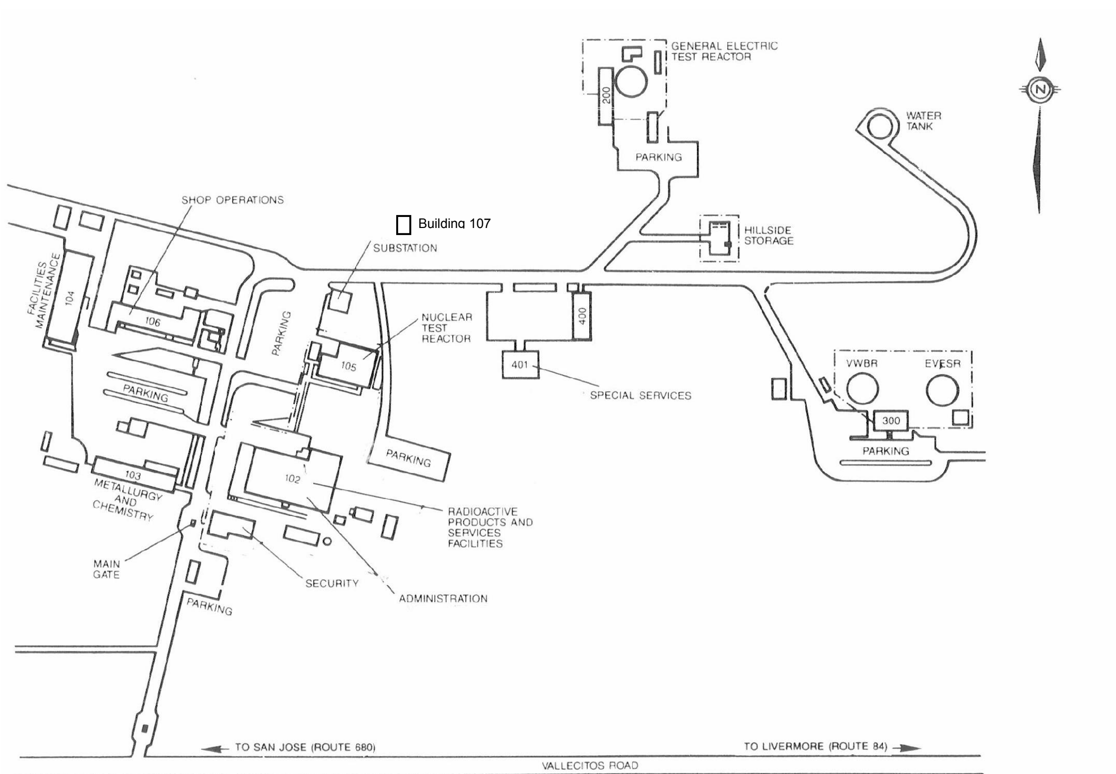
1.1.2.11 Building 300

This building is used for access to the shutdown reactors VBWR and EVESR. No activities associatied with SNM-960 are performed in building 300.

1.1.2.12 400 Area

The 400 Area consists of two buildings, 400 and 401. Building 401 is devoted chiefly to offices, and non-radioactive materials laboratories. The laboratories in Building 401 (operated by the contractor) use only minute quantities (less than accountable amounts) of special nuclear materials, which are maintained by GEH under this license.

FIGURE 1.2
Vallecitos Nuclear Center



SITE PLAN (Typical)

1.1.3 FACILITY RESISTANCE TO ENVIRONMENTAL EVENTS

The seasonal rainfall patterns and the presence of geological faults in the East Bay area constitute the primary environmental threats. Surface water on the property drains via drainage ditches to an unnamed tributary of the Vallecitos Creek. The property is not located within the 100 year flood zone.

Typically, no rain falls in the area from mid spring to fall. Starting in mid-May the grasses in the chaparral hills becomes dormant and dry, leading to a significant wildfire risk. Typically, the wildfires are fast moving and consume only the grasses in the fire zone. As this is a regional risk, fire departments and other public protection agencies have the experience and equipment necessary to effectively fight these grass fires. VNC has instituted several practices to minimize the risk of grassfires. 1) Grazing cattle until the grasses become dormant – minimizing the amount of fuel (grass) present 2) plowing firebreaks through the grasses along roadways and between sections of the property to increase the difficulty of fires to spread if started 3) minimizing the use of vehicles in the rural portions of the property to prevent starting fires due to vehicle operations and 4) training the on site fire brigade in the combat of grass fires as well as incipient stage building fires.

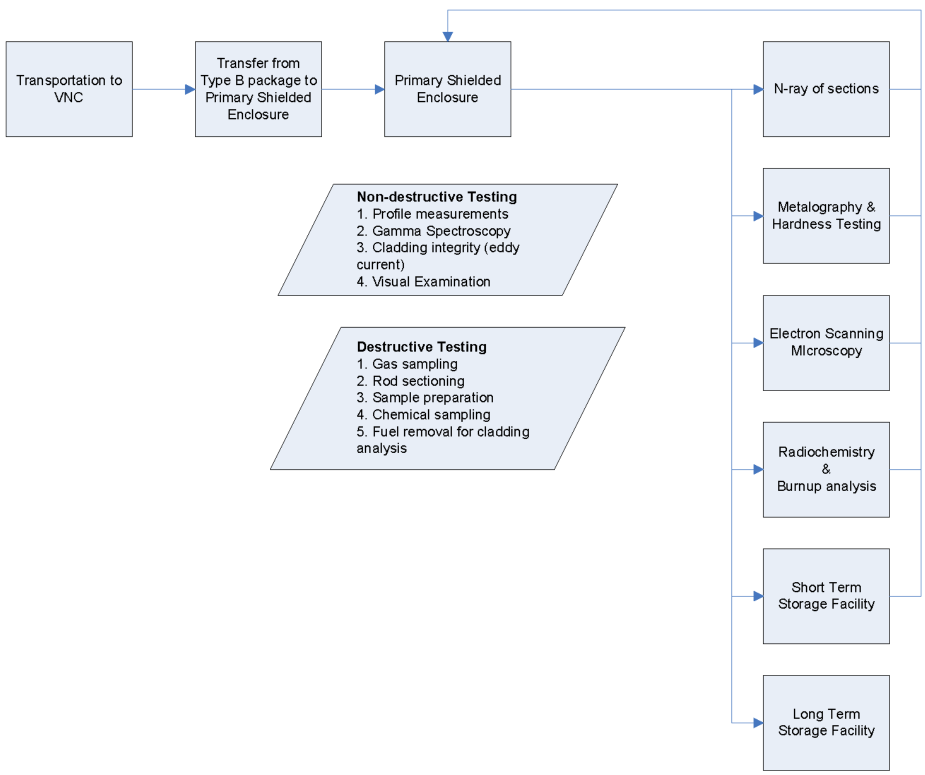
Earthquakes are an occasional occurrence in all parts of the Bay Area. VNC contains one fault through the property, which is a branch off of the Hayward fault. Typically, measurable earthquakes (specialized sensitive instruments are needed to measure most of the earthquakes) occur nearly daily in the area. Rarely are they of sufficient magnitude to be noticed without precision instruments. Area building codes account for this seismic activity and the buildings and structures at VNC were built to the codes in effect at the time of construction.

1.1.4 PROCESS DESCRIPTION

The operations authorized by this license consist of receiving small quantities of irradiated and unirradiated special nuclear material and performing a variety of tests (mechanical, nuclear and chemical) on the material for purposes of improving the design of nuclear fuel and the functioning of utilization facilities. These tests may also be performed as basic research. Most of the analysis centers around Boiling Water Reactor fuel and cladding, with a minor amount of focused on other fuel types. This analysis may focus on a leaking rod, a new fuel design, increased burn-up, or other inquiries. A single campaign typically contains several rods, some of which are of primary concern and others analyzed as a reference or control standard.

A typical analysis campaign will take between 12 to 18 months. Figure 1.3 provides a description typical for one of these campaigns, but may include other steps or omit some of these items.

FIGURE 1.3 - Typical Spent Nuclear Fuel Examinations



1.1.5 FACILITIES AND EQUIPMENT FOR HANDLING SNM

Routine handling of SNM occurs in specially designed shielded enclosures (also referred to as hot cells) with additional activities in various laboratories, more lightly shielded enclosures, storage locations and radiography facilities.

The radioactive material laboratory (RML) and hot cells are located in Building 102, which is a single-story facility with a basement, concrete and steel structure. 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, shielded hot cells, and other equipment as described in this section.

The current layout of the 100 Area and Surroundings is shown in figure 1.4.

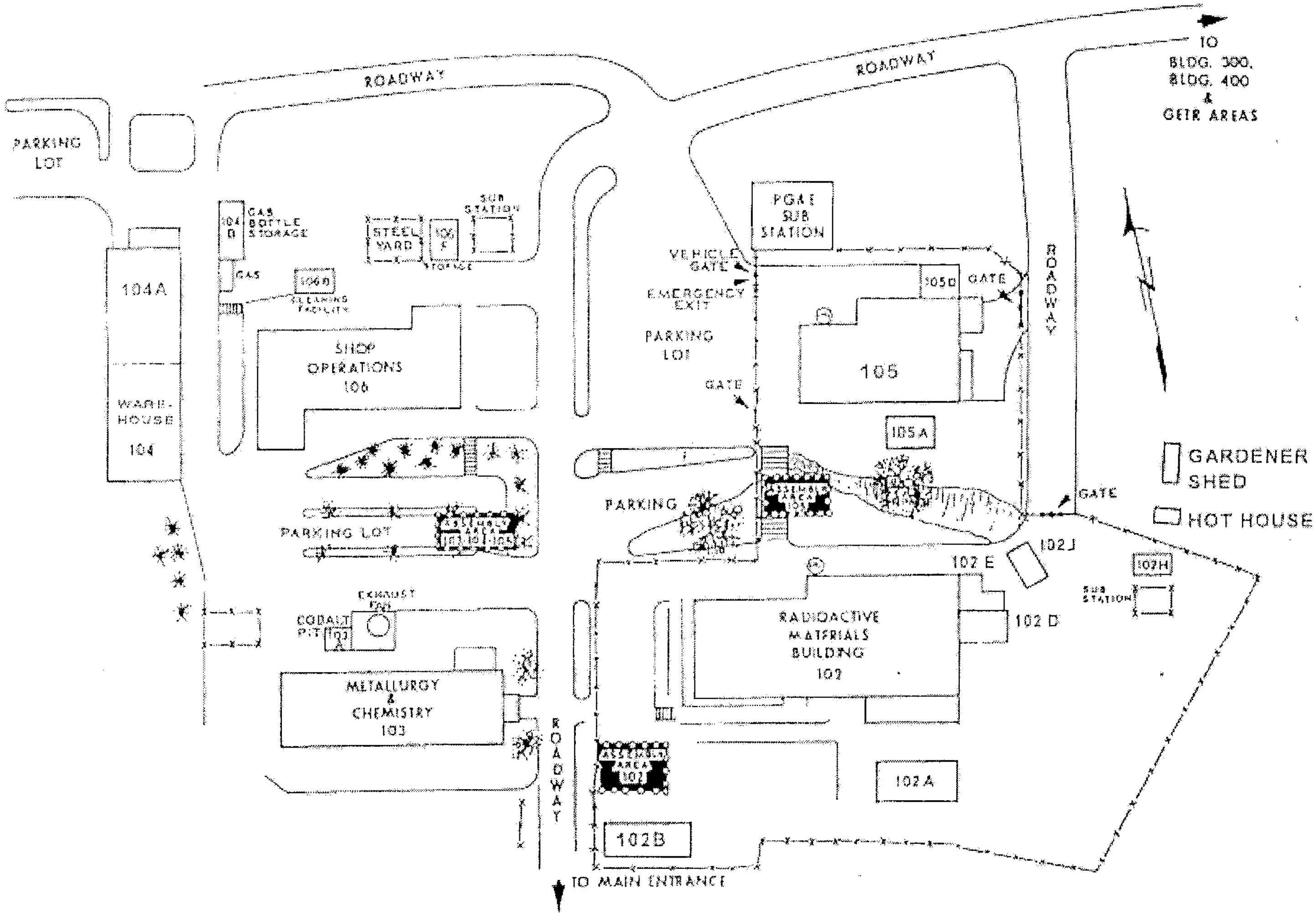


Figure 1.4 100 Area and Surroundings (Typical)

1.1.5.1 Laboratory Building 102

1.1.5.2 Location and general description

Building 102 is a single-story with basement, concrete and steel structure located as shown in Figure 1.5. 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 laboratory areas of the main floor are separated from general office areas by a firewall. Inherent in the building design are general service facilities such as the main ventilation system, decontamination rooms, and other areas that are described later in this section.

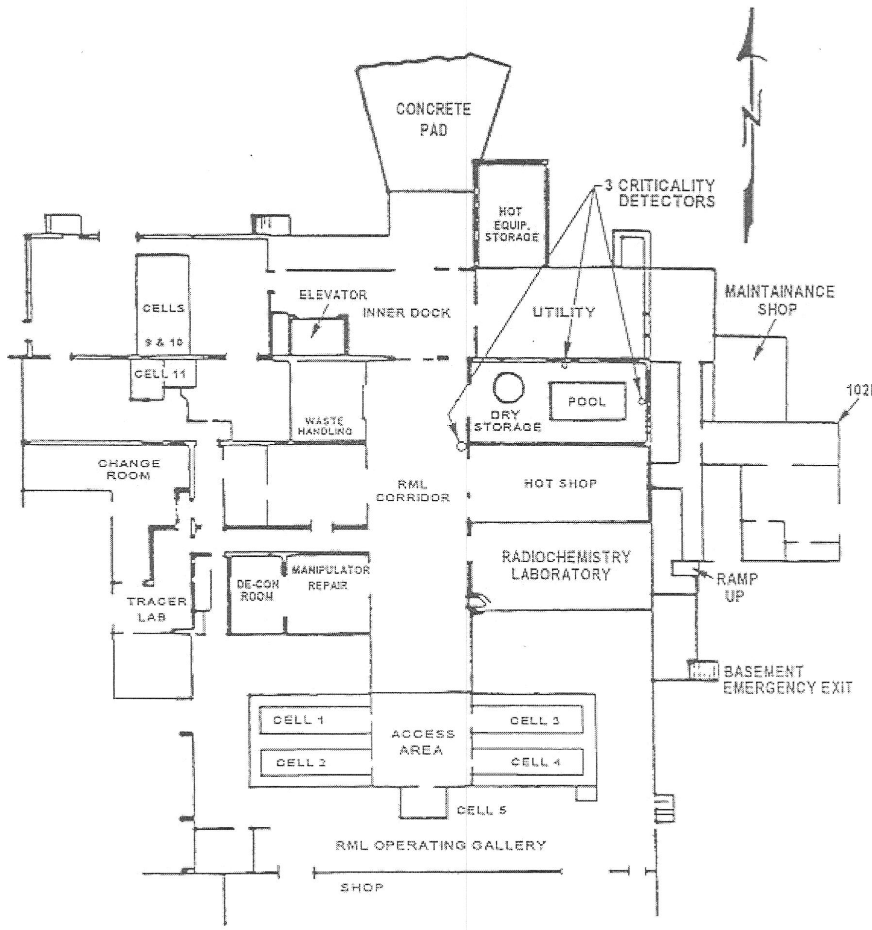


Figure 1.5 Building 102 Main Floor (Typical)

1.1.5.3 Receiving

Designated areas in the radioactive materials storage area (Hillside Storage) and the north side of Building 102 and the Building 102 dock are used for receiving incoming shipments from off site.

1.1.5.4 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.

A stainless steel lined water pool is available for the underwater container transfers, examination, repair, assembly, disassembly, and storage of irradiated materials. The pool is equipped with remote handling equipment including tongs, hooks, storage racks and an overhead crane. The pool water is circulated through strainers and ion exchange resins to maintain water clarity and minimize area dose rates.

An in-floor dry pit storage area is available for temporary shielded storage of irradiated fuel assemblies, rods or other materials. The storage area consists of recessed steel pipes placed in the concrete floor that extend downward below the floor surface. Removable one foot thick stepped shield plugs provide radiation shielding and access using remote handling tools and shielded transfer casks.

1.1.5.5 Shielded Enclosures (Hot Cells)

Inspection, use, storage, and packaging of irradiated special nuclear material is performed in specially designed shielded enclosures equipped with shielded viewing windows, shielded radiation locks, and as required remote handling capabilities.

Radiation shielding in a typical hot cell consists of up to three feet thick high density concrete walls and a three foot thick leaded glass viewing window, as shown in Figure 1.6. Each cell can be equipped with an overhead bridge crane and remote manipulators as necessary to provide remote handling capabilities of irradiated materials.

A typical hot cell is also equipped with a pair of hydraulically operated steel shield doors which form a radiation lock. The doors are operated and controlled at the cell operating face. The controls are interlocked so that the outer door normally cannot be opened when the inner door is open. A key lock override switch is provided for unusual circumstances such as cell decontamination activities.

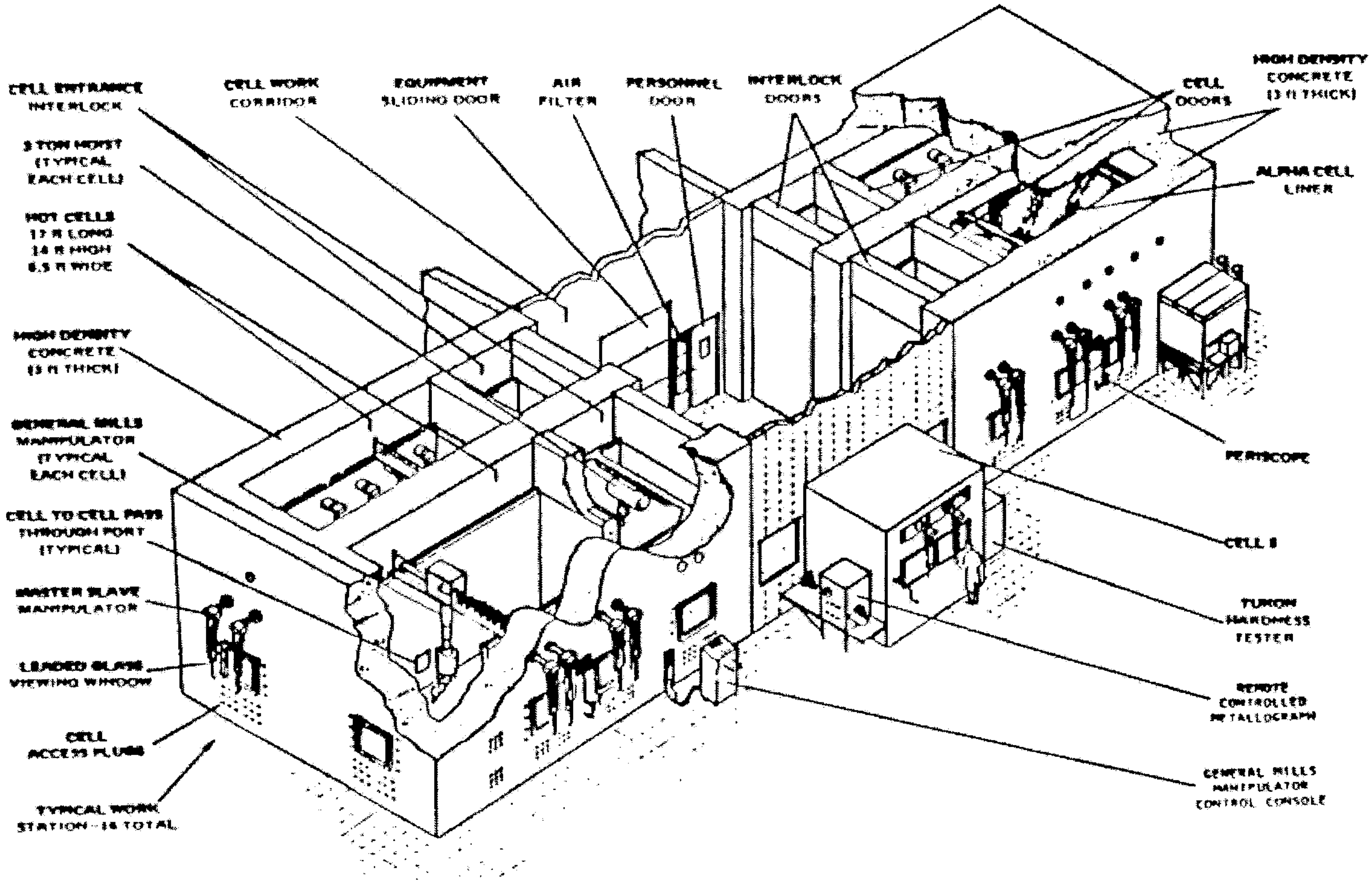


Figure 1.6 RADIOACTIVE MATERIALS LABORATORY HOT CELLS (Typical)

1.1.5.5.1 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, offices, equipment storage areas, and a manipulator repair room.

A hot shop facility occupies the area next to the radiochemistry laboratory. The area is used for the repair of contaminated equipment. Only traces of SNM are present in the contamination.

1.1.5.6 Ventilation System

Inlet air is typically provided by multiple air conditioning units furnishing filtered and tempered outside air to the operation and office areas. The inlet air is adjusted as necessary to maintain a balance with the exhaust system.

Room air ventilation is designed to provide air flow from areas of lesser potential contamination to areas of higher potential contamination. The system of directional airflow minimizes the possibility of accidental contamination of nonradioactive areas. Airflows are typically single pass and continuous except in a portion of the RML operating gallery where the room air is filtered and recirculated. The operating areas are typically supplied at rates of 6 to 40 changes per hour. A minimum of six air changes per hour (approximately) is being exhausted from the RML hot cell spaces. This amount of airflow to the cells is adequate for control of contamination.

Direction of air flow between areas is checked on a routine surveillance interval in accordance with written procedures or after significant changes to the ventilation system. Ventilation exhaust systems for facilities that routinely handle dispersible SNM in quantities in excess of the limits specified in Appendix C of 10CFR20 are monitored while in use as proof of filter performance.

1.1.5.7 Air Supply

The capacity of the inlet air system is rated at a nominal 65,000 cubic feet per minute, and can be adjusted as necessary to maintain a balance with the exhaust system.

1.1.5.8 Pressure and Flow Rates

Small static pressures (-0.01 to -0.03 inch of water) are typically 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. Glove box contamination control is typically affected by maintaining from 0.5 to 1.0 inch of water negative static pressure. At this static pressure, air velocities on the order of 125 linear feet per minute or greater are achieved through openings, an adequate rate to provide the necessary control.

1.1.5.9 Filtration and Exhaust

The main exhaust equipment is located in Building 102A. Typically 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 HEPA filters and then discharged to the atmosphere through a stack.

Exhaust air from the RML Hot Cells is prefiltered at the outlet of each cell and is routed to a filter bank of HEPA filters in the Building 102 basement before it joins the main exhaust stream at Building 102A. 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 filters are constructed of fire-resistant materials and are housed in noncombustible ductwork.

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

Fire protection is provided to the exhaust system by: (1) a C02 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.

1.1.5.10 Alternate Power for Ventilation Exhaust

In the event of an interruption of power provided by the local utility, necessary equipment, supplied through the main building switchboard, will be supplied with alternate power. This equipment includes two exhaust fans, fans supplying air to areas where radioactive materials are handled, and the main stack monitoring equipment.

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 provides for automatic switching of equipment during transitions between normal and alternate supply conditions. During normal and alternate power operations, 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.

In case a fire is identified by a detector in the exhaust ducts, the exhaust system is operated on reduced flow until the "fire” condition is cleared manually.

1.1.5.11 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 and records the data.

1.1.5.12 Noble Gas Monitor

The flow-through ion chamber is a detector to measure activity in gases. The gas sample 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 gas sample then passes into the 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 approximately 3 cfm.

1.1.5.13 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 drawn from the stack. The sampled air (approximately1.5 cfm) passes through high efficiency filter.

The beta-gamma activity collected on the particulate filter is monitored by a GM detector.

The high efficiency (Particulate) filter and, if required, a charcoal cartridge is 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.

The basic stack sampling system is shown in Figure 1.7.

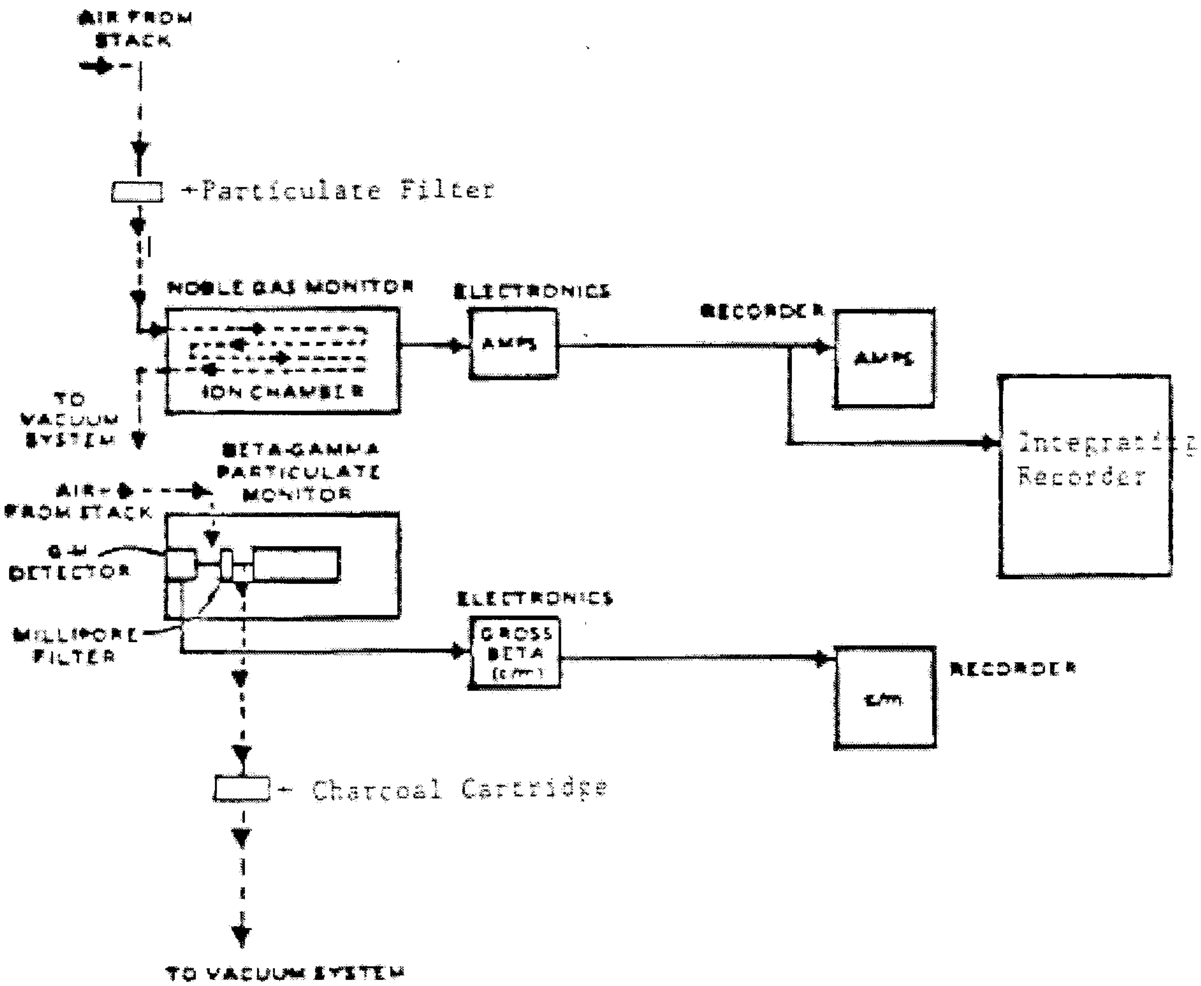


FIGURE 1.7 BASIC STACK SAMPLING SYSTEM FOR BUILDING 102 (Typical)

1.1.5.14 Enclosures And Localized Ventilation

Hoods and other localized ventilation designs are utilized to minimize personnel exposure to airborne contamination. Activities and process equipment that generate airborne contamination are designed with filtered enclosures, hoods, and other devices which maintain air concentrations of radioactivity in work areas such that personnel exposures are below 10 CFR 20 limits under normal operating conditions.

Air flows through hood openings and localized vents are maintained in accordance with Table 1.1

Additionally, differential pressure indicators are installed across exhaust system filters to monitor system performance. The flows and differential pressures are checked on a routine surveillance interval documented in written procedures and after significant changes to the ventilation system.

TABLE 1.1
SPECIFIC FACILITIES & CAPABILITIES OF VENTILATION SYSTEMS

Facility	Alarms, Interlocks & Safety Features	Purpose
Hoods	Air flow during typical operation $\geq$ 125 linear feet per minute	Prevents spread of radioactive materials
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials to environs
High Velocity Local Ventilation	Air flow designated to maintain an average of approximately 200 linear feet per minute	Prevents spread of radioactive materials from work area to immediate room area
Recirculating Air Systems & Exhaust Air Systems	Air filtered in potentially contaminated zones with HEPA filters	Removes essentially all contaminants from room and exhaust to environs
	Pressure drop indicator set to alarm at <0.8" & $\geq$4" H_2O ΔP across final filter	Maintains adequate circulation for removal of dust and contaminants from the room air
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials in environs

1.1.5.15 Exhaust System

Potentially contaminated air is exhausted through high efficiency particulate air (HEPA) filter media. HEPA filters in the exhaust system are equipped with a device for measuring differential pressure. Differential pressures greater than four inches of water are investigated. In no case will filters be operated at a differential pressure which exceeds the manufacturer's ratings for the filter.

Exhaust air from each hot cell is prefiltered at the outlet of each cell and is routed to a bank of high efficiency particulate air (HEPA) filters before it is exhausted. The exhaust connections have adjustable dampers where necessary to provide the capability for balancing flows throughout the system. The filtered air is discharged to the atmosphere through a nominal 66-inch-diameter, 75-foot-high stack.

Fire protection for the exhaust system is provided by a C02 suppression system, a water fog system for the Hot Cell basement exhaust filter bank, and a water fog spray for the main exhaust ducting from Building 102.

A schematic diagram of the ventilation exhaust system is shown Figure 1.8.

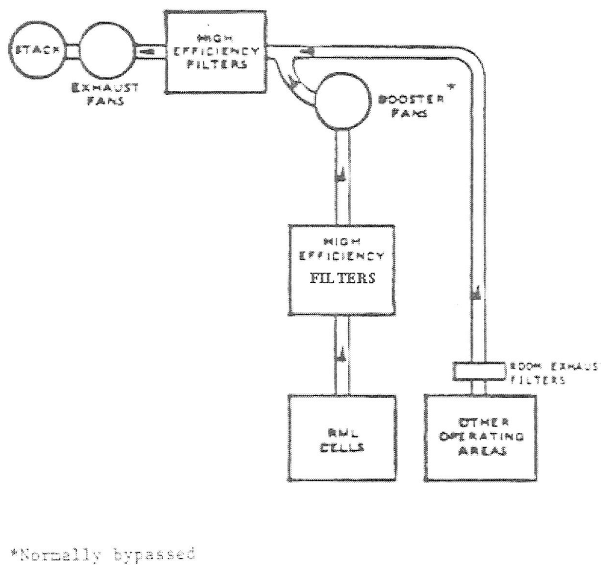


Figure 1.8 Building 102 Ventilation Exhaust System (Typical)

1.1.5.16 Backup Power Supply

In the event of the loss of utility power, alternate 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. Equipment necessary for contining operation of the hot cells and laboratories in building 102, site communication and site computer functions is currently provided with backup 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 also provides backup power to the criticality accident alarm system, the fire alarms, the telephone equipment, and an annunciator panel which transmits these alarms to the security building.

The generator is supplied with fuel from a 1,000-gallon (nominal capacity) 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.

1.1.6 CHEMISTRY, METALLURGY AND CERAMICS LAB – BUILDING 103

1.1.6.1 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. 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 currently located on one side of the corridor and offices on the other as shown in Figures 1.9 and 1.10.

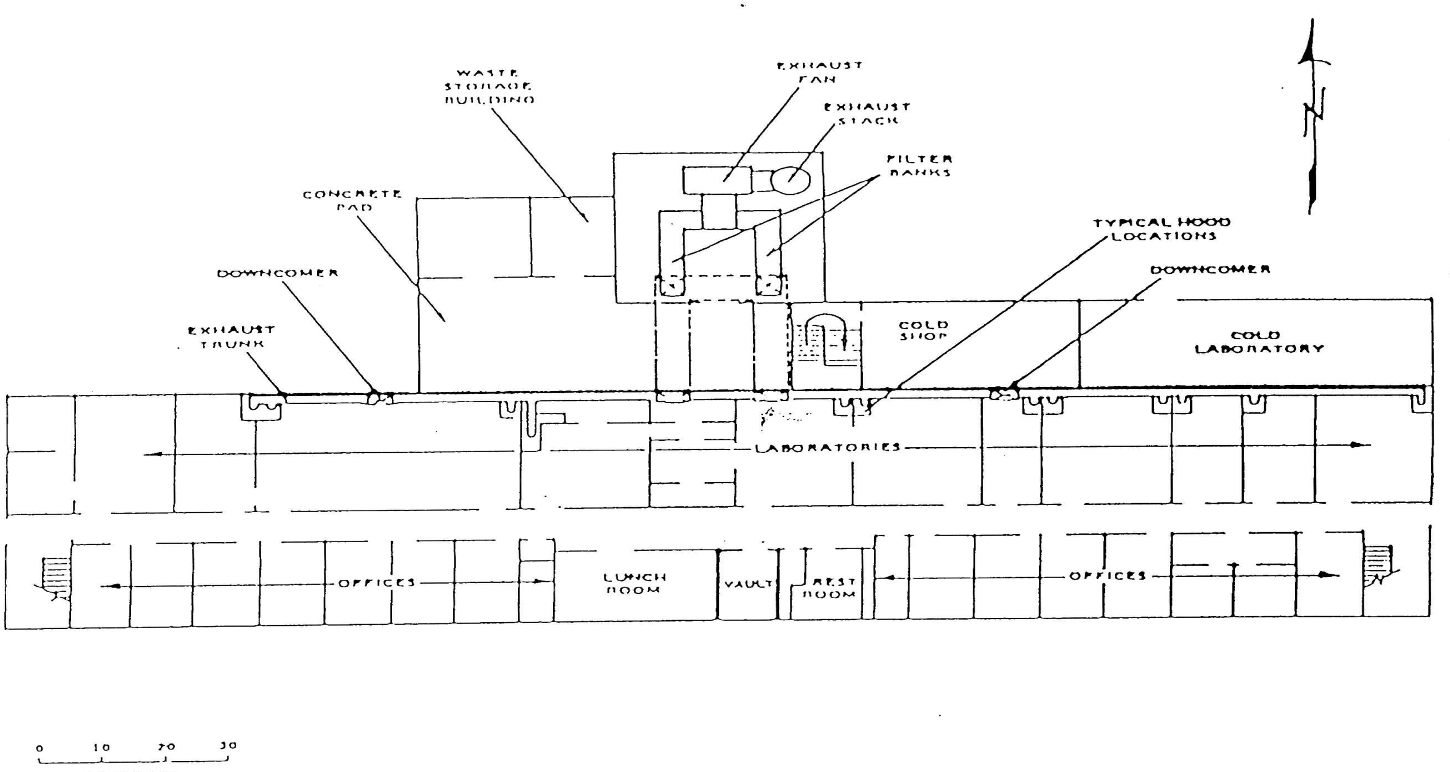


Figure 1.9– Building 103 First Floor Plan (Typical)

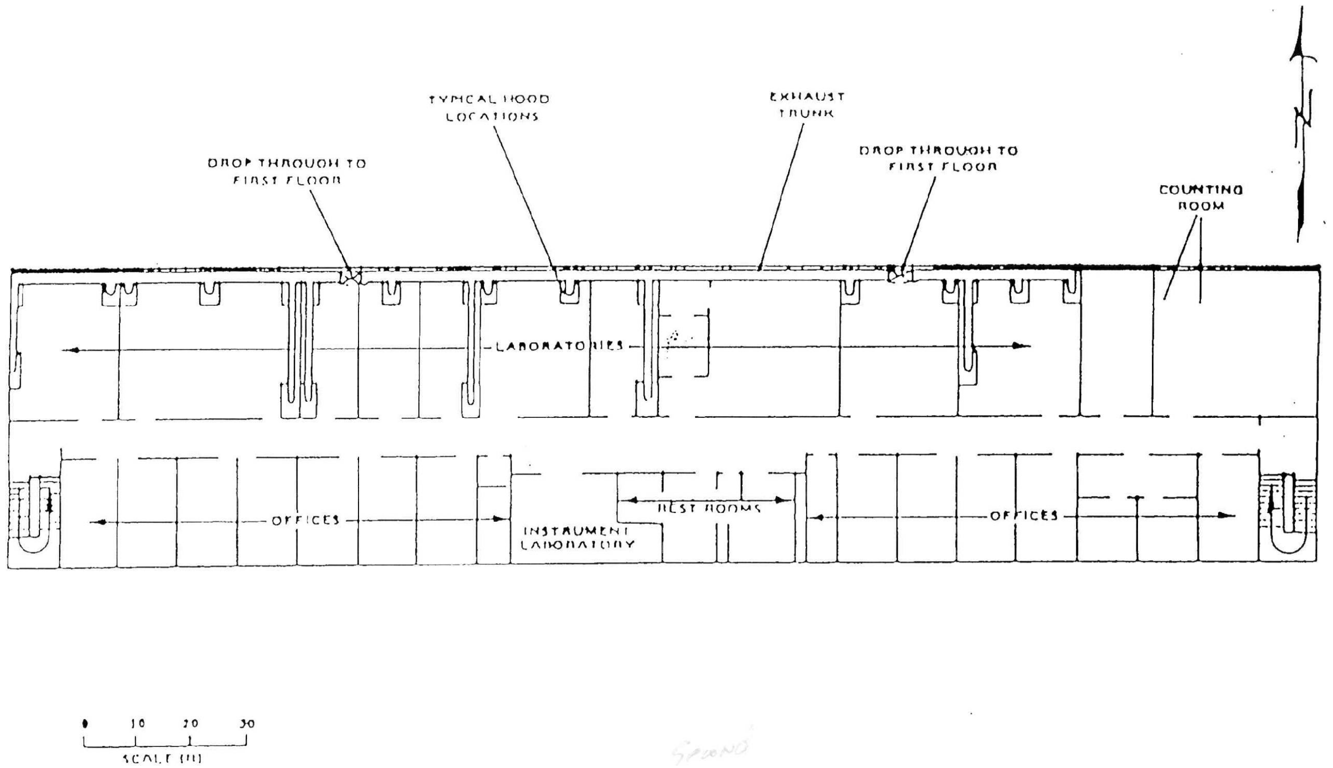


Figure 1.10 – Building 103 Second Floor Plan (Typical)

1.1.6.2 General Plans and Uses of Material

Special nuclear material is used in research and development activities including analytical, test specimen fabrication, maintenance, 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.

1.1.6.3 Laboratory Facilities and Equipment

Laboratory facilities include electron microscopes, x-ray equipment, microscopes, machine shop equipment of various kinds, ultrasonic equipment, electrochemical equipment, mechanical testing equipment, mounting presses, hoods, and similar experimental equipment deemed necessary.

The chemistry areas consist of typical chemical laboratories, a counting room, and an instrument room. Equipment in the chemistry laboratories currently includes 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 currently provided on the ground floor.

1.1.6.4 Fire Protection

Building 103 is provided with an automatic sprinkler system. Fire extinguishers also are located strategically throughout the area. VNC fire prevention procedures minimize the fire potential; however, extinguishing equipment and materials are provided at strategic locations in the building. 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 are typically used for pyrophoric materials.

1.1.6.5 Ventilation System

Inlet air supply for Building 103 is provided by air conditioning units that furnish filtered and tempered outside air to the building. Air typically passes from the office areas through 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.

Air is withdrawn through the hoods and glove boxes passing first through individual high-efficiency filters at each hood or glove box.. From the individual filters, the air is conducted through a second filtration in one of two parallel banks of high-efficiency filters and is discharged through a stack which is approximately 48-foot-high, and 5-foot-diameter.

The main exhaust blower operates at approximately 36,000 cubic feet per minute. If complete ventilation failure occurs, an evacuation alarm is sounded automatically.

The high-efficiency filters are fabricated of fiberglass or equivalent to provide resistance to fire. Filter frames are metal or chemically impregnated for resistance to fire, and permanent ductwork is typically metal or polyvinyl chloride. The main filter system has dual filter banks 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. The exhaust air is sampled continuously for iodine (if required) and particulate activity at a point approximately 24 feet below the top. The particulates are collected on a high efficiency filter; and iodine is collected on a charcoal cartridge. The filter and cartridge (if required) are evaluated on a weekly basis.

Each laboratory room used to conduct activities with radioactive materials is equipped with air sampling devices. Typically airflow rates are adequate to perform routine operations with nuclear materials without the use of respiratory protection. Approximately 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 inch of water with respect to the room. Appropriate instrumentation indicating airflow and/or differential pressure is available.

1.1.6.6 Radioactive Waste Facilities

Dry contaminated waste materials generated in Building 103 are packaged and stored in appropriate locations prior to being transferred to the site radioactive material storage facility. 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. Typically tank wastes are sent to the site waste evaporator for concentration and solidification.

1.1.6.7 Criticality Alarm

Building 103 currently is not monitored for a criticality accident. The quantity of SNM permitted in Building 103 is below the threshold requiring coverage by a criticality alarm system.

1.1.6.8 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 but do require a wide variety of forms, compounds, and concentrations.

Criticality control of these materials is based on controlling the amount of SNM available for use in any single laboratory room at any one time to a safe batch or less. Special nuclear materials normally are stored in a vault when not in laboratory use.

1.1.7 NUCLEAR TEST REACTOR FACILITY – BUILDING 105

Building 105 is located immediately north of Building 102 in the 100 Area. This building currently houses the Nuclear Test Reactor (NTR) and facilities leased to an external company. The current floor plan of Building 105 is shown in Figure 1.11.

1.1.7.1 NTR Facilities

The Nuclear Test Reactor (NTR) is licensed pursuant to 10CFR50 (NRC License R-33). Special nuclear material authorized by License SNM-960 is occasionally taken to the NTR facility primarily for purposes of neutron examination. Such material is in the form of sealed units, and less than a minimum critical mass is permitted in the NTR facility at any one time. The materials are handled pursuant to the safety provisions and requirements of License R-33.

1.1.7.2 Other Laboratory Areas (Leased Facilities)

Other laboratory areas in Building 105 use small quantities of SNM as sealed sources, standards, foils, or as electronic components in gram or microcurie quantities. These radioactive materials are controlled and inventoried by the Regulatory Compliance function.

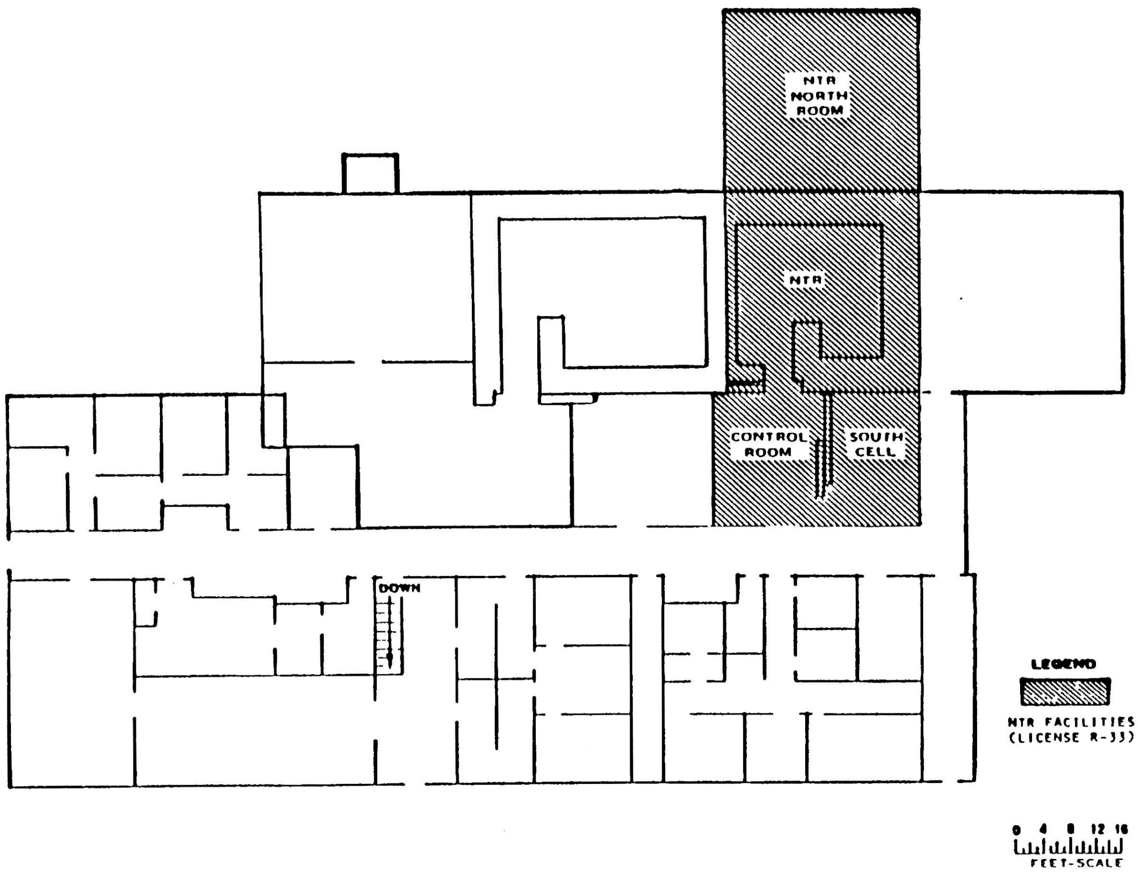


Figure 1.11 – Building 105 Floor Plan (Typical)

1.1.8 SITE RADIOACTIVE WASTE FACILITIES

1.1.8.1 Liquids

A liquid processing facility is provided with sufficient capacity and capability to enable collection, sampling, analysis, and disposition of radioactive liquids in accordance with the regulations. Liquid waste volumes are reduced using techniques such as evaporation, pH adjustment, sedimentation, filtration, ion exchange, solidification and packaging of the concentrates for disposal. These activities primarily occur at the waste evaporation plant.

The quantity of radioactive materials in liquid effluents discharged from the facility are reported to the NRC on an annual basis.

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

Other liquid non-radioactive waste flows through a piping system into state environmentaly permited waste collection system. After sampling and determining that radioactivity, if any, is within permissible discharge levels, the water in the basin is released.

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. 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.

1.1.8.2 Radioactive Liquid Waste Evaporator Plant

The Waste Evaporator Plant (WEP) 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 maintain 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.

Activities 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 WEP. Activities conducted at the WEP include concentration of these wastes, the discharge of processed effluents and the preparation for transfer of concentrated waste materials to a licensed waste disposal contractor. The estimated annual plant throughput is 100,000 gallons of liquid waste.

Liquid wastes are collected from the various site accumulation tanks and transported to the WEP by fork lift truck in a specially designed 1,500-gallon tank. The waste transfer tank normally 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., typically 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.

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

Storage of dry wastes at the WEP typically 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.

1.1.8.3 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. 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 about 3,000 cfm is driven by an electrically powered blower mounted at the base of the stack.

1.1.8.4 Monitoring Procedures

Monitoring and/or step-off procedures are observed at points where each potentially contaminated regulated area exits into the clean or non-regulated areas. In addition, survey instruments are provided at convenient locations for final surveying.

1.1.8.5 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. Solidified waste contained in drums is appropriately marked and stored awaiting removal from the Vallecitos site by a licensed waste disposal contractor or other approved methods. Waste drums are not buried on site.

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

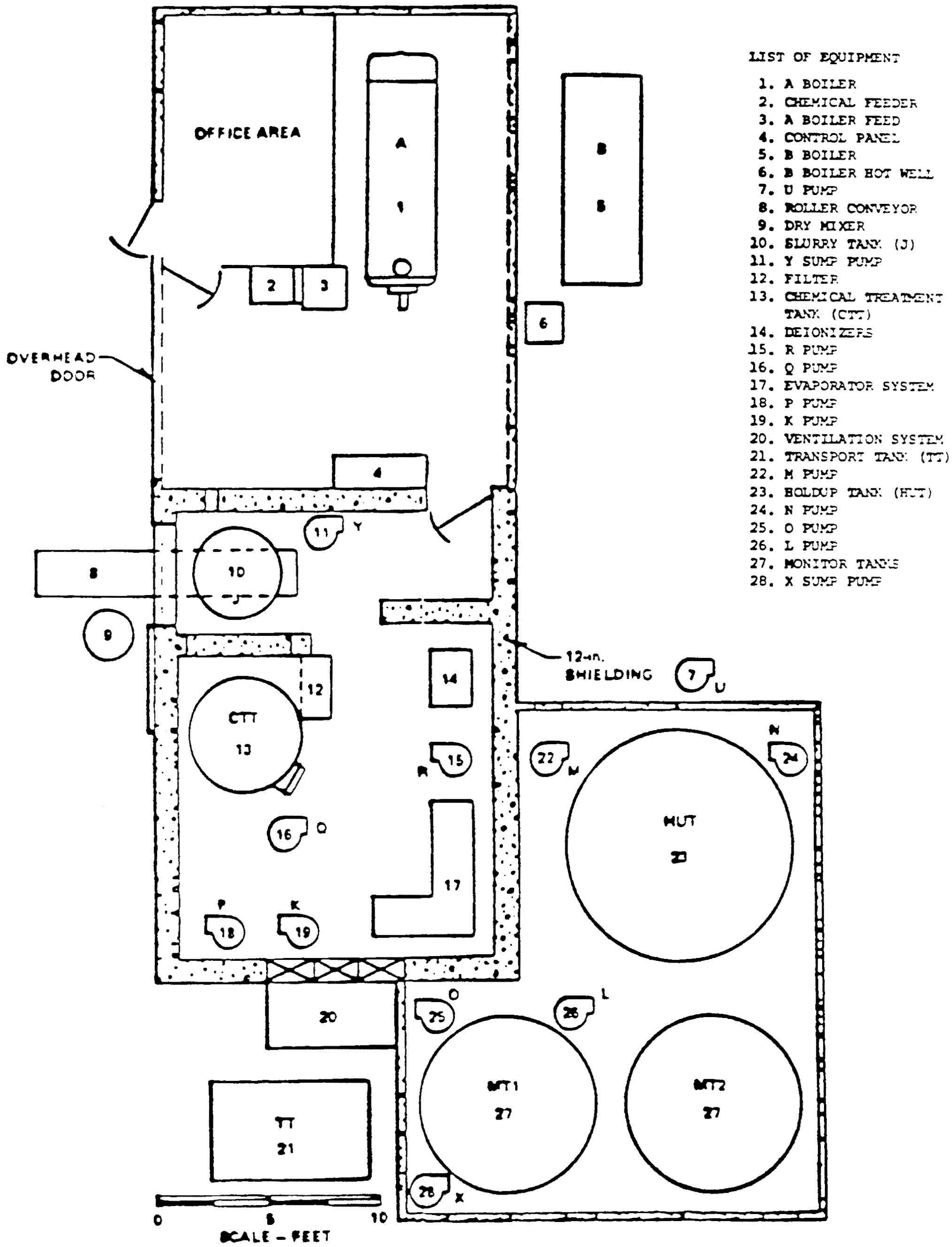


Figure 1.12 – Waste Evaporator Building (Typical)

1.1.9 SOLIDS

Solid radioactive waste management facilities are provided with sufficient capability to enable preparation, packaging, storage, and transfers to licensed disposal sites in accordance with the regulations.

Dry contaminated wastes are placed in sealable drums, tubes, boxes, or casks available at each facility where such wastes may be generated. Each facility maintains a designated area for temporary waste storage. Dry wastes are typically transferred to Building 102 for final inspection and repackaging as necessary. Irradiated fuel specimens that have been examined or analyzed in the Building 102 complex and other waste packages are transferred to the site radioactive material Hillside Storage Facility. The current layout of this facility is shown in Figure 1.13.

[REDACTED]

Figure 1.13 – Hillside Waste Storage (Typical)

[REDACTED]

[REDACTED]

The facility is covered by a prefabricated metal building.

1.1.9.1 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 by appropriate regulations.

1.1.9.2 Solid Waste Storage

Solid waste materials are stored in the Hillside Storage Facility. 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 waste storage 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

1.1.9.3 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 Regulatory Compliance function.

1.1.10 MEASURING AND MONITORING DEVICES

Portable radiation instrumentation used in the facility have the capability to measure alpha, beta, gamma or neutron radiation. A sufficient number of various types of instrumentation are available to ensure adequate radiation surveillance can be accomplished. Selection criteria of portable and laboratory counting equipment is based on the types of radiation detected, maintenance requirements, ruggedness, interchangeability and upper and lower limits of detection capabilities. The radiation safety function annually reviews the appropriateness of the types of instruments being used for each monitoring function.

1.1.11 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

In any Vallecitos Nuclear Center area in which licensed 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 chapter 5), 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 criticality accident alarm system (CAAS) is uniform throughout the required facilities for the type of radiation detected, the mode of detection, and alarm signal. Also, individual unit detectors are located to assure compliance with appropriate requirements of such as 10 CFR 70.24 (a)(1) and ANSI/ANS-8.3 (2003).

The CAAS initiates immediate evacuation of affected portions of the facility. Employees are trained in recognizing the evacuation signal. This system, and proper response protocol, is described in the Radiological Contingency and Emergency Plan for VNC.

The CAAS is maintained through routine response checks and scheduled functional tests conducted in accordance with internal procedures. In the event of loss of normal power, back-up power is automatically supplied to the criticality accident alarm system.

In the event that CAAS coverage is lost in an area required to have a monitor alarm system, compensatory measures such as limiting personnel access, halting special nuclear material movement or installing temporary detection equipment are used as an interim measure until the system is restored.

1.1.12 EMERGENCY EQUIPMENT

A vehicle is available to Radiation Safety and is normally equipped 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.

1.1.13 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 protection facilities and equipment that are available include portable extinguishers, sprinkler systems, and a wide range of typical industrial safety equipment.

1.2 INSTITUTIONAL INFORMATION

VNC's NRC license number is SNM-960 (Docket #70-754).

1.2.1 IDENTITY AND ADDRESS

This application for license renewal is filed by GE-Hitachi Nuclear Energy Americas, LLC, Vallecitos Nuclear Center. GEH's principal place of business is 3901 Castle Hayne Rd, Wilmington, NC, 28401

The full address for Vallecitos Nuclear Center is: Vallectios Nuclear Center, 6705 Vallecitos Road, Sunol, California 94586.

1.2.2 CORPORATE AND FINANCIAL

SNM-960 is held by GE-Hitachi Nuclear Energy Americas, LLC, which is wholly owned by GE-Hitachi Nuclear Energy Holdings, LLC. This parent, in turn, is controlled through a 60% voting ownership interest held by GENE Holding, LLC, which is controlled by the General Electric Company, a New York corporation with a principal place of business at 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.

GE-Hitachi Nuclear Energy Americas, LLC is a single member Delaware limited liability company. Its officers are a Chief Executive Officer (CEO), currently Caroline Reda, a citizen of the United States and permanent resident of the United States, and its Chief Financial Officer (CFO), currently Craig Steven, a citizen of the United States. The officer's business address is, 3901 Castle Hayne Rd, Wilmington, NC, 28401.

GE-Hitachi Nuclear Energy Americas, LLC is wholly owned by GE-Hitachi Nuclear Energy Holdings, LLC, a multi-member Delaware limited liability company. GE-Hitachi Nuclear Energy Holdings, LLC has a Board of Managers comprised of seven representatives, five of whom have voting rights. Three voting members are appointed by GE and are U.S. citizens, and two voting members are appointed by Hitachi. In addition, GE-Hitachi Nuclear Energy Americas, LLC's Chief Executive Officer and the Chief Executive Officer of Hitachi-GE Nuclear Energy Co., Ltd. serve on the Board but have no voting rights.

1.2.3 TYPE, QUANTITY, AND FORM OF LICENSED MATERIAL

- The following types, maximum quantities, and forms of special nuclear materials are authorized:
- [REDACTED] kilograms of U-235 contained in uranium enriched to a maximum enrichment of less than or equal to 10%, for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant byproduct and reactor-produced transuranics,
- [REDACTED] kilograms of U-235 enriched to more than 10 percent for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant byproduct and reactor-produced transuranics,
- [REDACTED] kilogram of U-235 in any unirradiated form,
- [REDACTED] of Plutonium in a contained or sealed form in addition to the irradiated quantities, and

- [REDACTED] U-233 in any form.

1.2.4 ACTIVITY

VNC complies with applicable parts of Title 10, Code of Federal Regulations, unless specifically amended or exempted by NRC staff.

Authorized activities at VNC include:

1.2.4.1 Laboratory Operations

- Analysis of the chemical and isotopic composition, concentration and behavior of special nuclear materials by wet chemistry and physical measurement techniques.
- Physical analyses and testing of physical and metallurgical properties of special nuclear materials.
- Measurements of radiation and its effects on instruments and on the structure and composition of materials.
- 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. Examination and temporary storage of irradiated enriched uranium fuel elements from the GEH Nuclear Test Reactor (NTR).
- Assembly, modification, cleaning and repair of unirradiated encapsulated experimental assemblies.
- Research and development activity associated with but not limited to those activities listed above.

1.2.4.2 General Services Operations

- Design, fabrication, and testing of equipment containing special nuclear materials and maintenance of such equipment.
- Storage of special nuclear materials other than wastes in shielded containers and or at locations as applicable and in designated general-purpose storage areas.

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

- Decontamination of equipment and facilities.

1.2.4.3 Waste Treatment

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

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

1.3 SPECIAL AUTHORIZATIONS AND EXEMPTIONS

1.3.1 ACTIVITIES NOT REQUIRING PRIOR NRC AUTHORIZATION BY LICENSE AMENDMENT

1.3.1.1 The licensee may make changes to the licensed operation provided they do not involve: major changes or additions to existing processes which may involve a significant increase in potential or actual environmental impact resulting from utilizing such changes or additions.

1.3.1.2 In determining whether a new process technology requires such prior authorization by license amendment, the following factors will be considered: (1) type of equipment utilized, (2) chemical reactions involved and (3) potential and/or actual environmental impact.

1.3.2 AUTHORIZED GUIDELINES FOR CONTAMINATION-FREE ARTICLES

Authorization to use the guidelines, contamination and exposure rate limits specified at the end of this Section, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," US NRC, April 1993 for decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use.

1.3.3 SEALED PLUTONIUM SOURCES

Sealed Plutonium sources shall be subject to the leak testing and actions specified in the attached specification "Leak Testing Sealed Plutonium Sources", dated April 1993, included at the end of this section.

1.3.4 EXEMPTION TO CRITICALITY MONITORING SYSTEM REQUIREMENTS

VNC is granted an exemption to the requirements of 10 CFR 70.24, "Criticality Accident Requirements", in accordance with 10 CFR 70.24(a). The following areas are exempt from monitoring:

- Areas where SNM is stored in locations within the United States provided that the SNM is fully packaged as for transport in containers meeting all of the general license requirements of 10 CFR 71 or in containers certified for transport under the provisions of 10 CFR 71 in accordance with the conditions of a Certificate of Compliance authorizing delivery of such containers to a carrier for Fissile Class I transport,
- Located in Building 102 storage pool and hot cells,
- For each area in which is stored one shipment of packages containing special nuclear material licensed pursuant to 10 CFR 71 for transport outside the confines of the Vallecitos Nuclear Center insofar as the requirements of Section 70.24 pertain to the material contained in such shipments,
- For each area where there is not more than one safe batch (as described in Section 5.4.3) under conditions which protect against the rearrangement of fuel bearing portions into more reactive configurations and
- For each area which meets the requirements of a subcritical area (as described in Section 5.4.3)

1.3.5 REACTOR PRODUCED TRANSURANICS

Notwithstanding 10 CFR 70.22(h)(1) and 70.22(j)(1) and in accordance with the licensed quantities and types of material in this license, the aggregate quantity of reactor produced transuranics distributed in spent fuel does not constitute a formula quantity of SNM.

1.3.6 NON-APPLICABILITY OF 10 CFR 70.61 THROUGH 70.76

Notwithstanding the requirements of 10 CFR 70.60, VNC is not required to perform an ISA as stipulated in 10 CFR 70.61 – 70.76 because VNC is not engaged in nor authorized by license to engage in the activities specified therein.

1.4 GENERAL PLANS, USES AND POSSESSION LIMITS 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 this Section; to receive and possess the special nuclear material and associated byproduct material produced by the irradiation thereof; and to use said special nuclear materials in research and development activities as defined in Section 70.4, in chemical and physical analysis, and examination and investigation of nuclear fuels, associated materials and devices at the Vallecitos Nuclear Center.

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 1.2.3.

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT
PRIOR TO RELEASE FOR UNRESTRICTED USE
OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE,
OR SPECIAL NUCLEAR MATERIAL

U.S. Nuclear Regulatory Commission
Division of Fuel Cycle Safety
and Safeguards
Washington, DC 20555

April 1993

The instructions in this guide, in conjunction with Table 1, specify the radionuclides and radiation exposure rate limits which should be used in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.
2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to the application of the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.
3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap that are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.
4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:
    a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.
    b. Provide a detailed health and safety analysis that reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey that establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, DC 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

    a. Identify the premises.
    b. Show that reasonable effort has been made to eliminate residual contamination.
    c. Describe the scope of the survey and general procedures followed.
    d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/ 100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

LICENSE CONDITION FOR LEAK TESTING

SEALED PLUTONIUM SOURCES

April 1993

A. Each plutonium source shall be tested for leakage at intervals not to exceed 6 months. In the absence of a certificate from a transferor indicating that a test has been made within 6 months prior to the transfer, the sealed source shall not be put into use until tested.

B. The test shall be capable of detecting the presence of 0.005 microcurie of alpha contamination on the test sample. The test sample shall be taken from the source or from appropriate accessible surfaces of the device in which the sealed source is permananetly or semipermanently mounted or stored. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Commission.

C. If the test reveals the presence of 0.005 microcurie or more of removable alpha contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with the Commission's regulations. Within 5 days after determining that any source has leaked, the licensee shall file a report with the Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, describing the source, test results, extent of contamination, apparent or suspected cause of source failure, and the corrective action taken. A copy of the report shall be sent to The Administrator of the nearest NRC Regional Office listed in Appendix D of Title 10, Code of Federal Regulations Part 20.

D. The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources expected from this test shall be tested for leakage prior to any use or transfer to another person unless they have been leak tested within 6 months prior to the date of use or transfer.

CHAPTER 2.0
ORGANIZATION AND ADMINISTRATION

2.1 POLICY

The Vallecitos Nuclear Center (VNC) policy is to maintain a safe work place for its employees, to protect the environment, and to assure operational compliance within the terms and conditions of special nuclear material licenses and applicable NRC regulations.

2.2 ORGANIZATIONAL RESPONSIBILITIES AND AUTHORITY

2.2.1 KEY POSITIONS WITH RESPONSIBILITIES IMPORTANT TO SAFETY (FIGURE 2.1)

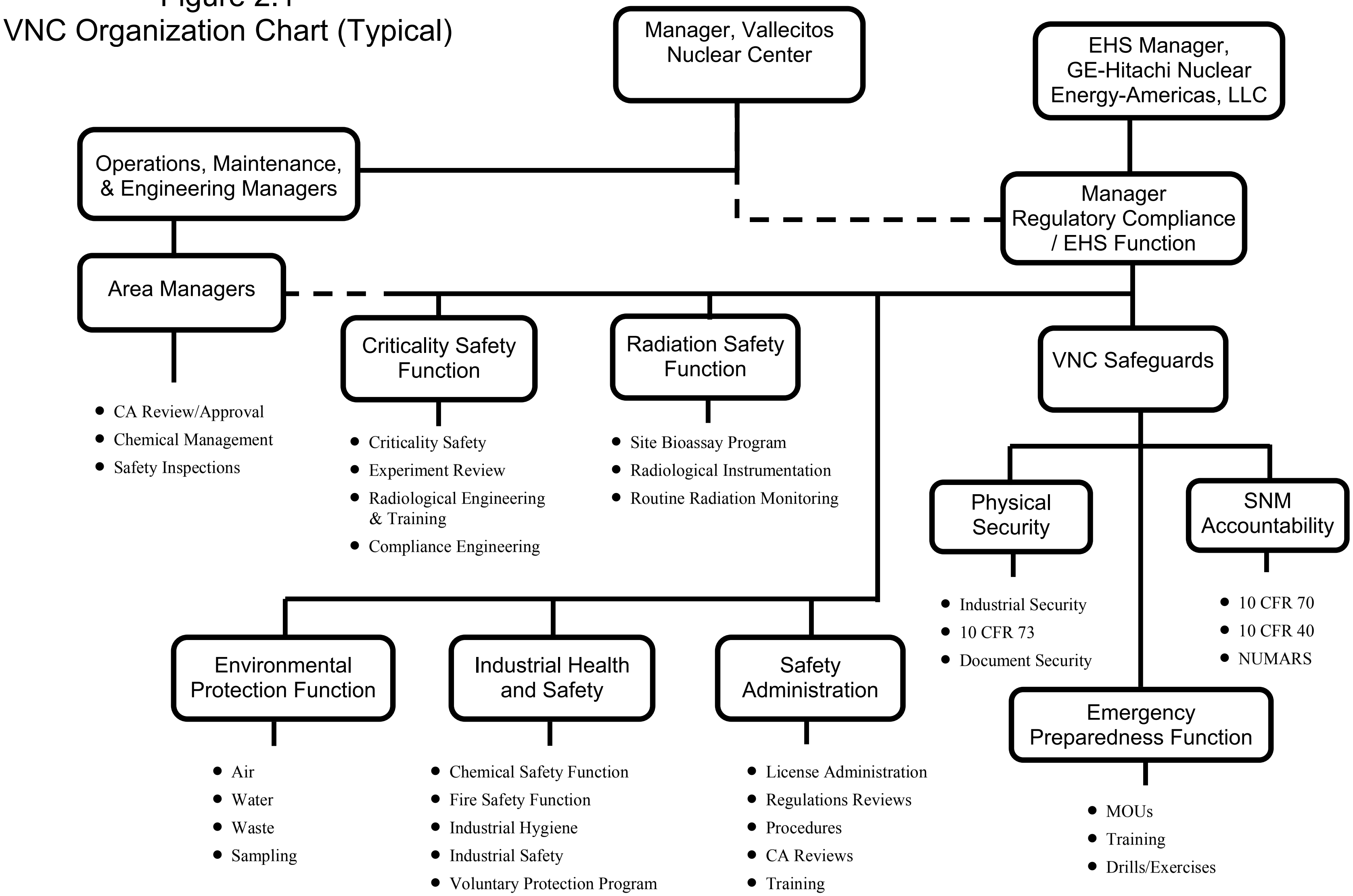
Responsibilities, authorities, and interrelationships among the VNC organizational functions with responsibilities important to safety are specified in approved position descriptions and in documented and approved practices.

2.2.1.1 Manager, Vallecitos Nuclear Center

The Manager, Vallecitos Nuclear Center is the individual who has overall responsibility for safety and activities conducted at the facility. The Manager, Vallecitos Nuclear Center directs operations by procedure, or through other management personnel. The activities of the Manager, Vallecitos Nuclear Center are performed in accordance with VNC's policies, procedures, and management directives. The Manager, Vallecitos Nuclear Center provides for safety and control of operations and protection of the environment by delegating and assigning responsibility to qualified Area Managers who are charged with maintaining and operating the facility in accordance with all applicable building codes & OSHA regulations.

The minimum qualifications of a Manager, Vallecitos Nuclear Center are a BS or BA degree and two years experience in nuclear operations. The Manager, Vallecitos Nuclear Center is knowledgeable of the safety program concepts as they apply to the overall safety of a nuclear facility and has the authority to enforce the shutdown of any process or facility. The Manager, Vallecitos Nuclear Center must approve restart of an operation they request be shutdown.

Figure 2.1
VNC Organization Chart (Typical)



2.2.1.2 Area Manager

The Area Manager is the designated individual who is responsible for ensuring that operations and activities necessary for safe operations and protection of the environment are conducted properly within their designated area of the facility. Designated Area Manager responsibilities include:

- Assure safe operation, maintenance and control of activities
- Assure safety of the environs as influenced by operations
- Assure application of assurance elements to safety controls, as appropriate
- Assure configuration control for safety controls for the assigned facility area, as required
- Assure use of approved written operating procedures which incorporate safety controls and limits
- Assure adequate operator training

The minimum qualifications of an Area Manager is a BS or BA degree in a technical field, and two years of experience in nuclear operations; or a high school diploma with five years of nuclear experience.

Area Managers shall be knowledgeable of the safety program procedures (including as applicable chemical, radiological, criticality, fire, environmental and industrial safety) and shall have experience in the application of the program controls and requirements, as they relate to their areas of responsibility. The Manager, Vallecitos Nuclear Center approves the assignment of individuals to the position of Area Manager, and the listing of Area Managers, by area of responsibility, is maintained current by the Manager of Regulatory Compliance and EHS.

The Area Manager shall incorporate the results of reviews, audits, inspections, assessments and investigations to improve operational controls, procedures and performance.

2.2.1.3 Regulatory Compliance Function

The Regulatory Compliance function is administratively independent of production responsibilities and has the authority to enforce the shutdown of any process or facility in the event that controls for any aspect of safety are not assured. This function has designated overall responsibility to establish the radiation safety, criticality safety, environmental protection, chemical safety, fire protection and emergency preparedness programs to ensure compliance with federal, state and local regulations and laws governing operation of a nuclear facility. These programs are designed to ensure the health and safety of employees and the public as well as protection of the environment.

The manager of the Regulatory Compliance function must hold a BS or BA degree in science or engineering and have five years of management experience in assignments involving regulatory activities. The manager of the Regulatory Compliance function must have appropriate understanding of health physics, nuclear criticality safety, environmental protection, and chemical and fire safety programs.

2.2.1.4 Criticality Safety Function

The criticality safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. This function must approve restart of an operation they request be shutdown.

Designated areas of oversight include:

- Establish the criticality safety program including design criteria, procedures and training
- Provide authoritative professional advice and counsel to Site employees and management on criticality safety control measures, including review and approval of operating procedures
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Specify criticality safety control requirements and functionality
- Perform methods development and validation to support criticality safety analyses
- Perform neutronics calculations, write criticality safety analyses and approve proposed changes in process conditions or equipment involving fissionable material
- Assess the effectiveness of the criticality safety program through audit programs

The criticality safety function manager shall hold a BS or BA degree in science or engineering, have at least three years experience in assignments involving regulatory activities, and have at least one year experience in the understanding, application and direction of nuclear criticality safety programs.

Minimum qualifications for a member of the criticality safety function are a BS or BA degree in science or engineering with at least one year of relevant criticality safety experience.

2.2.1.5 Radiation Safety Function

The radiation safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. This function must approve restart of an operation they request be shutdown.

Designated areas of oversight include:

- Establish the radiation protection and radiation monitoring programs
- Establish the radiation protection design criteria, procedures and training programs to control contamination and exposure to individuals
- Evaluate radiation exposures of employees and visitors, and ensure the maintenance of related records
- Conduct radiation and contamination monitoring and control programs
- Evaluate the integrity and reliability of radiation detection instruments
- Provide analysis and approval of proposed changes in process conditions and process equipment involving radiological safety
- Provide advice and counsel to Site employees and management on matters of radiation safety
- Assess the effectiveness of the radiation safety program through audit programs

A member of the radiation safety function shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the radiation safety function. The minimum qualifications of personnel assigned functional responsibility in the radiation safety function shall be:

1. The radiation safety function manager shall hold a BS or BA degree in science or engineering, have at least five years experience in assignments that include responsibility for radiation safety, and have experience in the understanding, application and direction of radiation safety programs. An

alternate minimum experience qualification is the professional certification in health physics (CHP).

2. A specialist in the radiation safety function shall hold a BS or BA degree in science or engineering, have at least two years experience in assignments that include responsibility for radiation safety or have eight years of experience in health physics or radiation protection.
3. A radiation monitoring technician (RMT) in the radiation safety function shall have a high school degree, or equivalent, 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 examinations covering radiation protection procedures.

2.2.1.6 Environmental Protection Function

The environmental protection function is administratively independent of production responsibilities and has the authority to shutdown operations with potentially uncontrolled environmental conditions. This function must approve restart of an operation they request be shutdown.

Designated areas of oversight include:

- Identify environmental protection requirements from federal, state and local regulations which govern the Vallecitos operation
- Establish systems and methods to measure and document adherence to regulatory environmental protection requirements and license conditions
- Provide advice and counsel to Site employees and management
- Evaluate and approve new, existing or revised equipment, processes and procedures involving environmental protection activities
- Provide environmental protection support for integrated safety analyses and configuration control
- Assure proper federal and state permits, licenses and registrations for non-radiological discharges from the facilities

2.2.1.7 Chemical and Fire Safety Function

The chemical and fire safety function is administratively independent of the production responsibilities and has the authority to shutdown operations with potentially hazardous health and safety conditions. This function must approve restart of an operation they request be shutdown.

Designated areas of oversight include:

- Identify fire protection requirements from federal, state, and local regulations which govern the Vallecitos operations
- Develop practices regarding non-radiological chemical safety affecting nuclear activities
- Provide advice and counsel to Site employees and management on matters of chemical and fire safety
- Provide consultation and review of new, existing or revised equipment, processes and procedures regarding chemical safety and fire protection
- Provide chemical and fire safety support for integrated safety analyses and configuration control

2.2.1.8 Site Security and Emergency Preparedness Function

The site security and emergency preparedness function is administratively independent of the production responsibilities. Designated areas of oversight include:

- Provide physical security for the site
- Establish and maintain the emergency preparedness program, including training and program evaluations
- Provide advice and counsel to Site employees and management on matters of physical security and emergency preparedness
- Maintain agreements and preparedness with off-site emergency support groups

2.2.2 MANAGEMENT CONTROLS

- Management controls for the conduct and maintenance of VNC's health, safety and environment protection programs are contained in documented procedures. An overview of the controls is provided in Chapter 11.

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 their componentin 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.

2.3 TRAINING

Personnel training is conducted as necessary to provide reasonable assurance individuals are qualified, continue to understand, and recognize the importance of safety while performing assigned activities.

Training is provided for each individual at VNC, commensurate with assigned duties. Training and qualification requirements are met prior to personnel fully assuming the duties of safety-significant positions, and before assigned tasks are independently performed.

Formal training relative to safety includes radiation and radioactive materials, risks involved in receiving low level radiation exposure in accordance with 10 CFR 19.12, basic criteria and practices for radiation protection, nuclear criticality safety principles, chemical and fire safety, maintaining radiation exposures and radioactivity in effluents As Low As Reasonably Achievable (ALARA), and emergency response.

The system established for management assurance, and record retention of training and retraining is described in Chapter 11.

2.3.1 NUCLEAR SAFETY TRAINING

Area Managers shall assure that new employees receive instruction in criticality safety, radiation safety, plant operating procedures, site emergency procedures and general industrial safety prior to their working with special nuclear materials in a criticality limit area (CLA). A criticality control training program is maintained to emphasize the need for following the criticality control procedures and to aid personnel in understanding the various parameters that are essential to the maintenance of subcritical conditions. The program may be conducted by the Criticality Safety function, some other portion of the Regulatory Compliance and EHS function, or combined with training performed by operating components. This training may be combined with Radiation Safety training. Each employee taking the criticality safety course completes a written test. Employees requiring criticality safety training receives refresher training annually.

2.3.2 OPERATIONAL TRAINING

Training of personnel is performance based and incorporates the structured elements of analysis, design, development, implementation, and evaluation. Emphasis is placed on safety requirements where human actions are important to safety. Job-specific training includes applicable procedures and safety provisions, and requirements.

2.4 SAFETY COMMITTEES

2.4.1 VALLECITOS TECHNOLOGICAL SAFETY COUNCIL

The functions of the Vallecitos Technological Safety Council (VTSC) include responsibility for the following:

- Review of major changes in authorized plant activities which may affect nuclear or non-nuclear safety practices
- Professional advice and counsel on environmental protection, and criticality, radiation, chemical and fire safety issues affecting the nuclear activities.
- Review of reportable incidents and the nuclear safety program
- An annual site safety and compliance program review which considers:
    - Programs and projects to maintain radiation exposures as low as reasonably achievable undertaken by the radiation safety function.
    - Programs for improving the effectiveness of equipment used for effluent and exposure control

The VTSC is an independent review body and consists of a minimum of five senior members of GE-Hitachi's technical and/or management personnel, appointed by the Manager, Vallecitos Nuclear Center, and includes members competent in physics, chemistry, engineering and radiation safety disciplines. Its proceedings, findings and recommendations are reported in writing to the Manager, Vallecitos Nuclear Center, Manager of Regulatory Compliance and EHS, and to appropriate functional managers responsible for operations, which have been reviewed by the committee. Such reports shall be retained for at least three years.

The committee holds at least three meetings each calendar year with a maximum interval of 180 days between any two consecutive meetings.

2.5 CHANGE MANAGEMENT AND AUTHORIZATION

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 (CA) procedure.

Change Authorization is prepared whenever the work involves changes to:

- Facilities, equipment, or processes so that safety or regulatory compliance considerations differ from those previously analyzed.
- Radioactive material limits.
- 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 appropriate operational and safety functions.

CHAPTER 3.0

INTEGRATED SAFETY ANALYSIS

3.1 REGULATORY STATUS

In accordance with 10 CFR 70.60 it has been determined that the requirements of 10 CFR 70.61 through 70.76 do not apply to VNC activities licensed under SNM-960. While VNC is licensed to possess greater than a critical mass, VNC is not engaged in any of the qualifying activities related to that material and is therefore excluded. Consequently there are no IROFS identified at the facility.

3.2 CURRENT SITUATION

Notwithstanding the exclusion by 10 CFR 70.60, the activities at VNC have been reviewed and a determination made that based on the safety program in place as defined in the chapters of this license application, none of the performance requirements stipulated in 10 CFR 70, Subpart H would be exceeded.

3.3 BASIS OF SAFETY

The basis of safety at VNC is defined by the organization and administrative structure defined in Chapter 2, the safety program elements and commitments of Chapters 4 through 10 and assured by the management measures identified in commitments in Chapter 11.

CHAPTER 4.0
RADIATION SAFETY

4.1 ALARA (AS LOW AS IS REASONABLY ACHIEVABLE) POLICY

VNC's standard of care for occupationally exposed individuals is to maintain exposures below the limits established by the U.S. Nuclear Regulatory Commission. Beyond the standard of care, VNC's radiation protection staff has a commitment to establish, maintain, and implement an effective radiation protection program. This includes a program commitment to maintain employee exposures As Low As Reasonably Achievable (ALARA) that is delineated by documented radiation protection program practices and procedures. Area Managers maintain worker exposures ALARA by proper use of procedures, equipment, and process design.

The radiation safety function ensures that occupational radiation exposures are maintained ALARA via timely exposure monitoring and interaction via Regulatory Compliance and ALARA Committee active participation with operations, maintenance and engineering personnel, and periodic ALARA program assessments with management.

The Vallecitos Technological Safety Council described in Chapter 2 also plays a role in the overall ALARA program at VNC.

4.2 RADIATION SAFETY PROCEDURES AND RADIATION WORK PERMITS

Routine work performed in Radiological Controlled Areas (RCA's) is administered by the use of standard practices and procedures described in Chapter 11. Non-routine activities, (e.g.: those not covered by documented procedures), performed by VNC and non-VNC employees, are administered by the Radiation Work Permit (RWP) system. The RWP system is described in documented plant practices and procedures.

RWPs are issued for non-routine operations not addressed by an operating procedure when special radiation control requirements are necessary. The RWP specifies the necessary radiation safety controls, as appropriate, including personnel monitoring devices, protective clothing, respiratory protective equipment, special air sampling, and additional precautionary measures to be taken.

Each affected individual reviews the RWP requirements. Work is monitored by the radiation safety function as required. RWPs have expiration dates and the status of issued RWPs is reviewed on a routine basis by a Radiation Monitor Technician (RMT), Area Manager or designated alternate.

4.3 AIR SAMPLING PROGRAM

Room air is continuously sampled in normally occupied areas in which dispersible SNM is handled. Samples are analyzed for gross alpha and gross beta-gamma. Samples used to determine worker intakes are collected in such a way that the concentrations of airborne radioactive material measured is representative of the air which workers breathe. Air sampling results are monitored by the radiation safety function to evaluate the effectiveness of personnel exposure controls.

Filters from air samplers are changed weekly during normal operating periods or at more frequent intervals following the detection of an event that may have released airborne contamination.

Air samplers may be equipped with a vacuum gage to indicate flow rate of air sampled. Air samplers flow indicators are calibrated on 6 months intervals.

Routine air sampling is supplemented by portable air sample surveys as required to evaluate non-routine activities or breaches in containment.

4.4 CONTAMINATION CONTROL

4.4.1 SURVEYS

Routine contamination survey monitoring is performed in RCAs and non-RCA's such as adjacent hallways and offices. Removable contamination measurements are made based on the potential for contamination in these areas and operational experience. Survey frequencies are determined by the radiation safety component. Survey results are compared to action guide values as specified in plant procedures and appropriate responses are taken.

For active RCAs, (inactive areas shall be surveyed prior to use and per this chart upon change of active status) the minimum survey frequencies and maximum removable contamination action levels are as follows:

Area	Frequency	Action Limit (dpm α/100 cm^2)	Action Limit (dpm β&γ/100 cm^2)
Radiological Controlled Areas (Floors & Other Readily Accessible Surfaces)	Weekly	100	5000

When contamination levels in excess of action limits are found, mitigating actions are taken within 24 hours.

Personnel exiting the RCAs require contamination surveys for external contamination on clothing and the body. If contamination is found in excess of established limits, the individual attempts self-decontamination at the facilities provided in the change rooms. If decontamination attempts are not successful, decontamination assistance will be provided by the radiation safety function. If skin or personal clothing is still found contaminated above background levels additional control measures are specified by the radiation safety function and the individual may not leave the area without prior approval of the radiation protection function.

4.4.2 ACCESS CONTROL

Routine access points to RCAs are established. Instructions controlling entry and exit from radiological controlled areas are typically posted at the entry points. Personnel survey meters are provided at the exit of the radiological controlled areas. Alternate access points to RCAs may be established for specific activities that are not accommodated by use of routine access points. Such access is governed by approved procedures, or Radiation Work Permits, which establish controls to prevent the spread of contamination.

4.4.3 PROTECTIVE CLOTHING

Protective clothing is provided to persons who are required to enter the RCAs where personnel contamination potential exists as determined by the radiation safety function. The amount and type of protective clothing required for a specific area or operation is determined by operational experience and the contamination potential. Available clothing includes caps, hoods, laboratory coats, coveralls, safety glasses, shoe covers, rubber and cloth gloves. Other specialized equipment can be employed if needed.

The protective clothing is removed upon exit from the contaminated area at the controlled area boundary.

4.5 EXTERNAL EXPOSURE

Deep-dose equivalent and shallow-dose equivalent from external sources of radiation are determined by individually assigned dosimeters. The radiation safety function makes a determination to issue personnel dosimetry to individuals based on work area surveys, occupancy time, or other exposure information such as area monitor results. Personnel dosimeters are processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited vendor. The capability exists to process dosimeters expeditiously if there is an indication of an exposure in excess of established action guides. Action guides for external exposures are documented in plant procedures. Radiation exposure action levels are specified in Section 4.8.

External exposures may be calculated by the radiation safety function on the basis of data obtained by investigation when the results of individual monitoring are unavailable or are invalidated by unusual exposure conditions.

4.6 INTERNAL EXPOSURE

Intakes are assigned to individuals based upon one or more types of measurements as follows: air sampling, invivo or whole body counting. Intakes are converted to committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) for the purposes of limiting and recording occupational doses. Action levels are established in plant procedures to prevent an individual from exceeding the occupational exposure limits specified in 10 CFR 20. Radiation exposure action levels are specified in Section 4.8.

4.6.1 IN VIVO PROGRAM

An in vivo program is available to evaluate the intake of alpha emitting radionuclides. Analysis may be performed on an as needed 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. In vivo monitoring may also be used to monitor individuals involved in non-routine operations or incidents.

4.6.2 WHOLE BODY COUNTING

VNC employees are scheduled to receive a whole body count as required by procedure. Baseline and termination counts are performed when feasible. An investigation shall be initiated if a whole body count result indicates an intake in excess of 10% of the applicable ALI.

4.7 SUMMING INTERNAL AND EXTERNAL EXPOSURE

Internal and external exposures determined as described in the preceding sections of this application are summed in accordance with the requirements of 10 CFR 20 for the purposes of limiting occupational doses and recording individual monitoring results.

4.8 ACTION LEVELS FOR RADIATION EXPOSURES

Work activity restrictions will be imposed when an individual’s exposure exceeds 80% of the applicable 10 CFR 20 limit.

4.9 RESPIRATORY PROTECTION PROGRAM

The respiratory protection program shall be conducted in accordance with the applicable portions of 10 CFR 20. Respiratory protection equipment specifically approved by the National Institute for Occupational Safety and Health (NIOSH) or equivalent is utilized.

4.9.1 QUALIFICATIONS OF RESPIRATOR USERS

Individuals designated to use respiratory protection equipment are evaluated by a physician or other licensed health care professional and periodically thereafter at a frequency specified by the medical function to determine if the individual is medically fit to use respiratory protection devices. If there are no medical restrictions precluding respirator use, the individual is provided respiratory training and fitting by a qualified instructor. Additional training on the use and limitations of self-contained breathing devices is provided to individuals that may be required to use them.

An adequate fit is determined for all face-sealing respirators using either a quantitative fit test method or a qualitative method. Qualitative fit testing is acceptable if (1) it is capable of verifying a fit factor of 10 times the assigned protection factor (APF) for face pieces operated in a negative pressure mode or (2) it is capable of verifying a fit factor of >100 for face pieces operated in a positive pressure mode. Mask fits are re-evaluated as necessary typically on an annual basis.

4.9.2 RESPIRATORY PROTECTION EQUIPMENT

Only NIOSH approved or equivalent respiratory protection equipment is utilized. Protection factors specified in 10 CFR 20, Appendix A are used for selecting the proper equipment and estimating personnel exposures.

4.9.3 EQUIPMENT MAINTENANCE

Respiratory protection equipment is cleaned, serviced, tested and inspected in accordance with the instructions specified by the manufacturer per the NIOSH certification and 10 CFR 20 for each respiratory protection device. Equipment maintenance is conducted in accordance with the applicable portions of 10 CFR 20.

4.10 INSTRUMENTATION

Appropriate radiation detection instruments are available in sufficient number to ensure adequate radiation surveillance can be accomplished. Selection criteria of portable and laboratory counting equipment are based on the types of radiation detected, maintenance requirements, ruggedness, interchangeability and upper and lower limits of detection capabilities. The radiation safety function annually reviews the appropriateness of the types of instruments being used for each monitoring function.

4.10.1 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	Alpha Proportional	3×10^{-15} µCi/cc
Stack Samples	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/cc
	Beta Proportional	3×10^{-6} µCi/cc
Water (Retention Basin)	Alpha Proportional	3×10^{-8} µCi/cc
	Beta Proportional	5×10^{-8} µCi/cc

*Typical value based on standard sample size and counting times.

4.10.2 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	0-500,000 cpm, beta-gamma
2. Ionization Chamber (low energy)	0-300 mrad/h, beta-gamma
3. Ionization Chamber (CP)	1-250,000 mR/h, gamma 4-1,000,000 mrad/h, beta
4. Ionization Chamber (gas multiplication)	1-1,000,000 mR/h, gamma 20-20,000,000 mrad/h, beta
5. Geiger Tube (telescopic)	1-1,000,000 mR/h, gamma
6. Micro-R Meter	0-5,000 µR/h, gamma
7. Scintillation Counter Sodium Iodide (Tl)	0-500,000 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

4.10.3 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 dosimeters located on the VNC site and at its perimeter. Four permanent environmental air sample stations also are located on site.

4.10.4 CALIBRATION

Portable instrumentation is calibrated before initial use, after major maintenance, and on a routine basis with a maximum interval of 12-months between calibrations.

Prior to each use, operability checks are performed on monitoring and laboratory counting instruments. The background and efficiency of laboratory counting instruments are determined on a daily basis when in use.

4.11 RADIATION PROTECTION PROCEDURES

A system of Vallecitos Safety Standards establishes the site radiation and criticality protection and regulatory compliance programs. The manager of the Regulatory

Compliance function issues the standards with review and comment of the managers of the major organizational components located on the site.

4.11.1 PERSONNEL WORK RULES

Requirements are established to prevent or minimize the hazards of radioactivity and radioactive materials. Food storage and consumption (including candy or beverages), the use of cigarettes, or the application of cosmetics are prohibited in posted radioactive materials areas where removable contamination above limits is present on surfaces of floors, walls, plant equipment, or furniture. Approval by persons responsible for radiation protection may be granted for these activities in a posted radiation area, which is shown by the survey to be free from general removable contamination and conditions are unchanging. Food containers may not be used for storing or handling radioactive material.

CHAPTER 5.0

NUCLEAR CRITICALITY SAFETY

5.1 NUCLEAR CRITICALITY SAFETY PROGRAM MANAGEMENT

5.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

The Double Contingency Principle as identified American National Standard ANSI/ANS-8.1 (1998) is the fundamental technical basis for design and operation of processes within the VNC operations where fissile materials are present. As such, "process designs shall incorporate sufficient margins of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." For each process that has accident sequences that could result in an inadvertent nuclear criticality, a defense of one or more system parameters provided by at least two independent controls is documented in the criticality safety analysis (CSA), which is reviewed and enforced.

The established design criteria and nuclear criticality safety reviews are applicable to:

- All new and existing processes, facilities or equipment that process, store, transfer or otherwise handle fissile materials, and
- Any change in existing processes, facilities or equipment that may have an impact on the established basis for nuclear criticality safety.

5.1.2 EVALUATION OF CRITICALITY SAFETY

5.1.2.1 Changes to Facility

As part of the design of new facilities or significant additions or changes in existing facilities, Area Managers provide for the evaluation of nuclear hazards, chemical hazards, hydrogenous content of materials (including firefighting materials), and mitigation of inadvertent unsafe acts by individuals. Specifically, when criticality safety considerations are impacted by these changes, the approval to operate new facilities or make significant changes, modification, or additions to existing facilities is documented in accord with established facility practices and conform to the configuration management process described in Chapter 11.

Change Authorizations (CAs) are processed in accordance with written plant configuration management practices described in Chapter 11. CAs which establish or involve a change in existing criticality safety parameters require a member of the criticality safety function or their delegate to disposition the proposed change with respect to the need for a criticality safety analysis.

If an analysis is required, the change is not placed into operation until a criticality safety analysis is complete and other preoperational requirements are fulfilled in accordance with established configuration management practices.

5.1.2.2 Role of the Criticality Safety Function

Qualified personnel as described in Chapter 2 assigned to the criticality safety function determine the basis for safety for processing fissile material. Assessing both normal and credible abnormal conditions, criticality safety personnel specify functional requirements for criticality safety controls commensurate with design criteria and assess control reliability. Responsibilities of the criticality safety function are described in Chapter 2.2.1.4.

5.1.3 OPERATING PROCEDURES

Procedures that govern the handling of fissile material are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 11.

5.1.4 POSTING AND LABELING

5.1.4.1 Posting of Limits and Controls

Nuclear criticality safety requirements for each major process system defined by the criticality safety function are made available to work stations in the form of written or electronic operating procedures, and/or clearly visible Criticality Limit Area (CLA) postings.

Posting may refer to the placement of signs, marking of floor areas or similar methods to summarize key criticality safety requirements and limits, to designate approved work and storage areas, or to provide instructions or specific precautions to personnel such as:

- Limits on material types and forms.
- Allowable quantities by weight or number.
- Allowable enrichments.
- Required spacing between units.
- Control limits (when applicable) on quantities such as moderation, density, or presence of additives.
- Critical control steps in the operation.

Storage postings are located in conspicuous places and include as appropriate:

- Material type.
- Container type.
- Number of items allowed.
- Mass, volume, moderation, and/or spacing limits.

Additionally, when administrative controls or specific actions/decisions by operators are involved, postings include pertinent requirements identified within the criticality safety analysis.

5.1.4.2 Labeling

Where practical, process containers of fissile material are labeled such that the material type, U-235 enrichment, and gross weights can be clearly identified or determined. Deviations from this process include, but are not limited to: large process vessels, fuel rods, shipping containers, waste boxes/drums, contaminated items, samples, containers of 1 liter volume or less, or other containers where labeling is not practical, or where the enrichment of the material contained is unknown (e.g. cleanout material).

5.2 ORGANIZATION AND ADMINISTRATION

5.2.1 GENERAL ORGANIZATION AND ADMINSTRATION METHODS

Information regarding General Organization and Administration is described in Chapter 2.

5.2.2 CRITICALITY SAFETY FUNCTION

Specific details of the Criticality Safety Function (CSF) responsibilities and qualification requirements for the manager and members are described in Chapter 2.

Criticality safety function personnel are specifically authorized to perform assigned responsibilities in Chapter 2. Nuclear criticality safety function personnel and their delegates have authority to shutdown potentially unsafe operations.

5.3 MANAGEMENT MEASURES

5.3.1 GENERAL CONFIGURATION MANAGEMENT

In accordance with ANSI/ANS-8.19 (2005), the Criticality Safety Analysis (CSA) is a collection of information that “provides sufficient detail and clarity, to allow independent judgment of the results.” The CSA documents the physical/safety basis for the establishment of the controls. The CSA is a controlled element of the plant configuration management process defined in Chapter 11.

Documented CSAs establish the nuclear criticality safety bases for a particular system under normal and credible abnormal conditions. A CSA is prepared or updated for new or significantly modified fissile units, processes, or facilities within VNC in accordance with established design Change Authorization (CA) system described in Chapter 11.

5.3.2 CSF CONFIGURATION MANAGEMENT

5.3.2.1 Auditing, Assessing and Upgrading the CSF Program

Details of the facility criticality safety audit program are described in Chapter 11. Criticality safety audits are conducted and documented in accordance with a written procedure and personnel approved by the criticality safety function. Findings, recommendations, and observations are reviewed with the Regulatory Compliance function manager to determine if other safety impacts exist. NCS audit findings are transmitted to Area Managers for appropriate action and tracked until closed.

Audits and assessments of the processes and associated conduct of operations within the facility, including compliance with operating procedures, postings, and administrative guidelines, are also conducted as described in Chapter 11.

The criticality safety function or other designated personnel independent of the VNC operations organization in accordance with Chapter 11 conduct nuclear criticality safety program reviews on a scheduled basis. This provides a means for independently assessing the effectiveness of the components of the nuclear criticality safety program.

5.3.2.2 Modifications to Operating and Maintenance Procedures

Procedures that govern the handling of enriched uranium are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through processes such as: postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 11.

5.3.2.3 Criticality Accident Alarm System (CAAS) Design and Performance Requirements

The criticality accident alarm system (CAAS) radiation monitoring unit detectors are located in required locations within the facility and are selected for the type of radiation detected, the mode of detection, the alarm signal and the system dependability. Individual unit detectors are located to assure compliance with appropriate requirements of 10CFR70.24(a)(1).

The installed CAAS initiate sounding of the evacuation alarm. Employees are trained in recognizing the evacuation signal. This system and proper response protocol is described in the VNC site emergency procedures.

The nuclear criticality accident alarm system is maintained through routine response checks and scheduled functional tests conducted in accordance with internal procedures. In the event of loss of normal power, emergency power is automatically supplied to the criticality accident alarm system.

(i) In the event that CAAS coverage is lost in an area required to have a monitor alarm system, compensatory measures such as limiting personnel access, halting special nuclear material movement or installing temporary detection equipment are used as an interim measure until the system is restored (by procedures).

5.3.2.4 Corrective Action Program

VNC commits to maintain a system to identify, track, investigate and implement corrective action for abnormal events (unusual incidents) as described in Chapter 11.

5.3.2.5 CSF Records Retention

Records of criticality safety analyses are maintained in sufficient detail and form to permit independent review and audit of the method of calculation and results. Such records are retained during the conduct of the activities and for six months following cessation of such activities to which they apply or for a minimum of three years.

A CSA is prepared or updated for each new or significantly modified unit or process system within VNC in accordance with established configuration management control practices defined in Chapter 11.

5.4 METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 CONTROL PRACTICES

Criticality safety analyses identify specific controls necessary for the safe and effective operation of a process. Prior to use in any enriched uranium process, nuclear criticality safety controls are verified against criticality safety analysis criteria. The configuration management program described in Chapter 11 implement performance based management of process requirements and specifications that are important to nuclear criticality safety.

5.4.1.1 Verification Program

The purpose of the verification program is to assure that the controls selected and installed fulfill the requirements identified in the criticality safety analyses. Processes are examined in the "as-built" condition to validate the safety design and to verify the installation.

Operations personnel are responsible for subsequent verification of controls through the use of functional testing or verification. Control calibration and routine maintenance are normally provided by the instrument and calibration and/or maintenance functions. Verification and maintenance activities are performed per established facility practices documented through the use of forms and/or computer tracking systems.

5.4.1.2 Maintenance Program

The maintenance program is described in Chapter 11.

5.4.2 MEANS OF CONTROL

The relative effectiveness and reliability of controls are considered during the criticality safety analysis process. Passive Engineered Controls are preferred over all other system controls and are utilized when practical and appropriate. Active Engineered Controls are the next preferred method of control. Administrative Controls are least preferred, however, augmented administrative controls are preferred over simple administrative controls. A criticality safety control must be capable of preventing a criticality accident independent of the operation or failure of any other criticality control for a given credible initiating event.

5.4.2.1 Passive Engineered Controls

A device that uses only fixed physical design features to maintain safe process conditions without any required human action. Assurance is maintained through specific periodic inspections or verification measurement(s) as appropriate.

5.4.2.2 Active Engineered Controls

A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action. Assurance is maintained through specific periodic functional testing as appropriate. Active engineered controls are designed to be fail-safe (i.e., meaning failure of the control results in a safe condition).

5.4.2.3 Administrative Controls

Either an augmented administrative control or a simple administrative control as defined herein:

- Augmented Administrative Control – A procedurally required or prevented human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions or otherwise add substantial assurance of the required human performance.
- Simple Administrative Control – A procedural human action that is prohibited or required to maintain safe process conditions.

Use of administrative controls is limited to situations where passive and active engineered controls are not practical. Administrative controls may be proactive (requiring action prior to proceeding) or reactive (proceeding unless action occurs). Proactive administrative controls are preferred. Assurance is maintained through periodic verification, audit, and training.

5.4.3 SPECIFIC PARAMETER LIMITS

The **safe mass** limit values of Table 5.1 below are specifically designated for use at VNC. Application of these limits is restricted to systems where the neutron reflection present does not exceed that due to full water reflection.

Other subcritical limits used shall be derived using approved analytic methods described in this chapter.

NOTE: The safe mass values specified for UO_2 of specific compounds may be adjusted when applied to other compounds by the formula:

kgs X = (kgs UO_2 • 0.88) / f

where,	kgs X	= safe mass value of compound 'X'
	kgs UO_2	= safe mass value for UO_2
	0.88	= wt. % U in UO_2
	f	= wt. % U in compound X

Table 5.1 Safe Mass Limits by Material Type

Material Form/Shape	***Safe Mass Limit for $k_{calc} + 2\sigma$ - bias* ≤ 0.95***
Homogeneous $U(5)O_2$ Sphere	28 kg UO_2 *(1230 g ^{235}U)*
Homogeneous $U(5)O_2$ Hemisphere	32 kg UO_2 *(1410 g ^{235}U)*
Sintered Pellets Sphere [$U(5)O_2$]	32 kg Pellets
Sintered Pellets Hemisphere [$U(5)O_2$]	36 kg Pellets
Optimal Heterogeneous $U(5)O_2$ Sphere	24 kg UO_2 *(1050 g ^{235}U)*
Optimal Heterogeneous $U(5)O_2$ Hemisphere	27 kg UO_2 *(1180 g ^{235}U)*
U-235 Sphere	575 g ^{235}U
U-235 Hemisphere	650 g ^{235}U

- Includes Bias Uncertainty.

Where applicable, an **equivalence factor** is used to calculate the U-235 equivalent fissile mass of fissile material for mixtures of uranium and plutonium isotopes. The U-235 equivalent fissile mass is used when implementing U-235 derived mass limits. Use of this equivalence factor assures that the calculated U-235 equivalent mass of fissile isotopes is no more reactive than the same mass of U-235. For Criticality Limit Areas (CLAs) in which both uranium and plutonium are present the following equation may be used to enforce U-235 derived mass limits (grams).

$$^{235}U\ Equivalent\ Fissile\ Mass = 1.7 \cdot Pu\ mass + {}^{235}U\ mass$$

where,

Pu mass = *Total Mass of Plutonium (All isotopes)*

^{235}U *mass* = *Mass of U-235*

A **safe batch** may also be used to establish maximum process mass limit in lieu of a safe mass. A safe batch means an accumulation of fissile material, which is 45% of the critical mass established considering enrichment, full water reflection, and optimum water moderation consistent with the form of material.

A **subcritical area** means a physically identified area or location involving special nuclear materials in quantities of less than the established Table 5.1 safe uranium mass limits (grams uranium-235 equivalent); 520 grams of uranium-233, or 450 grams of plutonium. A subcritical area is considered neutronically isolated when it meets the spacing requirements from other areas which special nuclear material is handled, used, or stored; or an unrelated building or structure which meets the criteria of 10CFR70.24(a).

5.4.4 CONTROL PARAMETERS

Nuclear criticality safety is achieved by controlling one or more parameters of a system within established subcritical limits. The internal configuration management process may require nuclear criticality safety staff review of proposed new or modified processes, equipment, or facilities to ascertain impact on controlled parameters associated with the particular system. Assumptions relating to processes, equipment, or facility operations including material composition, function, and operation, including upset conditions, are justified, documented, and independently reviewed.

Identified below are specific control parameters that may be considered during the CSF review process:

5.4.4.1 Geometry

Geometry may be used for nuclear criticality safety control on its own or in combination with other control methods. Favorable geometry is based on limiting dimensions of defined geometrical shapes to established subcritical limits. Structure and/or neutron absorbers that are not removable constitute a form of geometry control. At VNC, favorable geometry is developed conservatively assuming unlimited water or concrete equivalent reflection, optimal hydrogenous moderation, worst credible heterogeneity, and maximum credible enrichment to be processed. Examples include cylinder diameters, slab thickness, and sphere diameters.

Geometry control systems are analyzed and evaluated allowing for fabrication tolerances and dimensional changes that may likely occur through corrosion, wear, or mechanical distortion. In addition, these systems include provisions for periodic inspection if credible conditions exist for changes in the dimensions of the equipment that may result in the inability to meet established nuclear criticality safety limits.

5.4.4.2 Mass

Mass control may be used for a nuclear criticality safety control on its own or in combination with other control methods. Mass control may be utilized to limit the quantity of uranium within specific process operations or vessels and within storage, transportation, or disposal containers. Analytical or non-destructive methods may be employed to verify the mass measurements for a specific quantity of material.

Establishment of mass limits involves consideration of potential moderation, reflection, geometry, spacing, and material concentration. The criticality safety analysis considers normal operations and credible process upsets in determining actual mass limits for the system and for defining additional controls. When only administrative controls are used for mass controlled systems, double batching is considered to ensure adequate safety margin.

5.4.4.3 Moderation

Moderation control may be used for nuclear criticality safety control on its own or in combination with other control methods. When moderation is used in conjunction with other control methods, the area is posted as a ‘moderation control area’. For situations where moderation is not intentionally introduced as part of the process, the required number of controls for each credible failure mode must be established in accordance with the double contingency principle.

5.4.4.4 Concentration (or Density)

Concentration control may be used for nuclear criticality safety control on its own or in combination with other control methods. Concentration controls are established to ensure that the concentration level is maintained within defined limits for the system. When concentration is the only parameter controlled to prevent criticality, concentration may be controlled by two independent combinations of measurement and physical control, each physical control capable of preventing the concentration limit from being exceeded.

5.4.4.5 Neutron Absorber

Neutron absorbing materials may be utilized to provide a method for nuclear criticality safety control for a process, vessel or container. Stable compounds such as boron carbide fixed in a matrix such as aluminum or polyester resin; elemental cadmium clad in appropriate material; elemental boron alloyed stainless steel, or other solid neutron absorbing materials with an established dimensional relationship to the fissionable material are recommended. The use of neutron absorbers in this manner is defined as part of a passive engineered control.

For fixed neutron absorbers used as part of a geometry control, the following requirements apply:

- The composition of the absorber are measured and documented prior to first use.
- Periodic verification of the integrity of the neutron absorber system subsequent to installation is performed on a scheduled basis approved by the criticality safety function. The method of verification may take the form of traceability (i.e. serial number, QA documentation, etc.), visual inspection or direct measurement.

5.4.4.6 Spacing (or Unit Interaction)

Criticality safety controls based on isolation or interacting unit spacing. Units may be considered effectively non-interacting (isolated) when they are separated by either of the following:

- 12-inches of full density water equivalent, or
- The larger of 12-foot air distance or the greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers.

Techniques which produce a calculated effective multiplication factor of the entire system (e.g., validated Monte Carlo or S_n Discrete Ordinates codes) may be used and documented in the criticality safety analysis.

5.4.4.7 Material Composition (or Heterogeneity)

The criticality safety analysis for each process determines the effects of material composition (e.g., type, chemical form, physical form) within the process being analyzed and identifies the basis for selection of compositions used in subsequent system modeling activities.

It is important to distinguish between homogeneous and heterogeneous system conditions. Heterogeneous effects within a system can be significant and therefore must be considered within the criticality safety analysis when appropriate. Evaluation of systems where the particle size varies takes into consideration effects of heterogeneity appropriate for the process being analyzed.

5.4.4.8 Reflection

Most systems are designed and operated with the assumption of 12-inch water or optimum reflection. However, subject to approved controls that limit reflection, certain system designs may be analyzed, approved, and operated in situations where the analyzed reflection is less than optimum.

In criticality safety analysis, the neutron reflection properties of the credible process environment are considered. For example, reflectors more effective than water (e.g., concrete) are considered when appropriate.

5.4.4.9 Enrichment

Enrichment control may be utilized to limit the percent U-235 within a process, vessel, or container, thus providing a method for nuclear criticality safety control. Active engineered or administrative controls are required to verify enrichment and to prevent the introduction of uranium at unacceptable enrichment levels within a defined subsystem within the same area. In cases where enrichment control is not utilized, the maximum credible area enrichment is utilized in the criticality safety analysis.

5.4.4.10 Process Characteristics

Within certain manufacturing operations, credit may be taken for physical and chemical properties of the process and/or materials as nuclear criticality safety controls. Use of process characteristics is predicated upon the following requirements:

- The bounding conditions and operational limits are specifically identified in the criticality safety analysis and, are specifically communicated, through training and procedures, to appropriate operations personnel.
- Bounding conditions for such process and/or material characteristics are based on established physical or chemical reactions, known scientific principles, and/or facility-specific experimental data supported by operational history.
- The devices and/or procedures, which maintain the limiting conditions, must have the reliability, independence, and other characteristics required of a criticality safety control.

5.4.5 ANALYSIS METHODS

5.4.5.1 Keff Limit

Validated computer analytical methods are used to evaluate individual system units or potential system interaction. When these analytical methods are used, it is required that the effective neutron multiplication factors, including applicable bias and bias uncertainty corrections, for credible process upset (accident) conditions are less than or equal to the established Upper Subcritical Limit (USL), that is:

$$k_{eff} + 2\sigma \leq USL$$

Normal operating conditions include maximum credible conditions expected to be encountered when the criticality control systems function properly. Credible process upsets include anticipated off-normal or credible accident conditions and must be demonstrated to be critically safe. The sensitivity of key parameters with respect to the effect on Keff is evaluated for each system such that adequate criticality safety controls are defined for the analyzed system.

5.4.5.2 Analytical Methods

Methodologies currently employed by the nuclear criticality safety function include hand calculations utilizing published experimental data (e.g., ARH-600 handbook), Solid Angle methods (e.g., SAC code), and Monte Carlo codes (e.g., GEMER, GEKENO) which utilize stochastic methods to approximate a solution to the 3-D neutron transport equation. Additional Monte Carlo codes (e.g., such as SCALE, MCNP) or S_n Discrete Ordinates codes (e.g., ANISN, DORT, TORT or the DANTSYS code package) may be used after validation as described in Section 5.4.5.3 below has been performed.

5.4.5.3 Validation Techniques

The validity of the calculation method (computer code and nuclear cross-sectional data set) used for the evaluation of nuclear criticality safety must be demonstrated and sufficiently documented in a validation report according to written procedures to allow understanding of the methodology by a qualified and knowledgeable individual. The validation of the computer code will be performed consistent with the guidance outlined in section 4.3 of ANSI/ANS-8.1-1998 and include the code calculation bias, bias uncertainty, and the minimum margin of subcriticality using well-characterized and adequately documented critical experiments.

The following definitions apply to the documented validation report(s):

Bias - the systematic difference between the calculated results and the experimentally measured values of k_{eff} for a fissile system.

Bias Uncertainty - the integrated uncertainty in the experimental data, calculation methods and models, and should be estimated by a valid statistical analysis of calculated k_{eff} values for the critical experiments.

Minimum Margin of Subcriticality (MMS) - an allowance for any unknown (or difficult to identify or quantify) errors or uncertainties in the method of calculating k_{eff}, that may exist beyond those which have been accounted for explicitly in calculating the bias and bias uncertainty.

Validation methodologies are consistent with the guidance in ANSI/ANS 8.1-1998 and ANSI/ANS 8.24-2007, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations. In accordance with the requirements of these national consensus standards, the criteria to establish subcriticality requires the calculated k_{eff} to be less than or equal to an established USL, as presented in the validation report, for a system or process to be considered subcritical. The validation of the calculation method and cross-sections considers a diverse set of parameters that include, but are not limited to:

- Fuel enrichment, composition, and form of associated uranium materials,
- Homogeneity or heterogeneity of the system,
- Presence of neutron absorbing materials,
- Characterization of the neutron energy spectra,
- Types of neutron moderating materials,
- Types of neutron reflecting materials,
- Degree of neutron moderation in the system (such as, H/fissile atom ratio), and
- Geometry configuration of the system (such as, shape, size, spacing, reflector).

The selection of critical experiments for the criticality safety code validation for each identified area of applicability incorporates the following considerations:

- Critical experiments are assessed for completeness, accuracy, and applicability to the VNC operations prior to its selection and use as a critical benchmark.

- Critical experiments are selected to cover the spectrum of parameters spanning the range of normal and credible abnormal conditions anticipated for past, current, and future analyzed uranium systems for modeled systems.
- Critical experiments are drawn from multiple series and sources of critical experiments to minimize systematic error. The range of parameters characterized by selected critical experiments is used to define the area of applicability for the code.

The calculation bias, bias uncertainty and USL over the defined area of applicability are determined by statistical methods as follows:

- The normality of calculated keff values based on a set of critical experiments similar in the system configuration and nuclear characteristics are verified prior to the estimation of the bias and bias uncertainty.
- The calculation bias is determined either as a constant, if no trends exist or as a smooth and well-behaved function of selected characteristic parameters (e.g., hydrogen-to-fissile ratio, etc.) by regression analysis if trends exist with parameters statistically significant over the area of applicability. The bias is applied over its negative range and assigned a value of zero over its positive range.
- The bias uncertainty is estimated by a confidence interval of uniform width that ensures that there is at least a 95% level of confidence that a future keff value for a critical system will be above the lower confidence limit.
- The USL is established based on confidence interval with MMS for the area of applicability as follows:

$$USL = 1 + bias - bias\ uncertainty - MMS$$

At VNC, a minimum MMS = 0.05 shall be used to establish the acceptance criteria for criticality calculations.

The following acceptance criteria, considering worst-case credible accident conditions, must be satisfied when using k_{eff} calculations by Monte Carlo methods to establish subcritical limits for the VNC facility:

$$k_{eff} + 2\sigma \leq USL$$

where σ is the standard deviation of the k_{eff} value obtained with Monte Carlo calculation.

If parameters needed for anticipated applications is beyond the range of the critical benchmark experiments, the Area of Applicability (AOA) may be extended by extrapolation using the established trends in the bias. In general, if the extrapolation is too large, new factors that could affect the bias may be introduced as the physical phenomena in the system or process change. For conservatism, the extrapolation should be based on the following rules:

- The extrapolation should not result to a large underlying physics or neutronic behavior change in the anticipated application. If there is a rapid or non-conservative change in bias in the vicinity of the AOA range endpoints of a trending parameter, extra safety margin should be needed. Otherwise, critical experiments should be added for further justification.
- Statistical methods should be used to ensue that the extrapolation is not large. The leverage statistic, a measure of the distance between the extrapolation point x for a predication and the mean of trending parameter values in the critical benchmark data set can be used to determine if an extrapolation using the regression model is acceptable when making predications at x.

5.4.5.4 Computer Software & Hardware Configuration Control

The software and hardware used within the criticality safety calculation system is configured and controlled in accordance with approved procedures. Software changes are conducted in accordance with an approved configuration management program described in Chapter 11 that addresses both hardware and software qualification.

Software designated for use in nuclear criticality safety is compiled into working code versions with executable files that are traceable by length, time, date, and version. Working code versions of compiled software are validated against critical experiments using an established methodology with the differences in experiment and analytical methods being used to calculate bias and uncertainty values to be applied to the calculational results. Each individual workstation is verified to produce results identical to the development workstation prior to use of the software for criticality safety calculations demonstrations on the production workstation.

Modifications to software and nuclear data that may affect the calculation logic require re-validation of the software. Modifications to hardware or software that do not affect the calculation logic are followed by code operability verification, in which case, selected calculations are performed to verify identical results from previous analyses. Deviations noted in code verification that might alter the bias or uncertainty requires re-validation of the code prior to release for use.

5.4.5.5 Criticality Safety Analysis (CSA)

The scope and content of any particular CSA reflects the needs and characteristics of the system being analyzed and typically includes applicable information requirements as follows:

- **Scope** - This element defines the stated purpose of the analysis.
- **General Discussion** - This element presents an overview of the process that is affected by the proposed change. This section includes as appropriate; process description, flow diagrams, normal operating conditions, system interfaces, and other important to design considerations.
- **Criticality Safety Controls/Bounding Assumptions** - This element defines a minimum of two criticality safety controls that are imposed as a result of the analysis. This section also clearly presents a summary of the bounding assumptions used in the analysis. Bounding assumptions include; worst credible contents (e.g., material composition, density, enrichment, and moderation), boundary conditions, inter-unit water, and a statement on assumed structure. In addition, this section includes a statement that summarizes the interface considerations with other units, subareas and/or areas.
- **Model Description** - This element presents a narrative description of the actual model used in the analysis. An identification of both normal and credible upset (accident condition) model file naming convention is provided. Key input listings and corresponding geometry plot(s) for both normal and credible upset cases are also provided.
- **Calculation Results** - This element identifies how the calculations were performed, what tools or reference documents were used, and when appropriate, presents a tabular listing of the calculation result and associated uncertainty (e.g., Keff + 2σ) results as a function of the key parameter(s) (e.g., wt. fraction H_2O). When applicable, the assigned bias of the calculation is also clearly stated and incorporated into both normal and/or accident limit comparisons
- **Safety During Upset Conditions** - This element presents a concise summary of the upset conditions considered credible for the defined unit or process system. This section includes a discussion as to how the established nuclear criticality safety limits are addressed for each credible process upset (accident condition) pathway.

- **Specifications and Requirements for Safety** - When applicable, this element presents both the design specifications and the criticality safety requirements for correct implementation of the established controls. These requirements are incorporated into operating procedures, training, maintenance, quality assurance as appropriate to implement the specifications and requirements.

- **Compliance** - This element concludes the analysis with pertinent summary statements and includes a statement regarding license compliance.

- **Verification** - a senior engineer approved by the criticality safety function and who was not involved in the analysis verifies each criticality safety analysis.

- **Appendices** - Where necessary, a summary of information ancillary to calculations such as parametric sensitivity studies, references, key inputs, model geometry plots, equipment sketches, useful data, etc., for each defined system is included.

5.4.5.6 Technical Reviews

Independent technical reviews of proposed criticality safety control limits specified in criticality safety analyses are performed. A senior engineer within the criticality safety function is required to perform the independent technical review. The independent technical review consists of a verification that the neutronics geometry model and configuration used adequately represent the system being analyzed. In addition, the reviewer verifies that the proposed material characterizations such as density, concentration, etc., adequately represent the system. The reviewer also verifies that the proposed criticality safety controls are adequate.

The independent technical review of the specific calculations and computer models is performed using one of the following methods:

- Verify the calculations with an alternate computational method.

- Verify methods with an independent analytic approach based on fundamental laws of nuclear physics.

- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations.

- Verify the calculations using specific checks of the computer codes used, as well as, evaluations of code input and output.

Based on one of these prescribed methods, the independent technical review provides a reasonable measure of assurance that the chosen analysis methodology and results are correct.

5.4.6 REFERENCES

5-1. 10 CFR 70, *Domestic Licensing of Special Nuclear Material*, U.S. Nuclear Regulatory Commission, 2008.

5-2. Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, U.S. Nuclear Regulatory Commission, Revision 1, October 2005.

5-3. ANSI/ANS 8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, American Nuclear Society, January 1998.

5-4. ANSI/ANS 8.3-1997 (R2003), *Criticality Accident Alarm System*, American Nuclear Society, January 1997.

5-5. ANSI/ANS 8.19-2005, *Administrative Practice for Nuclear Criticality Safety*, American Nuclear Society, January 2005.

5-6. ANSI/ANS 8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*, American Nuclear Society, January 1995.

5-7. ANSI/ANS 8.22-1997 (R2006), Nuclear Criticality Safety Based on Limiting and Controlling Moderators, American Nuclear Society, January 1997.

5-8. ANSI/ANS 8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*, American Nuclear Society, January 1997.

5-9. ANSI/ANS 8.24-2007, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations, American Nuclear Society, 2007.

5-10. ANSI/ANS 8.26-2007, Criticality Safety Engineer Training and Qualification Program, American Nuclear Society, June 2007.

5-11. ARH-600, *Criticality Handbook*, R. D. Carter, G. R. Kiel, and K. R. Ridgway, Atlantic Richfield Hanford Co. Report, 1968.

CHAPTER 6.0
CHEMICAL SAFETY

6.1 CHEMICAL SAFETY PROGRAM

The chemical safety program is applicable to the chemicals produced from or coming in contact with licensed materials described in Chapter 1. Guidance regarding the VNC chemical safety program is provided in written, approved procedures.

Hazard evaluations are performed on nuclear and non-nuclear operations within the VNC where the potential exists for hazardous chemicals to be used in such a manner that they could affect the licensed materials. Typically chemicals which come in contact with licensed materials are in small quantities necessary to perform analytical or inspection procedures. This ensures appropriate controls are in place for adequate protection of the general public and safe use by employees, and that the use of chemicals does not create potential conditions that adversely affect the handling of licensed nuclear materials.

Employees using hazardous materials are trained to ensure safe handling, use, and disposal.

6.2 CONTENTS OF CHEMICAL SAFETY PROGRAM

The following management control elements are incorporated into VNC safety program:

6.2.1 CHEMICAL SAFETY

The VNC configuration management process assures that internal process descriptions are sufficiently detailed to allow an understanding of the chemical process hazards. Management assurances for chemical safety are included in Chapter 11.

6.2.2 CHEMICAL APPROVAL / EVALUATION

Prior to new hazardous materials being brought on-site or used in a process, they are approved through the environmental health and safety function and the chemical and fire safety function. The formal approval process typically consists of evaluations of the following potential hazards:

- Physical Hazards
- Health Hazards
- Fire / Explosive Hazards
- Potential Impact on handling of licensed nuclear material

The conclusions of this approval process may dictate the following assurance of chemical process safety:

- New procedures or changes in existing procedures
- Maintenance programs for control related equipment
- Configuration management
- Emergency Planning
- Training

6.2.3 LABELING & IDENTIFICATION

Hazardous materials or conveyance systems are labeled or identified to meet applicable regulations. The proper identification of hazardous materials decreases the likelihood of improper use, handling and disposal reducing potential negative consequences.

6.2.4 EMPLOYEE TRAINING & AWARENESS

Radiation workers receive nuclear safety training and other job related training, which includes safety information related to chemicals associated with nuclear material and chemicals in the area, which could impact the safety of the process.

6.2.5 CONDUCT OF OPERATIONS

Other elements of the chemical safety program are included in Chapter 11, "Management Measures".

CHAPTER 7.0
FIRE SAFETY

Appropriate combinations of fire prevention measures and response capabilities achieve VNC's fire protection. Such measures and capabilities are designed and maintained in accordance with federal, state, and local codes, appropriate industry standards and prudent practices.

7.1 FIRE PROTECTION PROGRAM RESPONSIBILITY

The Emergency Organization is comprised of functional groups capable of assisting and/or advising in the prevention, response to and controlling of emergency situations. The structure of the Emergency Organization is detailed in VNC written procedures.

7.2 FIRE PROTECTION PROGRAM

A fire hazard analysis is incorporated into the VNC's change authorization procedure.

Routine inspection and testing of the fire protection system is conducted by VNC personnel and/or contract personnel in accordance with VNC written procedures.

Responsibility for maintenance, operation, and engineering of the fire protection system and equipment is specified in written, approved VNC procedures.

The fire protection program equipment is maintained as part of the formal, planned preventative maintenance program at VNC.

Review and control of modifications of the facility or processes to minimize fire hazards is part of configuration management described in Chapter 11.

An approved cutting and welding procedure known as a hot work permit is provided to control welding and torch cutting activities as a means of fire prevention.

Basic fire protection training is provided as needed. Additionally, new employees and contractors are trained during orientation programs. A system is provided to enable reporting of fire incidents to the emergency response organization. Facilities with potential fire hazards are equipped with appropriate fire detection and/or suppression systems.

7.3 ADMINISTRATIVE CONTROLS

Administrative controls are described in Chapter 11.

7.4 BUILDING CONSTRUCTION

7.4.1 EXISTING BUILDINGS

The existing building's original design is in accordance with the local, state, federal and national codes, standards and/or regulations in effect at the time of construction.

7.4.2 PROCESS FIRE SAFETY

Potential fire hazards are determined, evaluated, and controlled by internal and external personnel using industry accepted methods, analysis, and procedures.

7.5 FIRE DETECTION AND ALARM SYSTEMS

7.5.1 DETECTION DEVICES

Areas where fire or explosion hazards are present, automatic detection equipment is installed. Equipment such as the following is utilized

- Smoke Detectors
- Heat Detectors

7.5.2 ALARMS

- Audible fire alarms are installed in specified locations throughout the site. A continuously manned, central control station is maintained that monitors fire detection system.

7.6 FIRE SUPPRESSION EQUIPMENT

Selection of equipment available for suppression of fire takes into account the severity of the hazard, the type of activity to be performed, the potential consequences of a fire, and the potential consequences of use of the suppression equipment.

Portable fire extinguishers are available and maintained throughout the site.

7.7 FIRE PROTECTION WATER SYSTEM

Commercial mains supply the fire protection water system.

Prime components of the fire protection system are as follows:

- Elevated tank capable of supplying dedicated water to the fire protection system.
- A fire main loops around primary facilities.
- A series of branch headers supplying fire protection water to sectionalized sprinkler system in each building.
- A supervised alarm and warning system providing full time coverage of prime fire protection safety auxiliaries such as sprinkler system supply valve closing indication, sprinkler system water flow, fire pump operations, smoke detection operation, etc.
- Fire hose on reels connected to the primary fire protection system.

7.8 EMERGENCY RESPONSE TEAM

Fire training of the Emergency Response Team is conducted for the response to incipient stage fires in accordance with emergency planning requirements. Outside agency fire departments are contacted for more serious fires such as structural fires.

In order to ensure emergency fire response readiness an annual fire drill is conducted.

CHAPTER 8.0

RADIOLOGICAL CONTINGENCY & EMERGENCY PLAN

VNC is not required to maintain a Radiological Contingency & Emergency Plan based on 10 CFR 70.22(i)(1)(i).

BACKGROUND

10 CFR 70.22(i)(1)(i) provides for an evaluation to be made to determine the need for an RC&EP based on an evaluation showing that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 1 rem effective dose equivalent or an intake of 2 milligrams of soluble uranium.

VNC does not handle or process UF_6 so special nuclear materials are typically insoluble and do not present a significant risk of intake of soluble uranium.

Quantities of material handled in the Building 102 and 103 facilities do not represent a risk of exceeding 1 rem effective dose equivalent to the public.

VNC maintains an active criticality safety program that ensures that any criticality event is highly unlikely.

In order to provide a boundary dose evaluation of a criticality accident, the fuel vault in Building 103 (generally the largest accumulation of fissile material) was selected. It was assumed that double the normal amount of material was in the vault in an optimally moderated and reflected condition resulting in a single burst criticality event. The radiation dose at the nearest site boundary continues to be nominally calculated near 0.35 rem effective dose equivalent and 0.30 rem to the thyroid. These values are significantly below the guidelines of 1 rem effective dose equivalent.

CHAPTER 9.0
ENVIRONMENTAL PROTECTION

9.1 ENVIRONMENTAL PROTECTION PROGRAM & REPORTS

The GE Hitachi Vallecitos Nuclear Center maintains an Environmental Protection Program for the site. A primary purpose of the Environmental Protection Program is to assure that exposures of the public and environment to radiation and hazardous materials used in facility operations are kept As Low As Reasonably Achievable (ALARA).

VNC submits annual environmental reports summarizing the effluent monitoring and environmental surveillance programs. This data demonstrates no significant impact to the environment.

As part of the design of new facilities or significant additions or changes in existing facilities, environmental considerations are assessed in accord with established facility practices. Change Authorizations are processed in accordance with configuration management requirements described in Chapter 11. Change Authorizations, which establish or involve a change in existing environmental controls, require an environmental review and disposition of the proposed change with respect to impact on established environmental protection programs.

9.2 AIR EFFLUENT CONTROLS AND MONITORING

Air effluent control systems are designed and operated to assure compliance with regulatory requirements. Operations that could potentially exhaust radioactive materials have air effluent controls that are monitored by representative stack sampling to demonstrate compliance with regulations. Samples are collected and analyzed so as to be representative of the discharges during production operations. Adequate controls and evaluations are in place to monitor, assess and take necessary protective actions that may be needed for circumstances not explicitly treated. The VNC ventilation and exhaust systems are described in Chapter 4.

Radioactivity in releases of radioactive materials in gaseous effluents from the facility is reported to the NRC on an annual basis.

9.3 LIQUID TREATMENT FACILITIES

A treatment facility, with sufficient capacity and capability to enable treatment, sampling, analysis, and discharge of liquids in accordance with the regulations, is provided and maintained in proper working order for the operation of the Site.

Liquid wastes are treated using techniques such as evaporation, chemical treatment, sedimentation, filtration, ion exchange, solidification and packaging of the concentrates for disposal.

Radioactivities in releases of radioactive materials in liquid effluents from the facility are reported to the NRC on an annual basis.

9.4 SOLID WASTE MANAGEMENT FACILITIES

Solid waste management facilities, with sufficient capability to enable preparation, packaging, storage, and transfers to licensed disposal sites in accordance with the regulations, is provided and maintained in proper operating condition as required to support the operation of VNC.

9.5 PROGRAM IMPLEMENTATION

VNC's environmental monitoring program includes the types of samples and monitored parameters listed in Figure 9.1. Analytical sensitivities (minimum detection levels) are also illustrated in Figure 9.1. Action levels will be included in documented procedures for environmental monitoring parameters as appropriate so that internal review and other actions are initiated. Such action levels provide guidance in assuring compliance within 10 CFR 20 limits. Environmental monitoring sample locations are shown for air (Figure 9.2), vegetation and dosimetry (Figure 9.3), surface water and sanitary/industrial discharge (Figure 9.4), and stream bottoms (Figure 9.5). For monitoring wells found not to contain water at time of sampling, an evaluation is performed by the Radiation Protection function to determine if alternate well sampling data may be used or other assessments will be used. These program elements, analytical sensitivities, and/or locations may be changed in accordance with the Change Authorization procedure described in Chapter 11.

9.6 EVALUATIONS

The Regulatory Compliance function performs a periodic evaluation of vendors contracted to analyze environmental samples. The evaluations consider applicable methods such as “spike” and “replicate sample” submittals.

9.7 OFF-SITE DOSE

Compliance with NRC 10 CFR 20, Subpart D for radiation dose to individual members of the public is demonstrated by assuring that the off-site annual dose to the maximum exposed individual does not exceed 100 mRem. Demonstration of the ALARA constraint in 10 CFR 20.1101(d) for off-site dose projections due to air emissions is demonstrated by assuring that off-site annual dose (due to air emissions of radioactivity) does not exceed 10 mRem.

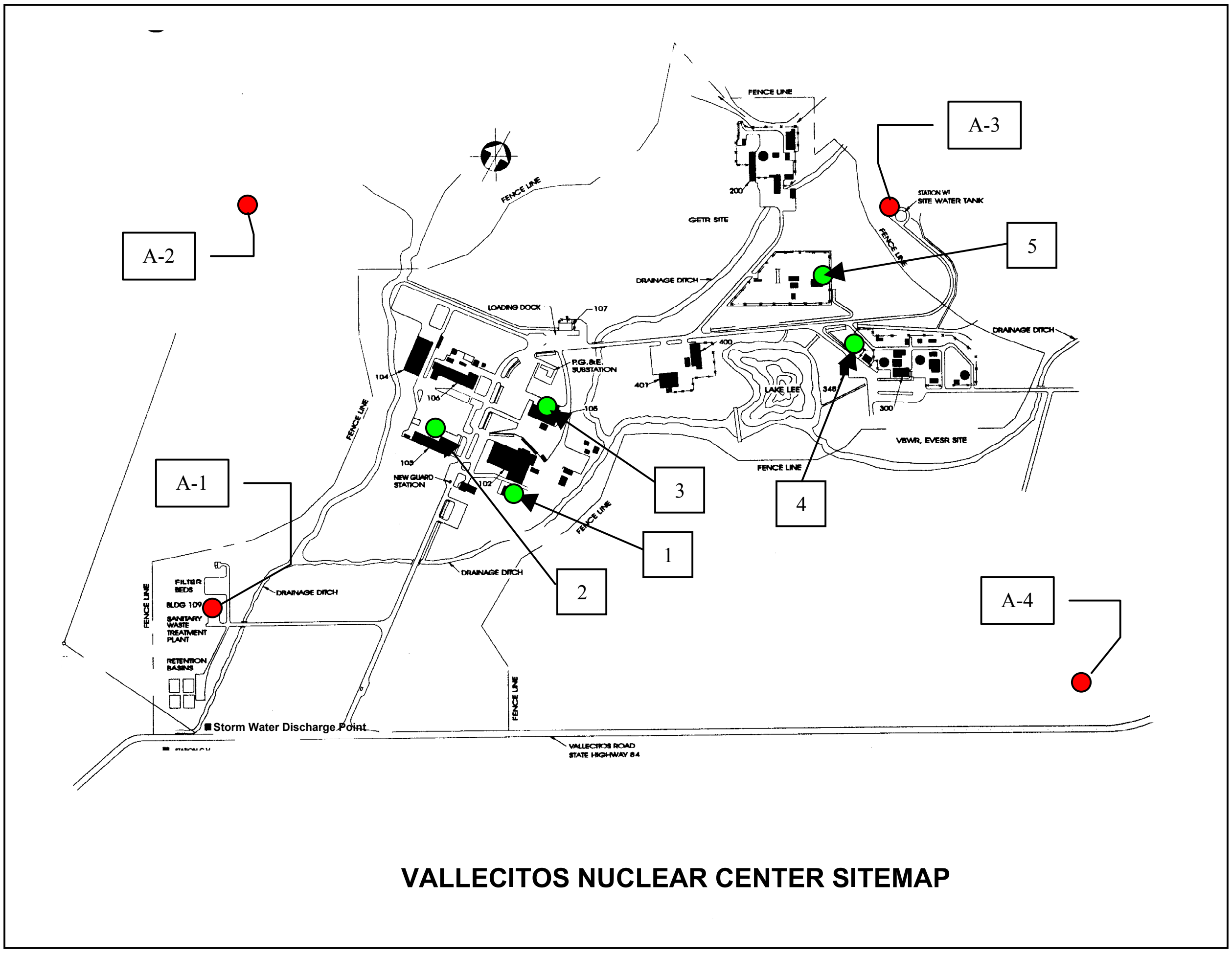
9.8 ALARA

Compliance and the ALARA concept are inherent in the Environmental Program in terms of comprehensive monitoring, analysis, and evaluation of air emissions, liquid effluents and disposition of solid waste. Management controls, quality assurance and program implementation provide (1) representative measurements of radioactivity in the highest potential exposure pathways and (2) verification of the accuracy of the effluent monitoring program of those environmental exposure pathways. Trends are assessed using monitoring results to evaluate licensed activities, in terms of “control-at-the-source” of contamination and the containment of radioactivity; the projections of potential dose to off-site populations; and the detection of any unanticipated pathways for the transport of radionuclides within the environment. Monitoring with periodic evaluations are summarized and presented to senior management on an annual basis.

FIGURE 9.1
VNC'S ENVIRONMENTAL MONITORING PARAMETERS

TYPE OF SAMPLE	ANALYSES	TYPICAL SAMPLING FREQUENCY	TYPICAL MINIMUM DETECTION LEVEL
Air Exhaust Stacks - Point Sources	Alpha, Beta, Gamma	Continuous (Collection Weekly)	$4x10^{-15}$ µCi/cc Alpha $7x10^{-15}$ µCi/cc Beta/Gamma
Ambient Air – Environmental On-Site	Alpha, Beta, Gamma	Continuous (Collection Weekly)	$4x10^{-15}$ µCi/cc Alpha $7x10^{-15}$ µCi/cc Beta/Gamma
Ground Water - On-Site	Alpha, Beta, Gamma	Quarterly	$2x10^{-8}$ µCi/cc Alpha $5x10^{-8}$ µCi/cc Beta/Gamma
Stream Bottom Sediments	Alpha, Beta, Cs-137, Co-60	Annual	30 pCi/gram Alpha 8 pCi/gram Co-60 15 pCi/gram Cs-137
Retention Basin Water	Alpha, Beta, Gamma	Prior to Release	$2x10^{-8}$ µCi/cc Alpha $5x10^{-8}$ µCi/cc Beta/Gamma
Environmental TLDs	Gamma	Continuous (Collection Annual)	1 mRem
Vegetation	Alpha, Beta Gamma	Annually	10 pCi/gram Alpha 50 pCi/gram Beta/Gamma

FIGURE 9.2
Air Sampling Sites (Typical)



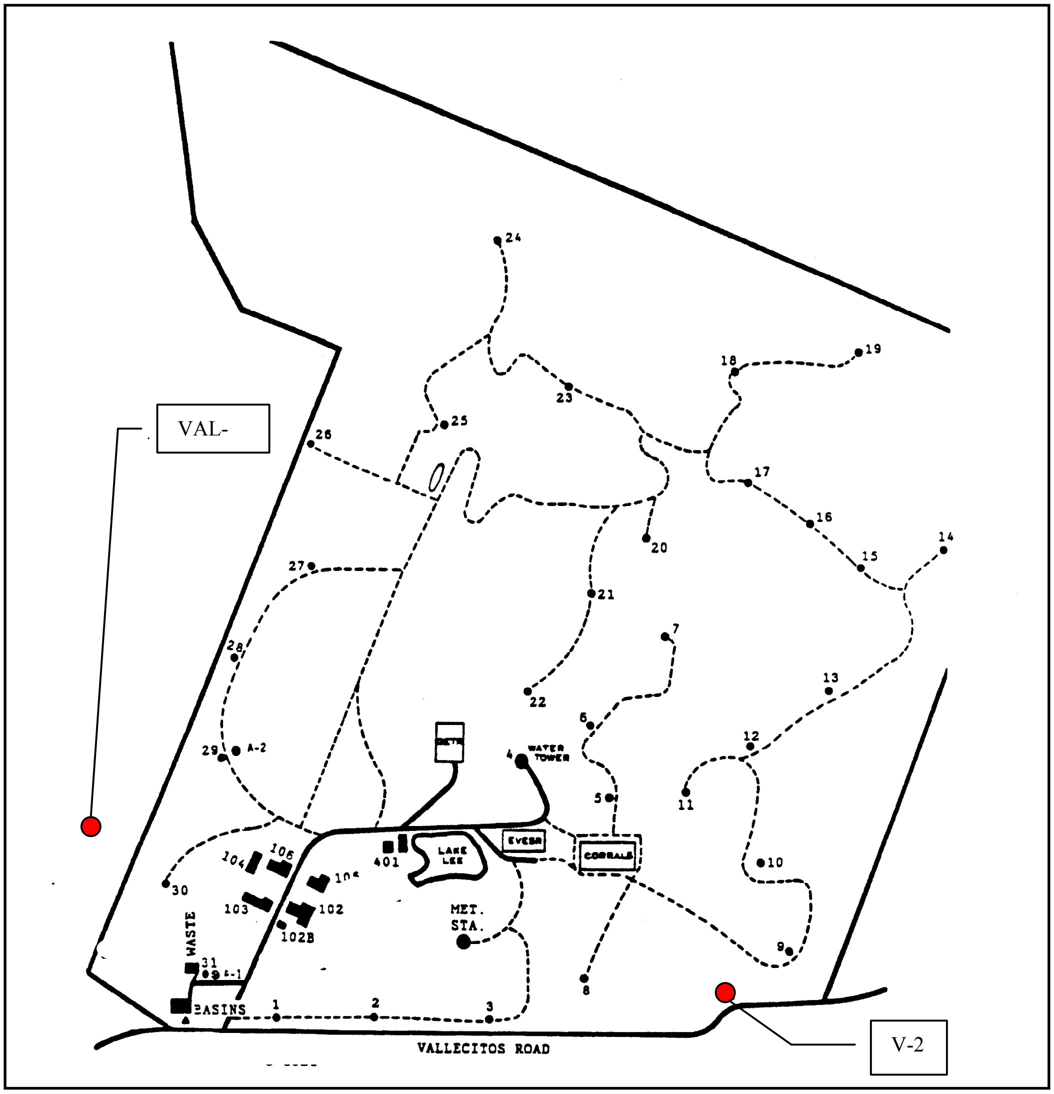
Environmental Air Monitoring Stations (A-1, A-2, A-3, A-4)

Stack Air Monitoring Stations

1. Bldg. 102A
2. Bldg. 103
3. Bldg. 105 (NTR)
4. Waste Evaporator
5. HSF Bunker

FIGURE 9.3

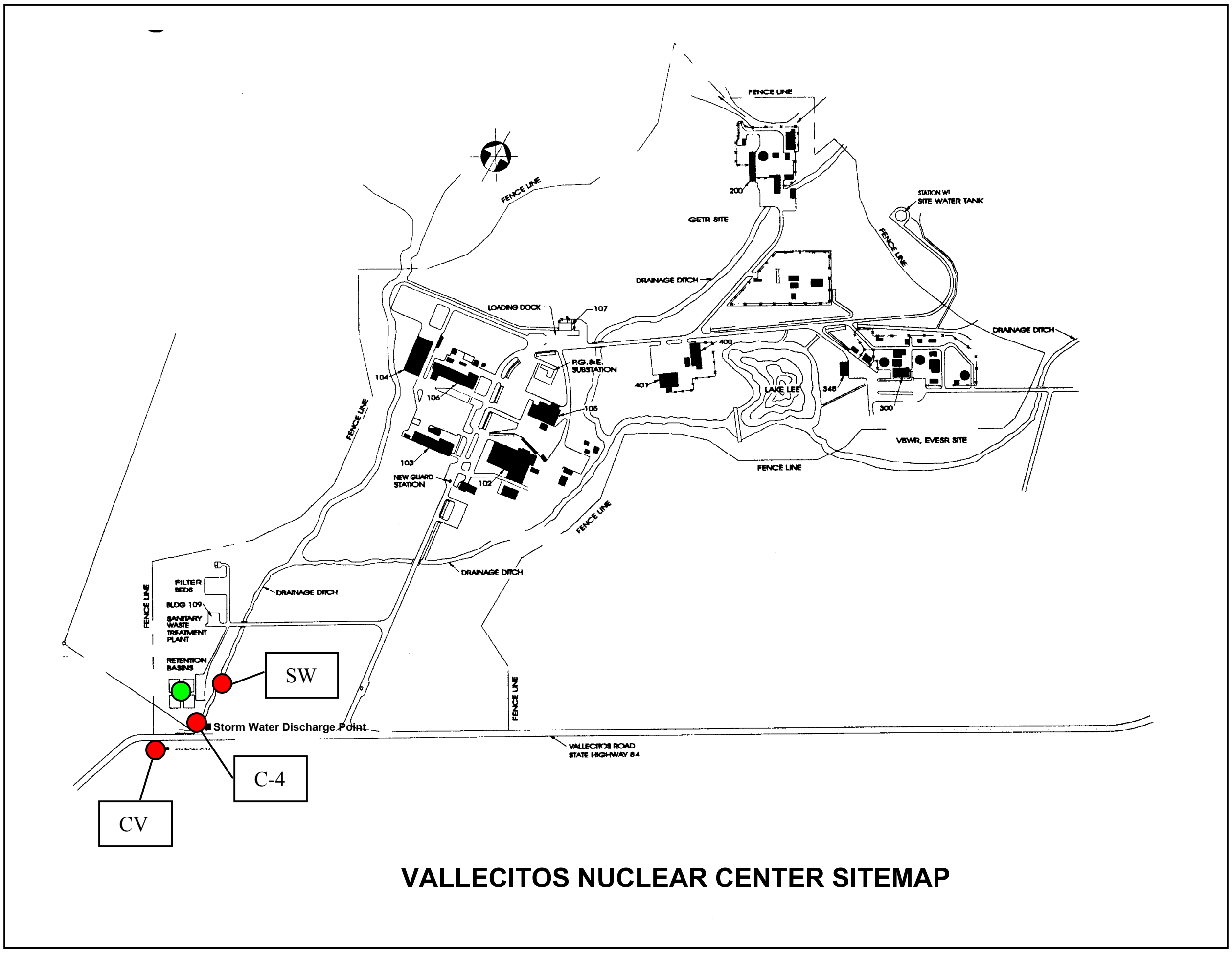
Environmental and Vegetation Sampling Points (Typical)



● Environmental Dosimeter Monitoring Stations (1-31)

● Environmental Vegetation Sampling Points (V-2, VAL-IV)

Figure 9.4
Location of Surface Water Monitoring Points (Typical)

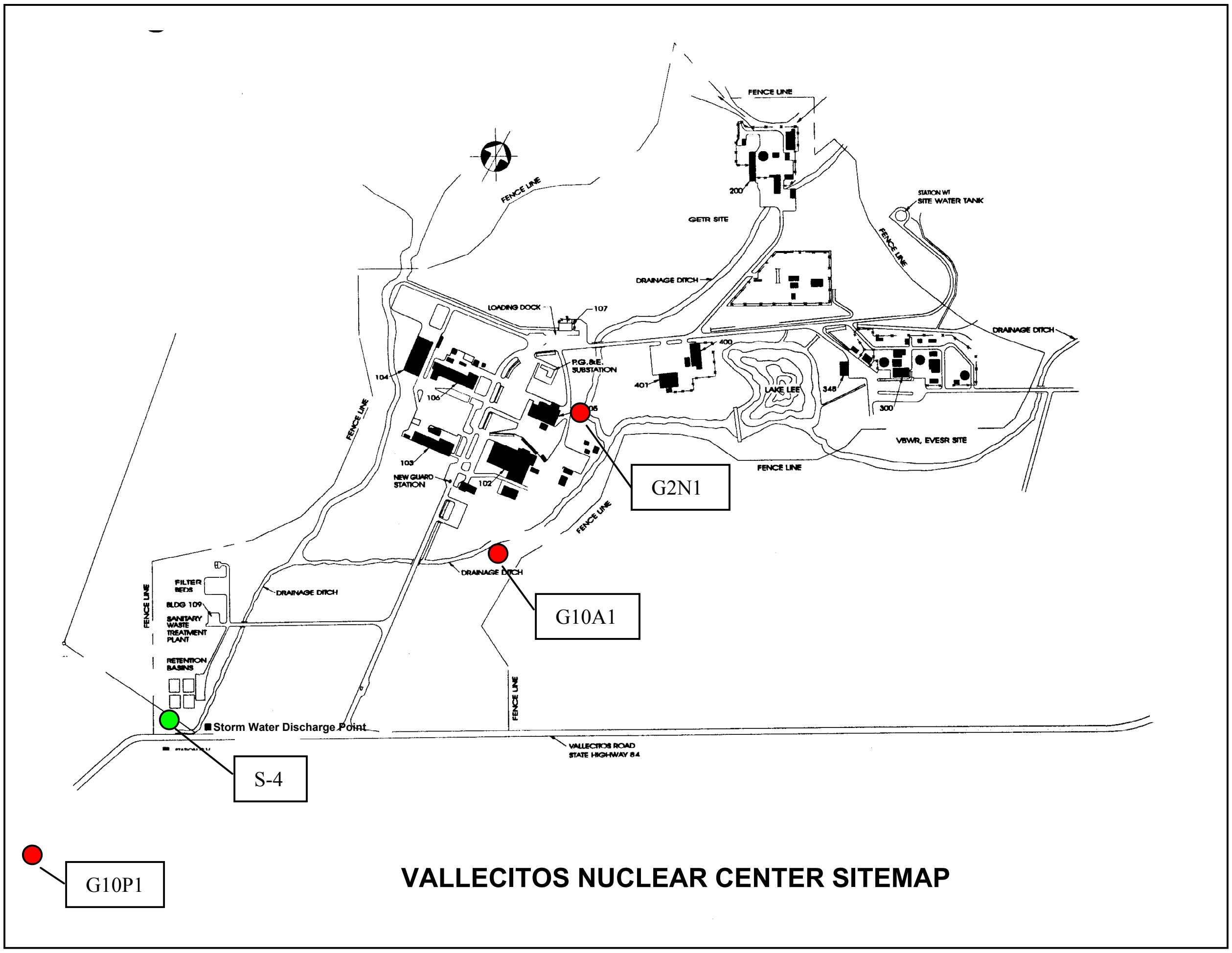


- Surface Water Monitoring Points
  - C-4
  - CV
  - SW

- Sanitary & Industrial Discharge Water Monitoring Points
  - Basin 1
  - Basin 2
  - Basin 3
  - Basin 4

4	1
3	2

Figure 9.5
Locations of Monitoring Well & Stream Bottoms Sampling Points (Typical)



● Radioactive Ground Water Monitoring Wells
G2N1
G10A1
G10P3

● Stream Bottoms Monitoring Point
S-4

CHAPTER 10.0
DECOMMISSIONING

The current Decommissioning and Closure Plan is dated July 2, 2009.

At the end of plant life, the facility shall be decommissioned in accordance with the then current Decommissioning and Closure Plan.

CHAPTER 11.0
MANAGEMENT MEASURES

11.1 MANAGEMENT MEASURES

11.1.1 REASONABLE ASSURANCE

VNC commits to apply *Management Measures* on a continuing basis to ensure safety systems are available and able to perform their function when needed.

11.1.2 GRADED APPLICATION OF MANAGEMENT MEASURES FOR SAFETY

VNC applies *Management Measures* in a graded approach as described in this chapter.

11.2 CONFIGURATION MANAGEMENT (CM)

11.2.1 CONFIGURATION MANAGEMENT POLICY

VNC commits to maintain a formal configuration management process, governed by written, approved practices, and ensures that plant design changes do not adversely impact safety, health, or environmental protection programs at VNC. This program is also referred to as the Change Authorization (CA) system. A Change Authorization is the VNC mechanism for independently reviewing and documenting changes. Any addition, alteration, deletion, modification or substitution involving licensed activities 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 previously approved operating instructions and may be used to assist in determining if the proposed activity constitutes a change. The following items are addressed prior to implementing a change:

- The technical basis for the change
- The impact of the change on safety, health and control of licensed material
- The impact of the change on industrial, health, environmental, chemical or fire safety involving materials licensed under SNM-960
- Modifications to existing operating procedures including any necessary training or retraining before operation
- Authorization requirements for the change
- For temporary changes, the approved duration (expiration date) of the change

The CA system ensures that the information used to operate and maintain safety controls is kept current. Safety controls are systems, structures, components and procedures that prevent and/or mitigate the risk of accidents.

The CA system includes the following activities:

- Maintenance of the design information for the plant
- Control of information used to operate and maintain the plant
- Documentation of changes
- Assurance of adequate safety reviews for changes

Any addition, alteration, deletion or substitution involving licensed activities 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 (CA) procedure. A Change Authorization is prepared for:

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 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 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 Regulatory Compliance function as appropriate.

11.2.1.1 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 Regulatory Compliance function 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 Regulatory Compliance function as appropriate 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 Regulatory Compliance function 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 Regulatory Compliance function 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 filled with the Regulatory Compliance function, with other copies distributed as appropriate.

11.2.2 DESIGN REQUIREMENTS

Written plant procedures define the development, application, and maintenance of the design specifications and requirements. Plant design specifications and requirements are maintained as controlled information. The specific content of the information depends on the age of the design and the requirements in place at the time of design. As a minimum, the information required for safe operation of the facility is available.

11.2.3 DOCUMENT CONTROL

Documented plant procedures define the control system, including creation, revision, storage, tracking, distribution and retrieval of applicable information including:

- Hazards analysis
- Operating procedures
- Drawings for equipment important for safety
- Specifications and requirements
- Calibration instructions
- Functional test instructions

The documented plant procedures describe the responsibilities and activities that maintain consistency between the facility design, the physical facility, and the documentation. They also describe how the latest approved revisions are made available for operations.

11.2.4 CHANGE CONTROL

VNC maintains written plant procedures describing the CA system for controlling design change, including approval to install and operate facility, process, or equipment design changes. The Area Manager assigns reviewers within the affected disciplines to ensure compliance with technical specifications, license, generally accepted safety practices and other applicable standards.

The written plant procedures also prescribe controls and define the distinction between types of changes, ranging from replacement with identical designs that are authorized as part of normal maintenance, to new or different designs that require specified review and approval.

11.2.5 ASSESSMENTS

Planned and scheduled internal and independent audits and evaluations are performed to evaluate the application and effectiveness of management controls and implementation of programs related to activities significant to plant safety. Audits and evaluations are performed to assure that operations are conducted in accordance with the operating procedures, and to assure that safety programs reflected in the operating procedures are maintained.

11.3 MAINTENANCE

The purpose of planned and scheduled maintenance of safety controls is to assure that systems are kept in a condition of readiness to perform the planned and designed functions when required.

Area Managers are responsible for assuring the operational readiness of safety controls in their assigned facility areas.

The maintenance function utilizes a systems-based program to plan, schedule, track and maintain records for maintenance activities. Maintenance instructions are an integral part of the maintenance system for maintenance activities. Key maintenance requirements for safety controls such as calibration, functional testing, and replacement of specified components are documented in site maintenance programs.

Maintenance activities generally fall into the categories described in the following sections.

11.3.1 CORRECTIVE MAINTENANCE

Corrective Maintenance refers to situations where repairs, replacements or major adjustments such as re-calibration take place.

VNC commits to promptly implement corrective actions as necessary to assure equipment or systems relied on, to maintain the safety of licensed material are available, to perform their function.

The maintenance records typically provide documentation of systems and components that have been repaired or replaced.

When equipment is repaired or replaced, the functionally is verified via post maintenance testing to assure that it has the capability to perform its planned and designed function when required.

If the performance of a repaired or replaced equipment is different from that of the original, the change is specifically approved under the CA system and pre-operationally tested to assure it is likely to perform its desired function when required.

11.3.2 PREVENTATIVE MAINTENANCE

Preventative Maintenance refers to activities that typically are performed as precautions to help ensure that systems remain operational and avoid unexpected failures. Examples of equipment important to safety included for scheduled preventive maintenance are:

- Radiation Measurement Instruments
- Criticality Detection Devices
- Effluent Measurement Devices
- Standby Power Generators
- Fire Detection and Control Systems
- Pressure Relief Valves
- Air Compressors
- Steam Boilers

11.3.3 SURVEILLANCE/MONITORING

Safety controls associated with passive engineered systems are typically fixed physical design features to maintain safe process conditions. Assurance is maintained through pre-operational audit and periodic verification of effectiveness as prescribed in the relevant analysis.

Safety controls relying on geometry-based controls, where the geometry is subject to undetected change in routine operation, are periodically verified on a schedule commensurate with the potential for change in the parameters of interest.

- Examples of active engineered controls – None at VNC
- Examples of passive engineered – Hot cell floors are not sealed preventing accumulation of liquids (moderator)

- Examples of geometry-based – Storage locations in fixed positions at the Hillside storage facility

11.3.4 FUNCTIONAL TESTING

VNC commits to perform post-maintenance testing to verify that the maintenance activity did not adversely affect the functionality of the safety controls associated with the maintenance work.

VNC commits to perform functional tests in accordance with written instructions that define the method for the test and the required acceptable results. The results of the tests are also recorded and maintained in accordance with record retention requirements.

11.4 TRAINING AND QUALIFICATIONS

11.4.1 ORGANIZATION AND MANAGEMENT OF THE TRAINING FUNCTION

Training programs at VNC for personnel who perform activities, relied on for safety, are provided through Regulatory Compliance/EHS, with additional assistance from program experts in other components. Area Managers are responsible for the content and effective conduct of training for operations personnel. Records are maintained on each employee's qualifications, experience, training, and retraining.

Facility administrative procedures establish the requirements for indoctrination and training of personnel performing activities relied on for safety and to ensure that the training program is conducted in a reliable and consistent manner throughout all training areas.

Training records are maintained to support management information needs associated with personnel training, job performance, and qualifications. Training records are retained in accordance with records management procedures.

11.4.2 FUNCTIONAL AREAS REQUIRING TRAINING

Training is provided for each individual at VNC, commensurate with assigned duties (or roles). Training and qualification requirements are met prior to personnel fully assuming the duties of positions listed in the license

Functional areas requiring training may be grouped into one of three broad categories:

- General employee training
- Technical training
- Developmental training

The objective of the training program is to ensure safe and efficient operation of the facility and compliance with applicable regulatory requirements. Training requirements shall be applicable to, but not restricted to, those personnel who have a direct relationship to the operation, maintenance, testing, or other technical aspects of the facility.

Continuing or periodic retraining courses shall be established when applicable to ensure that personnel remain proficient. Periodic training generally is conducted to ensure retention of knowledge and skills important to facility operations. The training may consist of periodic retraining exercises, instructions, or review of subjects as appropriate to maintain the proficiency of all personnel assigned to the facility.

11.4.2.1 General Employee Training

General employee training (GET) encompasses those quality assurance, radiation protection, industrial safety, environmental protection, emergency response, and administrative procedures established by facility management and applicable regulations. The industrial safety training for VNC complies with applicable section of the California Occupational Safety and Health (CalOSHA) regulations such as Title 8 and with 10 CFR 19 (Notices, Instructions, and Reports to Workers: Inspection and Investigations). Continuing training is conducted in these areas as necessary to maintain employee proficiency. All persons under the supervision of facility management (including contractors) must participate in GET; however, certain facility support personnel, depending on normal work assignment, may not participate in all topics of this training. Temporary maintenance and service personnel receive GET to the extent necessary to assure safe execution of their duties. Certain portions of GET may be included in new employee orientation program implementation.

GET topics are listed below:

- General administrative controls and procedures and their use
- Quality Assurance policies and procedures
- Nuclear Safety (Criticality/Radiological)
- Industrial, Chemical, Fire, Health and First Aid
- Emergency Plan and implementing procedures
- Fire protection and fire brigade
- New Employee Orientation
- Environmental Protection

11.4.2.2 Nuclear Safety Training

Training programs are established for the various job functions (e.g., operations, radiation protection technicians, contractor personnel) commensurate with criticality safety and radiation safety responsibilities. Visitors to VNC are trained in the formal training program or are escorted by trained personnel.

Formal Nuclear Safety training includes:

- Information about radiation and radioactive materials
- Risks involved in receiving low level radiation exposure in accordance with 10 CFR 19.12
- Basic criteria and practices for radiation protection
- Nuclear criticality safety principles (not verbatim, but in general conformance with applicable objectives contained in ANSI/ANS 8.19 and ANSI/ANS 8.20 national consensus standard guidance)

Training policy requires that employees must complete nuclear safety training prior to unescorted access in posted radiation areas. Methods for evaluating the understanding and effectiveness of the training includes passing an initial examination covering formal training contents and observations of operational activities during scheduled audits and inspections.

Such training is typically performed either with standardized training, computer based training, or by knowledgeable personnel. The Regulatory Compliance function reviews training program contents on a scheduled basis to ensure that training program contents remain current and adequate.

Employees who are allowed unescorted access to posted radiation areas receive retraining at least every two years. The effectiveness of the training is confirmed by either initial training exam or re-training exam. Visitors are trained commensurate with the scope of their visit and/or escorted by trained employees.

11.4.2.3 Industrial, Chemical, Fire, Health and First Aid

Industrial, Chemical, Fire Safety, Health and First Aid safety orientation of new or transferred employees is an important part of establishing the proper safety attitude among employees engaged in licensed activities and insuring that they are aware of safety procedures, rules and hazards involving assigned duties. New employee orientation in performance of duties may include, as appropriate, the review of:

OSHA General Duty Clause

- Employee Responsibilities
- Employer Responsibilities
- General Site Safety Rules
- Hazard Communication Training
- Fire Extinguisher Training
- Emergency Evacuation Procedure
- Job Hazards Analysis (JHA)
- Material Safety Data Sheets (MSDS)
- Lock-Out-Tag-Out Awareness

11.4.2.4 Technical Training

Technical training is designed, developed and implemented to assist facility operations and maintenance personnel in gaining an understanding of the applicable fundamentals, procedures, and technical practices associated with licensed activities. Technical training consists of initial training, on-the-job training, continuing training, and special training, as applicable to assigned technical duties of the job function (or role). This may include, but is not limited to, the following topics:

- On-the-Job Training
- Process Specific Training
- Mechanical Maintenance
- Controls, Instrumentation, Electrical Maintenance
- Chemistry

11.4.2.5 Development Training

Developmental Training is a broad category implemented to assist facility operations supervisory, and management personnel in gaining additional understanding of fundamentals and technical practices common to assigned job duties (or roles). Developmental training typically utilizes internal/external professionals via formal workshops, tutorials, and/or training programs.

11.4.3 POSITION TRAINING REQUIREMENTS

Operator training is performance based, and incorporates the structured elements of analysis, design, development, implementation, and evaluation commensurate with assigned duties.

Minimum training requirements are identified for positions whose activities are relied on for safety. Initial identification of job-specific training requirement is based on individual employee experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

Job-specific training is performance based and established with relevant technical EHS safety discipline and operations leadership to develop a list of qualifications for assigned duties (or roles). Changes to facilities, processes, equipment, or job duties are incorporated into revised lists of qualifications.

11.4.4 BASIS OF TRAINING AND OBJECTIVES

The training program is designed to prepare initial and replacement personnel for safe, reliable, and efficient operation of the facility. Emphasis is placed on safety requirements where human actions are important to safety.

11.4.4.1 Evaluation of Trainee Learning

Trainee understanding and proficiency is evaluated through observation/demonstration and/or oral or written examinations, as appropriate. Such evaluations measure the trainee's skill and knowledge of job performance requirements.

Operator training and qualification requirements are met prior to process safety-related tasks being independently performed or before startup following significant changes to safety controls.

11.4.4.2 Conduct of On-the-Job Training

On-the-Job training (OJT) is a method of providing the required job related skills and knowledge for a position. This training is conducted in the work environment. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The object of the program is to assure the trainee's ability to proficiently perform job duties as required for the assigned role.

Completion of on-the-job training is demonstrated through actual task actions using the conditions encountered during the performance of assigned duties (or roles) including references, tools, and equipment conditions reflecting the actual task to the extent practical.

11.4.3 EVALUATION OF TRAINING EFFECTIVENESS

Periodic evaluations of training program content and requirements are performed to assess program effectiveness. These evaluations identify program strengths and weaknesses, determine whether training content matches current job needs, and determines if corrective actions are needed to improve program effectiveness.

Independent audits of EHS safety disciplines may also be used to provide independent evaluations of overall training program effectiveness as it relates to the protection of the public, worker, and environment.

Evaluation objectives applicable to the overall organization and management of the VNC training programs may include, but are not limited to:

- Management and administration of training and qualification programs
- Design and development of training programs, content, and conduct of training, and trainee examinations / evaluations.
- Training program interface with facility configuration management practices
- Training program assessments and evaluations

11.4.4 PERSONNEL QUALIFICATION

The qualification requirements for key management positions are described in Chapter 2, Organization and Administration.

Qualification and training requirements for operations personnel are established and implemented in accordance with internal plant procedures (e.g., Human Resource).

11.5 PROCEDURES

Licensed material processing or activities are conducted in accordance with properly issued and approved procedures.

11.5.1 OPERATING PROCEDURES

Area Managers are responsible to assure preparation of written, approved and issued operating procedures incorporating control and limitation requirements established by the criticality safety function, the radiation safety function, the environmental protection function and the chemical and fire safety function. Operating procedures are initiated and controlled by the CA system. Area Managers ensure that operating procedures are made readily available in the work area and that operators are trained to the requirements of the procedures and that conformance is mandatory. Operators are trained to report inadequate procedures, and/or the inability to follow procedures.

Nuclear safety control procedure requirements are incorporated into the appropriate operating, maintenance and test procedures where SNM is used.

The safety program design requires the establishment and maintenance of documented procedures for environmental, health and safety limitations and requirements to govern the safety aspects of operations. Requirements for procedure control and approval authorities are documented. Procedure reviews for updating frequencies are as follows:

Document	Review Frequency (Typical)	Reviewing & Approving Functional Manager
Component Operating Procedures (OPs)	When changed (1)	Area Manager and Affected RC&EHS Discipline (Radiation, Criticality, Environmental, Industrial (3), or MC&A)
Operating Procedures (OPs)	Every 2 Years (2)	Area Manager and Affected RC&EHS Discipline (Radiation, Criticality, Environmental, Industrial (3), or MC&A)
Vallecitos Safety Standards Nuclear Safety Procedures, Site Emergency Procedures, Site Security Procedures, & Site Operating Procedures VNC EHS Manual	Every 3 Years (2)	RC&EHS

1) The safety awareness portions of these OPs are reviewed and updated by the appropriate environment, health, and safety (EHS) discipline when warranted based on process related facility change requests.
2) Frequencies are defined as 1year = maximum 13 months, 2 years = maximum 26 months and 3 years = maximum interval of 39 months.
3) EHS Discipline - Industrial means normal worker safety, chemical safety, and fire and explosion protection.

11.5.2 MANAGEMENT CONTROL PROCEDURES

Licensed material activities are conducted in accordance with management control programs described in administrative and general plant practices approved and issued by management at a level appropriate to the scope of the practice. These documented practices direct and control activities across the manufacturing functions, and assign functional responsibilities and requirements for these activities. These practices typically are reviewed for updating at least every two years.

11.6 AUDITS AND ASSESSMENTS

11.6.1 CRITICALITY, RADIATION, CHEMICAL, INDUSTRIAL, ENVIRONMENTAL AND FIRE SAFETY AUDITS

Criticality inspections are performed quarterly in Building 102 and annually for balance of site operations.

Radiation Monitors conduct inspections covering radiation, criticality, chemical, fire, industrial and environmental safety on a routine basis, documenting these inspections at least quarterly for active plant areas.

Area manager and safety committee inspections similarly cover active VNC areas quarterly.

Audits against EHS program areas are included in written audit procedures and integrated into the audits performed for radiation and criticality safety.

Audit results and inspection findings are communicated in writing or electronically to the Area Manager and to the Manager of RC&EHS. Required corrective actions are documented and approved by the Area Manager, and tracked to completion by the environment, health & safety function.

Records of the audit or inspection, instructions and procedures, persons conducting the audits or inspections, audit or inspection results, and corrective actions for identified violations of license conditions are maintained in accordance with document retention policies.

11.6.2 INDEPENDENT AUDITS

VNC commits to perform triennial independent audits of its safety program elements (radiation protection, criticality safety, chemical safety, fire protection, industrial safety and environmental protection). The audit team will typically consist of appropriately trained and experienced individuals who are not involved in the routine performance of the work or program being audited. The audit scope includes compliance to procedures, conformance to regulations and the overall adequacy of the safety program.

Audit results are reported to the Manager Vallecitos Nuclear Center, the Area Managers, the manager of the radiation safety function, and the manager of the criticality safety function, as appropriate. The findings of the audit are assigned to the appropriate safety function or Area Managers. The assigned responsible individual takes the necessary steps to investigate the finding and identify appropriate corrective actions to address and correct the finding.

The corrective actions resulting from the audit are entered into the appropriate tracking system and reported and tracked to completion by the applicable Manager.

11.6.3 WORKER CONCERNS

VNC commits to maintain a safety conscious work environment. All workers are encouraged to report potentially unsafe conditions to their supervisor, management or the safety organization. Reported concerns are promptly investigated, assessed and resolved.

11.7 INCIDENT INVESTIGATIONS

VNC commits to maintain a system to identify, track, investigate and implement corrective action for abnormal events (unusual incidents). Incident investigations typically include but are not limited to an evaluation of the following elements:

- Description of the event
- Timeline associated with the event
- Whether or not the system was operated in accordance with written procedures
- Human factors contributing to the event
- Severity and consequence of the event
- Corrective actions necessary to assure safety and prevent or mitigate future events
- Abnormal events are documented, tracked and reported to the Area Managers, the safety functions and facility management in accordance with incident investigation procedures.
- Each event is considered in terms of regulatory reporting criteria

- Records of the events and the documented evidence of closure are maintained in accordance with document retention policies.

11.8 RECORDS MANAGEMENT

Records appropriate for safety controls, criticality and radiation safety activities, training/retraining, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent safety activities are maintained in such a manner as to demonstrate compliance with license conditions and regulations.

Records of criticality safety analyses are maintained in sufficient detail and form to permit independent review and audit of the method of calculation and results. Such records are retained during the conduct of the activities and as required by record retention policies.

Records of material control and accountability are maintained in sufficient detail and form to permit independent review and audit of the inventory and transactions. Such records are retained as required by record retention policies. Records associated with personnel radiation exposures are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20. The following additional radiation protection records will typically be maintained for at least three years:

- Records of the safety review committee meetings
- Surveys of equipment for release to unrestricted areas
- Instrument calibrations
- Safety audits
- Personnel training and retraining
- Radiation work permits
- Surface contamination surveys
- Concentrations of airborne radioactive material in the facility
- Radiological safety analyses

Records associated with the environmental protection activities described in Chapter 9 are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20 and this license.

11.9 OTHER QA ELEMENTS

VNC performs a broad spectrum of work that requires the application of QA measures. This includes work-requiring conformance to 10 CFR 50, Appendix B, 10 CFR 71, Subpart H as well as certain aspects of 10 CFR 70. As a result of these over arching quality requirements, VNC's management system is structured to provide a full scope of QA elements and apply them as appropriate.

With regard to 10 CFR 70, particularly the identification and maintenance of safety controls and the management measures (discussed in this Chapter) that assure the availability of the these controls to perform their intended function when required, the following information outlines QA Elements and summarizes the manner in which they are applied for the operations. The following assurance elements are applied to safety controls and the management measurements at VNC:

- Organization – VNC operates to a documented organizational structure in which responsibility and authority is clearly identified
- Program – VNC operates to written policies, procedures and instructions
- Design Control – VNC policies and procedures outline a program to provide design control for safety controls including the management measures necessary to assure their successful operation
- Procurement Documentation Control – VNC policies and procedures require the definition of procurement specifications, review and approval of procurement to assure they are compatible with regulatory requirements
- Instructions, Procedures, and Drawings – VNC uses instructions, written procedures and drawings to document configuration, processes and methods for doing work
- Document Control – VNC implements document control as described in this Chapter
- Control of Purchased Materials, Equipment, and Services – VNC procedures require that purchased materials, equipment or services be secured from appropriately qualified vendors and that as appropriate vendor certifications or in-house dedication of the items or work are provided
- Control of Special Processes – VNC procures materials from qualified vendors to documented specifications that include where necessary control of

special processes. Internally the CA system, Radiation Work Permit and Temporary Operating Procedure routines control special situations.

- Internal Inspections – VNC uses pre-operational audits for safety controls to verify that parts, configuration and operations are as intended.
- Test Control – VNC implements a functional test program for safety controls as defined in this Chapter.
- Control of Measuring and Test Equipment – VNC maintains measuring and test equipment in accordance with procedures.
- Handling, Storage, and Shipping Controls –VNC process for procuring materials include, where appropriate, handling and shipping controls to ensure the validity of the items received. In addition where shelf life is important controls are implemented to ensure these limits are implemented for the item.
- Inspection, Test, and Operating Status – Where safety controls require this type of marking; items are so marked and maintained.
- Control of Nonconforming Materials, Parts, or Components - VNC maintains a non-conforming materials program.
- Corrective Action – VNC procedures for investigating the failure of safety controls require the definition of root cause and corrective action.
- Records – Where specific actions are required, VNC maintains records to demonstrate the action has been completed.
- Audits – VNC provides audits as defined in this Chapter.