Attachment 2

TAC 13-004

Revised License Renewal Application

(LRA for Vallecitos Nuclear Center)

Public Version

This is a non-security related version of Attachment 1, from which the security related information has been removed. Portions of the document that have been removed are identified by white space within double brackets, as shown here [[ ]].

CHAPTER 1.0

GENERAL INFORMATION

1.1 SITE LOCATION AND DESCRIPTION

The primary purpose of the GE – Hitachi Nuclear Energy Americas LLC (GEH) Vallecitos Nuclear Center (VNC) Special Nuclear Material (SNM) operation is research and development, engineering studies and storage of irradiated LWR fuel and components previously subjected to engineering tests and evaluations awaiting transfer to DOE. 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). The principal currently performed activity on site is the by-product material activities covered under the State of California license CA-0017-01 including sealed source manufacture and research and development

VNC is located near the center of the Pleasanton quadrangle of Alameda County, California. The site 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 72,000, located 4.1 miles to the north-northwest and Livermore with a population of approximately 8,000, 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. A Union Pacific railroad line lies about two miles west of the site. 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 property boundary is fenced with 3-strand barbed wire and posted “No Trespassing”. The front of the site is fenced with 8-foot commercial chain-link fencing which wraps around to the west past Building 103 and terminates to the rear at a major ditch and to the east completely around Buildings 102 and 102A. A

security gate at the entrance provides access control to the site. Hillside Storage is also fenced with 8-foot commercial chain link fencing. The Waste Evaporator Plant (WEP) is not fenced due to the small quantities of material processed there.

The VNC 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.

The site is located in the Livermore Upland physiographic 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 and situated between the 400 and 600-foot topographic contours are developed. The property is not located within a 100-year flood zone.

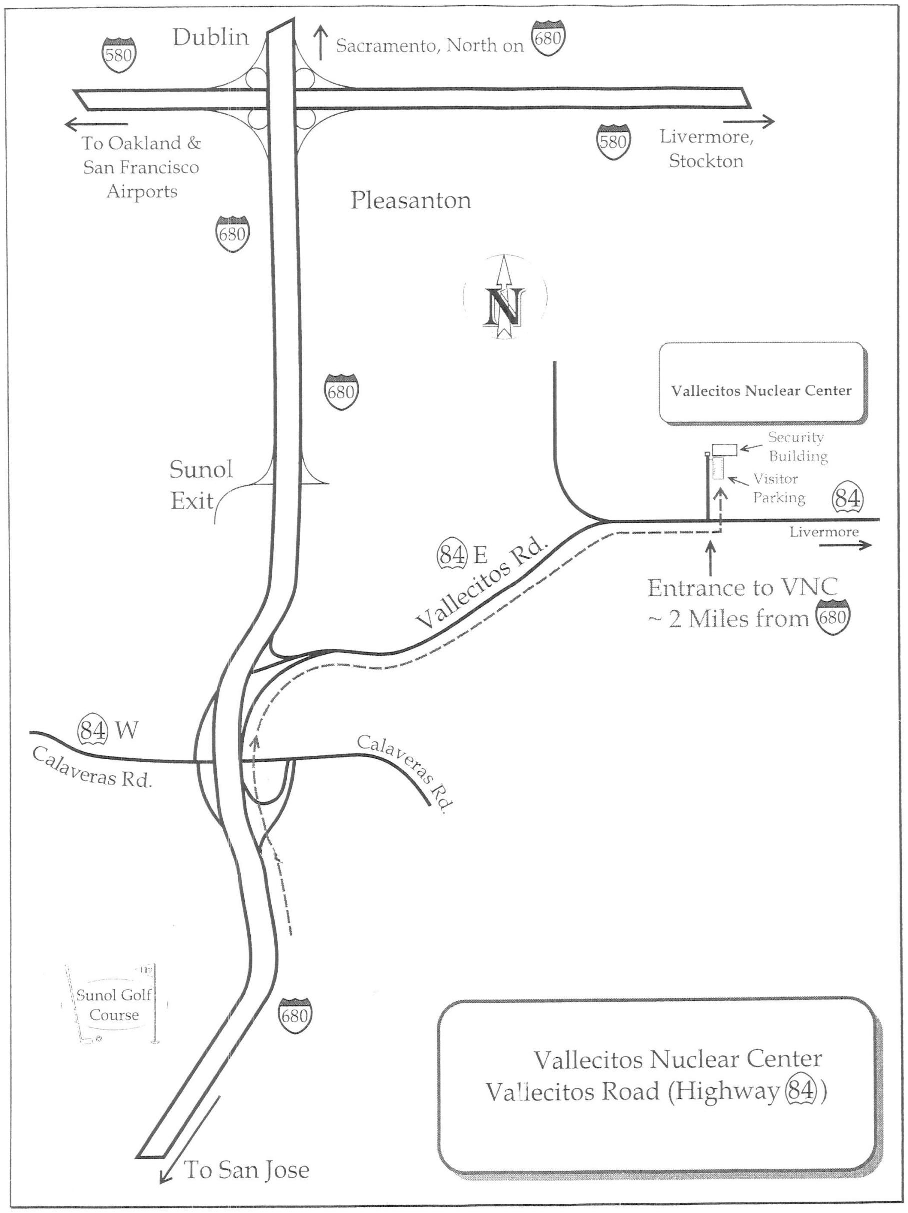
Geological faults exist in the East Bay area. Earth tremors are an occasional occurrence in all parts of the Bay Area. The VNC site property contains a branch of the Hayward fault. Very small tremors can be measured daily using very sensitive instruments, however, very seldom are tremors of a magnitude that they are noticed. The VNC facilities were built to building codes in effect at the time of construction that account for this seismic activity. There have been no significant damages at the VNC site from any of the earth tremors to date.

Seasonal rainfall in the area is responsible for one other minor environmental related threat to the site in that certain parts of the year are particularly dry. During these dry periods the grasses become dormant and dry and are subject to grass fires. At VNC cattle are pastured on a large portion of the site to keep the grasses to a minimum. Firebreaks are plowed along roadways. The use of vehicles on rural parts of the property is controlled and the site fire brigade is trained to combat grass fires and incipient stage building fires. The local fire departments are also well-trained and equipped to deal with these potential fire dangers.

The property on which the 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.

Pacific Gas and Electric supply electrical power to the main site substation for distribution to each building on the site.

FIGURE 1.1
Area Map
(Not to Exact Scale)



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

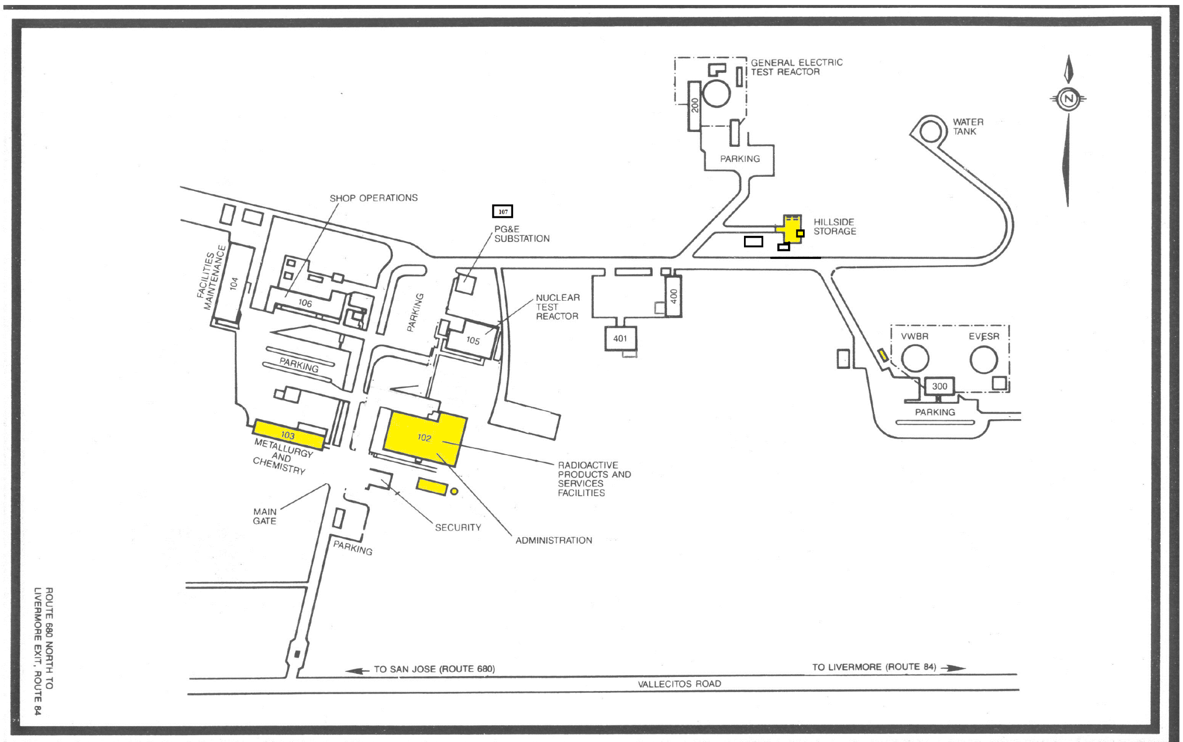
Industrial wastewater is processed in a system of surface impoundments and used as irrigation water for grasslands on site. Wastewater containing radioactive materials is evaporated to reduce the volume and then solidified for disposal in a licensed disposal facility.

1.1.1 VALLECITOS NUCLEAR CENTER FACILITIES

General descriptions of the principal buildings and laboratories on the overall VNC site are described in this section. Principal buildings are those buildings that typically appear on site drawings. Detailed discussions of the facilities, processes and equipment used for SNM-960 activities follow in subsequent sections. They are not discussed in detail here because this section is intended to provide a general overview of the site. The current locations of these facilities are shown in Figure 1.2 and 1.3.

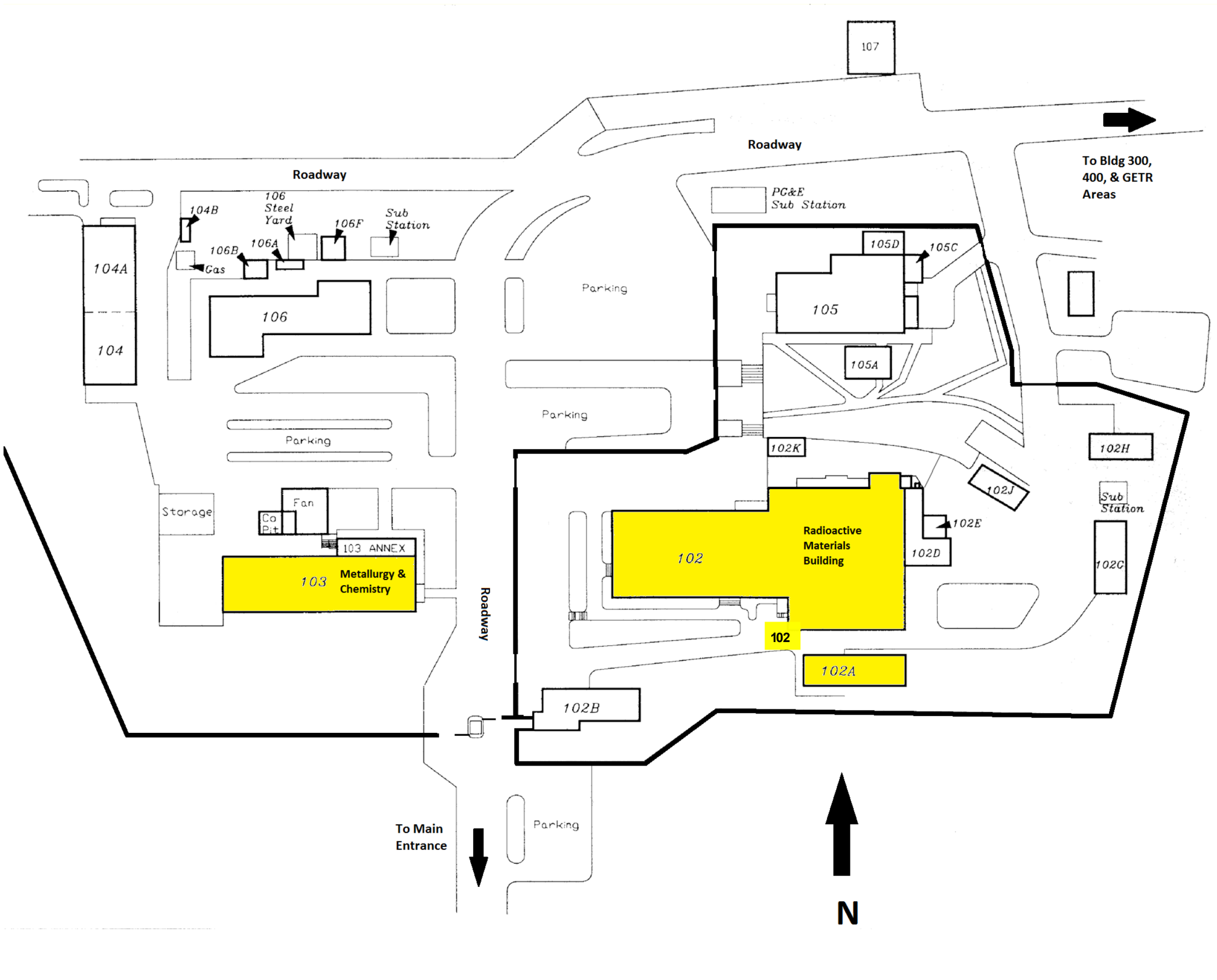
As discussed further below, there are four principal facilities covered by the SNM-960 license at the VNC site: (1) the Radioactive Materials Laboratory (RML) (Buildings 102 and 102A); (2) the Metallurgy, Chemistry and Ceramics Laboratory (Building 103); (3) the Hillside Storage Facility; and (4) the Waste Evaporation Plant (WEP) (Buildings 304 and 349). These are discussed first below, followed by brief descriptions of VNC site facilities outside the scope of SNM-960 for background and orientation.

FIGURE 1.2
Vallecitos Nuclear Center
(Not to Exact Scale)



Note: Yellow shaded areas denote principal SNM-960 licensed activity

FIGURE 1.3
100 Area And Surroundings
(Not to exact scale)



Note: Yellow shaded areas denote principal SNM-960 licensed activity

1.1.1.1 Radioactive Materials Laboratory (RML) - Buildings 102 & 102A

The first of the four principal SNM 960 licensed facilities on site is the RML. Previously SNM work on reactor fuel was performed in this building, however, the irradiated SNM has been removed to Hillside Facility for storage. Building 102 contains the hot cells and related activities. Building 102A contains the ventilation equipment. Currently the work at the RML involves by-product materials under a California license. Infrequently, however, there is the potential that small quantities of irradiated fuel are handled for engineering evaluation under SNM-960. The Hot Cells are highly shielded work cells that are operated with remote manipulators. The Radio Chemistry Laboratory within the RML is a laboratory area for performing radiochemical analysis on samples prepared in the RML. The Maintenance Shop includes decontamination, hot shop work and manipulator repair. The Waste Compactor processes dry active waste (< 200 mR@ contact and may contain trace amounts of Special Nuclear Material (SNM)) that is placed in drums or casks for disposal or storage. In addition there is a Storage Pool in the RML which is outside the scope of SNM-960 materials. See Photos 1.1 and 1.2. Additional details of the RML activities are provided in Sections 1.1.2 and 1.1.3.1.

Photo 1.1 – Building 102 east side

Photo 1.2 – Building 102A east side

1.1.1.2 Metallurgy, Chemistry, and Ceramics Laboratory– Building 103

The second of the four principal facilities supporting SNM-960 work 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 ^{235}U. The functions served by this facility are analytical chemistry services, and research and development. See Photos 1.3 and 1.4.

Photo 1.3 – Building 103 southeast corner view

Photo 1.4 – Building 103 typical lab room

1.1.1.3 Hillside Storage Facility

The third of the four principal facilities supporting SNM-960 work is the Hillside Storage Facility, which is located approximately midway between the deactivated VBWR and GETR areas. See Figure 1.10. Solid irradiated nuclear fuel and materials generated from the engineering evaluations and analytical work are stored in this facility . The Hillside Storage Facility includes earth-shielded horizontal tubes for storing liners and additional above ground space for lower level waste or other materials. See photo 1.5

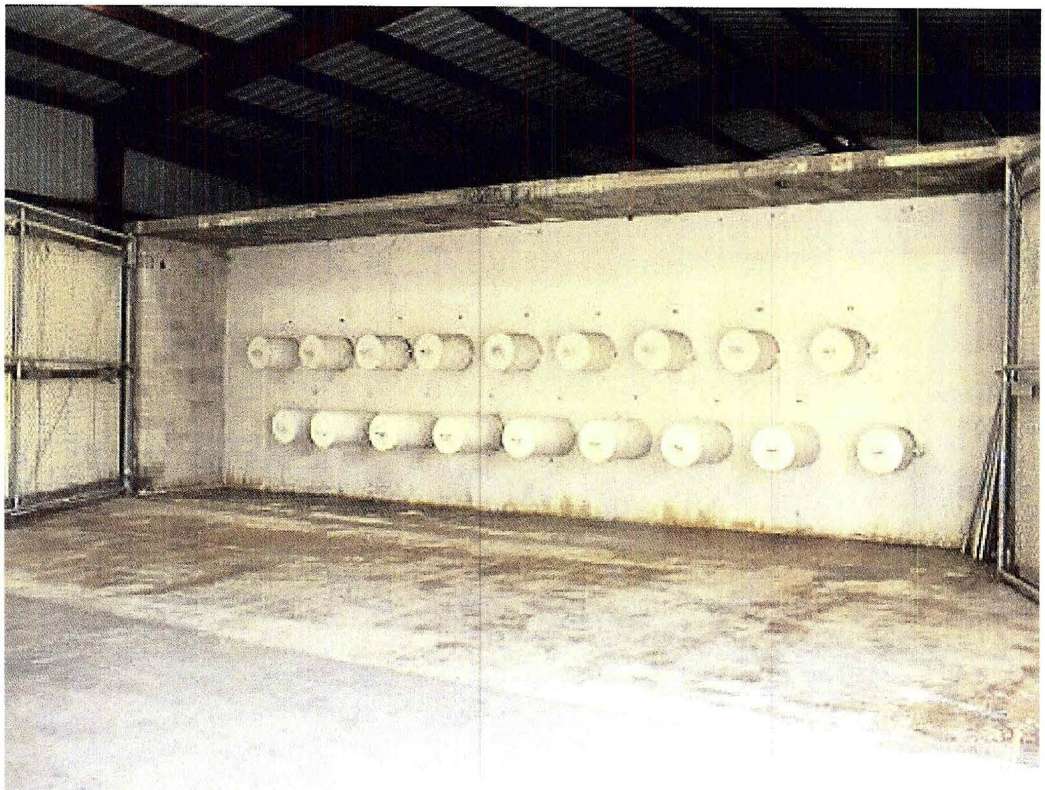


Photo 1.5 - Hillside Storage

1.1.1.4 Waste Evaporator Plant (WEP) – Buildings 304 and 349

The fourth of the four principal facilities supporting SNM-960 work is the WEP (Buildings 304 & 349), which is located adjacent to the deactivated VBWR site. The WEP is used to evaporate, concentrate and solidify liquid radioactive wastes generated at VNC prior to transfer to authorized waste disposal firms or waste burial sites. Such wastes may contain minute quantities of SNM associated with SNM-960 activities. Building 304 is used to store high radiation level waste (waste drums greater than 100 mR/hr) prior to transport for burial. The area is classified as a High Radiation Area and is posted accordingly. See Photo 1.6.

Photo 1.6 – WEP view of southside and 30,000 gallon tank

1.1.1.5 Building 106 Engineering Shop Building

Building 106 is not one of the principal SNM-960 facilities. It contains Engineering Shop Operations, which includes various maintenance shops (e.g., machine, carpentry, electric, maintenance support), and instrument calibration facilties. The only SNM-960 activity in this building involves handling plated plutonium calibration sources used for calibrating instuments.

1.1.1.6 Building 104 Warehouse and Training Building **(non SNM-960)**

Building 104 is used for warehousing and training and no activity associated with SNM-960 is conducted in it.

1.1.1.7 Building 105 NTR Building **(non SNM-960)**

Just north of Building 102 is Building 105. The principal facilities located in this building are the operational Nuclear Test Reactor, engaged primarily in neutron radiography under a separate NRC license (R-33).

1.1.1.8 Building 107 RCRA Waste Storage Building **(non SNM-960)**

Building 107 is the Hazardous Waste and Chemical Storage Building. No radiactive materials or SNM-960 licensed activites are allowed in this building. Activities in this building and the materials stored here pose no threat to the safe handling of nuclear material associated with SNM-960 authorized activities.

1.1.1.9 Reactors and Auxiliary Facilities **(non SNM-960)**

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

1.1.1.10 400 Area Offices and Laboratories **(non SNM-960)**

The 400 Area consists of two buildings, 400 and 401. It is devoted to offices, and non-radioactive materials laboratories.

1.1.2 SNM-960 RELATED PROCESS DESCRIPTIONS

1.1.2.1 Irradiated Fuel Evaluation (Building 102)

1.1.2.1.1 Hot Cells within the RML may be used for post irradiation examination of full length and partial length fuel rods. This includes visual and photographic examination, physical measurements such as bow and length, eddy current examinations, fission gas determinations, pellet clad interaction (PCI) investigations; and tensile and burst testing. Rods are sectioned, ground and polished for metallographic examination. Examinations using an electron microscope are conducted. Samples are prepared for additional measurements and evaluations in the radiochemistry laboratory and in the Metallurgy, Chemistry and Ceramics Laboratory (Building 103).

1.1.2.1.2 The Radio Chemistry Laboratory within the RML is used to perform calibrations and related laboratory activities, liquid scintillation analysis, burn-up analysis, and density and porosity determinations. Samples are also prepared for additional analytical work in the Metallurgy, Chemistry and Ceramics Laboratory (Building 103).

1.1.2.1.3 The Maintenance Shop is used for decontamination of equipment, standard contaminated equipment repair and the repair of the manipulators used in the Hot Cells.

1.1.2.1.4 The Dry Pit Storage is a below grade facility used for interim storage of highly radioactive material in metal canisters. See Section 1.1.3.1.3 for additional detail.

1.1.2.1.5 Waste processing involves only the compaction of dry active waste and the loading of canisters for storage in the Hillside Storage Facility.

1.1.2.2 Metallurgy, Chemistry and Ceramics Laboratory (Building 103)

The work in this laboratory involves both nuclear and non-nuclear materials. Within the scope of the SNM-960 license, the laboratory typically works with unirradiated uranium; however, a few samples from Building 102 are evaluated after the irradiation products have been significantly reduced.

Analytical chemistry and metallographic studies of UO2 powder and pellets constitute the majority of the activity in Building 103. This includes ICP spectrometry, mass spectrometry, gamma scanning, metallography, density and porosity, and electron microscope evaluations. See Section 1.1.3.2 for additional detail on building 103.

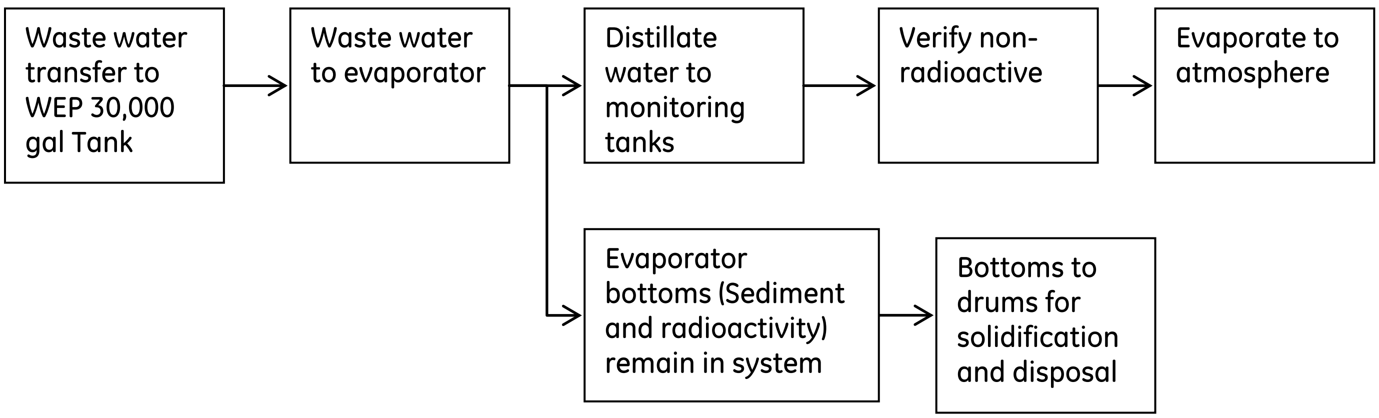
1.1.2.3 Irradiated Fuel Residues Storage (Hillside Storage Facility)

Waste containing irradiated uranium from engineering tests or reactor fuel and components is stored in a horizontal storage matrix containing sealed metal canisters. See Photo 1.5 and Section 1.1.3.3. Additional above ground space for lower level waste or other materials is available within the facility. This material is awaiting transfer to DOE.

1.1.2.4 The WEP

The waste evaporator receives liquid waste in a special 30,000-gallon tank and processes it through evaporation to reduce the volume. After concentration the liquid waste is solidified for disposal at a licensed disposal facility. See Figure 1.4 and Section 1.1.3.4 for additional detail.

FIGURE 1.4
WEP Process Flow Diagram



1.1.3 FACILITIES AND EQUIPMENT FOR HANDLING SNM

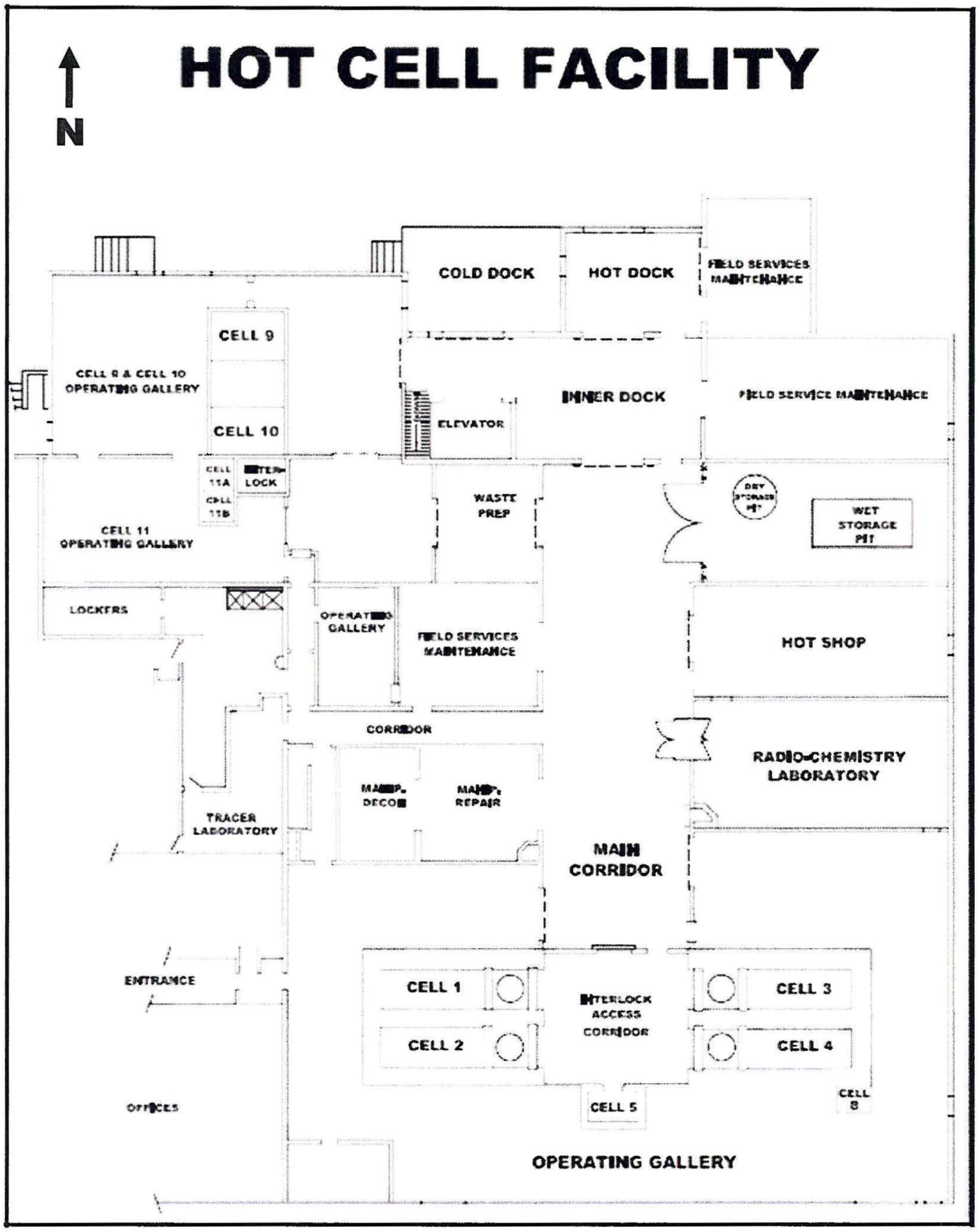
1.1.3.1 RML – Building 102 and 102A

1.1.3.1.1 Location and General Description

Building 102 is a single-story concrete and steel structure with a basement. Building 102A is a concrete and steel, two story industrial building designed to contain the ventilation equipment. The laboratory areas on the main floor of Building 102 are separated from the general office areas by a firewall. Radiological control procedures are in place for ingress and egress of the laboratory area. Included in the building design are general service facilities including a ventilation system, decontamination area, and general maintenance areas as described later. The hot cell facility is shown in Figure 1.5.

SNM-960 activities typically take place in Cells 3, 5 and 11, Dry Pit Storage and the Radiochemistry Laboratory, and are supported by the Main Corridor, Decon Room, Manipulator Repair, Hot Shop, Waste Handling, and the Building Ventilation System. See Sections 1.1.3.1.2 – 1.1.3.1.11 for additional detail on these facilities and equipment within the RML.

FIGURE 1.5
RML, Hot Cell Facility (Not to exact scale)



1.1.3.1.2 Shielded Enclosures (Hot Cells)

Cells 1-4, located on either side of the south end of the Main Corridor are designed to safely handle in excess of one million curies of 1-MeV Co^{60} gamma radiation. See Photo 1.7. The cell walls are constructed of Ferro-phosphorous aggregate type material poured to a density of 300 pounds per cubic foot. The walls are nominally 36 inches thick to a height of approximately 12 feet where they reduce to 24 inches where shielding is not as critical and to provide a set-back for the overhead manipulator rails. A nominally 3-foot thick concrete roof provides shielding in the vertical direction.

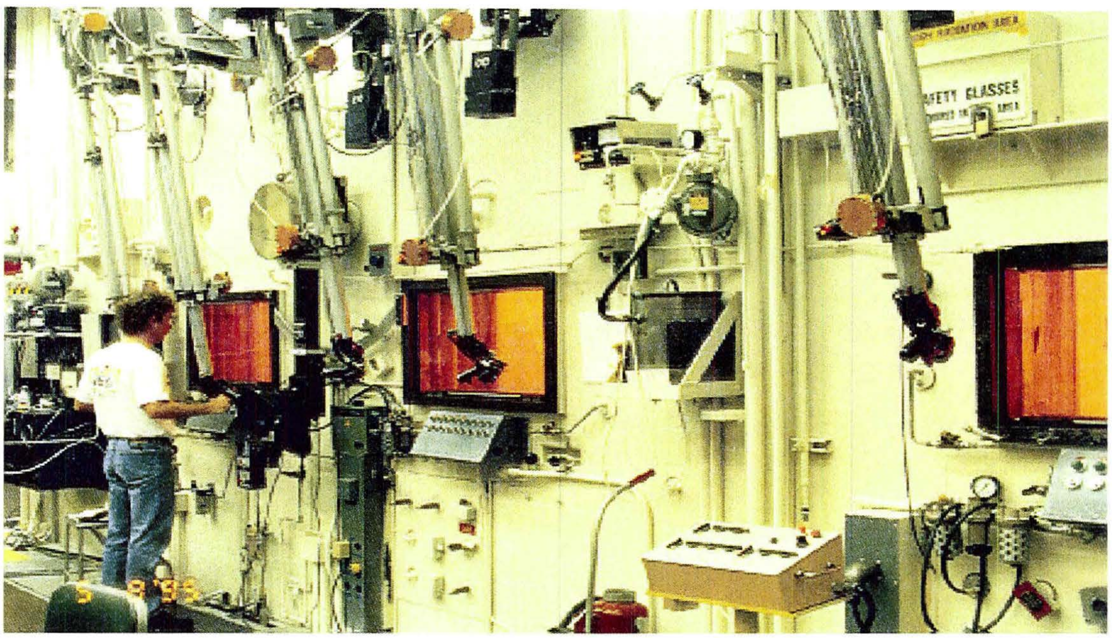


Photo 1.7 – Hot Cell

Each cell can be equipped with an overhead bridge-mounted manipulator and a 3-ton bridge crane running on the same rails to service the entire cell.

Each cell is fitted with four (4) nominally 3-foot thick lead glass windows with a minimum density of 6.2 gm/cc that provides shielding equivalent to the cell walls.

Each of these cells includes a 6-foot long radiation lock for entry from the access corridor. These radiation locks are formed by electronic controlled, hydraulically operated bi-parting steel shielding doors. The outer door is nominally 18-inches thick and the inner door is nominally 15-inches thick. The doors are interlocked by an electronic control system so that only one set of doors can be opened at any given

time. An override system is operated by key control and the key is maintained locked and accessible only by the area manager as described in approved operating procedures. Each radiation lock includes a nominally 3.5-foot diameter by 9-foot deep pit to accommodate large casks and long irradiated fuel assemblies.

Each cell has four operating stations. Each station is equipped with a pair of through-wall commercial CRL master-slave manipulators.

Water is supplied to the hot cells by an independent, 23-gallon pressurized storage tank located in the mezzanine. This tank is filled by a back-flow-prevented (air gap) water line. The system pressure is limited to approximately 60 psi.

Cell 5 is a smaller cell used primarily for metallographic examination and micro hardness. The cell walls are constructed of nominally 18-inch thick magnetite concrete and include two nominally 13-inch thick lead glass windows and a steel safe style door opening into the Main Corridor. The cell is equipped with two (2) through-wall master-slave manipulators.

Cell 11 contains a scanning electron microscope used to examine SNM bearing specimens prepared in Cell 5. No processing or storage of SNM takes place in this cell.

1.1.3.1.3 Dry Storage Pit

The in-floor Dry Storage Pit consists of 19-recessed pipe tubes fabricated of nominally 6-inch Schedule 40 steel pipe, of nominally 46. 5-inch length, attached to two horizontal circular steel plates of nominally 91. 25-inch diameter and 6 inch thickness. The device rests on a ledge cast in the concrete floor so that the top surface is flush with the floor and the tubes extend downward below the floor surface. The center-to-center spacing of the pipes is a nominal 18 inches. Twelve-inch thick stepped plugs provide top shielding. 1 MeV gamma sources of up to 500R/hr at the liner surface are shielded to a dose rate of 2.5 mR/hr at the surface of the dry pit storage. Materials can be stored in the Dry Storage Pit in sealed metal tubes called "waste liners". Movement of waste liners is accomplished using a shielded transfer cask with top and bottom entry ports. Handling tools that are attached to the waste liner remotely permit insertion and withdrawal. Each storage locator tube includes drainage holes to prevent the build-up of liquid.

1.1.3.1.4 Radiochemistry Laboratory

The Radiochemistry Laboratory, used primarily for the analysis of samples of irradiated fuel materials from the RML, is located immediately adjacent to the Hot

Cell area. These analyses are performed in hoods and glove boxes that are connected to the Building 102 ventilation system and maintained to provide a minimum of 125 linear feet per minute face velocity.

1.1.3.1.5 Main Corridor

The Main Corridor runs down the center of Building 102 from the concrete service pad to the hot cell access area for Cells 1 – 5 and includes access to the other function areas of the building. Movement of radioactive material and equipment in this area is facilitated, as needed with two forklifts – one a nominal 25, 000-pound capacity, propane powered unit and the other a nominal 4,000 pound capacity, electric powered unit. An air pallet is also used to move loads that require weight distribution to ensure that floor loading is not exceeded. See Photo 1.8.

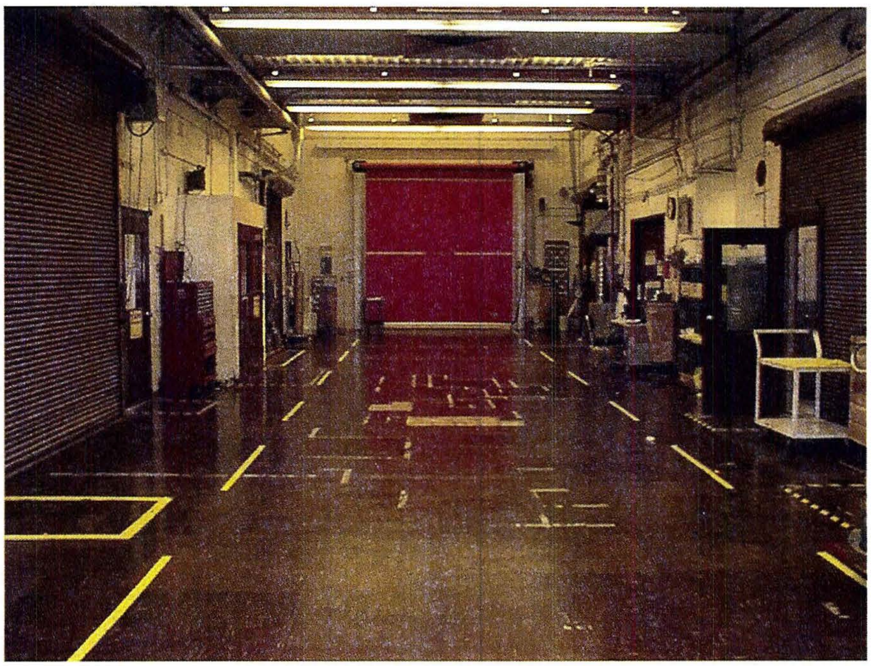


Photo 1.8 - Building 102 Main Corridor

1.1.3.1.6 Decon Room

The Decon Room is associated with manipulator repair and provides a location to decontaminate equipment parts before the actual repair procedure takes place. Any decon materials produced here either go to the Waste Handling Area or are routed to the Waste Evaporation Area.

1.1.3.1.7 Manipulator Repair

The Manipulator Repair area is used for the repair and routine maintenance associated with the manipulators used in the Hot Cells. No SNM processing occurs in this area.

1.1.3.1.8 Hot Shop

The Hot Shop is used for the general maintenance of equipment used in Building 102. Residues generated from equipment that requires decontamination during the maintenance process is routed to Waste Handling (if solid) or Waste Evaporation (if liquid). No SNM processing occurs in this area.

1.1.3.1.9 Waste Handling

At all locations where contaminated materials are processed, solid waste residues are collected in cans, drums casks, or other containers. Dry active Class A waste is shipped to a commercial off-site licensed disposal facility. Dry active Class B & C waste is stored in the hot cell canisters at the Hillside Storage Facility.

1.1.3.1.10 Building 102 Ventilation System

VNC uses, to the extent practical, process or other engineering controls (e.g., containment or ventilation) to control the concentration of radioactive material in air.

The Building 102 Ventilation System is a demand driven system based on six exhaust fans (2 of 6 provide normal operation) with a nominal exhaust flow rate of approximately 43,000 cubic feet per minute. There are 2 booster fans that can be switched on if additional flow is necessary.

Inlet air to the building is provided by multiple modulated, conditioned and filtered air-handling units, with a combined capacity of approximately 65,000 cubic feet per minute. The inlet air is adjusted as necessary to maintain the appropriate balance with the exhaust air. See Figure 1.6.

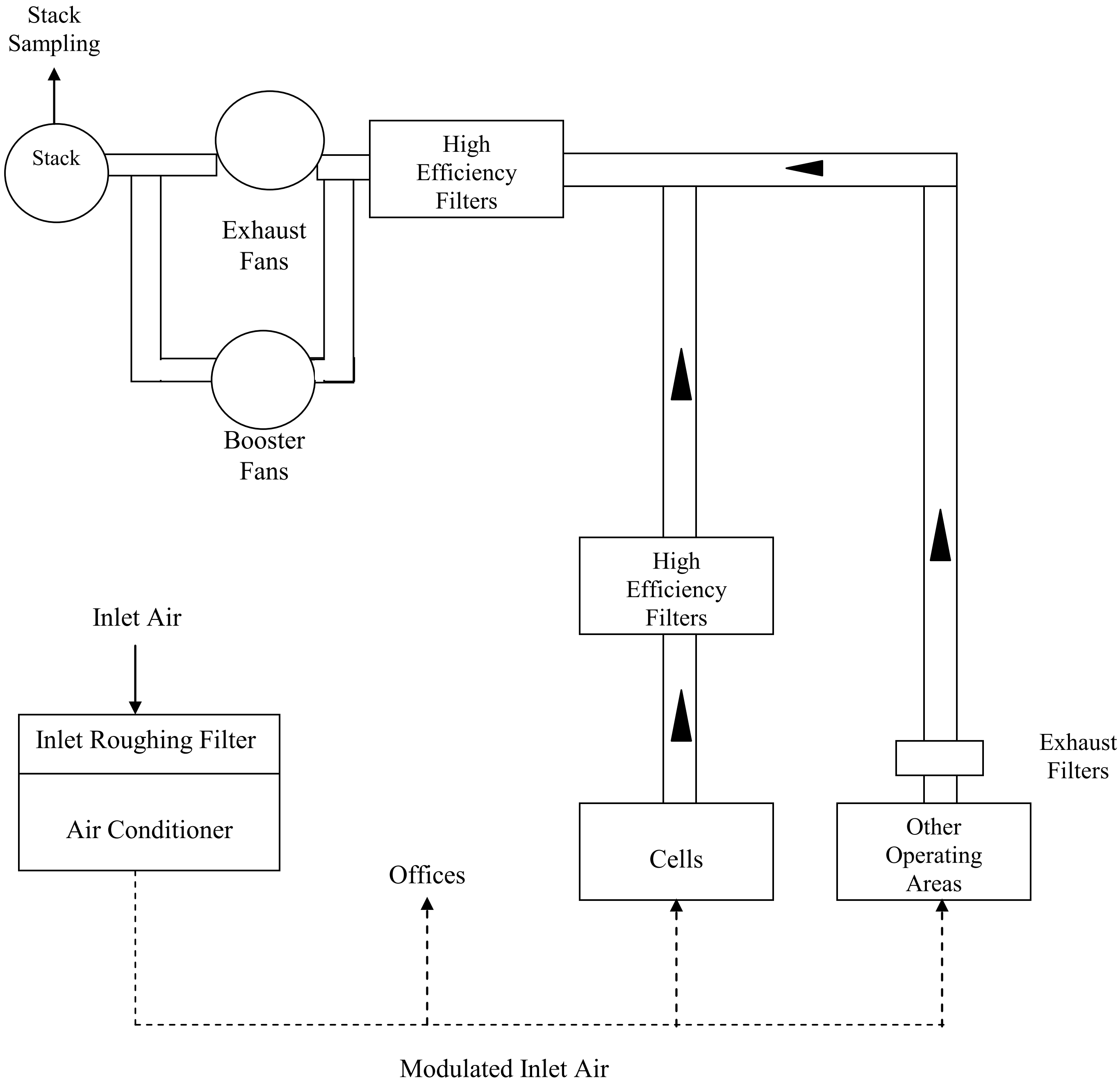
Airflows in the hot cell facility are generally single pass with the exception of a portion of the RML operating gallery that uses recirculated air, recirculated through absolute filters. Air balance is maintained so that air flows from areas of lower potential contamination to areas of higher potential contamination. Operational areas are generally managed to provide between 6 and 40 air changes per hour. Small static pressures, e.g. -0.01 to – 0.03 inches of water, are used in the isolation of thin-walled general work areas. Static pressures of approximately -0.02 to -0.2 inches of water are used to isolate the RML Hot Cells. Glove Box and hood contamination control

utilizes approximately -0.5 to -1.0 inches of water static pressure to maintain a minimum of 125 linear feet per minute or greater airflow across openings.

Exhaust air from the RML Hot Cells is prefiltered at the outlet of each cell and routed through a bank of 10 HEPA filters in the basement of Building 102 before joining the main exhaust stream in Building 102A. Fire protection is provided by CO2 suppression and water fog system for the HEPA filter bank.

Exhaust air is routed to Building 102A through overhead ducts. The ducting from Building 102 includes a fog water spray system for fire protection. In Building 102A there is a bank of 90 HEPA filters that the exhaust air passes through before discharge to the atmosphere through a nominally 66-inch diameter, 75-foot high stack.

FIGURE 1.6
Building 102 Ventilation System



Backup power is provided to the ventilation system (and other critical systems including the Switchboard, the stack monitoring equipment and the exhaust fans supplying air to areas where radioactive material is handled) by a nominal 335-KW diesel-driven generator. The generator is automatically started on loss of commercial power and is designed to reach maximum output within approximately 1-minute. The fuel supply is from a nominally 1, 000-gallon storage tank. The fuel consumption rate is on the order of 20 – 25 gallons per hour, and therefore the fuel supply should be sufficient for approximately 40 hours of continuous, full load operation.

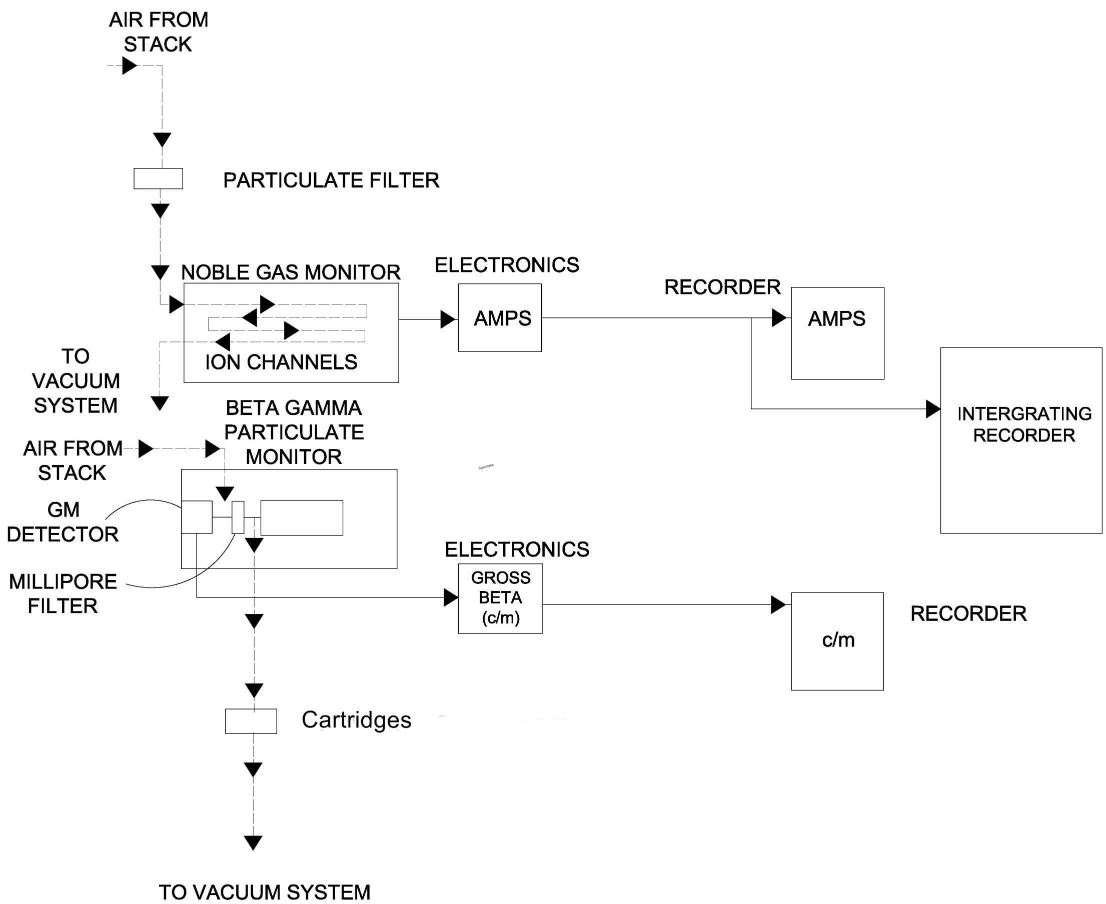
The exhaust fans and building alarm panel are supplied power through a Motor Control Center (MCC) that is of solid-state design and battery powered. This unit provides the logic to support switching between normal commercial power and emergency power. In addition, if an exhaust fan fails during operation, the MCC sounds an alarm with identification of the failed fan on the control panel, shuts down the failed fan and starts a replacement fan. In the case of a fire detected by the detectors in the ducts and filter housings, the MCC reduces the flow of air in the exhaust until the fire signal is cleared manually at the control panel.

The exhaust system is monitored for atmospheric discharges with systems designed to detect beta-gamma particulate and gaseous activities. The sampling system is shown in Figure 1.7. The beta-gamma monitor uses nominally 1.5 cubic feet per minute sample rate and passes the sample stream through a 47-mm high efficiency filter (~99.9% of all particles >0.3µm diameter). The 47-mm filter is monitored by a 2-inch diameter pancake-type GM detector having a window thickness of 7-8 mg/cm^2 located approximately ½-inch from the filter. The GM detector is calibrated against ^{36}Cl with a 4π efficiency of nominally 35%. The Noble Gas Monitor is designed to measure activity of gaseous radioactive discharges. The sampling rate is nominally 3 cfm. The sample gas passes through a pair of electrodes to remove ionization from previous decays. It then enters a nominally 16-liter volume where ions produced by radioactive decay are collected by electrodes. The design response for ^{133}Xe is 1.45×10^{-7} amp/µCi/cc.

The 47-mm high efficiency filter is periodically sent to the laboratory for alpha and beta-gamma analysis. The results of this analysis are used to document the atmospheric discharges from Building 102.

Hoods and other localized ventilation designs are utilized to minimize personnel exposure to airborne contamination. Activities and process equipment that generate

FIGURE 1.7
Basic Stack Sampling System For Building 102 (Typical)



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
BUILDING 102 VENTILATION SYSTEM EQUIPMENT AND CAPABILITIES

Equipment	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 environment
High Velocity Local Ventilation	Designed to maintain an average airflow 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 environment
	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 to environment

1.1.3.2 Metallurgy, Chemistry and Ceramics Lab – Building 103

1.1.3.2.1 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 hallway runs the length of the building on each floor. The laboratories are currently located on one side of the hall and offices on the other as shown in Figures 1.8 and 1.9.

FIGURE 1.8
Building 103 First Floor Plan

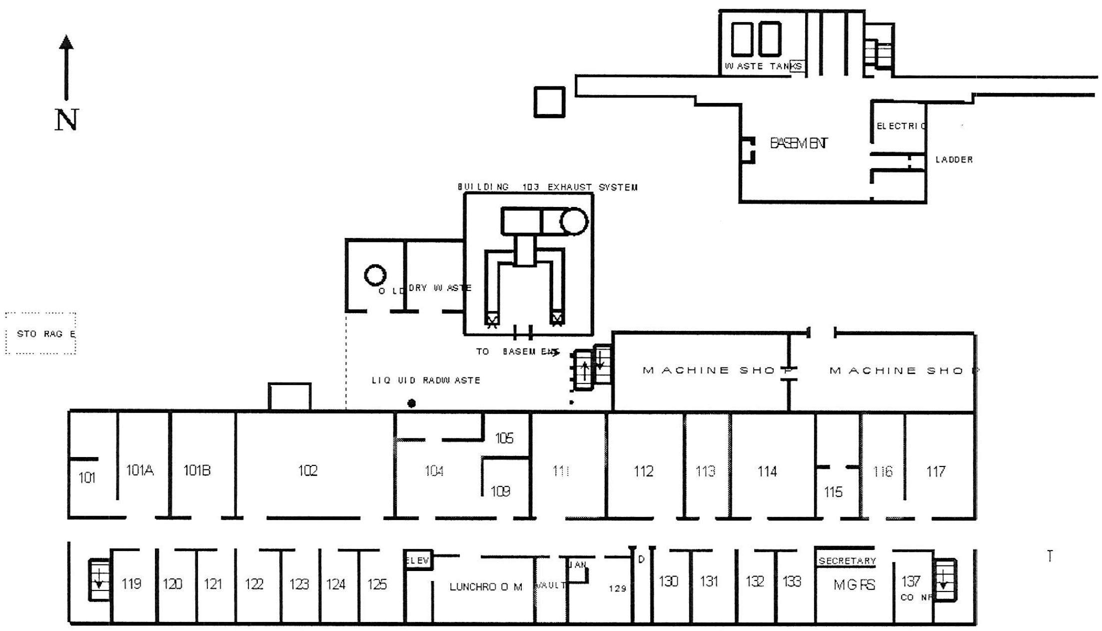
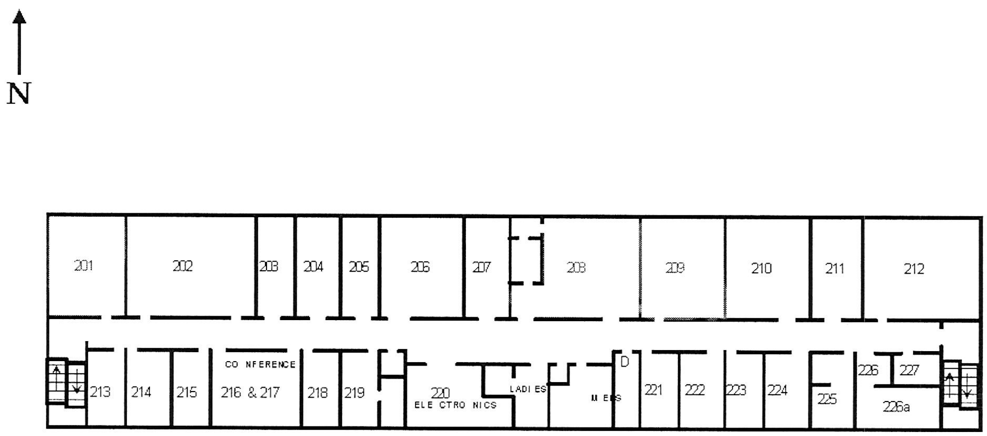


FIGURE 1.9
Building 103 Second Floor Plan



1.1.3.2.2 Laboratory Facilities and Equipment

Analytical and other laboratory work in Building 103 utilize samples containing very small quantities of radioactive material as is characteristic of this type of work. 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.

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. See Photo 1.4

1.1.3.2.3 Building 103 Ventilation System

VNC uses, to the extent practical, process or other engineering controls (e.g., containment or ventilation) to control the concentration of radioactive material in air.

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 process in one of two parallel banks of high-efficiency filters and is discharged through a stack which is approximately 48-feet-high, with a 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 particulate activity at a point approximately 24 feet below the top. The particulates are collected on a high efficiency filter. 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. Typically 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.

1.1.3.2.4 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 SNM are handled in hoods or glove boxes to minimize fire hazards in these enclosures. For example, metal containers are typically used for pyrophoric materials.

1.1.3.2.5 Radioactive Waste Handling – Building 103

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 6,000-gallon capacity, are provided. The tanks are equipped for representative sampling and for draining to drums or the site transport tank. Typically, tank wastes are sent to the WEP for concentration and solidification.

1.1.3.3 Hillside Storage Facility

FIGURE 1.10
Hillside Storage Facility

Note: The yellow shaded area denotes principal SNM-960 licensed activity

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Figure 1.11
Waste Liner (Metal Canister)

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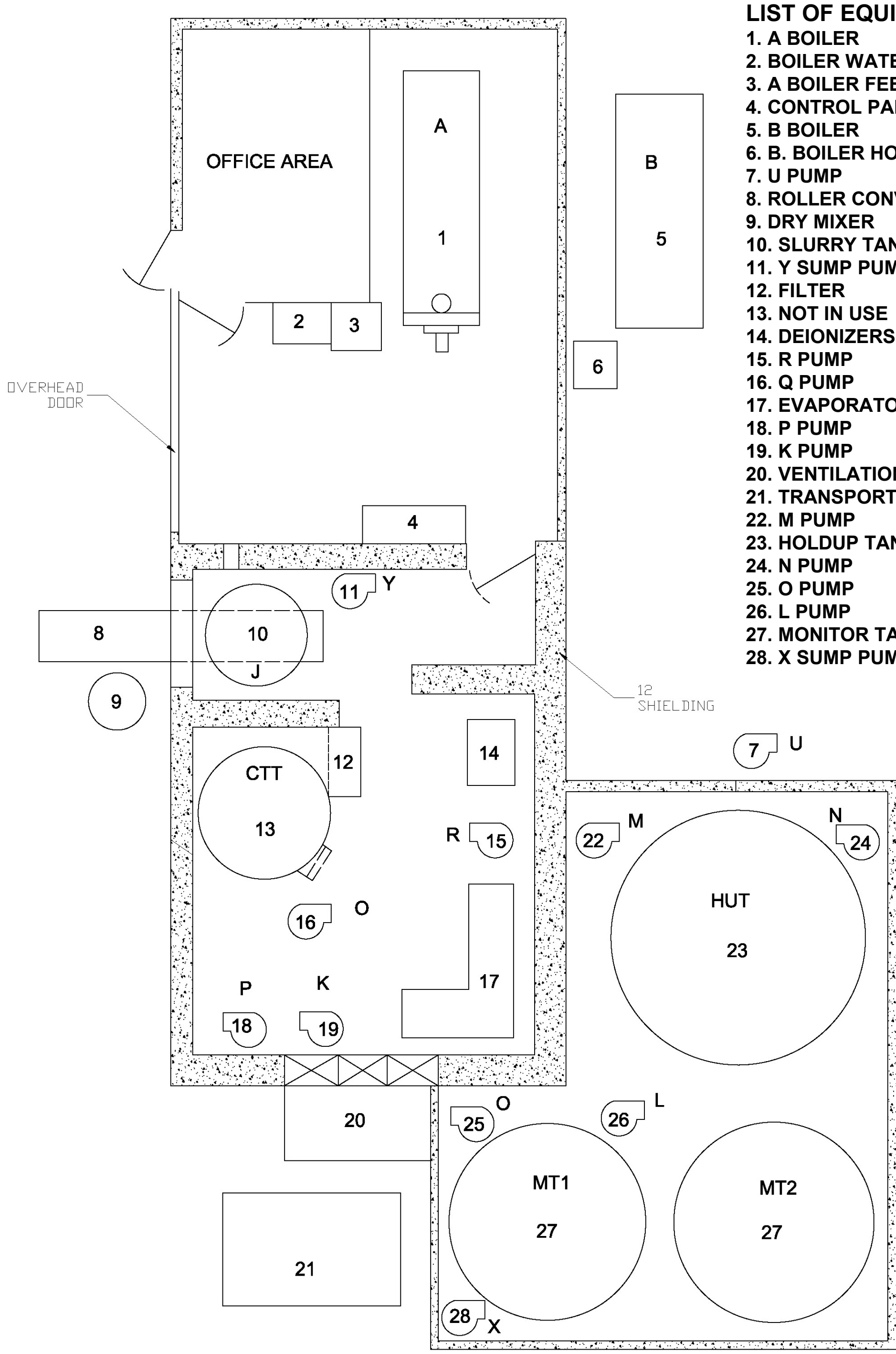
1.1.3.4 WEP – Buildings 304 & 349

A liquid processing facility (See Figure 1.12 and Photo 1.6) is provided with sufficient capacity and capability to enable collection, evaporation,sampling, analysis, and solidification for disposal in a licensed burial facility. Liquid waste volumes are reduced for disposal using techniques such as evaporation, pH adjustment, sedimentation, filtration, ion exchange, solidification and packaging of the concentrates. These activities primarily occur at the waste evaporation plant.

Known or potentially contaminated liquid wastes are routed from laboratory sinks and gravity drains to retention tanks located in each building where such wastes are generated. These wastes are transferred periodically to the waste evaporation plant for concentrating and solidifying the liquid wastes which are described in this application. The transfer takes place in a 1,500-gallon stainless steel transfer tank or smaller containers as appropriate.

FIGURE 1.12
Waste Evaporator Building



LIST OF EQUIPMENT

1. A BOILER
2. BOILER WATER
3. A BOILER FEED
4. CONTROL PANEL
5. B BOILER
6. B. BOILER HOT WELL
7. U PUMP
8. ROLLER CONVEYER
9. DRY MIXER
10. SLURRY TANK (J)
11. Y SUMP PUMP
12. FILTER
13. NOT IN USE
14. DEIONIZERS
15. R PUMP
16. Q PUMP
17. EVAPORATOR SYSTEM
18. P PUMP
19. K PUMP
20. VENTILATION SYSTEM
21. TRANSPORT TANK (TT)
22. M PUMP
23. HOLDUP TANK
24. N PUMP
25. O PUMP
26. L PUMP
27. MONITOR TANKS
28. X SUMP PUMP

Prior to transferring liquid wastes to the WEP, samples are taken for gross alpha and beta, total uranium and plutonium, and ^{235}U analyses. A log of the total quantity of ^{235}U 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 ^{235}U inventory is typically between 50 and 70 grams.

The 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 undertaken at the WEP involve the transfer of liquid radioactive wastes generated at various facility and laboratory installations atVNC, including those containing small quantities of SNM. The estimated annual total site throughput is 100,000 gallons of liquid waste. SNM-960 activities contribute a very small portion of this volume.

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 stainless steel tank. The waste transfer tank is equipped with internal vertical baffles to prevent cyclic shifting of the liquid material during transport. External protuberances such as pipes, valves, and gages, 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 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 and, as appropriate, discharge into 17-H 55-gallon drums or other approved container for solidifiction. 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 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 to Building 304 for storage prior to shipment to burial sites.

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 potential 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 cubic feet per minute is driven by an electrically powered blower mounted at the base of the stack.

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

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 VNC for licensed waste disposal. Waste drums are not buried on site.

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

1.2 INSTITUTIONAL INFORMATION

This application addresses NRC license number SNM-960 (Docket #70-754).

1.2.1 IDENTITY AND ADDRESS

This application for license renewal is for a period of 10 years and is filed by GE-Hitachi Nuclear Energy Americas LLC (GEH). GEH's principal place of business is 3901 Castle Hayne Road, Wilmington, NC, 28401. The full address for Vallecitos Nuclear Center is: Vallecitos Nuclear Center, 6705 Vallecitos Road, Sunol, California 94586.

1.2.2 CORPORATE AND FINANCIAL INFORMATION

The VNC site is corporately owned and SNM-960 is held by GEH, 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 its principal place of business at 3135 Easton Turnpike, Fairfield, Connecticut 06431. General Electric Company is not controlled by any alien, foreign corporation, or foreign government. General Electric Company is a publicly held corporation whose stock is traded on the principal stock exchanges. The remaining 40% is owned by Hitachi.

GEH is a single member Delaware limited liability company. Its officers are a President and Chief Executive Officer (CEO), a Chief Financial Officer (CFO) and a Secretary, all of whom are US Citizens and residents of the United States. Each officer's business address is 3901 Castle Hayne Road, Wilmington, NC, 28401.

GE-Hitachi Nuclear Energy Americas LLC has a Board of Managers comprised of seven representatives, five of whom have voting rights. Three voting members are appointed by General Electric Company and are U.S. citizens, and two voting members are appointed by Hitachi. In addition, Hitachi-GE Nuclear Energy Co., Ltd's CEO and the CFO serve on the GEH 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 material are authorized under SNM-960:
- [[

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- [[

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1.2.4 ACTIVITY

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

Authorized activities under SNM-960 includes possession and storage of the material listed in Section 1.2.3 at the Hill Side Storage Facility; and possession and use of limited quantities (350g) of SNM for laboratory analysis and engineering studies.

1.3 SPECIAL AUTHORIZATIONS AND EXEMPTIONS

1.3.1 ACTIVITIES NOT REQUIRING PRIOR NRC AUTHORIZATION BY LICENSE AMENDMENT

1.3.1.1 GEH is authorized to make changes to the site, structures, processes, systems, equipment, components, computer programs and activities of personnel without NRC approval provided the changes do not:

- Create new accident scenarios that unless prevented or mitigated could cause the licensee to violate the GEH requirements,
- Involve new processes, technologies or control systems for which the licensee has no prior experience.
- Violate any other NRC regulations, license condition or order.

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 EXEMPTION TO CRITICALITY MONITORING SYSTEM REQUIREMENTS

VNC has been 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 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,

- Each area in which is stored one shipment of packages containing SNM licensed pursuant to 10 CFR 71 for transport outside the confines of VNC insofar as the requirements of Section 70.24 pertain to the material contained in such shipments,

- Each area where there is an insufficient quantity of fissile material to form a critical mass.

1.3.4 REACTOR PRODUCED TRANSURANICS

In accordance with NRC correspondence dated July 24, 2008 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.

1.3.5 PYSICAL SECURITY

VNC maintains the Physical Security Plan approved by the NRC on July 24, 2008 as amended in accordance with regulatory requirements.

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.

1.3.7 MATERIAL CONTROL AND ACCOUNTING

A fundamental Nuclear Material Control Plan is not required for SNM 960 licensed activities because the inventory is less than 1-effective kilogram of SNM as provided in 10 CFR 74.31. However, VNC continues to maintain an acceptable material accountability program pursuant to 10 CFR 74.11, 74.13, 74.15, 74.19 and to ensure the SNM currently at VNC is adequately controlled and reported.

1.4 GENERAL PLANS, USES AND POSSESSION LIMITS OF SPECIAL NUCLEAR MATERIALS

This license renewal application requests authorization under Title 10, Code of Federal Regulations, Part 70, to receive, possess and store the SNM and associated byproduct material produced by the irradiation as described in Sections 1.2.3 and 1.2.4.

1.4.1 VALLECITOS NUCLEAR CENTER

The SNM used in connection with activities authorized by License SNM-960 at VNC 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, ^{235}U, 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 permanently or semi permanently mounted or stored. Records of leak test results shall be kept in units of micro curies 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 license SNM-960 and applicable NRC regulations. Employees are provided a simple mechanism to report and have safety concerns addressed.

2.2 ORGANIZATIONAL RESPONSIBILITIES AND AUTHORITY

2.2.1 KEY POSITIONS (FIGURE 2.1)

Responsibilities, authorities, and interrelationships among the VNC organizational functions with responsibilities for safe operations and design changes are specified in approved position descriptions and in documented and approved practices. A single individual may be responsible for more than one position or a position may be split between two or more individuals.

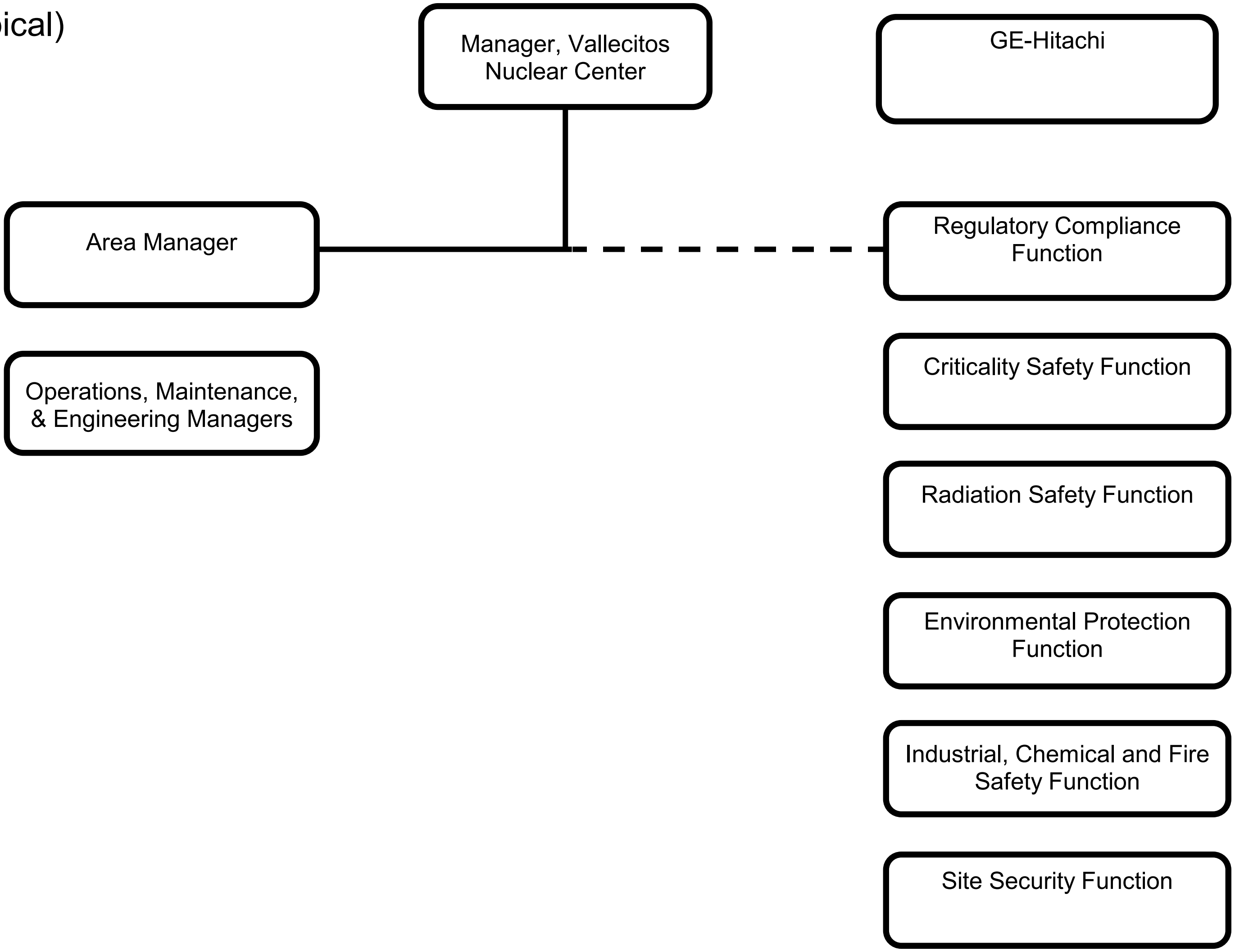
2.2.1.1 Manager, Vallecitos Nuclear Center

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 applicable building codes and regulations.

The minimum qualifications of the Manager, Vallecitos Nuclear Center are a bachelor's degree and two years experience in nuclear operations or a high school diploma and five years supervisory or technical experience in a nuclear, manufacturing or other technical field. 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 shutdown any process or facility. The Manager, Vallecitos Nuclear Center must approve restart of any operation shutdown.

Figure 2.1
VNC Organization Chart
(Typical)



2.2.1.2 Area Manager

An Area Manager is designated as the individual who is responsible for ensuring that operations and activities necessary for safe operations and protection of the environment are conducted properly within the assigned area of the facility. Designated Area Manager responsibilities include assuring that licensed activities conducted in accordance with properly issued and approved procedures. The Area Manager also assures that new employees receive instruction in criticality safety, radiation safety, site emergency procedures, general industrial safety and operating procedures commensurate with assigned duties. The Manager, Vallecitos Nuclear Center approves the assignment of the Area Manager.

The minimum qualifications of the Area Manager is one of the following three options:

Option 1, a combination of:

- Bachelor’s degree or equivalent in a science or engineering subject
- Two years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- One year of supervisory or technical experience in nuclear operations.

Option 2, a combination of;

- Bachelor’s degree or Associate degree;
- Three years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- One year of supervisory or technical experience in nuclear operations

Option 3, a combination of;

- High School diploma;
- Five years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- Two year of supervisory or technical experience in nuclear operations

The Area Manager 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.

2.2.1.3 Regulatory Compliance Function

The Regulatory Compliance function is administratively independent of production responsibilities and has the authority to shutdown any process or facility in the event that adequate controls for any aspect of safety may not be assured. This function has designated overall responsibility to ensure compliance with federal, state and local regulations and laws governing operation of the licensed activities.

The manager of the Regulatory Compliance function must hold a bachelor's degree or equivalent and have two years of management experience in assignments involving regulatory activities or a high school diploma and five years supervisory or technical experience in a nuclear, manufacturing or other technical field.

2.2.1.4 Criticality Safety Function

The site criticality safety function is administratively independent of production responsibilities, has oversight responsibility for the material storage area, and has the authority to shutdown potentially unsafe operations. The criticality safety function establishes the criticality safety program including design criteria, procedures and criticality safety training in accordance with Chapter 5.

Minimum qualifications for a member of the criticality safety function are a bachelor's degree or equivalent in a science or engineering subject or at least five years professional experience in applied criticality safety.

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

Designated areas of responsibility include:

- Establish the radiation protection and radiation monitoring programs, including the As Low As Reasonably Achievable (ALARA) program
- 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:

- The site radiation safety function leader shall hold a bachelor's degree or equivalent in a science or engineering subject, and have at least five years of professional experience in applied radiation protection. An alternate minimum experience qualification is a professional certification in health physics (CHP).
- A site radiation monitor technician (RMT) in the radiation safety function shall meet one of the following:
- Hold a bachelor's degree or equivalent in a science or engineering subject.
- Have a high school diploma and at least two years experience in Applied Radiation Protection or
- Have eight years experience in health physics or radiation protection.

- Have a high school diploma or equivalent with two years experience in handling radioactive materials or
- Two years of college and four months experience.

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

Designated areas of responsibility include:

- Identify environmental protection requirements from federal, state and local regulations which govern SNM-960 operations
- 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
- Assure proper federal and state permits, licenses and registrations for non-radiological discharges from the facilities

2.2.1.7 Industrial Health and Safety (Including Chemical and Fire Safety Functions)

Industrial Health and Safety maintains programs generally related to OSHA and Cal/OSH regulations. In regards to SNM-960 operations, functions specifically pertinent are the chemical and fire safety functions. The function is administratively independent of the production responsibilities and has the authority to shutdown operations with potentially hazardous health and safety conditions.

Designated areas of responsibility include:

- Identify industrial health, chemical safety and fire protection requirements from federal, state, and local regulations which govern the SNM-960 operations
- Develop practices regarding non-radiological chemical hazards that could affect the safety of licensed materials

- Provide advice and counsel to Site employees and management on matters of industrial health, chemical and fire safety
- Provide consultation and review of new, existing or revised equipment, processes and procedures regarding industrial safety, chemical safety and fire protection

2.2.1.8 Site Security Function

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

- Provide physical security for the site
- Provide advice and counsel to Site employees and management on matters of site security

2.2.2 MANAGEMENT CONTROLS

Management controls for the conduct and maintenance of VNC's health, safety and environment protection programs are contained in formally approved, written procedures prepared in compliance with a formal document control program.

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

Take all necessary steps to plan and organize the work within their area of responsibility, in accordance with approved procedures.

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.

Integrate the results of reviews, inspections, engineering assessments and investigations 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 their positions, and before assigned tasks are independently performed. Training relative to safety includes instructions to workers in accordance with 10 CFR 19.12, storage, transfer, and use of radioactive materials, minimization of exposures to radiation and limits, maintaining radiation exposures and radioactivity in effluents ALARA, radiation safety principles, radiation exposure reports, and risks involved in receiving low level radiation exposure. Training relative to safety is also provided on nuclear criticality safety principles, chemical and fire safety, emergency response, and the responsibility to promptly report any condition that may lead to, or cause a violation of regulations, license requirement, or create unnecessary exposure.

2.3.1 ALARA COMMITTEE

The ALARA Committee is described in Chapter 4, Section 4.2.

2.3.2 VALLECITOS TECHNOLOGICAL SAFETY COUNCIL

Oversight responsibilities of the Vallecitos Technological Safety Council (VTSC) includes SNM 960 licensed activities. The VTSC is an independent review body and consists of a minimum of five senior members of GE-Hitachi Nuclear Energy's technical and/or management personnel. Its proceedings, findings and recommendations are reported in writing to the Manager, Vallecitos Nuclear Center, manager Regulatory Compliance, 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.

2.4 CHANGE MANAGEMENT

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 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 the requirements of 10 CFR 70.61 through 70.76 do not apply to VNC activities licensed under SNM-960. VNC is not engaged in any of the qualifying activities related to that material and is therefore excluded. Consequently there is no requirement for an Integrated Safety Analysis, items relied on for safety (IROFS), or management measures as defined in 10 CFR Part 70. Nevertheless, this application describes applicable safety programs and controls utilized by GEH to ensure those programs are effectively implemented.

3.2 BASIS OF SAFETY

The basis of safety for SNM-960 licensed activities are defined by the organizational and administrative structure defined in Chapter 2, and the safety program elements and commitments in Chapters 4 through 10, and is assured by management measures that are in place in the operation of the organization.

CHAPTER 4.0
RADIATION SAFETY

4.1 RADIATION PROTECTION PROGRAM

A radiation protection program has been established to ensure compliance with the requirements of 10 CFR 20. A system of written Procedures establishes the site radiation protection and regulatory compliance programs. The manager of the Regulatory Compliance function issues the procedures with review and comment from the managers of the major organizational components located on the site. Requirements are established to prevent or minimize the hazards of radioactivity and radioactive materials. The useage of terminology in this Chapter is consistent with the definitions in 10 CFR 20.1003. Key personnel, minimum qualifications and radiation protection independence from operations are addressed in Chapter 2, Section 2.2.1. Ventilation and containment systems are addressed in Chapter 1, Sections 1.1.3.1.10 and 1.1.3.2.3.

The content and implementation of the radiation protection program is reviewed annually pursuant to 10 CFR 20.1101. The annual radiation protection program review considers means to enhance the effectiveness of the program.

Instructions to workers in accordance with 10 CFR 19.12 regarding the use of radiation and radioactive materials is described in Chapter 2, Section 2.3.

Records as required by 10 CFR 20.2102 and 20.2103 are maintained in such a manner as to demonstrate compliance with license requirements and regulations.

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 radiation protection activities are described in written procedures and their associated records will be maintained for at least three years:

- Safety review committee meetings
- Surveys of equipment for release to unrestricted areas
- Instrument calibrations
- Safety audits and other reviews of the radiation program

- Personnel training and retraining
- Radiation work permits
- Surface contamination and dose rate surveys
- Radiological safety analyses
- Facility changes involving licensed activities not requiring NRC approval

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.

4.2 ALARA (AS LOW AS REASONABLY ACHIEVABLE) PROGRAM

VNC's radiological protection program applicable to SNM 960 activities uses, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA) pursuant to 10 CFR 20.1101. Methods used to maintain exposures ALARA including the use of controls such as equipment and process design are documented in written procedures.

A VNC ALARA committee is established to review and recommend actions to minimize radiation exposures, consider alternative engineered controls, establish program goals and other dose reduction techniques. Committee members include personnel from radiation safety, operations, maintenance, and engineering as required to conduct periodic ALARA reviews in accordance with written procedures.

The ALARA committee reviews the ALARA program including an evaluation of radiation levels in the facility, contamination levels, worker exposures and effluent releases as appropriate. The review determines if exposures, releases and contamination levels are in accordance with the ALARA concept. Recommendations of the ALARA committee are documented and tracked to completion.

The ALARA committee meets at least quarterly (intervals not to exceed 110 days) and reports to the manager, VNC.

Dose rates in areas accessible to personnel are also maintained ALARA and controlled by written procedures or special approval documented in a radiation work permit.

4.3 RADIATION SAFETY PROCEDURES AND RADIATION WORK PERMITS

Radiation protection procedures are issued to ensure safe operation of routine work and compliance with state and federal regulations, permits and licenses. A process is established for procedure generation, modification, approval, distribution and training. The radiation safety function leader approves procedures related to radiation protection.

Non-routine activities, (e.g. those not covered by documented procedures), performed by VNC and non-VNC employees, are administered by a Radiation Work Permit (RWP) system. The RWP system is also described in documented procedures.

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 for non-routine operations not addressed by an operating procedure when special radiation control requirements are necessary. RWPs are approved by a member of the radiation safety function. 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.

4.4 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 sampler flow indicators are calibrated annually.

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

4.5 CONTAMINATION CONTROL

4.5.1 SURVEYS

Routine contamination survey monitoring is performed in potentially contaminated areas in accordance with 10 CFR 20.1501.

Removable contamination measurements are performed commensurate with the nature of the work being conducted, the quantities of material being used, and operational experience. Survey frequencies are determined by the radiation safety function and documented in procedures. Survey results are compared to action guide values as specified in plant procedures and appropriate responses are taken.

The removable surface contamination action levels for routine surveys in non-controlled areas are the removable contamination values in Table 1 as referenced in Section 1.3.2.

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

The contamination levels for release for unrestricted use of equipment and material are described in Chapter 1.

4.5.2 ACCESS CONTROL

An access control program has been established to ensure that routine access points to contaminated areas are properly posted and operative. Contaminated area boundaries are established to prevent the spread of contamination and are identified with the appropriate signs, step off pads, change facilities, protective clothing facilities, and personnel monitoring instruments in sufficient quantities and locations.

Alternate access points to contaminated areas may be established for specific activities that are not accommodated by use of routine access points. Such access is governed by approved procedures or RWP's which establish controls to prevent the spread of contamination.

4.5.3 PROTECTIVE CLOTHING

Protective clothing is provided as required to persons enter a contaminated area 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.

4.5.4 POSTING AND LABELING

Based on radiation surveys, air sampling results, process knowledge and radioactive material storage and usage conditions, areas are posted with the appropriate radiation and radioactivity caution signs in accordance with 10 CFR 20.1901, 1902 and 1903.

Containers of licensed material are labeled with the appropriate radioactive material caution information in accordance with 10 CFR 20.1904 and 1905. Prior to removal or disposal of empty uncontaminated containers to unrestricted areas, radioactive material labels are removed or defaced.

4.6 CONTROL OF ACCESS TO HIGH RADIATION AREAS

In accordance with 10 CFR 20.1601, each entrance or access point to a high radiation area has one or more of the following features:

1. A control device that, upon entry into the area, causes the level of radiation to be reduced below that level at which an individual might receive a deep-dose equivalent of 0.1 rem in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates;
2. A control device that energizes a conspicuous visible or audible alarm signal so that the individual entering the high radiation area and the supervisor of the activity are made aware of the entry;
3. Entryways that are locked, except during periods when access to the areas is required, with positive control over each individual entry; or

4. Continuous direct surveillance by an authorized individual capable of preventing unauthorized access.

4.7 EXTERNAL EXPOSURE

Individuals are monitored for exposures to radiation in accordance with 10 CFR 20.1502 to demonstrate compliance with the occupational dose limits specified in 10 CFR 20.1201. Deep-dose equivalent and shallow-dose equivalent from external sources of radiation are determined by individually assigned dosimeters. Dosimeters are issued to and used by individuals likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the applicable occupational exposure limits in 10 CFR 20.1201, and declared pregnant women likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem. 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. Dosimeters are also issued to and used by individuals entering a high radiation area or a very high radiation area.

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 4.10.

When the results of individual monitoring are unavailable or are invalidated by unusual exposure conditions, external exposures are estimated by the radiation safety function on the basis of data obtained by investigation.

4.8 INTERNAL EXPOSURE

Individuals are monitored for the occupational intake of radioactive material at levels sufficient to demonstrate compliance with the occupational dose limits of 10 CFR 20.1201. Monitoring for the intake of radioactive material is performed for individuals likely to receive, in one year, an intake in excess of 10 percent of the applicable ALI, and for declared pregnant women likely to receive, during the entire pregnancy, a committed effective dose equivalent in excess of 0.1 rem. For the purposes of assessing dose when required pursuant to10 CFR 20.1204, suitable and timely measurements are made of:

1. Concentrations of radioactive materials in air in work areas; or
2. Quantities of radionuclides in the body; or
3. Quantities of radionuclides excreted from the body; or
4. Combinations of these measurements.

When concentrations of radioactive material in air are used to assess dose it is assumed an individual inhales radioactive material at the airborne concentration in which the individual is present unless respiratory protective equipment is used. If respiratory protectve equipment is used the airborne concentration inhaled is equal to the airborne concentration in the area divided by the approved protection factor as described in Section 4.12.2

The committed effective dose equivalent is calculated by assuming that the inhalation of one ALI (stochastic value) results in a committed effective dose equivalent of 5 rem.

Action levels are established in plant procedures to prevent an individual from exceeding the occupational dose limits specified in 10 CFR 20.1201. Radiation exposure action levels are specified in Section 4.10.

4.8.1 IN VIVO PROGRAM

An in vivo program is available to evaluate the intake of alpha emitting radionuclides. Analyses are 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.

4.8.2 WHOLE BODY COUNTING

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

4.9 SUMMING INTERNAL AND EXTERNAL EXPOSURE

In accordance with 10 CFR 20.1202, internal and external doses as described in the preceding sections of this application are summed for the purposes of limiting occupational doses and recording individual monitoring results. Total effective dose equivalent is determined by summing the committed effective dose equivalent and the deep dose equivalent. Total organ dose equivalent is determined by summing the committed dose equivalent to the maximally exposed organ or tissue (other than the lens of the eye) and the deep dose equivalent.

4.10 COMPLIANCE WITH OCCUPATIONAL DOSE LIMITS

To ensure compliance with occupational dose limits, work activity restrictions are imposed when an individual's exposure exceeds 80% of the applicable 10 CFR 20.1201 occupational dose annual limit as follows:

	10CFR20 annual limit	Action Level
Total effective dose equivalent	5 rem	4 rem
The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye	50 rem	40 rem
Lens dose equivalent	15 rem	12 rem
Shallow-dose equivalent to the skin of the whole body or to the skin of any extremity.	50 rem	40 rem

4.11 DOSE REPORTS

In accordance with 10 CFR 19.13, an annual dose report is provided to each individual with monitoring results exceeding 100 mrem TEDE or 100 mrem to any individual organ

or tissue, during the calendar year in a format consistent with the NRC Form 5 occupational exposure report. Reports are also provided at the request of a monitored individual or by a worker formerly engaged in licensed activity.

4.12 RESPIRATORY PROTECTION PROGRAM

The respiratory protection program is conducted in accordance with the applicable portions of 10 CFR 20.1703 including written procedures for air sampling sufficient to identify the potential hazard, proper equipment selection, maintenance and testing, dose estimation and surveys or bioassays, as necessary, to evaluate actual intakes. Respiratory protection equipment specifically approved by the National Institute for Occupational Safety and Health (NIOSH) or equivalent is utilized.

To the extent practical, process or other engineering controls (e.g. containment or ventilation) are used to control the concentration of radioactive material in air.

4.12.1 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 protection equipment 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.

4.12.2 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.

4.12.3 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.

4.13 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.

4.13.1 ANALYTICAL LABORATORY COUNTING EQUIPMENT AND CAPABILITIES

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

Sample Type	Instrument	Minimum Detection Limit*
Air and Exhaust	Alpha Proportional	4×10^{-15} μCi/cc
Stack Samples	Beta Proportional	7×10^{-15} μCi/cc
Smears	Alpha Proportional	7×10^{-8} μCi/cc
	Beta Proportional	3×10^{-6} μCi/cc
Water (Retention Basin)	Alpha Proportional	1.5×10^{-8} μCi/cc
	Beta Proportional	3×10^{-8} μCi/cc

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

4.13.2 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

4.13.3 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 a derived air concentration (DAC) 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 100 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.

4.13.4 CALIBRATION

Portable instrumentation is maintained per manufactuer recommendations and 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.

4.14 PERSONNEL WORK RULES

Food storage and consumption (including candy or beverages), the use of cigarettes, or the application of cosmetics are prohibited in contaminated areas. 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 removable contamination and conditions are unchanging. Food containers may not be used for storing or handling radioactive material.

4.15 RADIATION PROTECTION EVENT REPORTS

VNC will notify the NRC of events involving radiation or radioactive material in accordance with 10 CFR 20.2201, 2202, 2203 and 10 CFR 70.50. These reports will be

made either in writing or as required, to the NRC Operations Center and will include as appropriate:

- Date, time, and location of the event
- Description of the event, including the radiological or chemical hazards involved, isotopes, quantities, and form of material
- Actual or potential health and safety consequences to workers, public and environment
- Sequence of occurrences leading to the event
- Response actions or corrective steps taken or planned to prevent against recurrence
- Notifications made or planned to local, State or other Federal agencies
- Plans for any press releases related to the event

4.16 FACILITIES AND EQUIPMENT

A description of the facilities, processes, equipment, types and quantities of material, authorized activities and special authorizations are described in Chapter 1, Sections 1.1, 1.2, and 1.3.

CHAPTER 5.0
NUCLEAR CRITICALITY SAFETY

5.1 NUCLEAR CRITICALITY SAFETY PROGRAM MANAGEMENT

5.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

GEH maintains the in-process inventory of SNM to a quantity less than that sufficient to form a critical mass (typically 350 grams U235).

The Hillside Storage Facility has been demonstrated safe for quantities of SNM greater than that required for a critical mass and the facility is maintained in accordance with the criticality safety program described in this chapter.

The Double Contingency Principle as identified in American National Standard ANSI/ANS-8.1 (1998) is the fundamental technical basis for design and operation of the Hillside Storage Facility. As such, "the design 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 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 facilities or equipment that store, transfer or otherwise handle fissile materials, and
- Any change in existing 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

Change Authorizations (CAs) are processed in accordance with written plant configuration management practices. CAs which establish or involve a change in existing criticality safety parameters require a member of the criticality safety function

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 and conditions of the license.

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 of 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, Section 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.

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

Documentation of the review, approval and operator orientation process is maintained within the configuration management system.

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, ^{235}U 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, where the enrichment of the material contained is unknown (e.g. cleanout material), or as exempted by 10 CFR 20.1905.

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 CSF members are described in Chapter 2, Section 2.2.1.4.

Criticality safety function personnel are specifically authorized to perform assigned responsibilities in Chapter 2, Section 2.2.1.4. 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 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.

Documented CSAs establish the nuclear criticality safety bases for a particular system under normal and credible abnormal conditions.

5.3.2 CSF CONFIGURATION MANAGEMENT

5.3.2.1 Auditing, Assessing and Upgrading the CSF Program

Criticality safety audits are conducted and documented in accordance with a written procedure and by personnel approved by the criticality safety function. Findings, recommendations, and observations are reviewed with the Regulatory Compliance Program Manager to determine if other safety impacts exist. NCS audit findings are transmitted to the Area Manager 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.

A nuclear criticality safety program review is conducted on a scheduled basis by nuclear criticality safety professionals independent of operations organization. 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.

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

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

A criticality accident alarm system (CAAS) is installed and maintained to energize an audible alarm in the event of a criticality. The 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 the appropriate requirements such as 10CFR70.24(a)(1) and with 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 site Radiological 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.

5.3.2.4 Corrective Action Program

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

5.3.2.5 CSF Records Retention

Records of CSAs 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 in accordance with established configuration management control procedures.

5.4 METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 CONTROL PRACTICES

CSAs 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 CSA criteria.

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 CSAs. 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. Calibrations and routine maintenance are 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 written procedures.

5.4.2 MEANS OF CONTROL

The relative effectiveness and reliability of controls are considered during the CSA 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 is a passive engineered control.

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 is an active engineered control. 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:

- <u>Augmented Administrative Control</u> – 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.
- <u>Simple Administrative Control</u> – 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. Limits presented in Table 5.1 are derived from documented criticality safety analyses (ref. 5-12).

NOTE: The safe mass values specified for UO2 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* $\leq$ 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)*
^{235}U Sphere	575 g ^{235}U
^{235}U Hemisphere	650 g ^{235}U

- Includes Bias Uncertainty.

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

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 no more than 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 ^{235}U 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 CSA 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. 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 CSA 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 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.

5.4.4.5 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 Sn Discrete Ordinates codes) may be used and documented in the CSA.

5.4.4.6 Material Composition (or Heterogeneity)

The CSA 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 CSA when appropriate. Evaluation of systems where the particle size varies takes into consideration effects of heterogeneity appropriate for the process being analyzed.

5.4.4.7 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 the CSA, 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.

5.4.4.8 Enrichment

Enrichment control may be utilized to limit the percent ^{235}U 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 enrichment is utilized in the CSA.

5.4.4.9 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 CSA 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 operations prior to their 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:

$$USL = 1 + \text{bias} - \text{bias uncertainty} - MMS$$

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:

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

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

If parameters needed for anticipated applications are 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 in 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 ensure 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 approved configuration management procedures 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 require 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 H2O). 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, and 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 CSA.
- **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 CSAs are performed. 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.

5-12. GEH Criticality Safety Analysis (CSA), Safe Mass Limits for Uranium Systems, September 2007 (as amended)

CHAPTER 6.0
CHEMICAL SAFETY

There are no hazardous chemicals produced from licensed materials at VNC. Based on the fact that the SNM is used in limited quantities or remains in storage, there is no reasonable risk of a chemical reaction that could endanger life or health. Likewise there are no postulated situations where Emergency Response Planning Guidelines (ERPG) or Acute Exposure Guideline Levels (AEGL) would be exceeded.

CHAPTER 7.0

FIRE SAFETY

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

Based on the fact that the SNM is used in limited quantities or remains in storage, there is reasonable assurance that the fire safety program provides adequate protection against fires and explosions that could affect the safety of licensed material and reasonably protect health and safety of the public and environment.

CHAPTER 8.0

RADIOLOGICAL CONTINGENCY & EMERGENCY PLAN

A Radiological Contingency & Emergency Plan (RC&EP) is not required for SNM 960 licensed activities because a release of radioactive materials would not cause a member of the public to exceed 1 rem effective dose equivalent or an intake of 2 milligrams of soluble uranium as provided in 10 CFR 70.22(i)(1)(i) and (i)(2). An evaluation demonstrating that an RC&EP is not required was provided to NRC on April 21, 1989 and NRC concurred with the evaluation on December 20, 1990. A recent review of this demonstration determined that the 1989 analysis remains valid and continues to be supported by the conditions in 10 CFR 70.22(i)(1)(i) and (2)(ii) and (vi). A copy of this review was submitted to NRC on October 18, 2013.

In addition, a recent evaluation was performed to address potential seismic activity and subsequent accidents at the Hillside Storage Facility. A copy of this evaluation was also submitted to NRC on October 18, 2013.

CHAPTER 9.0

ENVIRONMENTAL PROTECTION

9.1 ENVIRONMENTAL PROTECTION PROGRAM & REPORTS

GEH 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 ALARA.

GEH 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 accordance with established procedures. 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 ventilation and exhaust systems for SNM-960 facilities are described in Chapter 1, Sections 1.1.3.1.10 and 1.1.3.2.3.

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.

Radioactivity 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, are provided and maintained in proper operating condition as required to support operations.

9.5 PROGRAM IMPLEMENTATION

The environmental monitoring program includes the types of samples and monitored parameters listed in Figure 9.1. Analytical sensitivities (minimum detection levels) are also identified in Figure 9.1. Action levels are 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 or other assessments may be used. These program elements, analytical sensitivities, and/or locations may be changed in accordance with written procedures.

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 for 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.

9.9 APPLICABLE MANAGEMENT MEASURES

9.9.1 FACILITIES AND PROCESSES

Facilities, processes and equipment are discussed in Chapter 1, Section 1.1. Authorized materials are described in Chapter 1, Section 1.2.

9.9.2 ORGANIZATION

The organization is described in Chapter 2.

9.9.3 TRAIINING AND QUALIFICATION

Workers are trained using on-the-job training and experience supported by written procedures.

9.9.4 PROCEDURES

Work is performed using written and approved procedures..

9.9.5 RECORDS MANAGEMENT

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

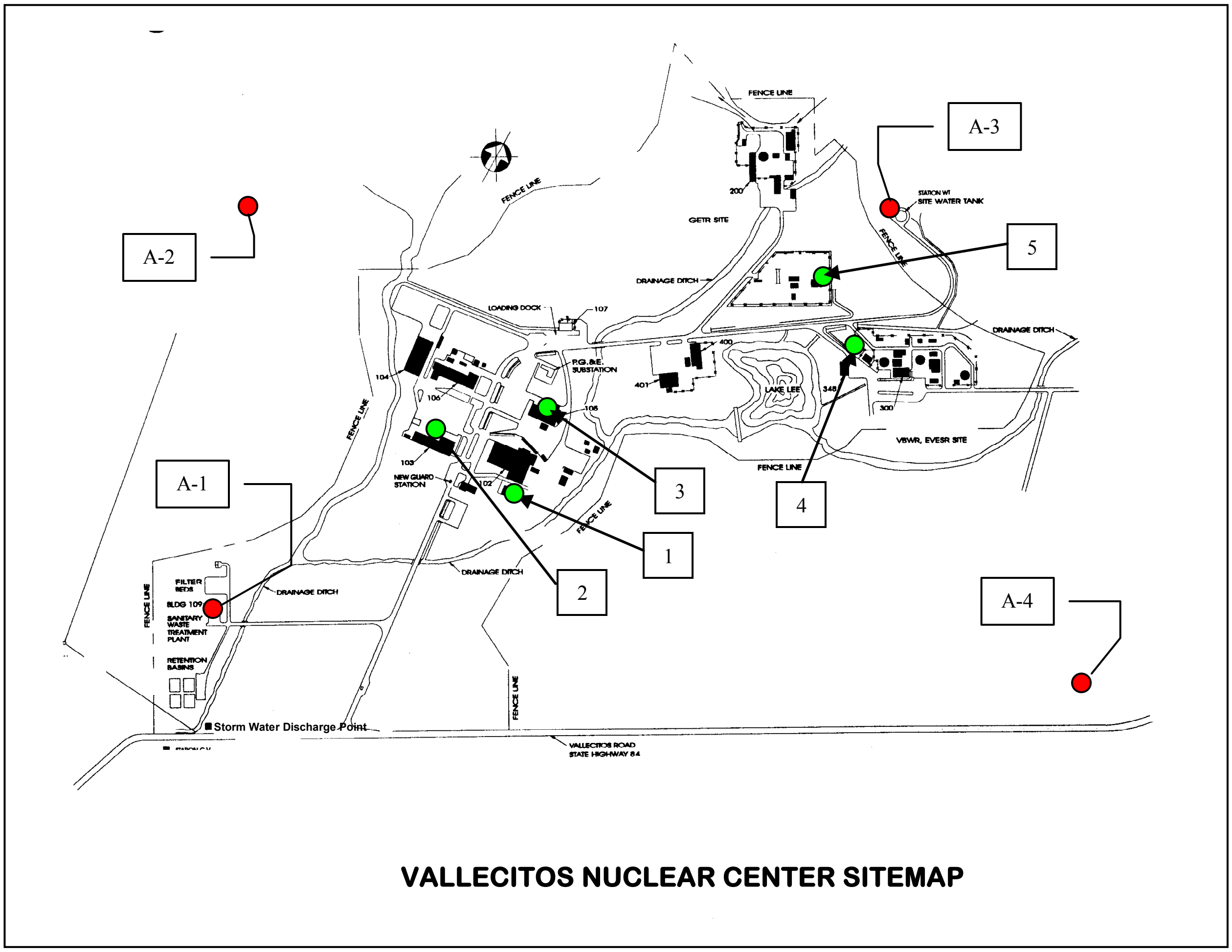
FIGURE 9.1
VNC'S ENVIRONMENTAL MONITORING PARAMETERS

Exposure Pathway and/or Sample	Type of Analysis	Sampling and Collection Frequency	Typical Minimum Detectable Concentration
1. Waterborne			
Sanitary/Industrial	a. Gross alpha, gross beta [1] b. Gross alpha, gross beta	a. When released b. Monthly/composite	1.5E-08 µCi/ml – gross □ 3.0E-08 µCi/ml – gross □
Groundwater - Onsite	Gross alpha, gross beta {1}	Quarterly	
Sediment (S-4)	Gross alpha, gross beta {1}	Annually	1.0E-07 µCi/gram – gross α 2.4E-06 µCi/gram – gross β
2. Ingestion			
Vegetation	Gross alpha, gross beta {1}	Annually	1.5E-07 µCi/gram – gross α 2.4 E-06 µCi/gram – gross β
3. Inhalation			
Ambient Air	Gross alpha, gross beta {1}	Continuous operation of the sampler with sample collection weekly	4.0E-15 µCi/ml {2} – α particulate 7.0E-15 µCi/ml {2} - β particulate
Stack	Gross alpha, gross beta {1}	Continuous operation of the sampler with sample collection weekly	
4. Direct Radiation	Gamma dose	Annual	1 millirem

Footnotes for Table 9-1

1. If gross α or gross β levels are greater than release limits; an isotopic analysis will be performed to identify Gamma isotopes.
2. Based on a 7-day sample period at ≥ 1 cubic foot per minute

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)

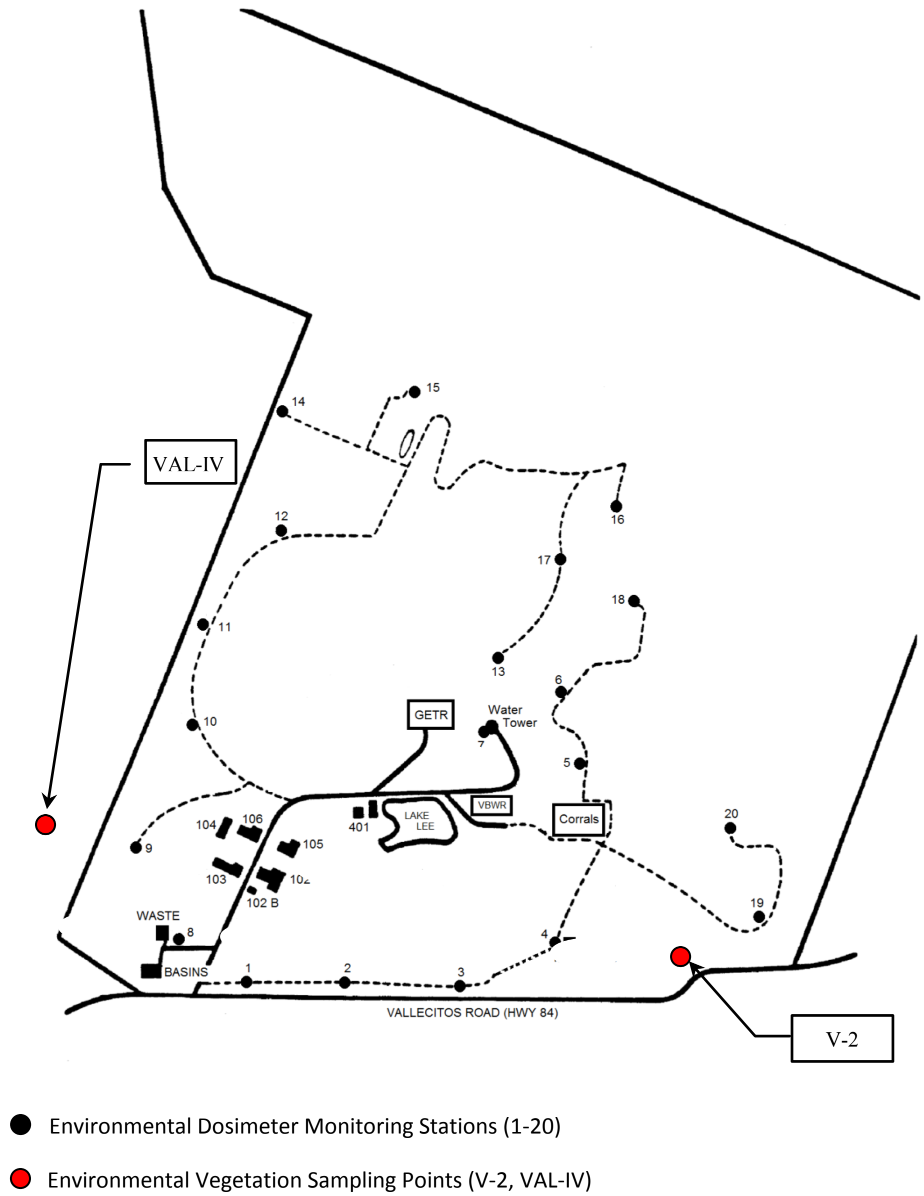
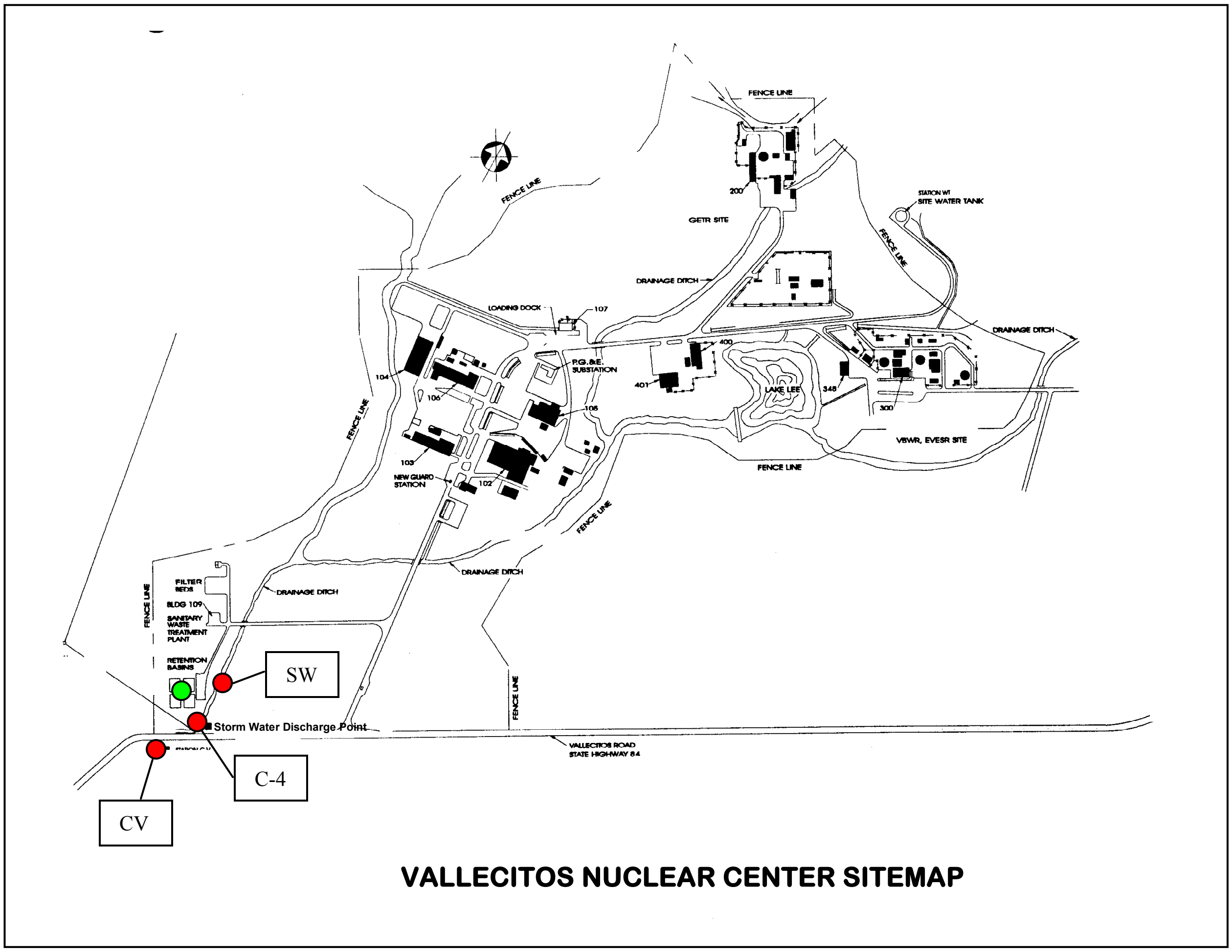


Figure 9.4
Location of Surface Water Monitoring Points (Typical)



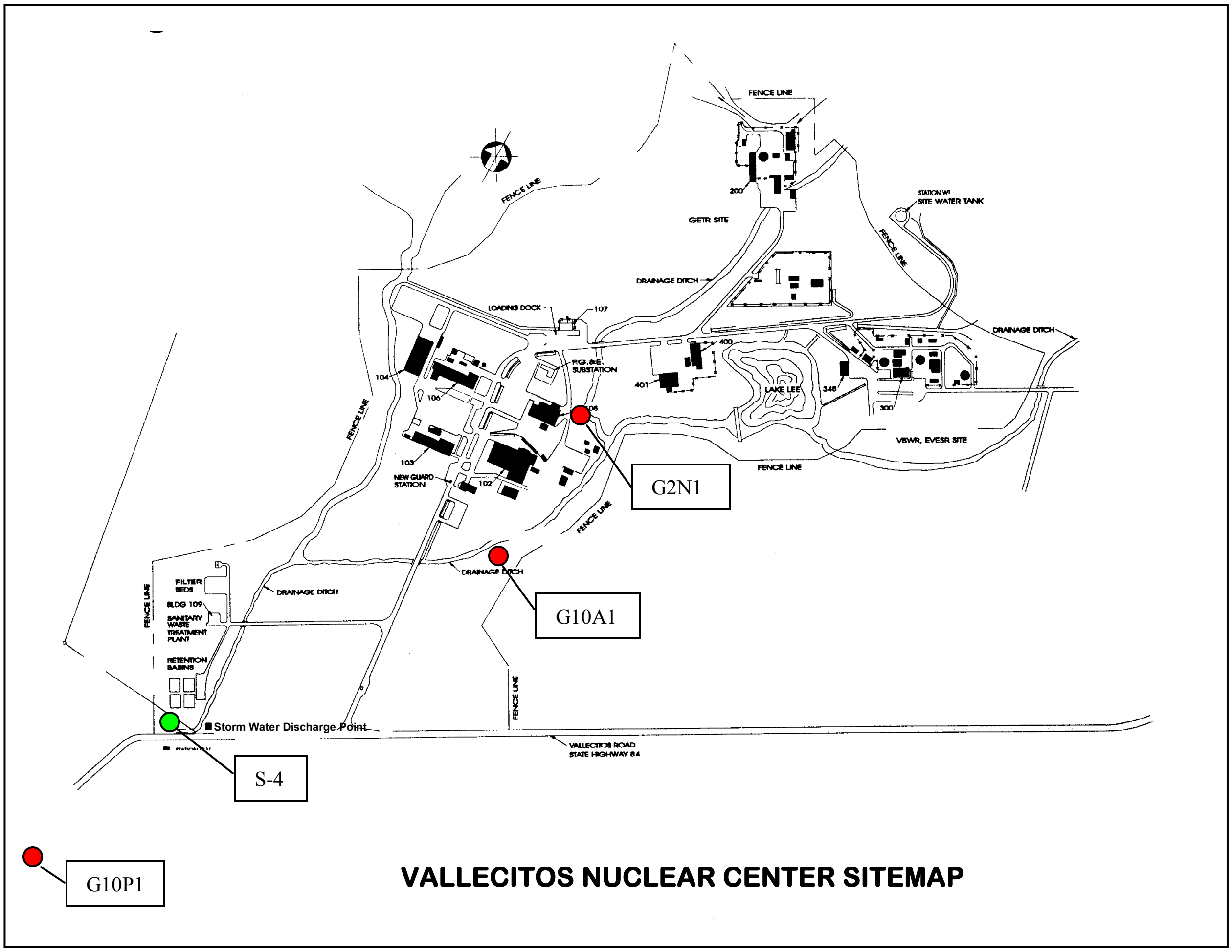
Surface Water Monitoring Points
C-4
CV
SW

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

10.1 DECOMMISSIONING FUNDING PLAN/COST ESTIMATE

A Decommissioning Funding Plan (DFP) and cost estimate has been prepared and submitted pursuant to 10 CFR 70.25 to demonstrate financial capability to support decommissioning and closure activities. In accordance with 70.25(f)(2), financial assurance is provided by GE as a parent company of GE-Hitachi Nuclear Energy Americas, LLC in the form of a parent guarantee.

The DFP establishes decommissioning criteria, describes key assumptions, outlines major technical approaches, and provides both a detailed site specific cost estimate and a certification of financial assurance for the decommissioning of facilities and equipment containing licensed radioactive material within the scope of SNM-960. The current DFP is dated November 30, 2012 and is scheduled to be updated by January 31, 2014 to be consistant with this application.

The GEH Vallecitos Nuclear Center has been in operations since 1957. The DFP was created in the 1970's and has been periodically revised in accordance with regulations and NRC guidance since that time. This Chapter and the DFP cost estimate provide information that is consistent with many aspects of NUREG/CR-1757 and is also consistent with the previously accepted requirements for providing this information. Also, the current process of determining decommissioning costs has been reviewed to ensure that the applicable evaluation criteria for unrestricted release listed in guidance from NUREG/CR-1757 have been incorporated. In addition, the DFP addresses the key elements of NUREG/CR-1757 including (1) Decommissioning Process (2) Characterization, Survey, and Free Release Criteria; and (3) Financial Assurance, Recordkeeping, and Timeliness. In 2003 as a publicly traded company, GE adopted the use of Financial Accounting Standard Board (FASB) guide 143 to account for decommissioning liability.

10.1.1 The cost estimate is reviewed and adjusted annually. The cost estimate is updated to reflect completed dismantlement activities, current contamination levels or events that could result in subsurface contamination requiring remediation, inflation, changes in waste volume and/or transportation and disposal costs, prices of goods and services, changes in decommissioning techniques, and any other relevant changes in facility

conditions. Checklists are used to validate the cost estimate taking into consideration specific factors to determine if changes are warranted. Examples include waste volumes, remediation activities that may have occurred that impacted waste volumes or labor, labor rates, disposal rates, transportation costs, inflation rates, and shared services (insurance, fees, and utilities). Every three years a more detailed review is performed that includes review and validation of assumptions. These reviews meet the applicable requirements of 10 CFR 70.25(e).

10.1.2 Cost estimates used in this DFP are based on documented and reasonable assumptions including actual costs that have been incurred performing liability reduction activities. Cost estimates are sufficient to allow an independent third party to assume responsibility for decommissioning the facility including labor, equipment, sampling, laboratory and miscellaneous expenses such as overhead and contractor profit.

10.1.3 The cost estimate does not take credit for: 1) any salvage value that might be realized from the sale of potential assets during or after decommissioning, or 2) reduced taxes that might result from payment of decommissioning costs or site control and maintenance costs.

10.1.4 The cost is based on a license termination without the need for continued surveillance. The cost is accreted on an annual basis to account for inflation.

10.1.5 The cost is based on the conditions expected to be present at the end of plant life. However, adjustments are made on an annual basis as described above.

10.1.6 All the major decommissioning tasks or activities outlined in NUREG -1757 are in the cost estimate and include the planning and preparation, decontamination and/or dismantling of radioactive facility components, final radiation survey, packing materials, shipping, waste disposal, equipment/supply, laboratory and miscellaneous costs. The key assumptions are discussed in Chapter 4 and of the DFP.

10.1.7 The plan is for unrestricted release of the facilities covered by the DFP. There are no known areas of confirmed soil or groundwater contamination associated with licensed activities covered by the DFP. Restoration of contaminated areas on facility grounds and site stabilization are assumed to not be required. Nonetheless, the DFP includes a conservative estimate for the removal and disposal of soil that provides shielding for the hillside material storage area and does not anticipate removal of large amounts of

soil. Therefore, no cost for restoration, stabilization or long-term surveillance is included.

10.2 COST ESTIMATE ASSUMPTIONS

The original estimates were made for all GE radioactive material licensed facilities at the Vallecitos site which included 4 reactors, the SNM-960 facilities and the State of California byproduct material license CA-0017-1.

All irradiated spent fuel in storage will have been removed from the site prior to initiation of decommissioning and closure activities. The US Department of Energy has contractual responsibility to dispose of this fuel and the cost of such disposal is separately covered under Standard Contracts entered into under the Nuclear Waste Policy Act. If, at the time of desired decommissioning continued storage is required, GEH, as necessary, will make appropriate arrangements to remove the fuel to an authorized recipient.

The estimated total cost provided in the decommissioning funding plan for the SNM-960 facility includes a 25-percent contingency to allow for unforeseen problems that might arise during the activity. The facility will be decommissioned such that the facilities can be released for unrestricted use.

The manpower requirements, timeframes and estimating equations discussed in both NUREGs were used to develop the detailed cost estimate. These estimates were based on interviews with site personnel, scaling factors from building volumes and foot prints, and comparisons to previous other decommissioning projects.

Since 2003 GEH has engaged in liability reduction activities across its facilities. These efforts have further validated that the prepared estimates are conservative and reasonable. In these activities the projected actual costs have consistently been in line with the estimated cost. These include the removal of over 10,000 cubic feet of debris from the former vaporization area of the Wilmington facility, the removal of more than 1,000,000 cubic feet of soil like material from the Wilmington facility, the removal of over 13,000 cubic feet of material from the VBWR in Vallecitos, CA and the removal of over 26,000 cubic feet of material from the process canyons in Morris, IL. All of these projects were accomplished by contract labor and the costs were comparable to the expected cost for labor, packaging, shipment and burial of the materials. The planning and professional cost associated with the future

decommissioning of the sites was unaffected and continued to accrete due to inflation over the period of material removal.

10.3 FINANCIAL SURETY

Appropriate financial assurance instruments are provided to demonstrate that sufficient funds will be available when needed for required decommissioning activities. GEH uses a General Electric Company parent- guarantee of funds for decommissioning costs. The most recent financial instruments and supporting documentation are shown in Chapter 8 of the DFP. This information is updated and submitted to NRC annually.