Attachment 2

License Renewal Application
For
Vallecitos Nuclear Center

NRC License SNM-960, Docket 70-754

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CHAPTER 1.0
GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

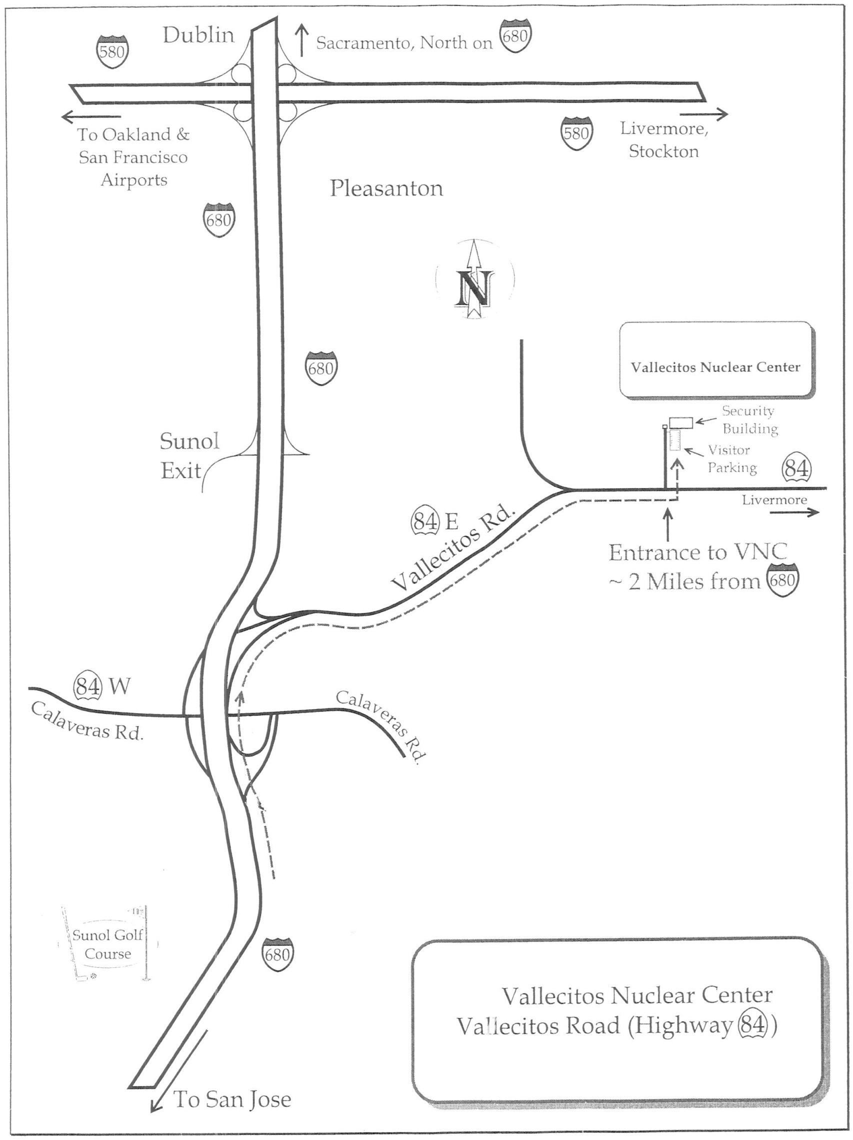
The primary purpose of the GE Hitachi (GEH) Vallecitos Nuclear Center (VNC) facility in Alameda County, California is the measurement, and engineering evaluation and support of light water reactor fuels, fuel materials, components and inspection tools. The SNM-960 authorized activities are co-located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

1.1.1 SITE DESCRIPTION AND LOCATION

VNC is located near the center of the Pleasanton quadrangle of Alameda County, California. The Laboratory is east of San Francisco Bay approximately 35 air miles east-southeast of San Francisco and 20 air miles north of San Jose. The site is indicated on the Area map, Figure 1.1. The nearest sizeable towns are Pleasanton with a population of approximately 67,000 located 4.1 miles to the north-northwest and Livermore with a population of approxiamtely 81,000 located 6.2 miles to the northeast. A United States Veterans Administration Hospital with a population of approximately 500 is located about 4 miles to the east.

The site is on the north side of Vallecitos Road (State Route 84), which is a two-lane paved highway. The laboratory site consists of approximately 1,600 acres, about one-third of which is relatively flat. Approximately 1,400 acres of the site are leased for raising feed crops and cattle grazing. There is light industrial activity within a 10-mile radius of the plant. San Jose (20 miles south), Oakland (30 miles northwest) and San Francisco (35 miles northwest) are major industrial centers. To the southwest there are no industries and very sparse population for 20 miles and beyond.

FIGURE 1.1
Area Map

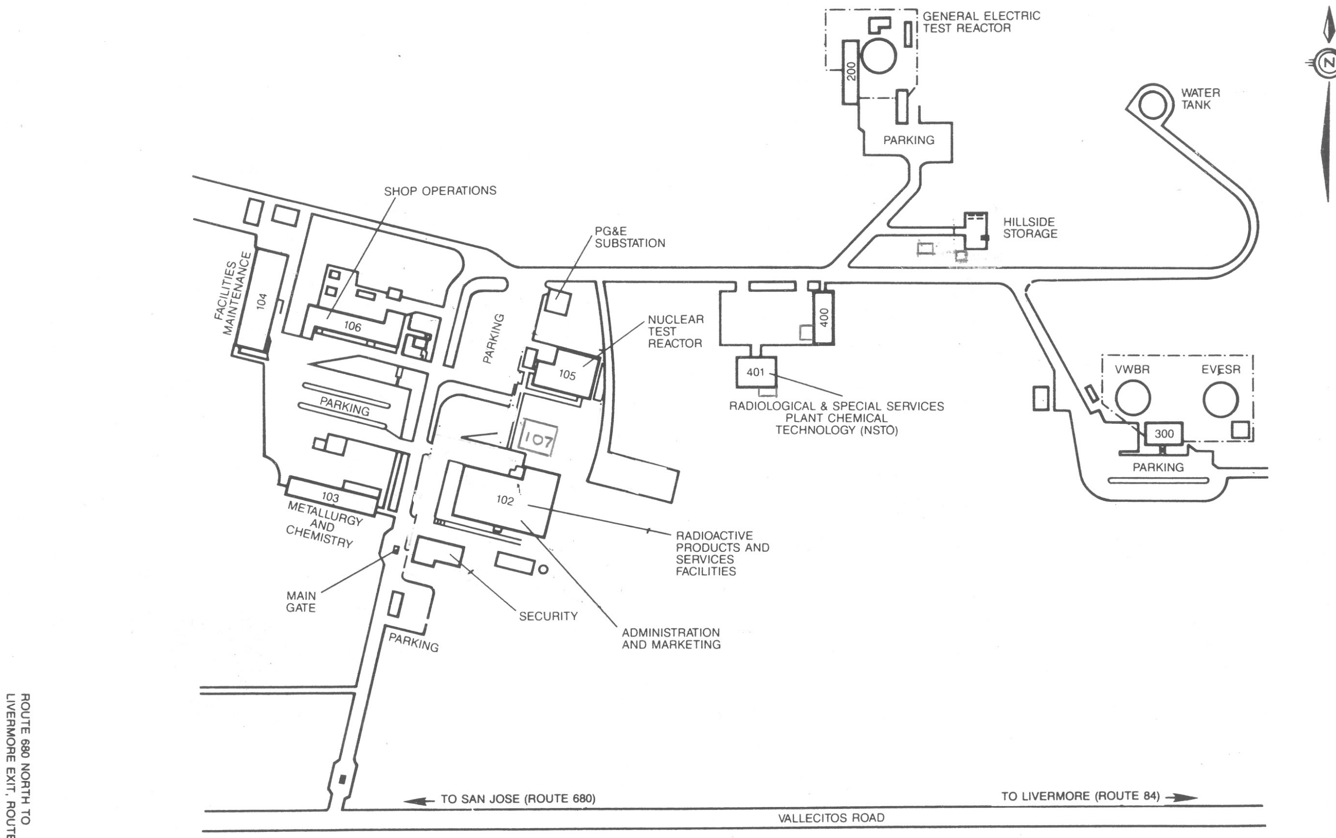


1.1.2 FACILITY DESCRIPTION

The location and arrangement of buildings at the VNC site is shown in Figure 1.2. Located on the property are the following major facilities associated with the activities authorized by SNM-960: (1) 400 Area consisting of Building 400 devoted to offices laboratories and Building 401 devoted to offices and laboratories, both buildings subleased by a non-GEH company that uses minute quantities of special nuclear material under the possession and control of VNC; (2) Building 102 consisting of the Radioactive Materials Laboratory and Hot Cells, Radiochemistry Laboratory and the Building 102 storage pool and dry pit storage; (3) Building 103 which includes the Metallurgy, Chemistry and Ceramics Laboratory, and offices; (4) Building 105 which houses the NTR (R-33) and laboratories subleased by a non-GEH company that uses minute quantities of special nuclear material under the possession and control of VNC; (5) Building 106 which includes the engineering shop and maintenance support; (6) The Solid Radioactive Waste Storage Facility that provides storage in a shielded facility; (7) The Waste Evaporator Plant used to concentrate and solidify liquid radioactive wastes; and (8) Building 104 used for warehousing. Building 300 is closed. The reactor and auxiliary facilities for GETR (TR-1), VBWR (DPR-1) and EVESR (DR-10) are in safe storage, and (9) Building 107 used to store site hazardous chemicals, including a loading dock and pad for temporary storage/loading of trailers.

FIGURE 1.2

Vallecitos Nuclear Center



SITE PLAN

1.1.3 FACILITY RESISTANCE TO ENVIRONMENTAL EVENTS

The seasonal rainfall patterns and the presence of geological faults in the East Bay area constitute the primary environmental threats.

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

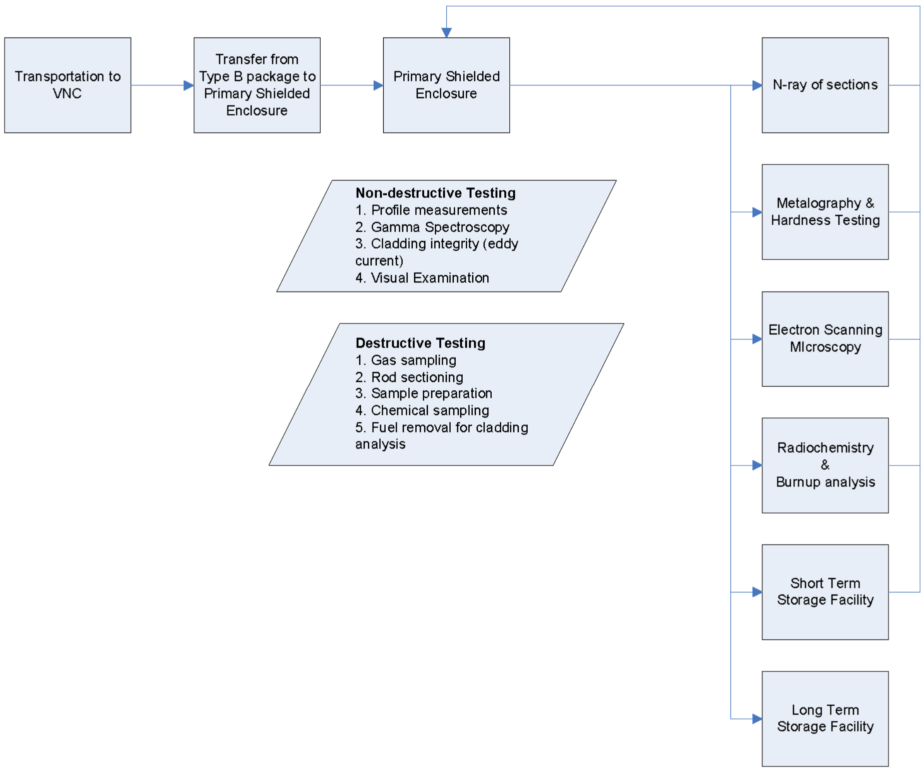
Earthquakes are an occasional occurrence in all parts of the Bay Area. VNC contains one fault through the property, which is a branch off of the Hayward fault. Typically, measurable earthquakes occur nearly daily in the area. Rarely are they of sufficient magnitude to be noticed without precision instruments.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

1.1.4 PROCESS DESCRIPTION

The operations authorized by this license consist of receiving small quantities of irradiated special nuclear material, and performing a variety of tests (mechanical, nuclear and chemical) on the material for purposes of improving the design of nuclear fuel and the functioning of utilization facilities. These tests may also be performed as basic research. The majority of this activity takes place in the Shielded Enclosures (also referred to as Hot Cells) with additional analysis of sub-sets of the material being used in various laboratories, more lightly shielded hot cells and radiography facilities. Handling of the material is conducted with remote handling techniques such as master-slave manipulators, through used of heavily shielded containers, or through reduced sample size. Most of the analysis centers around Boiling Water Reactor fuel and cladding, with a minor amount of focused on other fuel types. This analysis may focus on a leaking rod, a new fuel design, increased burn-up, or other inquiries. A single campaign typically contains several rods, some of which are of primary concern and others analyzed as a reference or control standard.

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

FIGURE 1.3 - Typical Spent Nuclear Fuel Examinations



1.2 INSTITUTIONAL INFORMATION

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

1.2.1 IDENTITY AND ADDRESS

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

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

1.2.2 TYPE, QUANTITY, AND FORM OF LICENSED MATERIAL

The following types, maximum quantities, and forms of special nuclear materials are authorized:

- ____ kilograms of U-235 contained in uranium enriched to a maximum enrichment of less than or equal to 10%, for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant byproduct and reactor-produced transuranics,
- ___kilograms of U-235 enriched to more than 10 percent for authorized activities. The material may be in the form of irradiated special nuclear material with its attendant byproduct and reactor-produced transuranics,
- Less than 1 effective kilogram of U-235 in any unirradiated form,
- _______ grams of Plutonium in a contained or sealed form in addition to the irradiated quantities, and
- ______ grams U-233 in any form.

1.2.3 ACTIVITY

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

Authorized activities at VNC include:

1.2.3.1 LABORATORY OPERATIONS

- Analysis of the chemical and isotopic composition, concentration and behavior of special nuclear materials by wet chemistry and physical measurement techniques.
- Physical analyses and testing of physical and metallurgical properties of special nuclear materials.
- Measurements of radiation and its effects on instruments and on the structure and composition of materials.
- Post-irradiation examination, testing, and analysis of fuel elements and materials in shielded enclosures by remote manipulative techniques; research and development and/or pilot plant activities. Examination and temporary storage of

irradiated enriched uranium fuel elements from the GEH Nuclear Test Reactor (NTR).

- Assembly, modification, cleaning and repair of unirradiated encapsulated experimental assemblies.
- Research and development activity associated with but not limited to those activities listed above.

1.2.3.2 GENERAL SERVICES OPERATIONS

- Design, fabrication, and testing of equipment containing special nuclear materials and maintenance of such equipment.
- Storage of special nuclear materials other than wastes in shielded containers and or at locations as applicable and in designated general-purpose storage areas.
- Inspection of packaging and preparation for shipment and/or transfer of special nuclear materials.
- Decontamination of equipment and facilities.
- Inspection, use and storage of irradiated special nuclear material is performed in specially designed shielded enclosures equipped with remote handling capabilities, and as required, shielded viewing windows and shielded air locks.

1.2.3.3 WASTE TREATMENT

- Concentration of the radioactive constituents of liquid wastes by evaporation, chemical treatment, sedimentation, filtration, and ion exchange; solidification and packaging of concentrates for disposal.
- Packaging and storage of wastes contaminated with or containing nonreclaimable special nuclear materials, excluding direct burial in soil.

1.3 SPECIAL AUTHORIZATIONS AND EXEMPTIONS

1.3.1 ACTIVITIES NOT REQUIRING PRIOR NRC AUTHORIZATION BY LICENSE AMENDMENT

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

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

1.3.2 AUTHORIZED GUIDELINES FOR CONTAMINATION-FREE ARTICLES

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

1.3.3 SEALED PLUTONIUM SOURCES

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

1.3.4 EXEMPTION TO CRITICALITY MONITORING SYSTEM REQUIREMENTS

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

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

1.3.5 REACTOR PRODUCED TRANSURANICS

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

1.3.6 NON-APPLICABILITY OF 10 CFR 70.61 THROUGH 70.76

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

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.

200 The licensee shall make a reasonable effort to eliminate residual contamination.

200 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.

200 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.

200 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:

200 Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

200 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.

200 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:

200 Identify the premises.

200 Show that reasonable effort has been made to eliminate residual contamination.

200 Describe the scope of the survey and general procedures followed.

200 State the findings of the survey in units specified in the instruction.

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

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

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

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

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

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

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

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

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

LICENSE CONDITION FOR LEAK TESTING

SEALED PLUTONIUM SOURCES

April 1993

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

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

C. If the test reveals the presence of 0.005 microcurie or more of removable alpha contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it ot 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 VNC policy is to maintain a safe work place for its employees, to protect the environment, and to assure operational compliance within the terms and conditions of special nuclear material licenses and applicable NRC regulations.

2.2 ORGANIZATIONAL RESPONSIBILITIES AND AUTHORITY

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

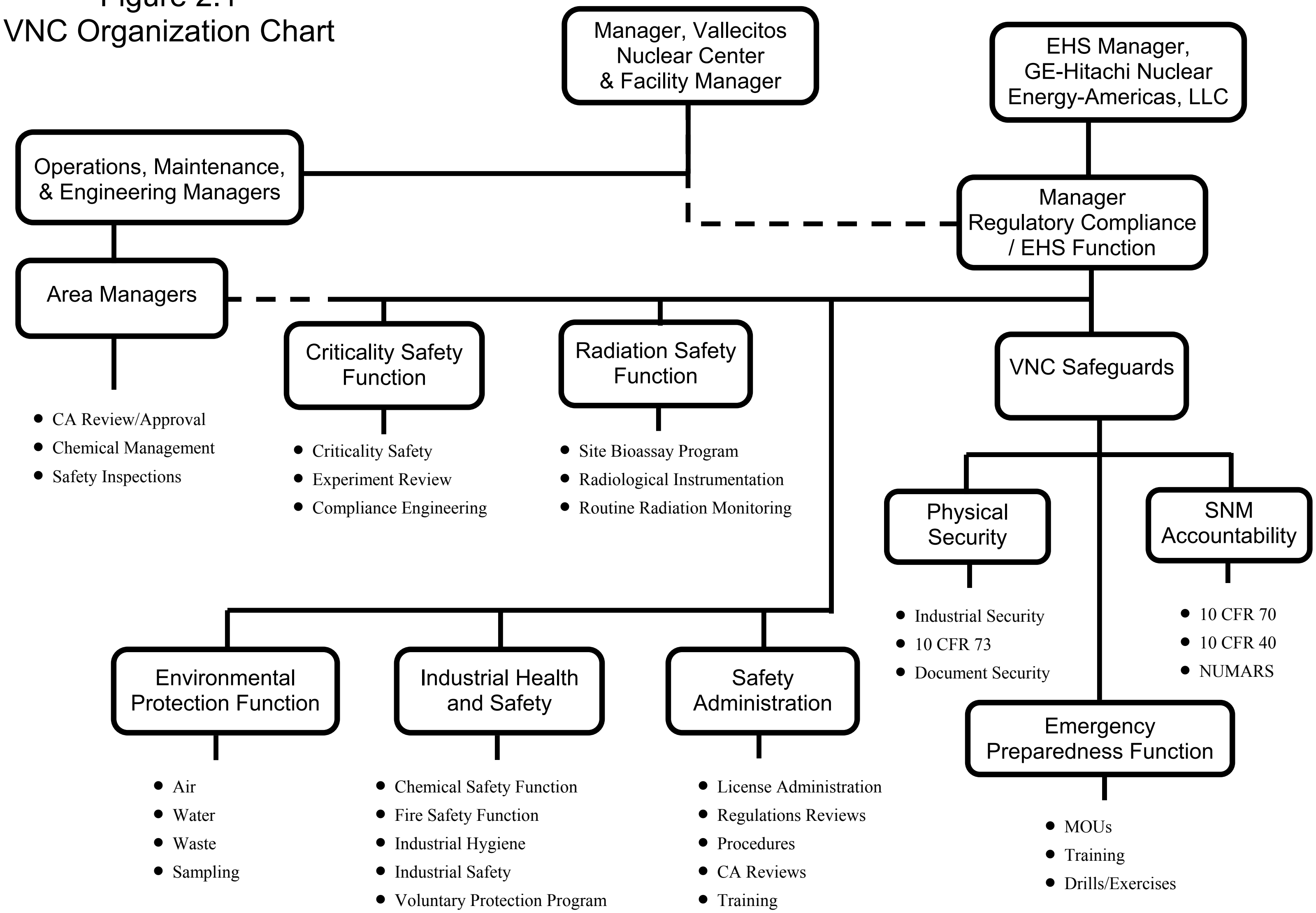
Responsibilities, authorities, and interrelationships among the VNC organizational functions with responsibilities important to safety are specified in approved position descriptions and in documented and approved practices.

2.2.1.1 VNC's Facility Manager

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

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

Figure 2.1
VNC Organization Chart



2.2.1.2 Area Manager

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

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

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

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

2.2.1.3 Regulatory Compliance Function

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

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

2.2.1.4 Criticality Safety Function

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

Designated responsibilities include:

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

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

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

2.2.1.5 Radiation Safety Function

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

Designated responsibilities include:

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

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

1. The radiation safety function manager shall hold a BS or BA degree in science or engineering, have at least five years experience in assignments that include responsibility for radiation safety, and have experience in the understanding, application and direction of radiation safety programs. An alternate minimum experience qualification is the professional certification in health physics (CHP).
2. A specialist in the radiation safety function shall hold a BS or BA degree in science or engineering, have at least two years experience in assignments that

include responsibility for radiation safety or have eight years of experience in health physics or radiation protection.

3. A radiation monitoring technician (RMT) in the radiation safety function shall have a high school degree, or equivalent, with two years experience in handling radioactive materials, or two years of college and four months experience; also, successful completion of a General Electric certification program, which includes written examinations covering radiation protection procedures.

2.2.1.6 Environmental Protection Function

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

Designated responsibilities include:

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

2.2.1.7 Chemical and Fire Safety Function

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

Designated responsibilities include:

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

2.2.1.8 Site Security and Emergency Preparedness Function

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

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

2.2.2 MANAGEMENT CONTROLS

Management controls for the conduct and maintenance of VNC's health, safety and environment protection programs are contained in documented plant practices described in Chapter 11, and approved by responsible management. Such practices are part of a controlled document system, and appropriately span the organizational structure and major plant activities to control interrelationships, and to specify program objectives, responsibilities and requirements. Personnel are appropriately trained to the requirements of these management controls, and compliance is monitored through internal and independent audits and evaluations.

Management controls documented in practices address requirements including:

- Configuration Management

- Radiation Safety
- Criticality Safety
- Environmental Protection
- Chemical Safety
- Fire & Explosion Safety
- Emergency Preparedness
- Quality Assurance
- Training
- Procedures
- Maintenance
- Audits
- Incident Investigation & Reporting
- Fissile Material Accountability and Control

2.3 TRAINING AND CONTINUING ASSURANCE

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

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

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

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

2.2.3 NUCLEAR SAFETY TRAINING

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

2.2.4 OPERATIONAL TRAINING

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

2.4 **SAFETY COMMITTEES**

2.4.1 VALLECITOS TECHNILOGICAL SAFETY COUNCIL

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

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

- Programs for improving the effectiveness of equipment used for effluent and exposure control

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

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

CHAPTER 3.0

INTEGRATED SAFETY ANALYSIS

3.1 REGULATORY STATUS

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

3.2 CURRENT SITUATION

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

3.3 BASIS OF SAFETY

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

CHAPTER 4.0

RADIATION SAFETY

4.1 ALARA (AS LOW AS IS REASONABLY ACHIEVABLE) POLICY

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

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

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

4.2 RADIATION SAFETY PROCEDURES AND RADIATION WORK PERMITS

Routine work performed in radiation-controlled areas is administered by the use of standard practices and procedures described in Chapter 11. Non-routine activities, performed by VNC and non-VNC employees, (i.e. those not covered by documented procedures), are administered by the Radiation Work Permit (RWP) system. The RWP system is described in documented plant practices and procedures.

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

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

4.3 VENTILATION REQUIREMENTS

4.3.1 INTER-AREA AIR FLOW DESIGN

Ventilation equipment is designed to provide airflow from areas of lesser potential contamination to areas of higher potential contamination. Direction of airflow between areas is checked on a routine surveillance interval documented in written procedures or after significant changes to the ventilation system. Installed ventilation system HEPA filters are fire resistant.

4.3.2 ENCLOSURES AND LOCALIZED VENTILATION

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

Air flows through hood openings and localized vents are maintained in accordance with Table 4.1. 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.

4.3.3 EXHAUST SYSTEM

Potentially contaminated air is exhausted through high efficiency filter media, which are rated to be 99.97% efficient for removal of 0. 3-micron particles. HEPA filters in the exhaust system are equipped with a device for measuring differential pressure. Differential pressures greater than four inches of water are investigated. In no case will filters be operated at a differential pressure, which exceeds the manufacturer's ratings for the filter.

4.4 AIR SAMPLING PROGRAM

Air is sampled continuously in normally occupied areas in which dispersible SNM is handled. Samples are analyzed for gross alpha and gross beta-gamma. Potential airborne radioactivity-producing operations shall utilize close-capture ventilation devices, (e.g., hoods or high-velocity local exhaust). Samples used to determine worker intakes are collected in such a way that the concentrations of airborne radioactive material measured is representative of the air which workers breathe. Air sampling results are monitored by the radiation safety function to evaluate the effectiveness of personnel exposure controls.

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

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

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

4.5 CONTAMINATION CONTROL

4.5.1 SURVEYS

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

The minimum survey frequencies and maximum removable contamination action levels are as follows:

Area	Frequency	Action Limit (dpm α/100 cm^2)	Action Limit (dpm β&γ/100 cm^2)
Controlled Areas (Floors & Other Readily Accessible Surfaces)	Weekly	100	5000
Non-controlled Areas	Monthly	≥20	1000

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

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

4.5.2 ACCESS CONTROL

Routine access points to radiological controlled areas are established. Each contaminated area is controlled by a step-off pad. Instructions controlling entry and exit from radiological controlled areas are posted at the entry points. Personnel survey meters are provided at the exit of the radiological controlled areas. Posted instructions address the use of the survey meters and appropriate decontamination methods.

Alternate access points to radiological controlled 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 Radiation Work Permits, which establish controls to prevent the spread of contamination.

4.5.3 PROTECTIVE CLOTHING

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

The protective clothing is removed upon exit from the contaminated area at the step off pad boundary.

4.6 EXTERNAL EXPOSURE

Deep-dose equivalent and shallow-dose equivalent from external sources of radiation are determined by individually assigned dosimeters. The radiation safety function makes a determination to issue personnel dosimetry to individuals based on work area surveys, occupancy time, or other exposure information such as area monitor results. Personnel dosimeters are processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited vendor. The capability exists to process dosimeters expeditiously if there is an indication of an exposure in excess of established action guides. Action guides for external exposures are documented in plant procedures. Radiation exposure action levels are specified in Section 4.9.

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

4.7 INTERNAL EXPOSURE

Intakes are assigned to individuals based upon one or more types of measurements as follows: air sampling (described in Section 4.4), urinalysis or whole body counting. Intakes are converted to committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) for the purposes of limiting and recording occupational doses. Action levels are established in plant procedures to prevent an individual from exceeding the occupational exposure limits specified in 10 CFR 20. Radiation exposure action levels are specified in Section 4.9.

4.7.1 URINALYSIS PROGRAM

A urinalysis program is available to evaluate the intake of alpha emitting radionuclides. Urinalysis may be performed on a job-by-job basis when radioactive materials which cannot be directly detected by the whole body counter and which are not tagged with isotopes detectable by the whole body counter are handled. All sample results above background are investigated. Urinalysis may also be used to monitor individuals involved in non-routine operations or incidents.

4.7.2 WHOLE BODY COUNTING

VNC employees are scheduled to receive a whole body count at least annually. Baseline and termination counts are performed when feasible. An investigation shall be initiated if a whole body count result exceeds 10% of an ALI.

4.8 SUMMING INTERNAL AND EXTERNAL EXPOSURE

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

4.9 ACTION LEVELS FOR RADIATION EXPOSURES

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

4.10 RESPIRATORY PROTECTION PROGRAM

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

4.10.1 QUALIFICATIONS OF RESPIRATOR USERS

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

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

4.10.2 RESPIRATORY PROTECTION EQUIPMENT

Only NIOSH approved 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.

4.10.3 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.

4.11 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. Table 4.2 lists examples of the types and uses of available instrumentation.

4.11.1 CALIBRATION

Portable instrumentation is calibrated before initial use, after major maintenance, and on a routine basis with a maximum interval of 12-months between calibrations. Calibration consists of a performance check on each range scale of the instrument with a radioactive source of known activity traceable to a recognized standard such as the National Institute of Standards and Technology (NIST).

TABLE 4.1
SPECIFIC FACILITIES & CAPABILITIES OF VENTILATION SYSTEMS

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

TABLE 4.2
TYPES & USES OF AVAILABLE INSTRUMENTATION (TYPICAL)

Type	Typical Range	Routine Use
DOSE RATE METERS		
gamma	1 – 50,000 mR/h	Area Dose Rate Survey, Shipment Survey
beta	4 – 200,000 mrad/h	Emergency Monitoring
Beta / gamma Survey Meter (GM)	Twice background to 40,000 cpm	Direct Personnel & Equipment Surveys
ALPHA SURVEY METERS	20 - 30,000 cpm	Direct Personnel & Equipment Surveys
NEUTRON METERS	1 – 5,000 mRem/h	Special Dose Rate Surveys
LABORATORY INSTRUMENTATION		
Automatic air sample counter	N/A	Lab Analysis
Fixed geometry proportional counter	N/A	Lab Analysis
Scintillation Counter	N/A	Lab Analysis
Whole Body Counter	N/A	Body Deposition Measurements

CHAPTER 5.0

NUCLEAR CRITICALITY SAFETY

5.1 NUCLEAR CRITICALITY SAFETY PROGRAM MANAGEMENT

5.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

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

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

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

5.1.2 EVALUATION OF CRITICALITY SAFETY

5.1.2.1 Changes to Facility

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

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

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

5.1.2.2 Role of the Criticality Safety Function

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

5.1.3 OPERATING PROCEDURES

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

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

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

5.1.4 POSTING AND LABELING

5.1.4.1 Posting of Limits and Controls

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

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

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

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

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

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

5.1.4.2 Labeling

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

5.2 ORGANIZATION AND ADMINISTRATION

5.2.1 GENERAL ORGANIZATION AND ADMINSTRATION METHODS

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

5.2.2 CRITICALITY SAFETY FUNCTION

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

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

5.3 MANAGEMENT MEASURES

5.3.1 GENERAL CONFIGURATION MANAGEMENT

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

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

5.3.2 CSF CONFIGURATION MANAGEMENT

5.3.2.1 Auditing, Assessing and Upgrading the CSF Program

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

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

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

5.3.2.2 Modifications to Operating and Maintenance Procedures

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

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

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

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

The criticality accident alarm system (CAAS) radiation monitoring unit detectors are uniform throughout required facilities for the type of radiation detected, the mode of detection, the alarm signal and the system dependability. Also, individual unit detectors are located to assure compliance with appropriate requirements of 10CFR70.24(a)(1).

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

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

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

5.3.2.4 Corrective Action Program

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

5.3.2.5 CSF Records Retention

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

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

5.4 METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 CONTROL PRACTICES

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

5.4.1.1 Verification Program

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

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

5.4.1.2 Maintenance Program

Details of the maintenance program are described in Chapter 11.

5.4.2 MEANS OF CONTROL

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

5.4.2.1 Passive Engineered Controls

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

5.4.2.2 Active Engineered Controls

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

5.4.2.3 Administrative Controls

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

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

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

5.4.3 SPECIFIC PARAMETER LIMITS

The **safe mass** limit values of Table 5.1 below are specifically licensed 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 Section 5.4.5.2 of this chapter.

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

kgs X = (kgs UO_2 $\bullet$ 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)*
U-235 Sphere	575 g ^{235}U
U-235 Hemisphere	650 g ^{235}U

- Includes Bias Uncertainty.

Where applicable, an **equivalence factor** is used to calculate the U-235 equivalent fissile mass of fissile material for mixtures of uranium and plutonium isotopes. The U-235 equivalent fissile mass is used when implementing U-235 derived mass limits. Use of this equivalence factor assures that the calculated U-235 equivalent mass of fissile isotopes is no more reactive than the same mass of U-235. For Criticality

Limit Areas (CLAs) in which both uranium and plutonium are present the following equation may be used to enforce U-235 derived mass limits (grams).

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

where,

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

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

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

A **subcritical area** means a physically identified area or location involving special nuclear materials in quantities of less than the established Table 5.1 safe uranium mass limits (grams uranium-235 equivalent); 520 grams of uranium-233, or 450 grams of plutonium. A subcritical area is considered neutronically isolated when it meets the spacing requirements of Section 5.4.4.6 from other areas which special nuclear material is handled, used, or stored; or an unrelated building or structure which meets the criteria of 10CFR70.24(a).

5.4.4 CONTROL PARAMETERS

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

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

5.4.4.1 Geometry

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

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

5.4.4.2 Mass

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

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

5.4.4.3 Moderation

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

5.4.4.4 Concentration (or Density)

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

5.4.4.5 Neutron Absorber

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

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

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

5.4.4.6 Spacing (or Unit Interaction)

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

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

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

5.4.4.7 Material Composition (or Heterogeneity)

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

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

5.4.4.8 Reflection

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

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

5.4.4.9 Enrichment

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

5.4.4.10 Process Characteristics

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

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

5.4.5 ANALYSIS METHODS

5.4.5.1 Keff Limit

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

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

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

5.4.5.2 Analytical Methods

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

5.4.5.3 Validation Techniques

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

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

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

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

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

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

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

- Geometry configuration of the system (such as, shape, size, spacing, reflector).

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

- Critical experiments are assessed for completeness, accuracy, and applicability to the VNC operations prior to its selection and use as a critical benchmark.
- Critical experiments are selected to cover the spectrum of parameters spanning the range of normal and credible abnormal conditions anticipated for past, current, and future analyzed uranium systems for modeled systems.
- Critical experiments are drawn from multiple series and sources of critical experiments to minimize systematic error. The range of parameters characterized by selected critical experiments is used to define the area of applicability for the code.

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

- The normality of calculated k_{eff} 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 k_{eff} 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:

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

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

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

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

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

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

- The extrapolation should not result to a large underlying physics or neutronic behavior change in the anticipated application. If there is a rapid or non-conservative change in bias in the vicinity of the AOA range endpoints of a trending parameter, extra safety margin should be needed. Otherwise, critical experiments should be added for further justification.

- Statistical methods should be used to ensue that the extrapolation is not large. The leverage statistic, a measure of the distance between the extrapolation point x for a predication and the mean of trending parameter values in the critical benchmark data set can be used to determine if an extrapolation using the regression model is acceptable when making predications at x.

5.4.5.4 Computer Software & Hardware Configuration Control

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

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

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

5.4.5.5 Criticality Safety Analysis (CSA)

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

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

system. This section includes a discussion as to how the established nuclear criticality safety limits are addressed for each credible process upset (accident condition) pathway.

- **Specifications and Requirements for Safety** - When applicable, this element presents both the design specifications and the criticality safety requirements for correct implementation of the established controls. These requirements are incorporated into operating procedures, training, maintenance, quality assurance as appropriate to implement the specifications and requirements.
- **Compliance** - This element concludes the analysis with pertinent summary statements and includes a statement regarding license compliance.
- **Verification** - a senior engineer approved by the criticality safety function and who was not involved in the analysis verifies each criticality safety analysis.
- **Appendices** - Where necessary, a summary of information ancillary to calculations such as parametric sensitivity studies, references, key inputs, model geometry plots, equipment sketches, useful data, etc., for each defined system is included.

5.4.5.6 Technical Reviews

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

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

- Verify the calculations with an alternate computational method.
- Verify methods with an independent analytic approach based on fundamental laws of nuclear physics.
- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations.
- Verify the calculations using specific checks of the computer codes used, as well as, evaluations of code input and output.

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

5.4.6 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

CHAPTER 6.0
CHEMICAL SAFETY

6.1 CHEMICAL SAFETY PROGRAM

The chemical safety program is applicable to the chemicals produced from licensed materials described in Chapter 1. VNC chemical safety program is documented in written, approved practices that are followed, and ensures that processes and operations comply with applicable federal and state regulations pertaining to chemical safety.

Hazard evaluations are performed on nuclear and non-nuclear operations within the VNC where the potential exists for hazardous chemicals to be used in such a manner that they could affect the nuclear and industrial safety program. This ensures appropriate controls are in place for adequate protection of the general public and safe use by employees, and that the use of chemicals does not create potential conditions that adversely affect the handling of licensed nuclear materials.

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

6.2 CONTENTS OF CHEMICAL SAFETY PROGRAM

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

6.2.1 CHEMICAL SAFETY

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

6.2.2 CHEMICAL APPROVAL / EVALUATION

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

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

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

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

6.2.3 LABELING & IDENTIFICATION

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

6.2.4 EMPLOYEE TRAINING & AWARENESS

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

6.2.5 CONDUCT OF OPERATIONS

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

CHAPTER 7.0

FIRE SAFETY

Appropriate combinations of fire prevention measures and response systems achieve VNC's fire protection. Such measures and systems are designed and maintained in accordance with federal, state, and local codes, appropriate industry standards and prudent practices. VNC used the more restrictive of the California Fire Code or NFPA as the standard practice and guidance in effect at the time of construction.

7.1 FIRE PROTECTION PROGRAM RESPONSIBILITY

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

7.2 FIRE PROTECTION PROGRAM

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

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

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

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

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

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

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

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

7.3 ADMINISTRATIVE CONTROLS

(See Chapter 11)

7.4 BUILDING CONSTRUCTION

7.4.1 EXISTING BUILDINGS

The existing building's original design is in accordance with the local, state, federal and national codes, standards and/or regulations in effect at the time of construction. The building and appurtenances used to process and store hazardous materials are designed to provide containment of such material under the conditions of fire and explosion.

7.5 PROCESS FIRE SAFETY

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

7.6 FIRE DETECTION AND ALARM SYSTEMS

7.6.1 DETECTION DEVICES

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

- Smoke Detectors
- Heat Detectors

7.6.2 ALARMS

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

7.7 FIRE SUPPRESSION EQUIPMENT

VNC's fire protection system is designed in accordance with the applicable and most restrictive of the California Fire Code or NFPA applicable at the time of construction.

Selection of equipment for suppression of fire takes into account the severity of the hazard, the type of activity to be performed, the potential consequences of a fire, and the potential consequences of use of the suppression equipment (including, risk of accidental criticality).

Portable fire extinguishers, of sufficient capacity, quantity and type of suppression agent used, are available and maintained throughout the site.

7.8 FIRE PROTECTION WATER SYSTEM

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

7.9 EMERGENCY RESPONSE TEAM

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

CHAPTER 8.0

RADIOLOGICAL CONTIENGIENCY & EMERGENCY PLAN

VNC is not required to maintain a Radiological Contingency & Emergency Plan based on 10 CFR 70.22(i)(1)(i) and as approved in a letter from Charles J. Haughney to General Electric Company, Vallecitos Nuclear Center dated December 20, 1990 wherein license condition 12 of SNM-960 was deleted.

BACKGROUND

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

NUREG-1140, "Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees", identifies the only accidents for which protective action guide doses, or the 2-milligram soluble uranium Intake could be exceeded to be a UF6 Cylinder rupture, a fire or a criticality accident.

VNC does not process or handle any UF6 so this scenario is not credible at VNC.

Release of radioactive material by a fire was most recently addressed in "General Electric Vallecitos Nuclear Center Safety Evaluation Report", pages 53-55,dated May 1984 that referenced "Accident Analysis for the General Electric Company Vallecitos Nuclear Center at Pleasanton, California", final draft dated October 1978. This 1978 document presents bounding scenarios of releases by fire with potential site boundary doses ranging from a low value less than 1 rem to an upper value of 50 rem.

Subsequent to this evaluation, the May 1984 renewal of SNM-960 significantly reduced the authorized activities at the site resulting in none of the operations contributing to the boundary scenarios existing in the current licensed activities, and no new activities have been licensed since May of 1984. Of the areas identified as having a potential for accidents due to fire, only the Radioactive Materials laboratory (RML) activities in Building 102 and 103 still utilize SNM. The ventilation and filtration systems, coupled with the small quantities of SNM used in the RML cells precludes the release of any significant quantity of SNM from a fire and therefore the site boundary effects would be less than 1 rem.

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

In order to provide a boundary dose evaluation of a criticality accident, the fuel vault in Building 103 (generally the largest accumulation of fissile material) was selected. It was assumed that double the normal amount of material was in the vault in an optimally moderated and reflected condition resulting in a single burst criticality event. Using Regulatory Guide 3.34, the radiation dose at the nearest site boundary was calculated to be 0.35 rem effective dose equivalent and 0.30 rem to the thyroid. These values are significantly below the guidelines of 1 rem effective dose equivalent.

CHAPTER 9.0

ENVIRONMENTAL PROTECTION

9.1 ENVIRONMENTAL PROTECTION PROGRAM & REPORTS

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

VNC's environmental assessment was most recently reviewed by NRC January 21, 1994 and found to have no significant impact in support of SNM-960's prior license renewal. Since that date VNC has submitted annual environmental reports summarizing the effluent monitoring and environmental surveillance programs. This data continues to support no significant impact to the environment.

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

9.2 AIR EFFLUENT CONTROLS AND MONITORING

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

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

9.3 LIQUID TREATMENT FACILITIES

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

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

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

9.4 SOLID WASTE MANAGEMENT FACILITIES

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

9.5 PROGRAM IMPLEMENTATION

VNC's environmental monitoring program includes the types of samples and monitored parameters listed in Figure 9.1. Analytical sensitivities (minimum detection levels) are also illustrated in Figure 9.1. Action levels will be included in documented procedures for environmental monitoring parameters as appropriate so that internal review and other actions are initiated. Such action levels provide guidance in assuring compliance within 10 CFR 20 limits. Environmental monitoring sample locations are shown for air (Figure 9.2), vegetation and dosimetry (Figure 9.3), surface water and sanitary/industrial discharge (Figure 9.4), and stream bottoms (Figure 9.5). For monitoring wells found not to contain water at time of sampling, an evaluation is performed by the Radiation Protection function to determine if alternate well sampling data may be used or other assessments will be used. These program elements, analytical sensitivities, and/or locations may be changed in accordance with the Change Authorization procedure described in Chapter 11 without prior NRC approval.

9.6 EVALUATIONS

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

9.7 OFF-SITE DOSE

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

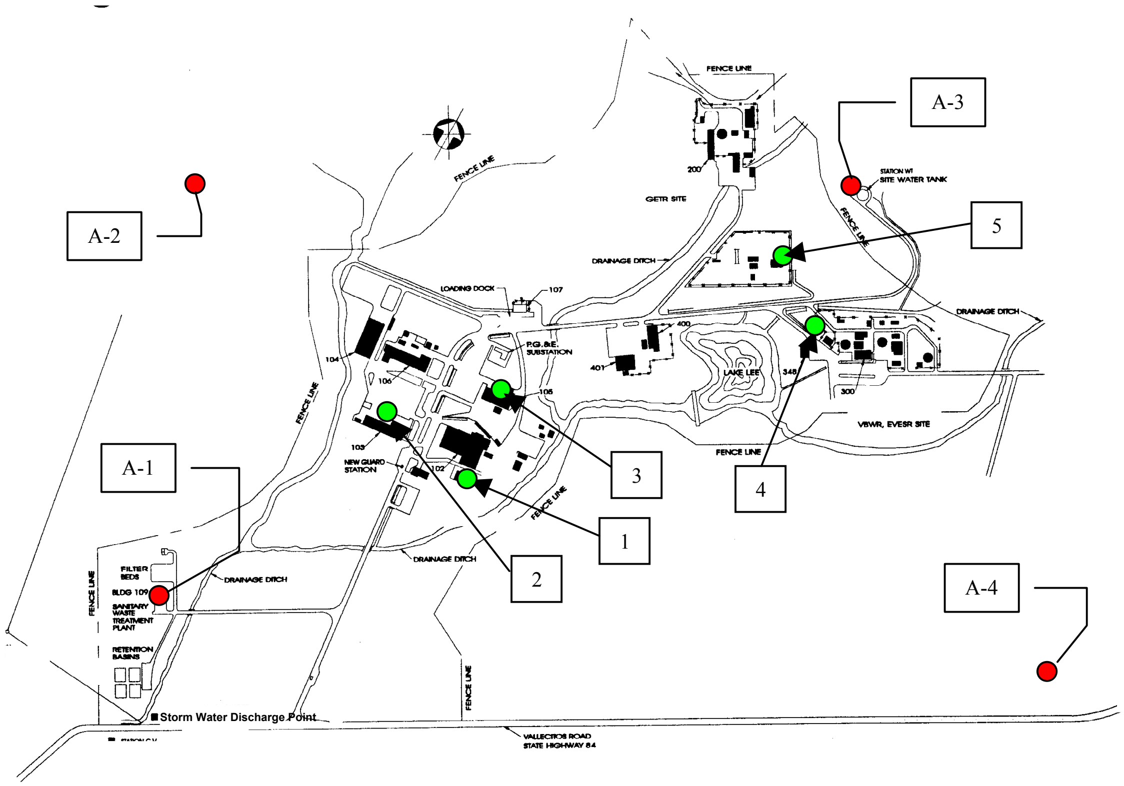
9.8 ALARA

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

FIGURE 9.1
VNC'S ENVIRONMENTAL MONITORING PARAMETERS

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

FIGURE 9.2
Air Sampling Sites (Typical)



VALLECITOS NUCLEAR CENTER SITEMAP

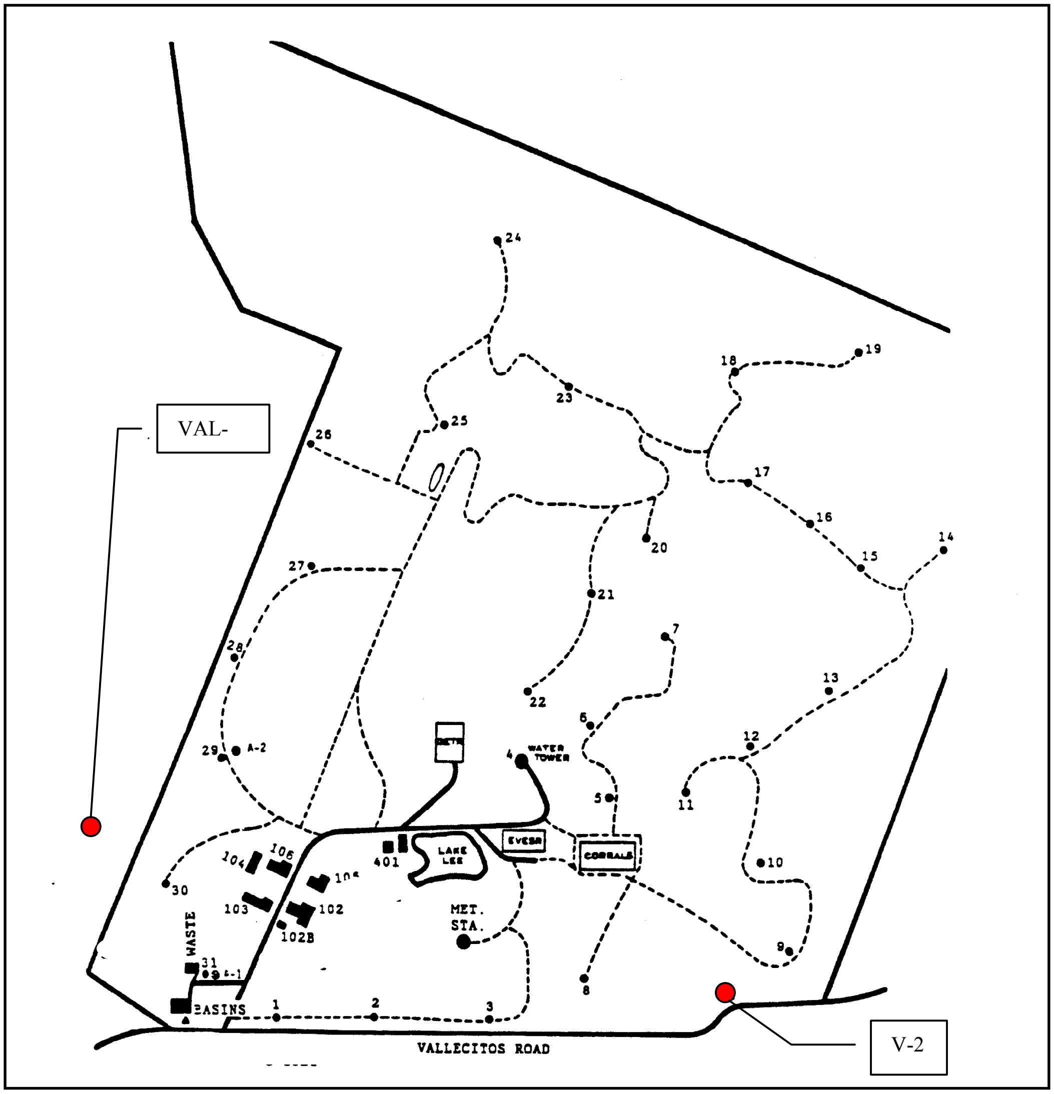
🔴 Environmental Air Monitoring Stations (A-1, A-2, A-3, A-4)

🟢 Stack Air Monitoring Stations

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

FIGURE 9.3

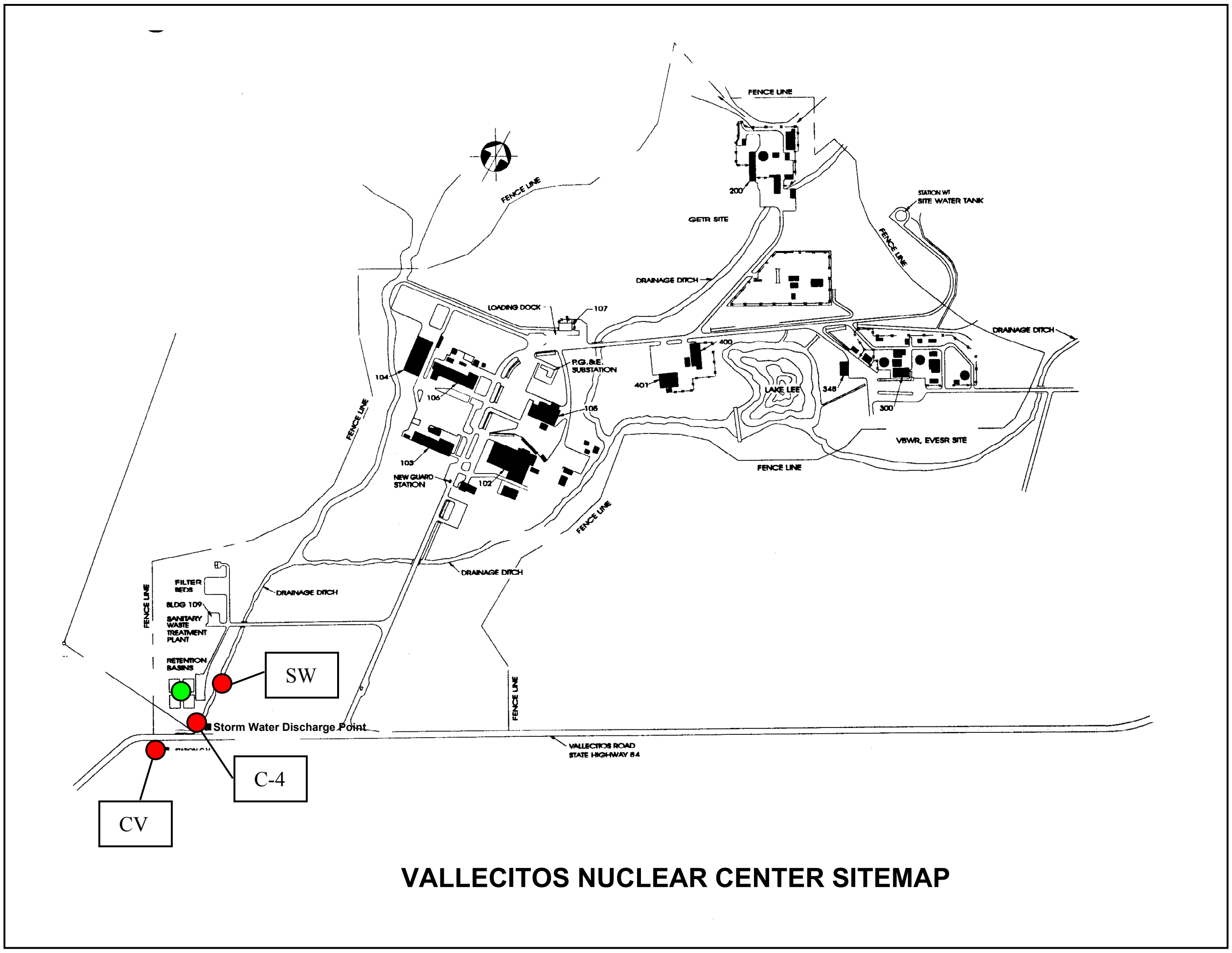
Environmental and Vegetation Sampling Points (Typical)



● Environmental Dosimeter Monitoring Stations (1-31)

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

Figure 9.4
Location of Surface Water Monitoring Points (Typical)

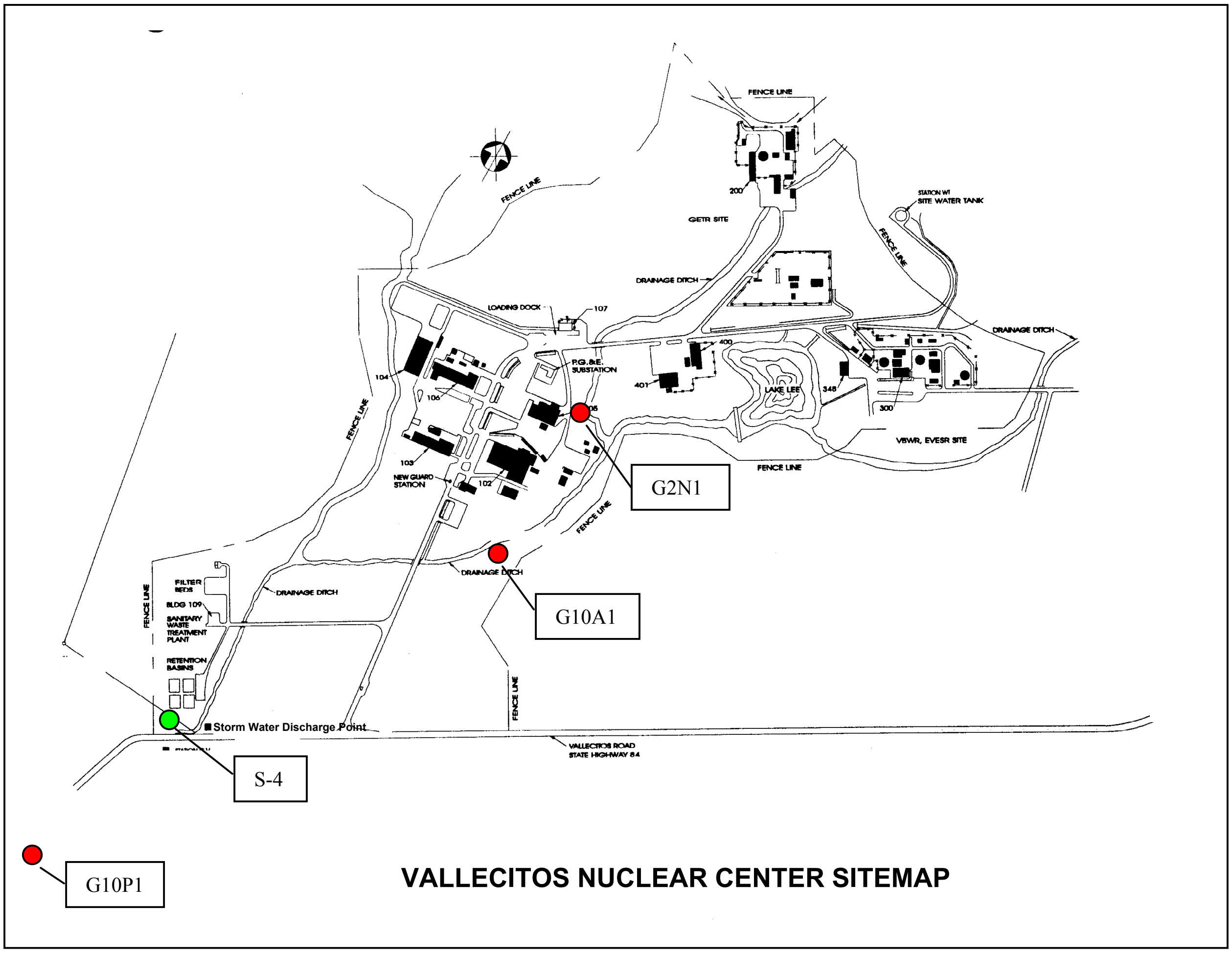


- Surface Water Monitoring Points
  - C-4
  - CV
  - SW
- Sanitary & Industrial Discharge Water Monitoring Points
  - Basin 1
  - Basin 2
  - Basin 3
  - Basin 4

4	1
3	2

Figure 9.5

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



- Radioactive Ground Water Monitoring Wells
  - G2N1
  - G10A1
  - G10P3

- Stream Bottoms Monitoring Point
  - S-4

CHAPTER 10.0
DECOMMISSIONING

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

The Decommissioning and Closure Plan for the facility was originally submitted to the NRC on February 17, 1982.

At the end of plant life, GE Hitachi through a parent company guarantee shall decommission the facilities and site in accordance with the then current Decommissioning and Closure Plan.

CHAPTER 11.0
MANAGEMENT MEASURES

11.1 MANAGEMENT MEASURES

11.1.1 REASONABLE ASSURANCE

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

11.1.2 GRADED APPLICATION OF MANAGEMENT MEASURES FOR SAFETY

VNC applies *Management Measures* as described in this chapter.

11.2 CONFIGURATION MANAGEMENT (CM)

11.2.1 CONFIGURATION MANAGEMENT POLICY

VNC commits to maintain a formal configuration management process, governed by written, approved practices, and ensures that plant design changes do not adversely impact safety, health, or environmental protection programs at VNC. This program is also referred to as the Change Authorization (CA) system. The following items are addressed prior to implementing a change:

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

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

The CA system includes the following activities:

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

11.2.2 DESIGN REQUIREMENTS

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

11.2.3 DOCUMENT CONTROL

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

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

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

11.2.4 CHANGE CONTROL

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

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

11.2.5 ASSESSMENTS

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

11.3 MAINTENANCE

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

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

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

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

11.3.1 CORRECTIVE MAINTENANCE

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

VNC commits to promptly perform corrective actions to remediate unacceptable performance deficiencies in safety systems.

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

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

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

11.3.2 PREVENTATIVE MAINTENANCE

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

- Radiation Measurement Instruments
- Criticality Detection Devices
- Effluent Measurement Devices
- Standby Power Generators
- Fire Detection and Control Systems

- Pressure Relief Valves
- Air Compressors
- Steam Boilers

11.3.3 SURVEILLANCE/MONITORING

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

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

- Examples of active engineered controls
- Examples of passive engineered
- Examples of geometry-based

11.3.4 FUNCTIONAL TESTING

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

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

11.4 TRAINING AND QUALIFICATIONS

11.4.1 ORGANIZATION AND MANAGEMENT OF THE TRAINING FUNCTION

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

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

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

11.4.2 FUNCTIONAL AREAS REQUIRING TRAINING

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

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

- General employee training
- Technical training
- Developmental training

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

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

11.4.2.1 General Employee Training

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

GET topics are listed below:

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

- Environmental Protection

11.4.2.2 Nuclear Safety Training

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

Formal Nuclear Safety training includes:

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

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

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

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

11.4.2.3 Industrial, Chemical, Fire, Health and First Aid

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

OSHA General Duty Clause

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

11.4.2.4 Technical Training

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

- On-the-Job Training
- Process Specific Training

- Mechanical Maintenance
- Controls, Instrumentation, Electrical Maintenance
- Chemistry

11.4.2.5 Development Training

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

11.4.3 POSITION TRAINING REQUIREMENTS

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

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

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

11.4.4 BASIS OF TRAINING AND OBJECTIVES

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

11.4.5 EVALUATION OF TRAINEE LEARNING

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

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

11.4.6 CONDUCT OF ON-THE-JOB TRAINING

On-the-Job training (OJT) is a systematic method of providing the required job related skills and knowledge for a position. This training is conducted in the work environment. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The object of the program is to assure the trainee's ability to proficiently perform job duties as required for the assigned role. Refer to Section 11.4.3.

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

11.4.7 EVALUATION OF TRAINING EFFECTIVENESS

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

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

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

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

11.4.8 PERSONNEL QUALIFICATION

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

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

11.5 PROCEDURES

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

11.5.1 OPERATING PROCEDURES

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

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

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

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

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

11.5.2 MANAGEMENT CONTROL PROCEDURES

Licensed material activities are conducted in accordance with management control programs described in administrative and general plant practices approved and issued by management at a level appropriate to the scope of the practice. These documented practices direct and control activities across the manufacturing functions, and assign functional responsibilities and requirements for these activities. These practices are reviewed for updating at least every two years (26 months).

11.6 AUDITS AND ASSESSMENTS

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

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

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

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

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

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

Records of the audit or inspection, instructions and procedures, persons conducting the audits or inspections, audit or inspection results, and corrective actions for identified violations of license conditions are maintained in accordance with procedural requirements for a minimum period of three years.

11.6.2 INDEPENDENT AUDITS

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

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

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

11.6.3 WORKER CONCERNS

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

11.7 INCIDENT INVESTIGATIONS

VNC commits to maintain a system to identify, track, investigate and implement corrective action for abnormal events (unusual incidents). The system includes the following requirements and features:

- The system operates in accordance with written procedures
- Abnormal events are documented, tracked and reported to the Area Managers, the safety functions and facility management
- Each event is considered in terms of regulatory reporting criteria
- Events are considered in terms of severity and consequence
- Events with little or no consequence (Category I, type 2 events) require investigation, a determination of root or most probable cause and the identification of required corrective action
- Reportable conditions or other events with moderate to high consequences require a formal, systematic determination of root cause (typically using an independent, qualified team), definition of corrective actions and a higher level management review and approval of the investigation and corrective actions
- Records of the events and the documented evidence of closure are maintained for a minimum of three years

11.8 RECORDS MANAGEMENT

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

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

Records of material control and accountability are maintained in sufficient detail and form to permit independent review and audit of the inventory and transactions. Such records are retained for a minimum of three years.

Records associated with personnel radiation exposures are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20. The following additional radiation protection records will be maintained for at least three years:

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

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

11.9 OTHER QA ELEMENTS

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

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

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

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