GENERAL ELECTRIC

AFFIRMATION

GENERAL ELECTRIC TEST REACTOR
LICENSE RENEWAL INFORMATION

TO THE BEST OF MY KNOWLEDGE AND BELIEF, THE INFORMATION CONTAINED IN THE ENCLOSED DOCUMENT IS ACCURATE.

R W Darmitzel

R. W. DARMITZEL, MANAGER
IRRADIATION PROCESSING OPERATION

SUBMITTED AND SWORN BEFORE ME THIS first DAY OF JULY, 1985.
Christine Ariizumi, NOTARY PUBLIC, IN AND FOR THE COUNTY OF ALAMEDA, STATE OF CALIFORNIA.

OFFICIAL SEAL
CHRISTINE ARIIZUMI
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My comm. expires APR 2, 1989

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REQUEST FOR AMENDMENT
TO
LICENSE RENEWAL APPLICATION

LICENSE TR-1

I. Introduction

On October 21, 1975, the General Electric Company filed a timely application requesting renewal of License TR-1 for the General Electric Test Reactor (GETR). The GETR was shut down in October, 1977, and has not operated since that time. General Electric hereby submits an amendment to the October 21, 1975, application for renewal of License TR-1 to request authorization for the possession but not operation of the reactor and the adoption of new proposed Technical Specifications (Appendix A to this document). Activities performed at the facility would include decontamination testing and decommissioning training exercises.

The primary radioactive material present at the GETR is Co-60. ALARA considerations are best served by adopting a possession/non-operating status to allow for the decay of this 5.2-year half-life isotope. Security and nuclear safety organizations at the Vallecitos Nuclear Center (VNC) will continue to serve the GETR facility.

II. Discussion

The GETR has not been operated since October, 1977. All GETR irradiated and unirradiated fuel, fueled experiments, and targets containing SNM have been removed from the reactor facility and shipped from the Vallecitos Nuclear Center (VNC). In addition, all contaminated resins have been removed from the various demineralizer facilities and shipped to a licensed waste disposal facility. Therefore, only activation and fission product contamination remain.

VNC utilizes a confinement approach at the GETR to minimize the possibility of contamination spreads and uncontrolled discharges. The confinement system consists of primary containers, piping, the ventilation system, and the reactor containment building. As this building no longer is maintained as a containment system, in the future it will be referred to as the "reactor building".

Beta-gamma continuous air monitors (CAMs) located in appropriate locations will monitor the effectiveness of the confinement technique. Continuous air sampling, contamination and radiation surveys, and a bioassay program also are used as a part of the radiation surveillance program.

Continuous air monitor and sampler filter changes, contamination and radiation surveys, and bioassay analyses are made on frequencies that are dictated by the contamination potential and the activity level for the involved area.

Exhaust air from the facility is sampled in the discharge stack after passing through a bank of particulate filters. The sample, collected on a membrane filter, is analyzed weekly. There has never been any discharge of quantities of airborne activity which would be of concern under 10CFR20.106 before or since shutdown in 1977.

A report of the current status of the facility is enclosed as Appendix B to this application. This report is for information only and does not represent a commitment to preserve this status. In fact, it is anticipated that the status will change as discussed below.

It is planned to utilize the facility for certain restricted activities involving activated or contaminated reactor facility structures, components or systems that could cause airborne radioactive material. All such activities will be conducted in accordance with the applicable limits of 10CFR20. These activities would include decontamination and decommissioning testing and training (D^2T^2) exercises. Work will be limited to equipment, components, or devices which could be removed, repaired, and reinstalled as part of "normal maintenance". In addition, all such work will exclude installation or reinstallation of any fuel, equipment, component or device for the purpose of restoring the facility to a condition where it would be capable of operating as a nuclear reactor.

All proposed restricted activities within the reactor facility will be documented and reviewed in accordance with license requirements and the site's established review procedures prior to conduct of the activity. Specific items that may be involved in future programs are discussed below.

1. Utilities

   Air, water and electrical power will be supplied as necessary to operate tools, decontaminate equipment, and perform similar supporting functions including the operation of monitoring and sampling equipment and systems.

2. Ventilation System

   A ventilation system will be provided for the reactor building. The system may or may not use components of the original system. However, all modifications will be conducted in accordance with the requirements of 10CFR50.59. Appropriate airborne radioactivity sampling/monitoring will be provided. As part of the radioactive material confinement system, the ventilation system will direct airflows from contaminated areas through the bank of particulate filters and out the reactor building stack.

3. Pool/Primary System

   A reduced water level is presently maintained in the pool/primary system to provide radiation shielding. This water level may be varied or the water completely removed.

4. Reactor Building

   Currently, the personnel airlock is padlocked closed and the equipment airlock is blocked closed from the inside. Authorized persons occasionally enter the building for periodic inspection. Access to the enclosure during the restricted activities programs will require provision of a modified personnel access control procedure and may require modification of the airlock closure methods. Criteria for such changes will be based on control of access to authorized personnel only as well as effective confinement of contamination.

5. <u>Reactor Facility Equipment and Components</u>

In order to develop decontamination test methods, to provide the necessary personnel training, and to conduct the requisite surveys of removable and nonremovable equipment and components including shielding, it will be necessary from time to time to remove and perhaps replace equipment and components with materials and devices more suitable for developing and testing decontamination methods, training of personnel, and monitoring of radiation and contamination fields and test results.

6. <u>Radiation Safety</u>

Existing equipment and procedures will be supplemented as necessary to ensure the facility is equipped and maintained to perform restricted activities in a manner providing a high degree of safety for persons employed at the facility and for residents of surrounding areas.

Similar work involving the dismantling and disposal of contaminated equipment is customary at the Vallecitos site. In 1960, the internal components of the VBWR reactor vessel were completely changed and the primary recirculation system was extensively modified. Again in 1962, sections of recirculation lines near the reactor vessel were removed and new sections welded in place. Deactivation of the VBWR and EVESR facilities involved modifying, dismantling, and disposing of radioactive systems and components.

IPO personnel have performed installation, modification, and removal of numerous experiment loops, heat exchangers, and piping systems at the General Electric Test Reactor and are continuously performing jobs at the Radioactive Material Laboratory (RML) involving radiation and contamination. All of these jobs involve similar machining, welding, cutting, and handling techniques and the same radiation safety practices that will be used in the proposed D^2T^2 programs or other restricted activities. Day-to-day operation of the RML also involves similar tasks, procedures, and practices.

Some dismantling and removal of equipment from the GETR as well as decontamination of the facility could result from the D^2T^2 type programs. Radiation and contamination levels have been measured periodically since the facility was shut down in 1977 providing an accurate record of the radiological conditions of the facility for appropriate planning of the D^2T^2 type activities.

A number of operating facilities remain on the VNC site. Accordingly, ongoing programs in the areas of environmental surveillance, radiation protection, and physical security are and will remain in effect. The site is manned on a 24-hour basis.

Typical major equipment and radiation protection procedures available for the protection of health, life, and property are set forth in Section III.

7. Administrative Considerations

A Facility Manager designated by the Manager, Irradiation Processing Operation, will be responsible for the safe and compliant conduct of restricted activities within the reactor facility and for the security, maintenance, and inspection of the deactivated facility.

III. Health and Safety

1. Radiation Protection Program

A system of Vallecitos Safety Standards establishes the site radiation protection and regulatory compliance program. The manager of the nuclear safety component issues the standards with review and comment of the managers of the major organizational components located on the site. Currently, there are about 50 standards dealing with radiation protection matters. The principal features of these are summarized below.

1.1 ALARA

Any controlled exposure to radiation (occupational or to the general public) must be justified by benefits that outweigh the requirements necessary to reduce or eliminate the exposure. For facility and procedure changes performed pursuant to 10CFR50.59, the reasons for accepting or permitting a particular level of exposure from an identified activity rather than reducing the exposure to a lower level are reviewed by an independent safety group. Planned exposures at the reactor facility which include those exposures associated with decontamination and decommissioning testing and training activities will be controlled as low as reasonably achievable (ALARA) through the iterative application of health physics practices and from managerial decisions which have qualitatively balanced the costs and benefits of alternative levels of control. General Electric Company management is fully committed in its support of the ALARA philosophy.

1.2 Limits of Radiation In Controlled Work Areas

All Vallecitos locations where there is a potential for radiation exposure are classified (radiation area, high radiation area, etc.) in accordance with the definitions of 10CFR20, Sections 20.202 and 20.203. General Electric's philosophy of protection is to keep radiation exposure at the lowest reasonably achievable level in all cases. Thus, design working limits substantially more conservative than the maximum permissible limits recommended by the National Committee on Radiation Protection are used as objectives of the radiation protection program. In no case may the calendar quarter limits of 10CFR20.101(a) be exceeded without written permission of the appropriate unit manager prior to the exposure and completion of data required by 10CFR20.101(b)(3). Managers of facilities are responsible for compliance with the above exposure limits. Work requiring skills of multiple organizational components is implemented by means of the Radiation Work Permit (RWP) procedure. The RWP is approved by the worker's direct supervisor and the supervisor of the area in which the work will be performed and by a person authorized to monitor and set time limits. The RWP is effective in preventing overexposure by assuring dose rates are known by monitoring measurements prior to the work. Employees who expect to receive a significant radiation dose maintain a current exposure estimate so that doses received by employees are known during film badge wearing periods. Exposures to dose rates in excess of 3 Rem per hour require appropriate management approvals as set forth in the site safety standards.

1.3 Respiratory Protection Program

With respect to operations which could produce airborne radioactive contamination, managers of facilities are responsible for providing ventilation equipment to meet the concentration limits of 10CFR20.103 without the necessity, or credit, for personal respiratory devices during routine operations. In certain nonroutine situations where adequate ventilation equipment would be impractical or could not ensure control of airborne material, respiratory protection of demonstrated integrity is utilized.

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

a. MSA Clearvue facepiece or Ultravue facepiece with a NIOSH-approved cartridge, or the MSA Ultra-Twin facepiece with a NIOSH-approved cartridge, or equivalent apparatus may be worn where concentrations do not exceed fifty times the concentrations of Appendix B, Table I. If radioactive iodine or other radioactive gases are present in concentrations in excess of concentrations of Appendix B, Table I, an air-supplied airline respirator normally is used.

b. NIOSH-approved plastic hoods with positive pressure, continuous flow air supply may be worn where concentrations may exceed fifty times the concentrations of Appendix B, Table I.

c. U.S. Divers Company's Survivair self-contained breathing apparatus (SCBA) or other NIOSH-approved SCBA having a back pack air supply, hose, harness, pressure demand regulator, and a facepiece may be worn where concentrations exceed the limits of protection provided by air-purifying and airline respirators.

VNC has adopted only equipment that is approved by NIOSH.

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

After use, masks are surveyed and thoroughly cleaned with detergents or soaps followed by a second survey when dry. No detectable contamination is permitted on the inside of the mask on the second survey, which must be conducted in a background of not more than 500 counts per minute (beta-gamma). Equipment parts and supplies are replaced as necessary by qualified technicians. The equipment is inspected to assure proper maintenance at all times. Masks which cannot be cleaned adequately or on which adequate surveys cannot be made are discarded to radioactive waste.

1.4 Personnel Monitoring

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

In order to determine the extent to which individuals may have internally accumulated any radioactive materials, two bioassay procedures are established. Persons who work routinely with radioactive materials, particularly alpha and beta emitters, are scheduled for urinalysis tests at least annually, or more frequently depending upon the potential for exposure to mechanisms which could cause ingestion, inhalation or absorption of such materials. Special analyses for other radioactive materials are performed whenever a need is indicated.

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

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

1.5 Surveys

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

Materials and equipment are surveyed prior to release from radioactive materials areas. Standards are established for each of two categories: Regulated Release and Unconditional Release. Regulated Release permits the removal of the item for reuse in another area at the site under conditions specified in writing on a tag attached to the item and permits the release of certified containers for off-site shipments. Unconditional Release permits reuse without restriction and requires survey using only the most sensitive detection instrumentation in background not exceeding 500 c/m beta-gamma or 200 d/m alpha. Unconditional Releases must be approved by radiation safety personnel.

1.6 Portable Monitoring Instruments

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

Portable Radiation Protection Instrumentation

	Instrument Type	Range
1.	Ionization Chamber G-M Region	0-500 k cpm, beta-gamma
2.	Ionization Chamber (Low Energy) Ionization Region	0-300 mrad/h, beta-gamma
3.	Ionization Chamber (CP) Ionization Region	1-250 k mR/h, gamma 4-1,000 k mrad/h, beta
4.	Ionization Chamber (Proportional Region)	1-1,000 k mR/h, gamma 20-20,000 k mrad/h, beta
5.	Ionization Chamber (Telescopic) G-M Region	1-1,000 k mR/h, gamma
6.	Scintillation Counter - Sodium Iodide (Tl)	0-500 k cpm, gamma
7.	Alpha Survey Probes (gas proportional and ZnS)	200-1,000,000 dpm, alpha
8.	Portable Air Samplers	0-8 cfm

1.7 Fixed Radiation Protection Equipment

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

- o Air samplers and monitors with moving and fixed filtering units. The monitors are capable of alarming at air concentrations equivalent to MPC's in less than four hours for most of the commonly encountered radioisotopes. Fixed filter units consist of 2-inch diameter Millipore filters and constant flow control regulators.
- o Fixed gamma monitors (Remote Area Monitors) that may be located in areas with potentially hazardous gamma fields.
- o Hand-and-shoe counters are provided at principal exit points for beta-gamma as required.
- o A whole body counter (shadow shield principle) utilizing a 5-inch by 5-inch NaI crystal is capable of detecting 0.01-0.1 of the maximum permissible body burden of several common gamma emitters and 1% for most other gamma emitters.
- o Environmental surveillance is provided by a number of TLD dosimeters located on the VNC site and at its perimeter. Four permanent environmental air sample stations also are located on site.

1.8 Protective Clothing

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

1.9 Film Badges and Pocket Dosimeters

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

1.10 Instrument Calibration

Routine calibrations of survey instruments are performed as follows:

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

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

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

All radiation monitoring instruments are calibrated as frequently as deemed necessary to assure reliability during use as based on some 30 years experience at the Vallecitos Nuclear Center. Portable radiation monitoring instruments are calibrated on an annual basis, before initial use, and after repair.

The reactor building stack particulate monitor system is calibrated by placing a uniformly distributed radioactive source in the same geometry as the filter paper used for collecting particulates.

1.11 Analytical Laboratory Counting Equipment and Capabilities

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

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

*Based on standard sample size and counting times.

1.12 Radiation Protection Design Features

Personnel performing work such as D^2T^2 activities at the GETR, other persons at the VNC site, and off-site personnel are protected against internal and external radiation exposures which could result from such activities. This protection is afforded by the reactor building, ventilation system, shielding structures, waste handling systems, radiation monitoring systems, and the use of radiation control standards and procedures.

The possibility of an event involving the dispersion of substantial quantities of radioactivity to the outside environment is essentially zero because of the lack of available source term, the nature of the planned D^2T^2 activities, and the radiation protection design features.

1.13 Radiation Safety Review

The design of new VNC facilities, or major changes in existing facilities and/or activities in which radioactive materials are to be handled, must be reviewed by the radiation safety function to insure that adequate health protection facilities are provided. Materials confinement, exposure reduction (both restricted activity and maintenance), design life, and assessments of potential hazards are considered in the design and planning phases; and, normally, preventive or mitigative systems are installed and tested prior to the completion of the independent review phase.

The facility manager is responsible for initiating requests for radiation safety review of proposed facilities and activities.

1.14 Reports and Records

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

1.15 Effluent Control

The allowable stack discharge dilution factor, i.e., the ratio of the effluent concentration in a stack to the concentration at the site boundary, for any given discharge stack on site is a function of, among other things, the actual annual meteorological conditions, the stack flow rate, and the distances from the stack to the site boundaries.

The method of calculating the allowable dilution factor is to:

(1) Collect and process the hourly site meteorological conditions for one or more years.

(2) Using the collected meteorological data, calculate and tabulate a dilution-dispersion factor, χ/Q, and wind direction for each hour at 16 equal sectors of the site boundary.

(3) Tabulate and analyze these data to determine a collection period average χ/Q value for each sector to the site boundary from the stack of interest. These factors represent the sector average air concentration ($\mu Ci/cc$) on the site boundary which would result from a continuous, steady release ($\mu Ci/sec$) from the stack. The largest χ/Q value from the 16 sectors then is selected as that stack's dilution-dispersion value.

(4) An accident condition dilution-dispersion factor, or short-term release, is determined by examination of all hourly χ/Q values for the year and picking the 95th percentile value.

Notwithstanding the above calculations, all effluent air from operations involving potential dust-producing activity is filtered; and, therefore, radioactive material concentrations have been and are expected to continue to be routinely well below permissible levels for unrestricted locations.

1.16 Environmental Program

Environmental surveys are made both within the VNC site and at off-site locations. These surveys involve samples of surface and ground water, soil, vegetation, and air and are used to verify compliance with the airborne concentration limits of 10CFR20.

2. Radioactive Waste Facilities

Industrial wastewaters which are not normally contaminated with radioactive materials by use at the reactor facility are piped to one of several 60,000-gallon retention basins near the site chemical treatment plant. Reactor facility sanitary wastes are transferred to the chemical treatment plant. After treatment, the sanitary waste effluent is sprinklered on GE property while the industrial waste is sampled, analyzed, and released off site. Existing facilities available for the storage and handling of reactor facility radioactive wastes are specified in the sections which follow.

2.1 Reactor Facility Tank Farm

The radioactive liquid waste storage system consists of storage tanks, filters, and resin beds to store liquid waste temporarily for later disposal, and to store and reprocess waters for continued use at the facility. The liquid waste system is located primarily in the tank farm (a fenced area approximately 40 feet by 80 feet) and an adjacent equipment building. The tank farm allows great flexibility, and water may be moved to any of the three underground hold tanks and pumped in a variety of ways for maximum processing capability, including transfer to the site liquid waste evaporator.

2.2 Site Waste Evaporator

Liquid wastes are routed from areas known to be or potentially contaminated through regulated pipe lines to the tank farm. Such wastes are transferred periodically to the radioactive liquid waste evaporator plant. In the direct evaporation process the liquids are pumped continuously from the feed tank or chemical treatment tanks directly into the evaporator. Here the wastes are concentrated through a vertical-tube, natural convection evaporator. The vapor is treated in associated equipment including a high-efficiency demister entrainment separator and a condenser. Effluent waters from the evaporator are collected in the monitoring tanks for analysis and disposal by evaporation. If further decontamination is necessary, the water can be re-routed to the feed storage tank for reprocessing or ion exchange treatment. Liquid waste concentrates from the evaporator are collected in a receiver and discharged into plastic-lined DOT Specification 17-H 55-gallon drums for solidification. After solidification, the drums are sealed and prepared for disposal as dry solid waste. This facility is regulated under the site's two materials licenses, SNM-960 and 0017-60, issued by the NRC and the State of California, respectively.

2.3 Hillside Storage Facility

Dry contaminated wastes are placed in sealable drums, tubes, boxes, or casks and transferred to the site radioactive material storage facility, Hillside Storage. The facility consists of two parts: (1) a vertical well facility for storage of 55-gallon drums of waste, and (2) a horizontal tube facility for storage of high level radioactive material contained in sealed encapsulations called "waste liners". The vertical well facility consists of 96 holes, four groups of 24, made by vertically stacking two 3-foot-long sections of 36-inch outside diameter concrete pipe with 3-inch-thick walls and with the top lip about 3 inches above grade. Each group of pipes is spaced in two separate rows of 12 each with about 1 foot between pipes and rows. Compacted earth filled in around the pipes keeps them firmly in place. Wooden lids and the 3-inch berms minimize entrance of water.

The horizontal tube facility is made of two rows of 40-foot-long concrete-lined steel pipes mounted horizontally and covered with earth. Eleven of the tubes have a 6-inch inside diameter, and seven have a 10-inch inside diameter. The tubes in either row are spaced on 3-foot centers, and the rows are spaced 3 feet apart with the tubes in the bottom row offset halfway between the tubes in the upper row. Shielding is provided on the top and sides of the facility by a minimum of 6 feet of compacted earth. Shielding at the front and back ends consists of 3 feet of concrete in which the pipes have been anchored, plus concrete-filled step plugs with a minimum of 3 feet of concrete shielding in the plug. Additional above-ground space for lower level waste or other materials is available within this fenced and posted facility. The Hillside Storage facility is regulated under licenses SNM-960 and 0017-60.

3. Industrial Safety Program

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

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

3.1 Industrial Safety Equipment

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

4. Vallecitos Emergency Plan

Plans for the prompt and rapid response to emergency situations are set forth in the Vallecitos Nuclear Center Radiological Emergency Plan. The plan is revised from time to time to reflect changes in equipment and organization at the site. The plan addresses emergencies which could arise from accidental criticality or release of radiation, as well as from fire, explosion, or earthquake, whether localized at a single facility or of site-wide significance.

4.1 Emergency Equipment

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

5. Conclusion

In view of the lack of materials which could be readily dispersed in such quantities as to present a problem at the site boundary and the site surveillance programs, the issuance of a possess but not operate license for the GETR reactor facility with provision for the conduct of restricted activities, such as D^2T^2 programs, would not result in any hazard to the public.