April 20, 2001

Mr. G. L. Stimmell, Manager
Vallecitos and Morris Operations
Vallecitos Nuclear Center
General Electric Company
6705 Vallecitos Road
Sunol, CA 94586

SUBJECT: ISSUANCE OF AMENDMENT NO. 21 TO FACILITY LICENSE NO. R-33 - GENERAL ELECTRIC NUCLEAR TEST REACTOR (TAC NO. MA0226)

Dear Mr. Stimmell:

The U.S. Nuclear Regulatory Commission has issued Amendment No. 21 for Facility License No. R-33. This amendment renews for 20 years from its date of issuance the General Electric Nuclear Test Reactor operating license. This is in response to the application for renewal dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000.

In accordance with our practice, we have restated the license in its entirety, incorporating all the changes and amendments made since issuance of the original license.

Enclosed with the amended license is a copy of the notice of renewal that is being sent to the Office of the Federal Register for publication, the Environmental Assessment, and the Safety Evaluation Report associated with the renewal.

Sincerely,

/RA/

Marvin M. Mendonca, Senior Project Manager
Events Assessment, Generic Communications and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-73

Enclosures:
1. Amendment No. 21
2. Notice of Renewal
3. Environmental Assessment
4. Safety Evaluation Report

cc w/enclosures:
Please see next page

cc:

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Service
P. O. Box 942732
Sacramento, CA 94234-7320

California Department of Health
ATTN.: Chief, Environmental Radiation
Control Unit
Radiological Health Section
714 P Street, Room 498
Sacramento, CA 95814

Mr. Chuck Bassett, Manager
Regulatory Compliance
Vallecitos Nuclear Center
General Electric Company
P.O. Box 460
Pleasanton, CA 94566

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-73

GENERAL ELECTRIC COMPANY

RENEWAL OF THE FACILITY OPERATING LICENSE FOR

THE GENERAL ELECTRIC NUCLEAR TEST REACTOR

Amendment No. 21
License No. R-33

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by the General Electric Company dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

   B. Construction of the General Electric Nuclear Test Reactor (the facility) was completed in substantial conformity with Construction Permit No. CPRR-19 dated October 24, 1957, the provision of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the pubic;

   H. The issuance of this license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of the byproduct, source, and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including Sections 30.33, 70.23 and 70.31.

2. Facility Operating License No. R-33 is hereby amended in its entirety to read as follows:

A. This license applies to the nuclear reactor designated by General Electric Company as the Nuclear Test Reactor (hereinafter the reactor or the NTR) which is owned by the General Electric Company and located at its Vallecitos Nuclear Center in Alameda County, California, and described in the application for license renewal dated September 30, 1997, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the General Electric Company (GE):

(1) Pursuant to Section 104c of the Atomic Energy Act of 1954, as amended (hereinafter the Act), and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the reactor as a utilization facility at the designated location in Alameda County, California, in accordance with the procedures and limitations described in the application and in this license, as amended.

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use in connection with the operation of the reactor:

a. 4 kilograms of contained U-235 as in-core reactor fuel;

b. 100 grams of plutonium for use in [including but not limited to] experimental devices, instrument check sources, and encapsulated fission foils;

c. 100 grams of uranium-233 for use in [including but not limited to] ionization chambers and experimental devices;

d. 700 grams of contained uranium-235 or 1500 grams of contained U-235 in uranium enriched to less than 4% U-235. This is not to be used as in-core fuel.

e. The limits in b.-d. above may include the types of materials authorized by Special Nuclear Material License SNM-960, as amended, Docket No. 70-754, and Reactor License TR-1, as amended, Docket No. 50-70, to be used in the reactor cell, south cell, north room, and control room, but not in experimental facilities of the NTR.

f. Such special nuclear material as may be produced by the operation of the reactor. The licensee is not authorized to separate this special nuclear material.

(3) Pursuant to the Act and Title 10, Chapter I Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” (a) to receive, possess and use 2,000 curies of either activated solids as contained in but not limited to such items as encapsulating materials, structural material and irradiated components or as contained materials; (b) any byproduct materials necessary for purposes of instrument calibration and startup sources; (c) 10 curies of tritium for pulsed neutron sources; and (d) to possess, but not to separate (except for byproduct material produced as allowed for experiments), such byproduct material as may be produced by the operation of the reactor.

(4) Pursuant to the Act and Title 10 CFR Part 40, “Domestic Licensing of Source Material,” to receive, possess and use 9.1 kg. of uranium and thorium as source material for experimental devices.

C. This license shall be deemed to contain and is subject to the conditions specified in Parts 20, 30, 40, 50, 51, 55, 70, and 73 of 10 CFR Chapter I, to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now, or hereafter in effect, and to the additional conditions specified below:

(1) Maximum Power Level

The licensee may operate the reactor at power levels not in excess of 100 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 21, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission’s approved physical security plan, including changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of the General Electric document, withheld from pubic disclosure pursuant to 10 CFR 2.790(d), entitled, “Security Plan for the Protection of Reactor Facilities,” submitted by letters dated October 13, 1992, as amended by letter dated September 28, 1994, April 25, and June 26, 1996, and April 16, 1998, under License R-33.

D. This license is effective as of the date of its issuance and shall expire 20 years from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications and
Non-Power Reactors Branch

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A, Technical
Specifications

Date of Issuance: April 20, 2001

NEDO-32765
Class 1
August 1997

TECHNICAL SPECIFICATIONS

FOR THE

GENERAL ELECTRIC NUCLEAR TEST REACTOR FACILITY LICENSE R-33

CONTENTS

TABLES

ILLUSTRATIONS

1.0 INTRODUCTION

1.1 PURPOSE

These Technical Specifications provide limits within which operation of the reactor will assure the health and safety of the public, the environment and on-site personnel. Areas addressed are Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features and Administrative Controls.

1.2 DEFINITIONS

1.2.1 Channel

The combination of sensors, lines, amplifiers and output devices which are connected for the purpose of measuring the value of a parameter.

1.2.2 Channel Calibration

A comparison and/or an adjustment of the channel so that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel if reasonable, including equipment actuation, alarm, or trip test and shall include the Channel Test.

1.2.3 Channel Check

A qualitative verification of acceptable performance by observation of channel behavior. This verification where possible shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.2.4 Channel Test

The introduction or interruption of a signal into the channel to verify that it is operable.

1.2.5 Experiment

Any operation, hardware or target (excluding devices such as detectors, foils, etc.), which is designed to investigate non-routine reactor characteristics or which is intended for irradiation in

an experiment facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.2.6 Experimental Facility

Any location for experiments which is on or against the external surfaces of the reactor main graphite pack, thermal column, or within any penetration thereof.

1.2.7 Explosive Material

Any chemical compound or mixture, the primary or common purpose of which is to function by an essentially instantaneous release of gas and heat.

1.2.8 Facility

That portion of the building and adjacent outside areas occupied by the reactor, reactor control room, and associated support areas.

1.2.9 Flammable

A flammable liquid is any liquid having a flash point under 100°F. A flammable solid is any solid material, other than one classified as an explosive, which is liable to cause fires through friction or which can be ignited easily and when ignited burns so vigorously and persistently as to create a serious hazard. Flammable solids include spontaneously combustible and water-reactive materials.

1.2.10 Licensed Operator

A person who is licensed as a reactor operator (RO) or senior reactor operator (SRO) pursuant to 10 CFR Part 55 to operate the controls of the Nuclear Test Reactor.

1.2.11 Limiting Conditions of Operation (LCO)

The lowest functional capability or performance levels of equipment required for safe operation of the facility.

1.2.12 Limiting Safety Systems Settings (LSSS)

Settings for automatic protective devices related to those variables having significant reactor safety functions.

1.2.13 Measured Value

The measured value of a parameter is the value as it appears at the output of a channel.

1.2.14 Operable

A system or component is operable when it is capable of performing its intended function.

1.2.15 Potential Excess Reactivity

That excess reactivity which can be added by the remote manipulation of control rods plus the maximum credible reactivity addition from primary coolant temperature change plus the reactivity worth of all installed experiments.

1.2.16 Reactivity Worth (Experiment)

The reactivity worth of an experiment is the maximum value of the reactivity change that would occur as a result of planned changes or credible malfunctions that alter experiment position or configuration.

1.2.17 Reactor Operating (Reactor Operation)

The reactor is considered to be operating when it is not secured or shut down (see 1.2.20 and 1.2.21).

1.2.18 Reactor Thermal Power

The reactor thermal power, as determined by a primary coolant system heat balance.

1.2.19 Reactor Safety Systems

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.2.20 Reactor Secured

The reactor is considered secured under either of the following two conditions:

1. The core contains insufficient fissile material to attain criticality under optimum conditions of moderation and reflection.

2. That overall condition where all of the following conditions are satisfied.

   a. Reactor is shut down.

   b. Console keylock switch is OFF and the console key is in proper custody.

   c. No work is in progress involving in-core components, installed rod drives, or experiments in an experimental facility.

1.2.21 Reactor Shutdown

That subcritical condition of the reactor where the negative reactivity of the Xenon-free core would be equal to or greater than the minimum shutdown margin and the reactivity worth of all experiments is limited in accordance with Specification 3.5.3.1.

1.2.22 Readily Available on Call (Senior Reactor Operator)

A senior reactor operator is readily available on call when all of the following conditions are satisfied:

a. Is within a reasonable driving time (½ hour) from the reactor facility.

b. Can be promptly contacted by telephone; and

c. Has informed the reactor operator on duty where he may be contacted.

1.2.23 Safety Limit (SL)

Limits upon important process variables which are found to be necessary to reasonably protect the reactor fuel.

1.2.24 Secured Experiment

Any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible natural phenomena or malfunctions.

1.2.25 Shutdown Margin

Shutdown margin shall mean the shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition, although the most reactive rod is stuck in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.2.26 Site

The area (approximately 1600 acres) within the confines of the Vallecitos Nuclear Center (VNC) owned and operated by General Electric.

1.2.27 True Value

The true value for a parameter is its actual value at any instant.

1.2.28 Unscheduled Shutdown

Any unplanned shutdown of the reactor caused by actuation of the scram channels, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation excluding shutdowns which occur during planned equipment testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS (SL)

2.1.1 Applicability

This specification applies to reactor thermal power level during either forced convection or natural circulation operation.

2.1.2 Objective

The objective of this specification is to specify a reactor power safety limit which provides the basis for the LSSS.

2.1.3 Specifications

The true value of the reactor thermal power shall not exceed 190 kW under any operating condition.

2.1.4 Basis

Safety Limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of the NTR fuel. The only accidents which could possibly cause fuel damage and a release of fission products from the NTR fuel are those resulting from large reactivity insertions. With the 0.76$ potential excess reactivity limit, a large reactivity insertion is not possible. Therefore, there is no mechanistic way of damaging the fuel and Safety Limits should not be required (refer to SAR, Sections 13.1 and 13.4.3).

The Code of Federal Regulations, however, requires a reactor to have Safety Limits. Therefore, a Safety Limit was chosen to restrict the ratio of the actual heat flux to the Departure from Nucleate Boiling (DNB) surface heat flux in the hottest fuel element coolant passage below l.5 to preclude any subsequent fuel damage due to a rise in surface temperature. Thermal-hydraulic analyses show that the DNB heat flux for the NTR is not significantly affected

by the core flow rate or the core inlet temperature. Reactor power is the only significant process variable that needs to be considered (refer to SAR, Section 13.7).

The safety limit for the reactor operating under steady-state or quasi steady-state conditions is 190 kW. A DNB ratio equal to 1.5 was selected as a conservatively safe operating condition for the steady- and quasi steady-state. The reactor thermal power level when the DNBR=1.5 is 190 kW (refer to SAR, Section 13.7).

Another Safety Limit under Reactor transient conditions is not required. Conservative transient analyses show that with the potential excess reactivity limit of 0.76$, fuel damage does not occur even if all scrams fail to insert the safety rods. Although the power level may safely attain 4000 kW during this transient event (refer to SAR, Section 13.7), the Safety Limit of 190 kW was conservatively selected to apply to the transient condition.

2.2 LIMITING SAFETY SYSTEM SETTINGS (LSSS)

2.2.1 Applicability

This specification applies to the scram set point for the linear neutron channels which monitor reactor power level.

2.2.2 Objective

The objective of this specification is to insure that automatic action will prevent the most severe postulated or anticipated transient from causing fuel damage.

2.2.3 Specification

The linear neutron power monitor channel set point shall not exceed the measured value of 125 kW.

2.2.4 Bases

Transient analyses presented in Subsection 13.4 of the SAR were performed assuming greater than 0.76$ maximum potential reactivity and an overpower scram set point at 150 kW. None of

the anticipated abnormal occurrences or postulated accidents resulted in fuel damage using these values. The LSSS of 125 kW is extremely conservative for the NTR.

3.0 LIMITING CONDITIONS FOR OPERATION (LCO)

3.1 REACTOR CORE PARAMETERS

3.1.1 Applicability

This specification applies to the reactivity condition of the reactor and to the reactivity worths of control rods, safety rods, manual poison sheets, experiments and the coolant temperature coefficient of reactivity.

3.1.2 Objective

The objective of this specification is to ensure the reactor can be safely controlled at all times and shut down when required.

3.1.3 Specifications

3.1.3.1

The reactor configuration shall be controlled to ensure that the potential excess reactivity shall be 0.76$. If it is determined that the potential excess reactivity is >0.76$, the reactor shall be shut down immediately. Corrective action shall be taken as required to ensure the potential excess reactivity is 0.76$.

3.1.3.2

The reactor shall be subcritical whenever the four safety rods are withdrawn from the core and the three control rods are fully inserted.

3.1.3.3

The minimum shutdown margin with the maximum worth safety rod stuck out shall be 1$.

3.1.3.4

Each manual poison sheet used to satisfy the requirements of Specification 3.1.3.1 shall be restrained in its respective graphite reflector slot in a manner which will prevent movement by more than ½ inch relative to the reactor core.

3.1.3.5

The temperature coefficient of reactivity of the reactor primary coolant shall be negative above a primary coolant temperature measured value of 124°F.

3.1.4 Bases

Operation in compliance with Specification 3.1.3.1 ensures that there would not be any mechanism for addition of reactivity greater than 0.76$. Detailed analyses have been made of reactivity insertions in the NTR Safety Analyses Report (SAR) Section 13. The analyses show that a reactivity step addition of 0.76$ will not cause significant fuel degradation.

Operation in accordance with Specification 3.1.3.2 ensures that criticality will not be achieved during safety rod withdrawal. Adherence to the 0.76$ limit also ensures that the reactor will not go critical during safety rod withdrawal.

Operation in accordance with Specification 3.1.3.3 ensures that the reactor can be brought and maintained subcritical without farther operator action under any permissible operating condition even with the most reactive safety rod stuck in its most reactive position.

Operation in accordance with Specification 3.1.3.4 ensures that the manual poison sheets will not be removed from the reactor core during the maximum postulated seismic event.

Operation in accordance with Specification 3.1.3.5 ensures there is no significant positive reactivity feedback from coolant temperature change during reactor power transients.

3.2 REACTOR CONTROL AND SAFETY SYSTEM

3.2.1 Applicability

This specification applies to the reactor safety rods, control rods and reactor safety systems.

3.2.2 Objective

The objective of this specification is to specify the lowest acceptable level of performance to reasonably ensure proper operation of the reactor safety rod, control rod and reactor safety systems.

3.2.3 Specifications

3.2.3.1

Reactor operation shall be permitted only when all safety and control rods are operable. The reactor shall be shut down immediately if it is known that a safety or control rod is not operable.

3.2.3.2

No more than one safety rod at a time shall be allowed to be moved in an outward direction.

3.2.3.3

The rate of withdrawal of each safety rod during reactor operation shall be less than 1-1/4 inches per second.

3.2.3.4

The rate of withdrawal of each control rod during reactor operation shall be less than 1/6 inch per second.

3.2.3.5

The average scram time (inflight time) of the four safety rods shall not exceed 300 msec.

3.2.3.6

Reactor operation shall be permitted only when the reactor safety system is operable in accordance with Tables 3-1 and 3-2.

The reactor shall be shut down immediately if any portion of the reactor safety system malfunctions, except as provided for in Tables 3-1 and 3-2.

3.2.4 Bases

Operation in accordance with Specification 3.2.3.1 ensures that during normal operation adequate shutdown margin is provided.

Operation in accordance with Specification 3.2.3.2 and specification 3.2.3.3 limits the rate of reactivity addition during safety rod withdrawal to that from one safety rod. This value is easily controlled by the operator.

Operation in accordance with Specification 3.2.3.4 limits the rate of reactivity addition during control rod withdrawal. Experience has shown that this is a value which is easily controlled manually by the operator. This rate is also less than the value analyzed in the rod withdrawal accident in the SAR.

Operation in accordance with Specification 3.2.3.5 ensures that the safety rod system performs satisfactorily. The specified time is approximately the inflight time originally established for this type reactor when higher potential excess reactivities were permitted. With the current limit on potential excess reactivity (see Technical Specification 3.1.3.1), a scram is not required during postulated events to prevent significant fuel degradation (see SAR, Section 13.4.3). Maintaining the safety rod system, then, is conservative.

Table 3-1

REACTOR SAFETY SYSTEM – SCRAM

Item No.	System	Condition	Trip Point*	Function
1.	Linear	High reactor power	No higher than 125 kW	Scram (2 out of 3 or 1 out of 2)
		Loss of positive high voltage to ion chambers (if ion chambers are used)	No less than 90% of operating voltage	Scram (2 out of 3 or 1 out of 2)
2.	Log N	Fast reactor period	No less than +5 sec	Scram
		Amplifier Mode switch not in operate	N/A	Scram
		Loss of positive high voltage to ion chambers (if ion chambers are used)	No less than 90% of operating voltage	Scram
3.	Primary Coolant Temperature	High core outlet temperature	No greater than 222ºF	Scram
4.	Primary Coolant Flow	Low Flow	No less than 15 gpm when reactor power is >0.1 kW	Scram
5.	Manual	Console button depressed	N/A	Scram
6.	Electrical Power	Reactor console key is off position (loss of ac power to the console)	N/A	Scram

*Trip points are the nominal measured values and need not take into account the uncertainty in the channel

Table 3-2
REACTOR SAFETY SYSTEM – FORMATION

Item No.	System	Condition	Set Point*	Function
1.	Reactor Cell Pressure	Low Differential pressure	>0.5 in. water ΔP	Visible and audible alarm; audible alarm may be bypassed after recognition.
2.	Fuel Loading Tank Water Level	Low Level	<3-ft below the overflow	Visible and audible alarm; audible alarm may be bypassed after recognition.
3.	Primary Coolant Temperature	High core outlet temperature	<200ºF	Visible and audible alarm; audible alarm may be bypassed after recognition.
4.	Primary Coolant Temperature	Core Delta temperature	N/A	Provide information for the heat balance determination.
5.	Stack Radioactivity	High Level	At a level to ensure compliance with Specs. 3.4.3.3 and 3.4.3.4	Visible and audible alarm; audible alarm be reset after recognition.
6.	Linear Power	Low Power indication	$\geq$2% on any scale	Safety or control rods cannot be withdrawn (2 out of 3 or 1 out of 2).
7.	Control or Safety Rod	Rods not in	N/A	Safety rod magnets cannot be reenergized.
8.	Safety Rod	Rods not out	N/A	Control rods cannot be withdrawn; safety rods must be withdrawn in sequence; may be bypassed to allow withdrawal of one control rod, or one safety rod (drive) out of sequence for purposes of inspection, maintenance and testing.

*Setpoint values are the nominal measured values and need not take into account the uncertainty of the channel

Operation in accordance with Specification 3.2.3.6 ensures that the reactor safety system is adequate to control operation of the facility, measure operating parameters, warn of abnormal conditions, and scram the reactor automatically if required.

The bases for items listed in Table 3-1 are as follows:

The linear high reactor power scram will be set no higher than the LSSS. Scram action as a result of a predetermined decrease of positive high voltage to ion chambers for the linear channels provides assurance that the high voltage power supply is functioning and the ion chambers are operating on a flat portion of the I-V curve.

The fast period scram limits the rate of rise of the reactor power to periods which are manually controllable. The Log N amplifier mode switch scram ensures that the Log N amplifier is in the Operate Mode. Scram action as a result of loss of positive high voltage to the ion chamber for the Log N channel provides assurance that the high voltage power supply is functioning and the ion chamber is operating on a flat portion of the I-V curve.

The primary coolant high core outlet temperature scram provides assurance that the reactor will be shut down if the primary coolant outlet temperature is high.

The primary coolant low-flow scram provides diversification in the safety system to ensure, when the reactor is at power levels which require forced cooling, that the reactor will be shut down if sufficient primary coolant flow is not maintained.

The manual console scram button provides a method for the reactor operator to manually shut down the reactor if an unsafe or abnormal condition should occur and the automatic reactor protection action as appropriate does not function. The loss of electrical power with the reactor console key in the off position (loss of ac power to the console) means that the reactor cannot be operated because ac power is no longer provided to the reactor safety system.

The bases for items listed in Table 3-2 are as follows:

The reactor cell low differential pressure alarm gives adequate assurance that operation of the reactor will be in compliance with specification 3.4.3.1.

The fuel loading tank low water level alarm gives adequate assurance that operation of the reactor will be in compliance with specification 3.3.3.1.

The primary coolant high core outlet temperature alarm gives adequate assurance that warning will be given to the operator of high primary coolant core outlet temperature.

The stack radioactivity high level alarm gives adequate assurance that operation of the reactor will be in compliance with specification 3.4.3.2.

The control rods “not in” interlock ensures that the reactor will be started up by withdrawing the four safety rods prior to withdrawing the control rods.

The safety rods “not-out” interlock ensures that the method of reactivity control is with the control rods during reactor operation.

3.3 REACTOR COOLANT SYSTEM

3.3.1 Applicability

This specification applies to the reactor primary coolant system.

3.3.2 Objective

The objective of this specification is to minimize the adverse effects on reactor components and to ensure the proper conditions of the coolant system for reactor operation.

3.3.3 Specifications

3.3.3.1

Above 0.l kW the reactor shall be cooled by light water forced coolant. At or below 0.1 kW forced coolant flow is not required.

3.3.3.2

Reactor operation shall not be permitted unless the core tank is filled with water. If during operation of the reactor it is determined or suspected that the core tank is not filled with water, the reactor will be shut down immediately and corrective action will be taken as required.

3.3.3.3

The specific conductivity of the primary coolant water shall be maintained less than 10 µmhos/cm except for time periods not exceeding 7 consecutive days when the specific conductivity may exceed 10 µmhos/cm but shall remain less than 20 µmhos/cm. If the specific conductivity exceeds 10 µmhos/cm, steps shall be taken to assure the specific conductivity is reduced to less than 10 µmhos/cm.

3.3.4 Bases

During a complete loss of primary coolant flow without a reactor scram, fuel damage does not occur (SAR, Section l 3.4.5). Natural convection cooling is sufficient. Requiring forced coolant flow above 0.l kW, then, is extremely conservative.

Operation in accordance with Specification 3.3.3.2 ensures that there will be no reactivity insertions due to the removal of voids or the sudden addition of water into the core tank during reactor operation.

The minimum corrosion rate for aluminum in water (< 50°C) occurs at a pH of 6.5. Maintaining water purity below 10 µmhos/cm will maintain the pH between 4.8 and 8.7. These values are acceptable for NTR operation. High specific conductivity can be tolerated for shorter durations

during unusual circumstances. Operation in accordance with Specification 3.3.3.3 ensures aluminum corrosion is within acceptable levels and that activation of impurities in the primary water remain below hazardous levels.

3.4 REACTOR CELL AND VENTILATION SYSTEM

3.4.1 Applicability

This specification applies to the reactor cell and ventilation system.

3.4.2 Objective

The objective of this specification is to ensure the release of airborne radioactive materials is below authorized limits.

3.4.3 Specifications

3.4.3.1

Reactor power shall not be increased above 0.1 kW unless the reactor cell is maintained at a negative pressure of not less than 0.5 in. of water with respect to the control room. If during operation of the reactor above 0.1 kW, the negative pressure with respect to the control room is not maintained, the reactor power shall be lowered to 0.1 kW immediately and corrective action shall be taken as required.

3.4.3.2

The limits for radioactive material discharged through the reactor ventilation system to the atmosphere shall be as specified in Table 3-3.

3.4.3.3

Alarm points for particulate and noble gas continuous monitors shall not exceed a value corresponding to the annual average release rate limit shown in Table 3-3.

Table 3-3

STACK RELEASE RATE LIMITS

Isotope Group	Annual Average
Halogen, > 8d T1/2	180 mCi/wk
Particulate, > 8d T1/2	
Beta-Gamma	870 microcuries/wk
Alpha	8.7 microcuries/wk
All other (including Noble Gas)	18 Ci/wk

3.4.3 4

During operation of the reactor above 0.1 kW or the performance of activities that could release radioactivity to the ventilation system, the stack particulate activity monitor and the gaseous activity monitor shall be operating.

If either the gas or particulate monitor is not operable, the reactor shall be shut down, or the activity involving releases shall be terminated, or the unit shall be promptly repaired or replaced with one of comparable monitoring capability. During this period, any indication of abnormal reactor operation shall be cause to shut down the reactor immediately.

3.4.4 Bases

Operation in accordance with Specification 3.4.3.1 and 3.4.3.2 ensures that potentially contaminated reactor cell air due to reactor operation is released and monitored through the ventilation system.

The ventilation system release limits in Specification 3.4.3.3 are based on the following:

The annual average dilution factor from the NTR stack to the site boundary based on 1976 and 1977 meteorological conditions and stack flow rate of 1,800 cu ft/min equals approximately 33,000. That is, the concentration at the site boundary from a continuous uniform release from the NTR stack will be $\leq$1/33,000 of the concentration at the stack when averaged over 1 year.

The above listed annual average limit contains a reduction factor of 2 to account for discharges from other VNC stacks.

The alarm points in Specification 3.4.3.3 are set for the annual average release rate limit of the most restrictive isotope in all categories which except noble gas uses the most probable isotope, Ar-41.

3.5 EXPERIMENTS

3.5.1 Applicability

This specification applies to reactor experiments.

3.5.2 Objective

The objective of this specification is to prevent an experiment from resulting in a hazard to the operating personnel or the general public or damage to the reactor.

3.5.3 Specifications

3.5.3.1

The reactivity worth of all experiments shall be limited so that the sum of the reactivity worths of all experiments performed at any one time shall be limited to comply with Specification 3.1.3.1.

3.5.3.2

The maximum amount of explosive material permitted in the NTR facilities is:

a. South Cell, $W \leq (D/2)^2$ with $W \leq 9$ lbs and $D \geq 3$ ft.

b. North room (without Modular Stone Monument), $W \leq D^2$ with $W \leq 16$ lbs and $D \geq 1$ft.

c. North Room (with Modular Stone Monument), $W \leq 2$ lbs in the MSM, 16 lbs in the north room.

d. Setup Room, $W \leq 25$ lbs.

where:

W = Total weight of explosives in pounds of equivalent TNT.

D = Distance in feet from the South Cell blast shield or the north face of the North Room wall.

3.5.3.3

Experimental objects shall not be allowed inside the core tank when the reactor is at a power greater than 0.1 kW.

3.5.3.4

Experimental objects located in the fuel loading chute shall be secured to prevent their entry into the core region during reactor operation.

3.5.3.5

A maximum of 10 Ci of radioactive material and up to 50 g of uranium may be in storage in a neutron radiography area where explosive devices are present (i.e., in the South Cell or North Room). The storage locations must be at least 1.5 m (5 ft) from any explosive device. Radioactive materials, other than those produced by the neutron radiography of the explosive devices and imaging systems, are not permitted in the Setup Room if explosive material is present.

Exception. Devices containing not more than 10 grams TNT equivalent of explosives with up to 200 mCi of tritium in the form of tritiated metal (hydride) are permitted. No more than one device may be in a neutron radiography area or the setup room at any one time, and no other explosive material may be in the same area at that time.

3.5.3.6

Unshielded high frequency generating equipment shall not be operated within 50 feet of any explosive devices.

3.5.3.7

Experimental capsules to be utilized in the experimental facilities shall be designed or tested to ensure that any pressure transient produced by chemical reaction of their contents and/or leakage of corrosion or flammable materials will not damage the reactor.

3.5.3.8

Experimental fuel elements containing plutonium to be utilized in the experimental facilities shall be clad and other experimental devices containing plutonium shall be encapsulated.

3.5.3.9

The maximum possible chemical energy release from the combustion of flammable substances contained in any experimental facility shall not exceed 1000 kW-sec. The total possible energy release from chemical combination or decomposition of substances contained in any experimental capsule shall be limited to 5 kW-sec, if the rate of the reaction in the capsule could exceed 1 W. Experimental facilities containing flammable materials shall be vented external to the reactor graphite pack.

3.5.3.10

A written description and analysis of the possible hazards involved for each type of experiment shall be evaluated and approved by the facility manager, or his designated alternate, before the experiment may be conducted.

3.5.3.11

No irradiation shall be performed which could credibly interfere with the scram action of the safety rods at any time during reactor operation.

3.5.3.12

The radioactive material content, including fission products, of any singly encapsulated experiment to be utilized in the experimental facilities shall be limited, so that the complete release of all gaseous, particulate, or volatile components from the encapsulation could not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR Part 20. This dose limit applies to persons occupying unrestricted areas continuously for 2 hours starting at time of release or restricted areas during the length of time required to evacuate the restricted area.

3.5.3.13

The radioactive material content, including fission products, of any doubly encapsulated or vented experiment to be utilized in the experimental facilities shall be limited so that the materials at risk from the encapsulation or confining boundary of the experiment could not result in a dose to any person occupying an unrestricted area continuously for a period of 2 hours starting at the time of release in excess of 0.5 rem to the whole body or 1.5 rem to the thyroid or a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 rem to the whole body or 30 rem to the thyroid.

3.5.4 Bases

Operation in accordance with Specification 3.5.3.1 ensures that there would not be any mechanism for addition of reactivity greater than 0.76$, including experiments. See the bases for Specification 3.1.3.1.

Specifications 3.5.3.1 through 3.5.3.11 are intended to reduce the likelihood of damage to the reactor components and/or radioactivity releases resulting from experiment failure and serve as a guide for the review and approval of new and untried experiments by the facility personnel.

Specifications 3.5.3.5 assures that any radiological effects in storage areas will not pose hazards to the public.

Specifications 3.5.3.12 and 3.5.3.13 ensure the radiological effects of experiment failures do not pose a hazard to the general public or to operating personnel.

4.0 SURVEILLANCE REQUIREMENTS

4.1 REACTIVITY LIMITS

4.1.1 Applicability

This specification applies to the surveillance requirements for reactivity limits.

4.1.2 Objective

To ensure that the reactivity limits of Specification 3.l are not exceeded.

4.1.3 Specification

4.1.3.1

Potential excess reactivity will be calculated before each startup. Actual critical rod position shall then be used to verify that the measured value is ≤0.76$.

4.1.3.2

Neutron multiplication will be observed throughout each startup. Safety rod withdrawal shall be stopped if it appears criticality will be reached before all safety rods are out.

4.1.3.3

The minimum shutdown margin shall be determined by calculation or measurement whenever a decrease in the reactivity worth of a safety rod is suspected.

4.1.3.4

Each manual poison sheet in the core region of the reactor shall be verified to be properly restrained upon insertion.

4.1.3.5

The temperature coefficient of reactivity of the reactor primary coolant shall be verified to be negative above 124 °F whenever changes made to the reactor could affect the temperature coefficient.

4.1.4 Bases

Operation in accordance with Specification 4.1.3.1 will ensure that the reactor is not operated with a potential excess reactivity of >0.76$.

Operation in accordance with Specification 4.1.3.2 will ensure that the reactor will be subcritical when all the safety rods are in the full-out position and the control rods are inserted.

Minimum shutdown margin is assured when the potential excess reactivity is limited to 76¢ and safety rod reactivity worths are unchanged. The shutdown margin, then, should be determined as specified in Specification 4.1.3.3 when changes to the reactor are made which could decrease the reactivity worth of a safety rod.

Verification that the manual poison sheets are properly restrained as specified in Specification 4.1.3.4 ensures that they cannot be ejected during any postulated natural phenomena or operational occurrence.

Compliance with Specification 4.1.3.5 ensures that the temperature coefficient is negative above 124°F. It is not affected by reactor configuration and fuel burnup and is therefore not expected to vary significantly with core life (but could be affected by fuel, core or moderator design changes).

4.2 REACTOR CONTROL AND SAFETY SYSTEM

4.2.1 Applicability

This specification applies to the surveillance requirements for the reactor control and reactor safety systems.

4.2.2 Objective

The objective of this specification is to specify the minimum surveillance requirements to reasonably ensure proper performance of the safety rod, control rod and safety systems.

4.2.3 Specifications

4.2.3.1

Each safety rod and control rod drive shall be tested for operability annually.

4.2.3.2

The interlock which restricts safety rod withdrawal to one rod at a time shall be tested annually.

4.2.3.3

The rate of withdrawal of each safety rod shall be measured annually.

4.2.3.4

The rate of withdrawal of each control rod shall be measured annually.

4.2.3.5

The safety rod scram time (inflight time) shall be measured semi-annually. The scram time (inflight time) shall additionally be measured after any work is performed which could affect the scram time or rod travel time.

4.2.3.6

Checks, tests and calibrations of the reactor safety system shall be performed as specified in Tables 4-1 and 4-2.

4.2.3.7

A thermal power verification shall be performed monthly when the reactor is operating above 50 kW.

Table 4-1

SURVEILLANCE REQUIREMENTS OF REACTOR SAFETY SYSTEM SCRAM INSTRUMENTS

Item No.	Item	Surveillance	Frequency*
1.	Linear System	Channel Check (neutron source check) Channel Test (high level trip test) Channel Check (comparison against a heat balance) Channel Calibration	Daily Daily Semi-annual Annually
2.	Log N System	Channel Test Channel Check Channel Calibration	Daily Monthly Annually
3.	Primary Coolant Temperature	Channel Test Channel Calibration	Daily Annually
4.	Primary Coolant Flow	Channel Check Channel Test Channel Calibration	Daily Daily Annually
5.	Manual	Channel Test	Daily
6.	Electrical Power	Channel Test	Daily

*Prior to placing into service an instrument which has been repaired, the instrument check, or test or calibration, as appropriate will be performed.

Table 4-2
SURVEILLANCE REQUIREMENTS OF REACTOR SAFETY SYSTEM INFORMATION INSTRUMENTS

Item No.	Item	Surveillance	Frequency*
1.	Reactor Cell Pressure	Channel Test	Quarterly
2.	Fuel Loading Tank Water Level	Channel Test	Quarterly
3.	Primary Coolant Temperature	Channel Test Channel Calibration	Quarterly Annually
4.	Primary Coolant Conductivity	Channel Check Channel Calibration	Quarterly Annually
5.	Primary Coolant Core Temperature	Channel Check Channel Calibration	Monthly Annually
6.	Reactor Cell Radiation Monitor	Channel Check Channel Test Channel Calibration	Daily Monthly Annually
7.	Stack Radioactivity (Gas and particulate channels)	Channel Check Channel Test Channel Calibration	Daily Monthly Annually
8.	Linear Power	Channel Test	Monthly

*Prior to placing into service an instrument which has been repaired, the instrument check, test or calibration, as appropriate, shall be performed.

4.2.4 Bases

Specification 4.2.3.1 ensures that each safety and control rod is maintained operable.

Specification 4.2.3.2 ensures that the safety rod interlock preventing the simultaneous withdrawal of more than one safety rod functions properly.

Specifications 4.2.3.3 and 4.2.3.4 ensure that the control and safety rod withdrawal rates are within limits.

Specification 4.2.3.5 provides for the periodic measurement of safety rod insertion times to ensure they are within limits.

Specification 4.2.3.6 ensures that the safety system is periodically tested and checked to maintain all instruments operable.

4.3 REACTOR COOLANT SYSTEM

Specifications regarding surveillance requirements of the reactor coolant system are included in the reactor safety system, Specification 4.2, Tables 4-1 and 4-2.

4.4 REACTOR CELL AND VENTILATION SYSTEM

4.4.1 Applicability

This specification applies to the surveillance requirements for the reactor cell and ventilation system.

4.4.2 Objective

The objective of this specification is to ensure that the reactor ventilation system is in satisfactory condition to provide adequate confinement and to control the release of radioactivity to the environment.

4.4.3 Specification

4.4.3.1

The reactor cell negative pressure, with respect to the control room, shall be verified prior to the first reactor startup of each day.

4.4.3.2

Surveillance requirements of the instrumentation and equipment required to comply with Specifications 3.4.3.2, 3.4.3.3 and 3.4.3.5 shall be as listed in Specification 4.2, Table 4-2.

4.4.4 Bases

Operation in accordance with Specification 4.4.3.l ensures that contaminated reactor cell air is exhausted through the ventilation system. This minimizes the possibility of airborne contamination release to surrounding areas.

Operation in accordance with Specification 4.4.3.2 ensures that all required channels are operational and that proper notification and surveillance will occur.

4.5 EXPERIMENTS

Specific surveillance activities shall be established during the review and approval process as specified in Section 6.2.3 “Review Function” and are not part of the Technical Specifications.

4.6 FREQUENCY OF TESTING

4.6.1 Applicability

This specification applies to all surveillance requirements in Section 4 of these Technical Specifications.

4.6.2 Objective

The objective of this specification is to establish maximum time intervals for surveillance periods. It is intended that this specification provides operational flexibility and not reduce surveillance frequency.

4.6.3 Specifications

4.6.3.1

Time intervals used elsewhere in these specifications shall be defined as follows:

a. Biennially – Interval not to exceed 30 months.

b. Annually – Interval not to exceed 15 months.

c. Semi-annual – Interval not to exceed 32 weeks.

d. Quarterly – Interval not to exceed 18 weeks.

e. Monthly – Interval not to exceed 6 weeks.

f. Weekly – Interval not to exceed 10 days.

g. Daily – Must be done prior to the first startup of the calendar day following a shutdown greater than 12 hours.

4.6.3.2

Surveillance tests (except those required for safety while the reactor is shut down) may be deferred during a reactor shutdown. Deferred surveillance tests must be completed prior to reactor startup.

4.6.3.3

Surveillance tests scheduled to occur during reactor operation, which cannot be performed with the reactor operating, may be deferred until the subsequent scheduled reactor shutdown.

4.6.4 Bases

Specification 4.6.3.1 establishes maximum time intervals for surveillance requirements which define the terms and makes them objectively quantifiable.

Specification 4.6.3.2 permits deferring tests which are not required if the reactor will not be operating.

Specification 4.6.3.3 permits deferring tests which might require a reactor shutdown for the sole purpose of performing the test.

5.0 DESIGN FEATURES

5.1 SITE AND FACILITY DESCRIPTION

5.1.1

The Nuclear Test Reactor (NTR) facility shall be located on the site of the Vallecitos Nuclear Center (VNC) which is owned and controlled by the General Electric Company.

5.1.2

The minimum distance from the reactor to the posted site boundary shall be approximately 488 meters (1600 feet). The restricted area, as defined in 10 CFR Part 20 of the Commission's regulations, shall be the Vallecitos Nuclear Center.

5.1.3

The fuel assemblies shall be positioned in a reel assembly inside the core tank. The core reel assembly shall be rotated only when the reactor is shut down and by manual operation of a crank inside the NTR cell.

5.1.4

The control system shall consist of four scrammable, spring-actuated safety rods, three nonscrammable control rods, and a number of manual poison sheets. When the poison rods and sheets are inserted, they shall be located in the graphite reflector at the outer periphery of the core tank. The safety and control rods shall be boron carbide clad in stainless steel. The manual poison sheets shall contain metallic cadmium.

5.1.5

The discharge of the gaseous effluent stack shall be approximately 14 meters (45 feet) above grade level of Building 105.

5.2 REACTOR PRIMARY COOLANT SYSTEM

The reactor coolant system shall be protected from overpressure by a vent line to the atmosphere of the cell.

5.3 REACTOR CORE AND FUEL

The core shall consist of 16 fuel element assemblies. Each fuel element assembly shall consist of 40 disks spaced on an aluminum support shaft. Other nominal specifications of the assemblies shall include the following:

a.	Fuel	23.5% by weight uranium - 76.5% by weight aluminum
b.	Enrichment	Approximately 93% U-235 (unburned)
c.	Cladding	Aluminum, 0.022 in. thickness
d.	Fuel disk active diameter	2.685 in.
e.	Fuel disk spacing on shaft	0.35 to 0.45 in., center-to-center

5.4 FISSIONABLE MATERIAL STORAGE

Fuel including fueled experiments and fuel devices not in the reactor shall be stored in a geometrical array where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION AND STAFFING

6.1.1 Structure

The NTR shall be owned and operated by the General Electric Company with management and operations organization as shown in Figure 6-l or equivalent.

6.1.2 Responsibilities

6.1.2.1

The Level 3 manager shall be responsible for the NTR facility license.

6.1.2.2

The Level 2 manager (Operations) is designated the facility manager and shall be responsible for the overall safe operation and maintenance of the facility.

6.1.2.3

The Level l manager (if utilized) is responsible for the routine safe operation and maintenance of the facility in accordance with the license, regulations and established written procedures. In the absence of this position, the Level l Reactor Supervisor or the Facility Manager shall assume the Level l manager responsibilities.

6.1.2.4

The Level I Reactor Supervisor (if utilized) is the individual responsible for supervising the daily operations. In the absence of this position, the Level I manager or the Facility Manager is responsible for supervising the daily operations.

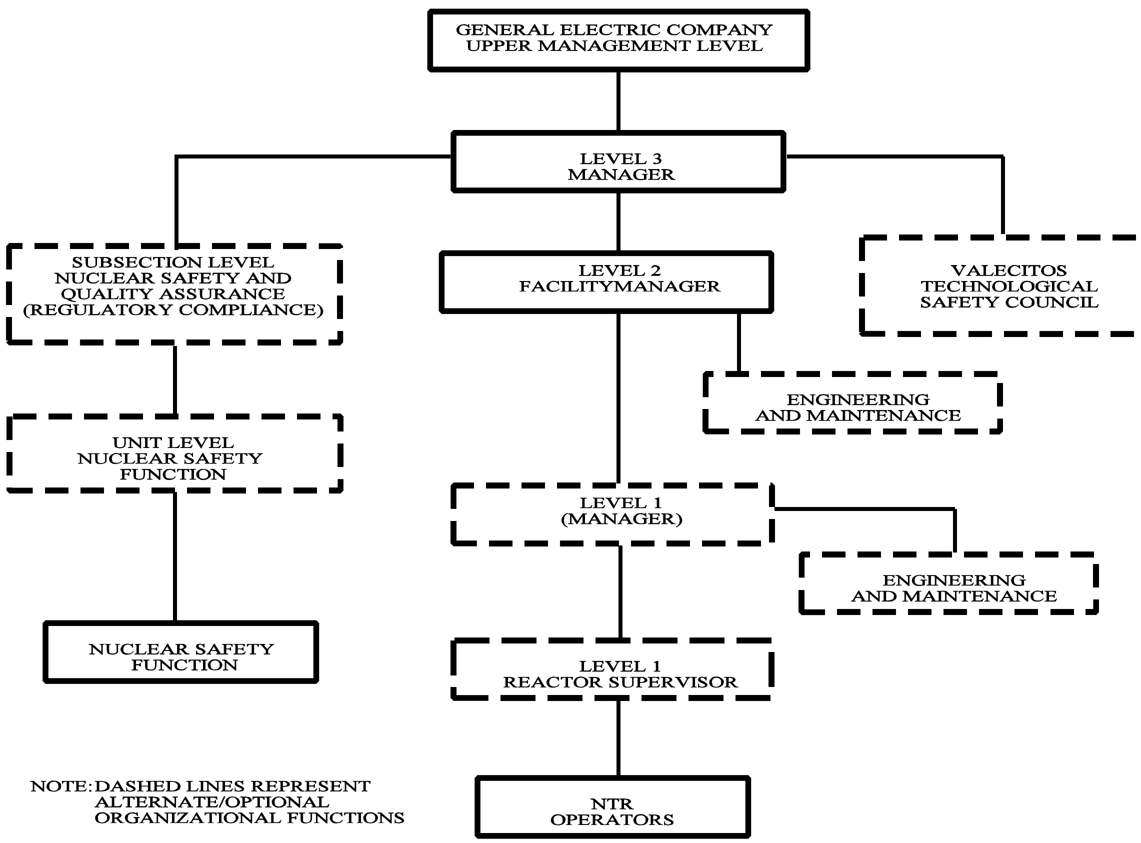

Figure 6-1. Facility Organization

6.1.2.5

Responsibilities of one level may be assumed by alternates when designated in writing.

6.1.2.6

Functions performed by one level may be performed by a higher level, provided the minimum qualifications are met (e.g., Senior Reactor Operator’s license).

6.1.3 Staffing

6.1.3.1

The minimum staffing when the reactor is not secured shall be composed of:

a. A licensed operator in the control room.

b. A second person present at the site familiar with NTR Emergency Procedures and capable of carrying out facility written procedures.

c. A licensed Senior Reactor Operator shall be present at the NTR Facility or readily available on call.

6.1.3.2

A licensed Senior Reactor Operator shall be present at the NTR Facility during the following events:

a. During the recovery from an unscheduled shutdown.

b. During reactor fuel loading or reactor fuel movement.

c. During any experiment or facility changes with a reactivity worth greater than one dollar.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS 15.4-1977, Sections 4 through 6; Title 10 of the Code of Federal Regulations, Part 55; and the latest revision of the Facility Operator Requalification Program.

6.2 INDEPENDENT REVIEWS

6.2.1

Independent reviews are performed by the cognizant Nuclear Safety Review Groups responsible to the Level 3 manager.

6.2.2

The independent review function shall be performed under a written charter or directive containing the following information as a minimum:

a. Subjects reviewed

b. Responsibilities

c. Authorities

d. Records

e. Other matters as may be appropriate

6.2.3

Activities requiring independent review shall include the following:

a. Proposed types of tests and experiments (or substantive changes thereto) including safety evaluations, that could affect core reactivity or result in an uncontrolled release of radioactivity, to be conducted without prior NRC approval, pursuant to 10 CFR 50.59,

to verify the proposed activity does not constitute a change in the Technical Specifications or an unreviewed safety question.

b. Proposed changes to the procedures or the facility, as described in the Safety Analysis Report, including safety evaluations, to be completed without prior NRC approval, pursuant to 10 CFR 50.59, to verify the activity does not constitute a change in the Technical Specifications or an unreviewed safety question.

c. All new procedures and revisions thereto having safety significance required by the specifications in Section 6.3.

d. Proposed changes to the Technical Specifications or the facility operating license.

e. Violations of the Federal Regulations, Technical Specifications, and facility license requirements.

f. Unusual or abnormal occurrences which are reportable to the NRC under provisions of the Federal Regulations or the Specifications in Section 6.6.

g. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect, or could affect, nuclear safety.

6.2.4

Independent periodic examination and verification shall be performed of facility operations, maintenance and administration. These periodic examinations and verifications shall be performed by staff that do not have direct responsibility for the safe operation of the reactor.

6.3 RADIATION SAFETY

6.3.1

The radiation safety program must achieve the requirements of 10 CFR Part 20.

6.3.2

Safety is foremost at the facility. Regulatory compliance personnel have the authority to intercede and suspend activities which could involve or result in radiologically hazardous situations.

6.3.3

The ALARA program shall be applied to all facility staff, facility users, visitors, the public and the environment.

6.4 PROCEDURES

6.4.1

Written procedures shall be prepared for the following activities as required:

a. Startup, operation, and shutdown of the reactor.

b. Defueling, refueling, and fuel transfer operations, when required.

c. Preventive or corrective maintenance which could have an effect on the safety of the reactor.

d. Off-normal conditions relative to reactor safety for which an alarm is received.

e. Response to abnormal reactivity changes.

f. Surveillance testing, and calibrations required by the Technical Specifications.

g. Emergency conditions involving potential or actual release of radioactive materials.

h. Radiation protection consistent with 10 CFR Part 20 requirements.

i. Review and approval of changes to all required procedures.

j. Security plan, the operator requalification program, and emergency procedures.

k. Operation and maintenance of experiments that could affect reactor safety or core reactivity.

6.4.2

The facility manager shall approve all procedures (including revisions) required by Specification 6.4.1 before implementation.

6.4.3

Minor changes to the original procedures which do not change their original intent may be made by the Level 1 Reactor Supervisor or Level I manager. These changes must be subsequently approved by the facility manager.

6.4.4

Temporary deviations from established procedures may be made by a Licensed Senior Reactor Operator in order to deal with special or unusual circumstances. These deviations shall be documented and reported to the facility manager.

6.5 REQUIRED ACTIONS

6.5.1 Action to be taken in the event of an occurrence of the type identified in Section 6.5.2.

1. Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the facility manager.

2. Occurrence shall be reported to the facility manager and to the NRC addressed in accordance with 10 CFR 50.73(d).

3. Occurrence shall be reviewed by Regulatory Compliance.

6.5.2 Action to be Taken in Case of Safety Limit Violation

1. The reactor shall be shut down, and reactor operations shall not be resumed until authorized by Level 3 management.

2. The safety limit violation shall be promptly reported to the facility manager.

3. The safety limit violation shall be reported to the NRC.

4. A safety limit violation report shall be prepared. The report shall describe the following:

    a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

    b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

    c. Corrective action to be taken to prevent recurrence.

The report shall be reviewed by Regulatory Compliance and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6 REPORTS

6.6.1 Operating Reports

Annual operating report(s) shall be submitted to the NRC Document Control Desk. The report(s) shall include the following:

a. A narrative summary of reactor operating experience including the hours the reactor was critical and total energy produced.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. A summary report in accordance with 10 CFR 50.59.

e. A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge.

f. Summarized results of environmental surveys performed outside the facility.

g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

6.6.2 Special Reports

Special reports are used to report unplanned events as well as planned major facility and administrative changes. The following special reports shall be forwarded to the NRC addressed in accordance with 10 CFR 50.73(d):

a. There shall be a report not later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance, to be followed by a written report within 14 days, that describes the circumstances of any of the following events:

   1. Release of radioactivity from the site above allowed limits.

   2. Any of the following:

      – Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the technical specifications.

      – Operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken.

      – A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the technical specifications, the

failure of the extra components or systems or not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

– An unanticipated or uncontrolled change in reactivity greater than 0.50$.

– Abnormal and significant degradation in reactor fuel, cladding, or coolant boundary, which could result in exceeding prescribed radiation limits for personnel or the environment.

– An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

b. A written report within 30 days to the NRC for the following:

1. Permanent changes in the facility organization involving Level 2 or Level 3 management.

2. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7 RECORDS

Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single, or multiple records, or a combination thereof.

6.7.1

Records to be retained for a period of at least five years or for the life of the Component, whichever is less:

a. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc., shall be maintained for a period of at least one year).

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by the Technical Specifications.

e. Reactor facility radiation and contamination surveys where required by applicable regulations.

f. Experiments performed with the reactor.

g. Fuel inventories, receipts, and shipments.

h. Approved changes in operating procedures.

i. Records of meeting reports of the review groups.

6.7.2

Records of the requalification programs shall be maintained in accordance with 10 CFR 55.59(c)(5).

6.7.3

Records to be Retained for the Lifetime of the Reactor Facility.

(Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

a. Gaseous and liquid radioactive effluents released to the environs.

b. Radiation exposure for all personnel monitored.

c. Drawings of the reactor facility.

UNITED STATES NUCLEAR REGULATORY COMMISSION

NOTICE OF RENEWAL OF FACILITY OPERATING LICENSE FOR

THE GENERAL ELECTRIC NUCLEAR TEST REACTOR

DOCKET NO. 50-73

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Facility Operating License No. R-33 issued to the General Electric Company (the licensee). This license amendment renews the license for the General Electric Nuclear Test Reactor in Sunol, California. The facility is a non-power reactor that has been operating at power levels not in excess of 100 kilowatts (thermal). The renewed Operating License will expire 20 years from the date of amendment issuance.

Opportunity for hearing on this amendment was afforded in the notice of the proposed issuance of this renewal in the Federal Register on September 22, 1999, at 64 FR 51340. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The amended license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I. Those findings are set forth in the license amendment.

The Commission has prepared a safety evaluation for the renewal of the license, which is attached to the amendment. The Commission has, based on that safety evaluation, concluded that the facility can continue to be operated by the licensee without endangering the health and safety of the public.

The Commission has also prepared an environmental assessment and finding of no significant impact for the renewal of the license. The environmental assessment and finding of no significant impact was published in the Federal Register on April 20, 2001, at 66 FR 20339. Continued operation of the reactor will not require alteration of buildings or structures, will not lead to significant changes in effluents released from the facility to the environment, will not increase the probability or consequences of accidents, and will not involve any unresolved issues concerning alternative uses of available resources. The Commission has concluded that this action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see (1) the application for amendment dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000, (2) the notice of the proposed issuance, (3) Amendment No. 21 to Facility Operating License R-33, (4) the Commission's related safety evaluation report, and (5) the environmental assessment and finding of no significant impact. These documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will also be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this April 20, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications, and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs

Office of Nuclear Reactor Regulation

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

GENERAL ELECTRIC COMPANY

DOCKET NO. 50-73

NUCLEAR TEST REACTOR

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an amendment to Facility Operating License No. R-33, issued to the General Electric Company (the licensee or GE) for operation of the General Electric Nuclear Test Reactor (NTR or the facility) located in Sunol, California.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action

The proposed action would renew the license for the NTR for 20 years from the date of issuance of the license amendment. The proposed action is in accordance with the licensee's application for amendment dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000. The licensee submitted an Environmental Report for license renewal.

Need for the Proposed Action

The proposed action is needed to allow continued operation of the NTR beyond the current term of the license in order to continue research and development using neutrons for experimental purposes.

Environmental Impact of the Proposed Action

The NTR is in Building 105 within the approximately 1600 acre (6.4 square kilometers) Vallecitos Nuclear Center (VNC) near Pleasanton, California. GE owns the VNC site for nuclear research and development. GE normally leases about 1500 acres (6.1 km^2) of the site for grazing and for cattle feed crops. The land surrounding the site is primarily used for agriculture and cattle raising. Building 105 has laboratories, offices and workshops and is surrounded by similar facilities in the immediate area.

On October 24, 1957, the U. S. Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-19, to GE. This permit authorized GE to construct the NTR at its VNC site in Southern Alameda County, California. On October 31, 1957, the AEC issued Facility Operating License No. R-33, authorizing GE to operate the reactor at steady-state power levels up to 30 kW(t). The reactor first reached criticality on November 15, 1957. On July 22, 1969, the license was amended authorizing GE to operate the reactor at steady-state power levels not in excess of 100 kW(t), and renewing the license. The facility license was renewed again on December 28, 1984, with an expiration date of October 31, 1997. The licensee applied for renewal on September 30, 1997, and, in accordance with 10 CFR 2.109, the license remains in effect. At each renewal, the facility description, organization and safety evaluation were updated. The reactor has operated about 139 megawatt-days for the first 39 years since initial licensing (Safety Analysis Report (SAR) section 4.4.1). Facility modifications have been minor. The licensee has not indicated any plans to change the design or usage significantly. The radioactive releases from the NTR have been well within regulatory limits of 10 CFR Part 20. The facility typically has 1 liter per year of radioactive liquid waste (SAR section 11.1.1.2) that is due to sampling. This liquid waste is transferred to monitored tanks. Solid radioactive releases are estimated to be less than 3 cubic feet or 0.085 cubic meters per year (SAR section

11.1.1.3). The radioactive content of this waste is measured in the millicurie or 10^8 becquerels range. Solid waste is transferred to separate State and NRC licenses held by the GE. Liquid and solid radioactive material has been transferred and disposed of in accordance with the requirements of the licensee's byproduct license. Any necessary releases will be similarly treated. Currently, the licensee has no plans to change any operating or radioactive release practices or characteristics of the reactor during the license renewal period.

The NRC concludes that conditions are not expected to change and that the radiological effects of the continued operation will continue to be minimal. The radiological exposures for facility operations have been and are expected to remain within regulatory limits.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released off site, and there is no significant increase to occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential non radiological impacts, the proposed action does not involve any historic sites. It does not affect non radiological facility effluents and has no other environmental impact. Therefore, there are no significant non radiological environmental impacts associated with the proposed action.

In addition, the environmental impact associated with operation of research reactors has been generically evaluated by the staff and is discussed in the attached generic evaluation. This evaluation concludes that no significant environmental impact is associated with the operation of research reactors licensed to operate at power levels up to and including 2 megawatts thermal. The NRC staff has determined that this generic evaluation is applicable

to operation of the NTR and that there are no special or unique features that would preclude reliance on the generic evaluation.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the "no-action" alternative). If the NRC denied license renewal, NTR operations would stop and decommissioning would be required with no significant benefit to the environment. The environmental impacts of the proposed action and alternative are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the safety analysis and evaluation for operating license renewal in 1984 and the "Environmental Assessment for the General Electric Company - Nuclear Test Reactor License No. R-33 Docket No. 50-73," dated November 9, 1984.

Agencies and Persons Contacted

On October 24 and 27, 2000, the staff consulted with the California Department of Health Official, Steve Hsu, regarding the environmental impact of the proposed action. The State official had no comment.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will also be accessible electronically from the ADAMS

Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 13th day of April, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications, and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Attachment

ENVIRONMENTAL CONSIDERATIONS REGARDING THE LICENSING OF RESEARCH REACTORS AND CRITICAL FACILITIES

Introduction

This discussion deals with research reactors and critical facilities which are designed to operate at low power levels, 2 MWt and lower, and are used primarily for basic research in neutron physics, neutron radiography, isotope production, experiments associated with nuclear engineering, training and as a part of a nuclear physics curriculum. Operation of such facilities will generally not exceed a 5-day week, 8-hour day, or about 2000 hours per year. Such reactors are located adjacent to technical service support facilities with convenient access for students and faculty.

Sited most frequently on the campuses of large universities, the reactors are usually housed in already existing structures, appropriately modified, or placed in new buildings that are designed and constructed to blend in with existing facilities. However, the environmental considerations discussed herein are not limited to those which are part of universities.

Facility

There are no exterior conduits, pipelines, electrical or mechanical structures or transmission lines attached to or adjacent to the facility other than for utility services, which are similar to those required in other similar facilities, specifically laboratories. Heat dissipation is generally accomplished by use of a cooling tower located on the roof of the building. These cooling towers typically are on the order of 10' x 10' x 10' and are comparable to cooling towers associated with the air-conditioning systems of large office buildings.

Make-up for the cooling system is readily available and usually obtained from the local water supply. Radioactive gaseous effluents are limited to Ar-41 and the release of radioactive liquid effluents can be carefully monitored and controlled. Liquid wastes are collected in storage tanks to allow for decay and monitoring prior to dilution and release to the sanitary sewer system. Solid radioactive wastes are packaged and shipped offsite for storage at NRC-approved sites. The transportation of such waste is done in accordance with existing NRC-DOT regulations in approved shipping containers.

Chemical and sanitary waste systems are similar to those existing at other similar laboratories and buildings.

Environmental Effects of Site Preparation and Facility Construction

Construction of such facilities invariably occurs in areas that have already been disturbed by other building construction and, in some cases, solely within an already existing building. Therefore, construction would not be expected to have any significant effect on the terrain, vegetation, wildlife or nearby waters or aquatic life. The societal, economic and aesthetic impacts of construction would be no greater than those associated with the construction of a large office building or similar research facility.

Environmental Effects of Facility Operation

Release of thermal effluents from a reactor of less than 2 MWt will not have a significant effect on the environment. This small amount of waste heat is generally rejected to the atmosphere by means of small cooling towers. Extensive drift and/or fog will not occur at this low power level.

Release of routine gaseous effluents can be limited to Ar-41, which is generated by neutron activation of air. Even this will be kept as low as practicable by using gases other than air for supporting experiments. Yearly doses to unrestricted areas will be at or below established guidelines in 10 CFR Part 20 limits. Routine releases of radioactive liquid effluents can be carefully monitored and controlled in a manner that will ensure compliance with current standards. Solid radioactive wastes will be shipped to an authorized disposal site in approved containers. These wastes should not require more than a few shipping containers a year.

Based on experience with other research reactors, specifically TRIGA reactors operating in the 1 to 2 MWt range, the annual release of gaseous and liquid effluents to unrestricted areas should be less than 30 curies and 0.01 curies, respectively.

No release of potentially harmful chemical substances will occur during normal operation. Small amounts of chemicals and/or high-solid content water may be released from the facility through the sanitary sewer during periodic blowdown of the cooling tower or from laboratory experiments.

Other potential effects of the facility, such as aesthetics, noise, societal or impact on local flora and fauna are expected to be too small to measure.

Environmental Effects of Accidents

Accidents ranging from the failure of experiments up to the largest core damage and fission product release considered possible result in doses that are less than 10 CFR Part 20 guidelines and are considered negligible with respect to the environment.

Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation involve the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of these unavoidable effects.

Alternatives to Construction and Operation of the Facility

To accomplish the objectives associated with research reactors, there are no suitable alternatives. Some of these objectives are training of students in the operation of reactors, production of radioisotopes, and use of neutron and gamma ray beams to conduct experiments.

Long-Term Effects of Facility Construction and Operation

The long-term effects of research facilities are considered to be beneficial as a result of the contribution to scientific knowledge and training. Because of the relatively small amount of capital resources involved and the small impact on the environment, very little irreversible and irretrievable commitment is associated with such facilities.

Costs and Benefits of Facility Alternatives

The costs are on the order of several millions of dollars with very little environmental impact. The benefits include, but are not limited to, some combination of the following: conduct of activation analyses, conduct of neutron radiography, training of operating personnel, and education of students. Some of these activities could be conducted using particle accelerators or radioactive sources which would be more costly and less efficient. There is no reasonable alternative to a nuclear research reactor for conducting this spectrum of activities.

Conclusion

The staff concludes that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 MWt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO

FACILITY OPERATING LICENSE NO. R-33 FOR

THE GENERAL ELECTRIC COMPANY NUCLEAR TEST REACTOR

DOCKET NO. 50-73

1.0 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated September 30, 1997, as supplemented on June 18, 1999, August 23, 1999, June 1, 2000, and October 5, 2000, the General Electric Company (GE or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a timely application for a renewal of the Class 104c Facility Operating License R-33 (NRC Docket No. 50-73). Such a renewal would authorize continued operation for an additional 20 years from the date of issuance for the General Electric Nuclear Test Reactor (NTR) at the Vallecitos Nuclear Center (VNC) Site in Alameda County, California. Until the NRC staff completes action on the renewal request, the licensee may operate the NTR under the conditions authorized in accordance with Title 10, Part 2, section 109, U.S. Code of Federal Regulations (10 CFR Part 2.109).

The NRC staff reviewed the licensee's renewal application, supporting supplemental information, and responses to NRC staff requests for additional information (RAIs). The renewal application included financial statements, the safety analysis report (SAR), an Environmental Report and Technical Specifications (TS). The NRC staff evaluated the facility against the applicable requirements, NRC Regulatory Guides (Reg Guides), and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series.

The purpose of this safety evaluation (SE) is to summarize the findings of the NRC staff's radiological safety review for continued operation of the NTR research reactor. This SE will be the basis for renewing the NTR license for operation at thermal power levels up to and including 100 kilowatts (kW).

In this SE, the NRC staff evaluates the following topics:

♦ Section 1 contains a summary and conclusions regarding the principal safety considerations of the NRC staff review. It also contains the history and a general description of the reactor

facility, information on shared facilities and equipment, comparisons with similar facilities, and how the licensee complies with the Nuclear Waste Policy Act of 1982.

- ♦ Section 2 evaluates the site and applicable site characteristics, including geography, demography, meteorology, hydrology, geology, seismology and interaction with nearby installations and facilities.

- ♦ Section 3 considers the design basis of facility structures, systems and components and the responses to environmental factors on the reactor site.

- ♦ Section 4 evaluates prior facility use focusing on reactor fuel, safety rods and control rods.

- ♦ Section 5 assesses the design bases and the functional characteristics of the reactor core and its components. In this section, the safety considerations and the features of the reactor are discussed.

- ♦ Section 6 considers the design bases and the function of the reactor coolant and associated systems, including the primary and secondary coolant systems, and coolant makeup and purification systems.

- ♦ Section 7 evaluates the design bases and the function of engineered safety features (ESFs) that may be required to mitigate the consequences of postulated accidents at the facility.

- ♦ Section 8 assesses the design bases and the function of the instrumentation and control (I&C) systems and subsystems of the facility. This section emphasizes consideration of safety-related systems and safe reactor shutdowns.

- ♦ Section 9 evaluates the design bases and the functions of the normal and emergency electrical power systems at the facility.

- ♦ Section 10 considers the design bases and the functions of the auxiliary systems, such as fuel handling and storage, compressed air, warning and communications and fire protection.

- ♦ Section 11 assesses the design bases and the functions of the experimental facilities. The NTR is designed with irradiation capabilities for research and development. This section discusses the characteristics of experiment and irradiation facilities, and experimental programs.

- ♦ Section 12 considers the design bases and the functions of the radiation protection and the radioactive waste management programs at the facility. The description of the radiation protection program includes health physics staffing and procedures, monitoring programs for personnel exposure and effluent releases, and assessment and control of radiation doses both for workers and the public. The facility program to maintain radiation exposures and releases as low as reasonably achievable (ALARA) is discussed in this section. The program for radioactive waste management is addressed including the control and disposal of radiological waste from both reactor operations and experimental programs.

- ♦ Section 13 evaluates the bases and the function of plans and procedures for the conduct of facility operations. These include discussions of the management structure, personnel training and evaluation, and provisions for safety review and auditing of operations by the safety committee. Section 13 also includes other required functions, such as reporting, security, and emergency preparedness.

- ♦ Section 14 considers the bases, scenarios and analyses of accidents at the reactor facility including accidents associated with experiments. The radiological consequences from analyzed accidents to the facility staff and members of the public are assessed.

- ♦ Section 15 evaluates the TSs, which specify operating limits, conditions and other requirements for the facility to ensure acceptable protection of the health and safety of the public and environment.

- ♦ Section 16 concerns financial qualifications of the licensee for continuing operations and decommissioning.

- ♦ Section 17 contains the major conclusions of the NRC staff review of the NTR renewal application.

This SE was prepared by Mr. Marvin M. Mendonca, of the NRC staff, and James R. Miller, A. Francis DiMeglio and Daniel E. Hughes of the Idaho National Engineering and Environmental Laboratory (INEEL) under contract to the NRC.

1.2 Summary and Conclusions Regarding the Principal Safety Considerations

In its evaluation, the NRC staff considered the information submitted by the licensee (including past operating history recorded in the licensee's annual reports to the NRC), as well as inspection reports prepared by the NRC personnel and first-hand onsite observations. From this evaluation, the NRC staff reached the following findings:

(1) The design, testing and performance of the NTR structures, systems, and components important to safety during normal operation were acceptable. Safe operation of the facility can reasonably be expected to continue.

(2) The licensee's management organization, training and research activities and security measures continue to be acceptable to ensure safe operation of the facility and the protection of its special nuclear material.

(3) The licensee and NRC staff have conservatively considered the expected consequences of postulated accidents. As a result, the NRC staff determined that the calculated potential radiation doses meet regulatory requirements and are, therefore, acceptable.

(4) Exposures from and releases of radioactive materials and wastes from the facility are not expected to result in concentrations beyond the limits specified by the Commission's regulations and are consistent with ALARA principles.

(5) The license and TSs, which state limits controlling operation of the facility, give a high degree of assurance that the licensee will operate the facility in accordance with the assumptions and analyses in the SAR. There has been no significant degradation of equipment, and the license and TSs will continue to ensure that the licensee acceptably addresses potential problems.

(6) The financial data submitted with the application demonstrate that the licensee has acceptable access to sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

(7) The licensee's procedures for training its reactor operators and the plans for operator requalification give assurance that the reactor will continue to be operated acceptably.

(8) The licensee's emergency plan provides acceptable assurance that the licensee continues to be prepared to assess and respond to emergency events.

(9) Continued operation of the NTR poses no significant radiological risk to the facility's employees, the public or the environment.

On the basis of these findings, the NRC staff concludes that GE may continue to operate the NTR in accordance with its application, without endangering the health and safety of the public.

1.3 History

On October 24, 1957, the U. S. Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-19, to GE. This permit authorized GE to construct the NTR at its VNC site in Southern Alameda County, California. On October 31, 1957, the AEC issued Facility Operating License No. R-33, authorizing GE to operate the reactor at steady-state power levels up to 30 kW(t). The reactor first reached criticality on November 15, 1957. On July 22, 1969, the license was amended authorizing GE to operate the reactor at steady state power levels not in excess of 100 kW(t).

The facility license has been renewed several times since its issuance on October 31, 1957. At each renewal, the facility description, organization and safety evaluation were updated. The last such renewal was dated December 28, 1986.

Since the December 28, 1986, license renewal, the licensee has made no significant changes to the facility. One license change was approved August 19, 1992, concerning the by-product material limit. One Technical Specification change was approved on June 2, 1989, concerning the separation of explosives and radioactive materials.

1.4 Reactor Description

The NTR is a heterogeneous, tank type reactor. The core contains enriched uranium fuel that is graphite moderated and reflected. The core is cooled either by natural or forced flow of water circulated in a primary system constructed primarily of aluminum. The reactor coolant flows through an external heat removal and purification system. The reactor's experimental systems

include a central sample tube, penetrations through and into the reflector, the reactor surfaces and neutron beams and tubes from any of these facilities.

The NTR fuel is high enriched, uranium-aluminum alloy disks, clad with aluminum. The reactor exhibits a negative void and temperature coefficient of reactivity above 124°F (51°C), which is approximately the steady state operating temperature. Reactivity is controlled by up to six manually positioned cadmium sheets, four safety rods filled with boron carbide and three control rods, also filled with boron carbide.

1.5 Shared Facilities and Equipment

The NTR shares Building Number 105 at the VNC with other laboratories. It also shares many facilities and equipment including potable water, fire protection, emergency supplies and support, heating, ventilation and air-conditioning systems, electrical distribution, and compressed air. Walls to delineate the NTR facility from other laboratories separate the shared building spaces. Other separations are installed to isolate the shared facilities and equipment.

1.6 Comparison with Similar Facilities

The design of the NTR resulted from the evolution of a series of reactors designed by the scientists at the GE Knolls Atomic Power Laboratory (KAPL) and operated at several locations in the United States. Three similar reactors were built and operated at varying times from 1953 to the mid-eighties. The instrumentation and controls (I & C) are similar in principle to those at most non-power reactors.

1.7 Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research reactor, that the applicant shall have reached an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level wastes and spent nuclear fuel. In a letter dated July 13, 1983, Thomas S. Keefe of the DOE informed R. W. Darmitzel of GE that Contract Number DE-CRO1-83NE44426 had been executed. By entering into such a contract with the DOE, GE has satisfied the requirements of the Waste Policy Act of 1982 as they apply to the NTR.

2.0 SITE CHARACTERISTICS

2.1 Reactor Site

The NTR is in Building 105 within the 1590 acre (6.4 square kilometers (km^2)) VNC near Pleasanton, California. GE owns the VNC site for nuclear research and development. The licensee normally leases about 1500 acres (6.1 km^2) of the site for grazing and for cattle feed crops. The land surrounding the site is primarily used for agriculture and cattle raising.

The VNC site consists of a quadrilateral bound on the west, north and east by hilly terrain. In some places, the hills are about 200 meters above the general site elevation. General site elevation is between 120 and 150 meters above sea level. State Highway 84 forms the

southern boundary of the site from which an expanse of gently rolling grassland extends for about 5 kilometers (km).

Approximately one third of the site is gently sloping or rolling terrain. The remainder consists primarily of the southwest slope of a ridge serrated by several small draws. The NTR is in the southern part of the site, and this part of the VNC is relatively flat.

2.2 Demography

Nine houses and a BMX bicycle race track are within a 1-km radius of the NTR. The closest houses are at 0.27 km. Eight more houses are within a 2-km radius.

A reservoir for San Francisco lies within the 4-km radius. Minor land development has occurred within the 6-km radius. The Hetch-Hetchy underground aqueduct is in this area running approximately in the east west direction.

The Livermore Division of the Veterans Administration Palo Alto Health Care System (population 400) is approximately 7 km to the east.

The 8-km radius contains the town of Pleasanton (population 54,347) northwest of the site. In this radius, the outskirts of the town of Livermore (population 60,000) are over a 365-meter mountain range. Livermore is largely a bedroom community with some light industry and agricultural activities.

The rate of population growth in the area surrounding the VNC is expected to continue to be slow based on recent history. The City of San Francisco owns a considerable portion of the land. Substantial portions have been placed into the Alameda County Land Preserve Program under the California Land Conservation Act of 1965. The remaining land is rugged terrain that does not attract industrial or residential development.

2.3 Nearby Industrial, Transportation and Military Facilities

While the towns of Livermore and Pleasanton contain some light industry, they are not considered industrial centers. Highway 680 runs within the 4-km area in approximately a north-south direction. The nearest major airport is in San Jose, approximately 32 km from the reactor site. No major air traffic flight patterns cross the site. A rail line runs in the 4-km area, about half way in the north-south direction, and then east-west. No military bases are within 8 km of the site.

The NRC staff concludes that because of the lack of significant industrial facilities and the distance of the highway, airport and rail lines from the NTR, no significant risk is posed to the continued safe operation of the NTR.

2.4 Climatology and Meteorology

The climatology of the NTR site, which includes information on precipitation, winds and temperature, is discussed in the following sections. The sources of meteorological data to be used in case of an emergency are also discussed.

2.4.1 Climatology

A Mediterranean-type climate generally characterizes the San Francisco bay area with warm, dry summers and mild winters with moderate rainfall. The climate of the eastern, inland portion of the bay area is controlled by the degree of isolation from bay influences. The more isolated locations experience a larger annual and diurnal temperature range, less winter rainfall, and less summer low cloudiness.

The greatest recorded daily rainfall is about 3.47 inches (8.8 centimeters). Precipitation averages about 18 inches (46 centimeters) per yr.

Long term conditions are defined by the meteorological data collected by GE at the VNC site.

2.4.2 Temperature and Wind Variability

The annual average temperature is 60^{o}F(15.1^{o}C). January temperatures average 46^{o}F (7.8^{o}C) with a range of 21^{o}F (11.4^{o}C). July temperatures average 72^{o}F (22.1^{o}C) with a range of 35^{o}F (20^{o}C).

The VNC site is infrequently subjected to violent storms. Thunderstorms occur less than 5 days per year and are not intense. Strong winds greater than 60 mph (27m/s) occur a few times during the fall and winter. The main consequence of such storms is the possible interruption of electrical power for which the reactor is shut down automatically.

Major flooding is a remote likelihood at the VNC. However, substantial sheet flow is a potential consequence of heavy rainfall and runoff from the surrounding hills. All roadways and facilities are constructed with drainage to preclude damage from such an occurrence. Surface waters drain away from the site to several natural ravines and man-made channels that empty into Vallecitos Creek.

2.4.3 Sources of Meteorological Data for Emergencies

VNC has its own meteorological instrumentation, which records wind direction and velocity and precipitation. From measurements dating from 1963, the distribution of wind direction and velocity has been determined as well as the frequency of distribution of stability classes.

Local meteorological measurements are available from these instruments for use in evaluating potential accidental gaseous releases from the NTR.

2.4.4 Conclusions

The meteorological characteristics of the NTR site and vicinity are mildly variable in both temperature and wind conditions. The procedure established by the licensee for collecting meteorological information to be used during a facility emergency continues to be acceptable to the NRC staff. The NRC staff concludes that no unique meteorological conditions have or could reasonably be expected to cause a significant risk to the continued safe operation of the NTR.

2.5 Geology and Seismology

Comprehensive geological and seismological studies have been conducted at and near the VNC in the years 1977-1980. The geology of the area surrounding the VNC is the result of a complex history of deposition and tectonic activity accompanied by folding, faulting and erosion. The VNC is in the Vallecitos Valley, in the southwest corner of the Livermore Valley. The area is covered by recent alluvium consisting of Livermore gravels that overlap tertiary and older rocks.

The NTR is on the lower edge of the gently sloping colluvial hills in the Vallecitos Valley. A shallow side hill cut and fill provided a level surface for the NTR facility. The building foundation is supported on native material.

California is a seismically active state. The San Francisco Bay Area contains many active faults with some near the VNC. Faults surround the Livermore Valley. The largest fault near the VNC is at little more than 3.2 km from the NTR**.**

A hypothetical accident postulates that a collapsing of the core along its three axes occurs from external forces (See Section 14.10 of this SE). The analysis indicated that fuel will not melt. Accordingly, nonvolatile fission products will not be released as a result of a postulated seismic event that produces the assumed dynamic environment.

2.6 Hydrology

The hydrology of the VNC was studied by a private company in 1976 and by the U.S. Geological Survey. Surface water at the VNC is carried away by the Vallecitos Creek. U.S. Geological Survey studies of the flood flow in this creek indicate that the water capacity of the creek is greater than the 100-year flood would provide. Therefore, the 100-year flood would not affect the VNC site.

The depth of the ground water underlying the VNC site varies from 10 to 27 feet (300 to 810 centimeters). Few wells are found in the Vallecitos Valley. Two wells have been drilled on the VNC property. Neither well is presently in use except for water quality monitoring and for watering shrubbery.

The Hetch-Hetchy Aqueduct, which provides water also to the city of San Francisco, normally supplies water to the VNC site and nearby homes. The Calaveras Reservoir, is about 13 km south of the VNC, provides a backup water supply.

VNC discharges no radioactive liquids from their facilities as discussed in Section 12 of this SE.

2.7 Conclusions

From the above considerations regarding both natural and man-made hazards, the NRC staff concludes that the site is not a significant risk associated with the NTR; therefore, continued operation of the reactor is acceptable.

3.0 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Reactor Facility Description

The NTR is in a separate portion of Building 105 in a thick walled reinforced concrete cell, which, with the control room, north room, setup room and south cell, houses the NTR facility. Normal access to the reactor cell is from the control room through two doors in the south wall: one a large, thick, shielded, concrete door and the other, a thick paraffin door covered with aluminum.

The control room contains the control console and space for experimental preparation and equipment and an operator work area. The south cell is a concrete shielded room that provides access to the thermal column and the horizontal facility. The north room provides space for performing experiments using the horizontal facility neutron beam and the control and instrument test facility.

The reactor cell is designed to function as a confinement type structure. It provides for controlled release of any airborne radioactivity through an exhaust stack, which is 45 feet (14 meters) above grade level and 9 feet (2.7 meters) above the highest point of Building 105. The reactor cell has internal dimensions of 22 feet (6.6 meters) wide by 23 feet (6.9 meters) long by 24 feet (7.2 meters) high encompassing a free air volume of about 10,500 cubic feet (ft^3) (315 cubic meters (m^3)). The normal ventilation rate is 1800 ft^3/minute (54 m^3/minute) with the reactor cell door closed. Radioactive gases and particulates released from the reactor cell are monitored continuously and the reactor is shut down if required to reduce emissions below release limits.

The reactor core is contained on a core reel assembly in a closed, water filled fuel container consisting of an inner and outer cylindrical skin. Openings are provided in the fuel container for primary coolant inlet and outlet lines, the drive mechanism for the reel assembly and the fuel-loading chute. Tubes for the control and safety rods, and the neutron source run horizontally along the outside surface of the fuel container. The entire container is enclosed in a graphite pack that serves as a reflector.

3.2 Wind and Water Damage

As summarized in Section 2.4, the meteorological conditions at the site are generally mild. Adverse conditions are not extreme. It is unlikely that the reinforced concrete reactor cell and control room will fail.

Under extreme conditions, the ventilation system may be damaged since the blower is on the roof. For this situation the reactor will be shut down. This system is not required since no credible accident can cause a fuel melt or significant release of fission products.

In addition, as discussed in Section 2.4 of this SE, the NTR is on an area that slopes so that the whole site can accommodate significant runoff. Drainage ditches were added during the construction of the site to prevent water from entering buildings.

Therefore, the NRC staff concludes that the design to mitigate water or significant wind damage to the NTR facility is acceptable.

3.3 Seismically Induced Reactor Damage

California is a seismically active area and some active faults are near the VNC. Seismic studies have shown the possibility of 0.8 times the force of gravity vibratory ground motion at the VNC. Major structures and systems have been structurally analyzed and determined capable of surviving this vibratory ground motion. These structures and systems include:

(1) The reactor cell and roof.
(2) The lead shield wall on the north side of the reactor graphite pack.
(3) The safety rod and control rod support structure.
(4) The reactor cell bridge crane.
(5) The fuel loading tank.
(6) The concrete shielding slab on top of the reactor graphite pack.

The remaining structures have not been seismically analyzed. However, safety analysis has been performed for catastrophic failure of the NTR facility resulting in compaction of the core following a seismic event. Compaction of the core is mechanistically impossible, because the roof of the reactor cell, the overhead crane, and the concrete shielding slab on top of the reactor will withstand the design basis earthquake. Even so, core compaction would not cause the reactor to become critical since moderator loss, increased self-shielding in the fuel, and geometry change due to flattening of the cylindrical core are all negative reactivity effects. Loss of coolant water was also acceptably analyzed and is discussed in Section 14.10 of this SE.

An analysis has also been performed for the withdrawal of all control rods and experiments from the core region. The poison sheets are manually positioned entirely within the graphite reflector and will not move during a seismic event. As discussed in Section 14 of this SE, no significant radiological consequences are foreseen because no fuel melt or gross dispersal of radioactive material can occur.

3.4 Mechanical Systems and Components

The mechanical systems important to safety are:

(1) The neutron absorbing cadmium poison sheets.
(2) The two boron filled motor-driven coarse control rods.
(3) One boron carbide filled motor-driven fine control rod.
(4) The four boron carbide filled motor-driven safety rods with an electromagnet that attaches the poison section with scram force provided by compressed springs.

The cadmium poison sheets are manually positioned within the graphite reflector, have no drive mechanism and, once latched will not move even in a seismic event.

The boron filled control rods are designed for precise position control required for experimental work. They move horizontally and do not provide a scram of the reactor.

Each of the four safety rods moves horizontally. Each safety rod is designed for rapid insertion on a reactor scram. Rapid insertion is affected through de-energizing the electromagnet for each safety rod and inserting the rod with a constant force spiral spring.

The control rods and safety rods are approximately 2 feet (61 centimeters) above the reactor floor. In the unlikely event of flooding, shorting or grounding of electrical systems would likely scram the reactor.

The motors, electromagnets, springs, switches and wiring are all accessible for visual inspection, testing and maintenance. The licensee has a program in place to ensure that these systems and components meet the performance requirements of the TSs.

3.5 Conclusions

On the basis of the above considerations, the NRC staff concludes that the NTR was acceptably designed and built to withstand all credible and likely wind, water and seismic damage associated with the site. The design and performance of the safety systems have been acceptable through more than 39 years of operation. Accordingly, the NRC staff concludes that the reactor systems and components are acceptable to continue to provide acceptable assurance that continued operation will not cause significant radiological risk to the health and safety of the public.

4.0 PRIOR FACILITY UTILIZATION

During the review for license renewal, the NRC staff considered whether prior operation (about 40 years) would cause significant degradation in the capability of components and systems of the facility to continue to perform their safety functions. Because fuel cladding is the component most responsible for preventing release of fission products to the environment, the NRC staff considered mechanisms that could possibly lead to detrimental changes in cladding integrity. Prominent among these considerations are:

(1) Radiation effects on cladding integrity.
(2) Fuel temperatures or temperature cycling effects on mechanical properties of the cladding.
(3) Corrosion or erosion effects on cladding physical properties
(4) Mechanical effects of handling or experimental use.
(5) The condition of safety structures, systems and components.

The NRC staff reached the following findings.

(1) The reactor fuel is uranium-aluminum alloy with aluminum cladding. This type of fuel (uranium-aluminum alloy with aluminum cladding) has had an acceptable 40-year history at this reactor and has been used at higher total radiation doses at many other reactors. The experience with this type of fuel has shown acceptable performance to the end of useful life. After that, the licensee will have to get new fuel that meets the design and safety requirements to replace the spent fuel. Therefore, fuel performance is expected to continue to be acceptable during the license renewal term.

(2) The fuel disk cladding temperature in the hottest channel is 195°F (90°C). The fuel temperature is 1°F (0.55°C) higher than the cladding temperature. These temperatures are within the extensive experience with uranium-aluminum alloy, aluminum clad research reactor fuel. Temperature cycling in the NTR is modest as shown by the fact that the temperature cycles only from about room temperature to the temperatures presented above.

(3) The coolant flow rate in the NTR is lower than that used in other reactors with aluminum clad fuel. Significant erosion problems have never been observed at these flow rates. Corrosion is kept to an acceptable minimum by controlling the conductivity and pH of the primary coolant water. When water flow, quality and materials in contact with the fuel element are controlled (as they are at the NTR), no significant degradation of cladding or fuel has been observed.

(4) Fuel elements are inserted into the core via a chute during fuel relocation manipulations. Because of the low duty cycle of the reactor and the unique design of the core and the reactivity control poison sheets, fuel handling is infrequent. Refueling has never occurred and is not anticipated before 2007. No significant fuel or core damage has been identified in the past and none is expected in the future. Experiments are not performed inside the fuel element container when the reactor is operating above 0.1 kW.

(5) The NTR staff performs regular preventive and corrective maintenance and replaces components as necessary. Nevertheless, some equipment malfunctions have occurred. The staff's review, however, indicates that most of these malfunctions have been one-of-a-kind, typical of industrial-quality electrical and mechanical instrumentation and components. There is no indication of significant degradation of the instrumentation and components, and there is strong evidence that any future degradation will be met with prompt remedial action by the NTR staff. Therefore, there is acceptable assurance that there will be no significant increase in the likelihood of occurrence of a reactor accident as a result of component malfunctions during continued operation under the renewed license.

(6) The NRC staff evaluated the licensee's analytical assumptions, methods and results for potential reactor accidents (See Section 14 of this SE). None of the credible accidents postulated would lead to the failure of the cladding of any fuel assembly or the uncontrolled release of fission products. However, the licensee hypothesized an enveloping event involving a failure in a fueled experiment. This event would lead to the maximum potential radiation hazard to personnel. The licensee makes no assumptions as to the cause of the failure. The licensee evaluated only the potential consequences of this event and not the likelihood or mechanisms for occurrence. The NRC staff has designated this event to be the maximum hypothetical accident and found that the radiological consequences were acceptable.

In addition to the considerations discussed above, the NRC staff reviewed licensee event reports and inspection reports. On the basis of this review and the above considerations, the NRC staff concludes that there has been no significant degradation of equipment. The NRC staff further concludes that facility management will continue to maintain and operate the

reactor so that there is no significant increase in the radiological risk to facility employees or the public.

5.0 REACTOR

The NTR is a fixed-core, tank-type research reactor that uses light water as the coolant, water and graphite as the moderator and graphite as the reflector. The fuel is highly enriched uranium. The reactor has been licensed to operate in the steady state mode at power levels up to and including 100 kW. The reactor is 14 feet (4.26 meters) long (including the thermal column and the control rod drives) by 6 feet (1.82 meters) wide and 6 feet (1.82 meters) high. The reactor core is an annular cylinder in shape. The core is centered in a 5-foot (1.52-meters) cube of AGOT (nuclear) grade graphite. The core container is a horizontal aluminum cylinder 20 inches (50.8 centimeters) in length and 18 inches (45.7 centimeters) in diameter with an inner annulus diameter of 11.5 inches (25.9 centimeters). The 11.5-inch diameter cylindrical space is filled with graphite and traversed by a horizontal experimental facility. The annulus formed between the inner and outer aluminum is fitted with 16 fuel assemblies as described in Section 5.1 of this SE.

Reactor control is achieved by inserting or withdrawing three control rods and four safety rods. In addition, up to six poison sheets may be manually positioned. All control elements are arrayed around the outside of the fuel container. Heat generation from fission is transferred from the fuel to the water in the fuel container. The water is normally pumped between the end plates of the core container to an external heat exchanger. The reactor may be operated at or below 0.1 kW without forced convection cooling.

5.1 Reactor Core

The NTR core is composed of 16 highly enriched uranium fuel assemblies loaded into a core reel assembly in the fuel container can. A reel drive mechanism is provided to rotate the entire reel assembly to a desired position with respect to a loading chute through which the fuel assemblies are loaded individually onto the reel. The fuel assemblies are securely and accurately placed onto the reel.

The fuel, the core reel assembly and the reactor cooling water are the only devices inside the fuel container can. All movable neutron absorbers for reactivity adjustments are on the outside periphery of the fuel container can.

5.1.1 Fuel Assemblies

All the fuel disks in the core are from the original fuel load fabricated in 1957. Each fuel assembly consists of 40 aluminum clad 2.75 inches (6.98 centimeters) outside diameter uranium-aluminum fuel disks. Each disk consists of a fuel bearing flat doughnut-shaped sandwich and an inner and outer edge ring. The three pieces are brazed together to clad the uranium-aluminum alloy meat. Each disk contains approximately 6.6 grams of approximately 92 percent enriched uranium. The disks are spaced on an aluminum shaft with an aluminum spacer between each fuel disk and an additional aluminum washer in every other space. This arrangement produces an active length of about 15.25 inches (38.7 centimeters). Motion of the

disks on the aluminum shaft is prevented by lock nuts placed on each end of the shaft. A 0.75-inch (1.69-centimeter) length of each end of the aluminum shaft is machined to provide a tip suitable for supporting and positioning the fuel assembly accurately in the core reel. This tip extends past the ends of the core reel into a raceway. It is this section of the fuel assembly that is engaged by a tool during fuel handling.

5.1.2 Control Rods

The reactor control system operates by inserting seven neutron absorbers about the outside periphery of the fuel container. All seven absorbers are boron carbide filled rods. Two motor-driven rods are for coarse control; one motor-driven rod is for fine control. Four, motor-driven rods are safety rods with an electromagnet attached to the poison section. This safety rod poison section can be driven into the core by a spring upon receipt of a scram signal from the safety system. The average in-flight scram time of the four safety rods is less than or equal to 300 milliseconds. Upon receipt of a scram signal, all control and safety rod drives run to the fully inserted position provided alternating current (ac) power to the console is maintained.

The rods move in guides that extend from the south end of the fuel container through the reflector and shield to the north face of the reactor. The guides place the center of the poison about 0.6 inches (1.5 centimeters) from the outside edge of the reactor core.

The control and safety rods have horizontally mounted drive mechanisms. These drive mechanisms are supported from the north face of the reactor on a 5-foot (1.5-meter) high aluminum support plate located about 4.5 feet (1.4 meters) in front of the north face. Drive motors are actuated from the control console for remote positioning of the rods.

Because of the lack of symmetry in the arrangement of the poisons around the core and the possibility of strong shadowing effects, the reactivity worth of the individual safety rods varies. In the normal core, the reactivity worth of the most effective safety rod is about 1$. The minimum worth of all four-safety rods is about 2$.

Calibrations of the control rods by the rising period method indicate that the total worth of all control rods is about 2.3$. The withdrawal speed of these rods is controlled so that if all three rods were withdrawn simultaneously (not allowed by the control system), the reactivity addition rate would be about 0.02$/second, which is readily controllable.

Means are provided for automatic and manual scrams, drive reversal and drive inhibits to maintain the reactor in a safe operating range or for safe shutdown.

Manually positioned, aluminum clad, cadmium poison sheets are a third type of reactivity control. The sheets are inserted and removed in the graphite around the fuel container through access holes provided in the north shield while the reactor is shut down. They are used to limit the reactivity available to the operator or to increase the shutdown margin.

The manual poison sheets are equipped with a spring-loaded handle, which provides positive restraint. Only positions with this positive restraint are utilized. The reactivity worth of a full sheet is approximately 1$. The worth of all six is about 3$. In a typical core configuration, the

worth of the manual poison sheet is 0.7$. This allows for a typical excess reactivity available from the control rods of about 0.3$.

5.1.3 Neutron Moderator and Reflector

The NTR is designed to operate with a fixed neutron moderator and reflector. It is a 5-foot (1.52-meter) cube of reactor grade graphite which not only serves as neutron moderator and reflector but also as physical support for the fuel container. The reflector is contained and supported by an aluminum box and base.

Many items penetrate the reflector such as the fuel loading chute, control and safety rod guide tubes, manually positioned poison sheets, core reel drive shaft and experimental facilities. In addition, two special sections of the reflector are removable making it possible to inspect the fuel container without disturbing the rest of the reflector.

5.1.4 Neutron Source

The NTR is required to ensure that the reactivity status of the reactor can be determined at all times, including when the reactor is shut down. This is achieved using an encapsulated 0.2-curie, radium-beryllium neutron source that travels in a guide tube identical to those used for control rods. Following a reactor scram, the source automatically runs to the fully inserted position provided ac power to the console is maintained.

The source-detector arrangement provides the minimum neutron flux signal required for the nuclear instrumentation for startup. The safety rod magnets cannot be energized until the source is full in.

5.1.5 Conclusions

The NRC staff has reviewed the information pertaining to the design, construction, and function of the NTR fuel, moderator, reflector, neutron source, control-safety rods and sheets, and rod drives. On the basis of this review, the NRC staff has concluded that the design of these core-related components will acceptably continue to permit safe operation and shutdown of the reactor.

5.2 Reactor Cell

The reactor is in a thick walled, rectangular shaped concrete cell. Normal access to the cell is through a large doorway in the south wall. During operation, a thick, motor-driven, sliding concrete door lined with steel normally closes the doorway. Power for moving the cell door is interlocked with a key operated switch under the control of the reactor operator.

A manually operated, paraffin door covered with aluminum, located just inside the reactor cell, is used to reduce further the radiation dose outside the reactor cell door.

A large removable equipment hatch is provided in the cell roof.

Penetrations into the reactor cell are provided for reactor and experiment services.

The reactor is in a high density concrete alcove in the reactor cell so that the 5-foot (1.52-meter) thick alcove walls provide acceptable shielding on the east and south sides. On the west side of the reactor, the fuel storage tank provides 4 feet (1.21 meters) of water shielding in addition to the 3-foot (0.9-meter) alcove wall. A lead shield wall provides shutdown gamma shielding for the control rod drive area. A removable shield slab of heavy density concrete covers the top of the reactor.

5.3 Core Support Structure

The fuel container rests on the sections of the graphite cube pack beneath it. The graphite cube pack is enclosed on all but two sides with aluminum. In addition, a cadmium liner is provided for the north and east sides of the box.

The box containing the graphite cube rests on an aluminum plate fastened to a framework of aluminum I-beams. The I-beam base is clamped to steel support plates anchored to the reactor cell floor.

5.4 Reactor Instrumentation

The reactor instrumentation is similar to that found in research reactor installations at other laboratories. The instrumentation gives the operator the information necessary for manipulation of the controls. The following instrument channels are part of the provided reactor instrumentation system.

- Source range channel.
- Three linear channels operating as information and safety channels with 2 out of 3 logic to provide power level scram.
- Log N power level and period.
- Cooling system temperatures.

5.5 Biological Shield

The reactor cell and alcove provide biological shielding as discussed in Section 5.2 of this SE. After reviewing the reactor cell and alcove design, and operational experience at the NTR, the NRC staff concludes that the shielding was acceptably designed to reduce external radiation exposures to acceptable levels.

5.6 Dynamic Design Evaluation

To ensure safe and responsive operation, the reactor is provided with multiple control and safety rods and nuclear instrumentation. The reactor exhibits a negative void coefficient of reactivity and a negative temperature coefficient of reactivity above 124^{o}F (51^{o}C). Thus, in the unlikely event of inadvertent escalation of the power leading to high temperatures, the negative coefficient will limit the reactor power.

5.6.1 Core Thermal and Hydraulic Characteristics

The maximum power level of the NTR is 100 kW. For powers above 0.1 kW, forced convection cooling with water is used to transfer heat from the core. At power levels less than or equal to 0.1 kW, operation is permitted with natural convection cooling. The coolant system is designed so that in the event of a loss of flow at any power level, the reactor will continue to be cooled by natural circulation.

The proposed safety limits and limiting safety system settings for forced and natural convection cooling were established on the basis of analyses and experiments as discussed in the following sections.

5.6.1.1 Forced Convection Cooling

The limiting criteria for safety are the integrity of the fuel and the cladding. Consequently, for purposes of analysis, the licensee has assumed that fuel or cladding integrity is compromised because of fuel and cladding melting and for burnout of the cladding. The following conservative assumptions formed the basis of the NTR analysis:

- The fuel container is filled with water.
- The primary coolant flow rate is 20 gallons/minute (76 liters/minute).
- The coolant inlet temperature is typically 90°F (32°C).
- The coolant average outlet temperature is typically 120°F (49°C).
- The coolant exit temperature for the hottest channel is 150°F (65°C).
- The overall power-peaking factor is 1.58.

The saturation temperature of the coolant corresponding to the average reactor pressure of 20 pounds per square inch gage (psig) (0.27 atmospheres (atm)) is 228°F (109°C). Thus, the state of the coolant is far removed from boiling at the design operating condition.

The licensee calculated that the burnout heat flux in the hottest channel of the core is 227,000 Btu/h-ft^2 (716,185 W/m^2) for the hydraulic conditions at which the channel is operating. The actual, maximum heat flux in this channel for 100 kW operation is 10,300 Btu/hr-ft^2 (32,496 W/m^2). While this burnout ratio is 22, the licensee has conservatively selected a DNB ratio of 1.5.

Thermal hydraulic analyses show that the DNB heat flux is not significantly affected by the core flow rate or the core inlet temperature over the possible range of these parameters.

On the basis of these calculations, the licensee proposed and applied a safety limit of 190 kW. This safety limit is included in TS 2.1.1.

In accordance with 10 CFR Part 50.36, the licensee proposed a limiting safety system setting (LSSS) designed to ensure that automatic protective action (a reactor shutdown) would occur in sufficient time to prevent the safety limit from being exceeded. For forced convection coolant flow conditions, this LSSS is conservatively set at 125 kW and is included in TS 2.2.3. The licensee conservatively selected the LSSS to allow for the most adverse combination of

uncertainties in the monitored parameter, including the response time of instruments and accuracy of measurements.

The NRC staff has reviewed the NTR safety analysis report and found these analyses appropriate and very conservative for application to the NTR. The staff, therefore, concludes that the conditions established by the specified safety limit and LSSS given in the NTR's TSs, Appendix A to the Operating License, give acceptable assurance that the fuel cladding integrity will be maintained.

5.6.1.2 Natural Convection Cooling

The licensee has analyzed an instantaneous loss of flow accident while the reactor is operating at 100 kW. For the analysis, the low flow scram is assumed to malfunction and the reactor continues to operate. The maximum fuel temperature during this transient is 238^{o}F(114^{o}C) and the equilibrium value of the burnout ratio is approximately 26.

Based on this analysis, the licensee has conservatively determined that above 0.1 kW the reactor will be cooled by forced convection and at or below 0.1 kW, forced convection cooling is not required. The applicable TS is given in TS 3.3.3.1.

The NRC staff concurs with the NTR analysis and evaluation, and has concluded that operation of the NTR with natural convection cooling at power levels up to and including 0.1 kW poses no significant risk of fuel or cladding damage resulting from high temperatures.

5.6.2 Shutdown Margin

The proposed TS 3.1.3.3 prescribes a minimum reactivity shutdown margin of 1$ with the non-scramming control rods out and the maximum worth safety rod stuck out. The reactivity worth of the most effective safety rod is about 1$. The minimum worth of all four-safety rods is 2$. The potential excess reactivity of the core is 0.76$. (Potential excess reactivity includes that which can be added by the manipulation of control rods, plus the maximum credible reactivity addition from primary coolant temperature, plus the reactivity worth of all installed experiments). Therefore, as long as the total excess reactivity loaded into the core, including that resulting from experiments, is no more than the TS limit of 0.76$, the shutdown margin can be achieved.

The NRC staff concludes that the shutdown margin of 1$ with the maximum worth safety rod stuck out is sufficient to ensure that the reactor can be acceptably shut down under all credible conditions.

5.6.3 Excess Reactivity

The potential excess reactivity that the licensee is authorized to load into the NTR is 0.76$. This amount provides for the various positive and negative reactivity effects associated with operation and use of the reactor, as well as operational flexibility. Low power operation of the NTR makes reactivity changes from fuel burn-up and fission product poisoning small. To account for these small changes, enough cadmium is removed from the remaining cadmium sheets to maintain normal operation and still ensure that the potential excess reactivity is 0.76$. The licensee has performed tests on the latching mechanism that hold the poison sheets in

place. The licensee has concluded that it is not credible that any of the cadmium sheets could be ejected from the core to result in a step reactivity insertion potentially greater than 0.76$.

Imposing a limit on excess reactivity helps ensure that the safety analysis report assumptions and analyses are applicable to the operating core. The NRC staff has concluded that the potential excess reactivity limit of 0.76$, when considered with the shutdown margin, is sufficient to ensure that the reactor can be acceptably controlled and shut down under all credible conditions.

5.6.4 Experiments

A movable experiment is one that can be inserted, removed and manipulated while the reactor is operating.

A non-secured experiment is one that is not mechanically held in position with sufficient force to overcome the expected effects of pneumatic or other forces that are normal to the operating environment of the experiment, or forces arising from likely credible malfunctions. These experiments are not designed to be moved during reactor operation.

A secured experiment is one that is mechanically held in a stationary position relative to the reactor. The restraining forces are substantially greater than those to which the experiment might be subjected by pneumatic forces or other forces that are normal to the operating environment of the experiment or forces arising as a result of credible malfunctions.

Experiments are limited so that the sum of the reactivity worth of all experiments shall be within the potential excess reactivity limit of 0.76$. This limitation applies to all experiments whether movable, non-secured or secured. The applicable TS is given in TS 3.5.3.1.

Experimental objects are not allowed inside the core tank when the reactor is at a power level greater than 0.1 kW. The applicable TS is given in TS 3.5.3.3.

Experimental devices in the fuel-loading chute are secured to prevent their entry into the core region during reactor operation. The applicable TS is given in TS 3.5.3.4.

No experimental object may be moved during reactor operation unless its potential reactivity worth is less than 0.5$ and the operation is performed with the knowledge of the licensed operator at the console. Experience at the NTR facility has shown that this worth is acceptable for experimental needs and that operator action provides easy control of any change in reactor power resulting from the removal and insertion of such an experiment.

All power operated, remotely controlled mechanisms for moving an object into the reactor core shall be energized from the reactor console. However, movement of the object may be initiated from another location. All manually operated mechanisms for moving an object into the reactor graphite pack shall be done with the knowledge and consent of the reactor operator at the controls of the reactor.

The potential reactivity worth of any component, which could be ejected from the reactor by a chemical reaction, shall be less than 0.50$.

Experiments may contain explosive materials or materials that can produce a chemical reaction. The licensee's TSs and safety criteria for explosives and combustible materials have been reviewed by the NRC staff (see Section 14) and determined to be acceptable.

The NRC staff reviewed the proposed limitations on the worth of experiments. On the basis of this review, the NRC staff concludes that these limitations are conservative. Specifically, because of the limitations based on potential reactivity worth, failure of experiments resulting in a positive reactivity insertion would not result in damage to the fuel or reactor components.

5.7 Functional Design of the Reactivity Control System

This section of the SE discusses the functional design of the reactivity control system, which includes three non-scramming control rod drives and four scrammable safety rod drives and the scram logic circuitry that initiates scrams. The control rods and the safety rods move horizontally.

5.7.1 Control Rod Drives

The two coarse control rods and one fine control rod each have drive units. The poison sections of the coarse control rods are contained in a stainless steel tube that ultimately is attached to a yoke that is positioned by the drive mechanism. This yoke is fastened to a lead screw that runs through a sprocket and nut assembly connected through a chain drive to a gear motor. Pushbutton switches at the console permit manual control. A scram signal runs the rods to their fully inserted position, provided ac power to the controls is maintained. Two limit switches on each control rod drive mechanism provide rod in and rod out information. The rod in switch energizes the green light at the console, prevents energizing the electromagnets for the safety rods unless all control rods are fully inserted, and interrupts the motor circuit for the full in position. The rod out switch energizes the yellow light at the console and interrupts the motor circuit at the outer limit of the stroke.

The poison section of the fine control rod is contained in a stainless steel tube that connects to a nut block that travels on a lead screw. The lead screw is rotated through a right angle gearbox by a gear motor with an electrically operated brake. Pushbutton switches on the console are used for remote manual control. Two limit switches perform the identical functions discussed above for the coarse control rods.

5.7.2 Safety Rod Drives

The poison section of each safety rod is contained in a stainless steel tube that connects to an extension rod culminating in an armature assembly (electromagnet). Two springs are attached to the extension rods so that withdrawal of the safety rod compresses the spring to store energy. The safety rod is held to the rod drive by an electromagnet that engages the armature attached to the extension rod. The electromagnet is attached to a drive nut that moves horizontally on a lead screw. Rotation of the lead screw is accomplished with an electric motor through a belt and pulley drive. Remote manual control by the operator is by pushbutton switches at the console. A circuit is provided to run the carriage automatically to the fully inserted position following a reactor scram, provided ac power to the console is maintained.

Upon initiation of a scram signal, the electromagnets are de-energized, the constant force spiral springs cause the armature to separate from the magnet and rapidly insert the safety rods. An automatic signal runs all rod carriages to the fully inserted position provided ac power to the console is maintained.

The limits regarding the stroke of the safety rods are set by adjustable limit switches mounted on the rod drives. These switches also provide interlocks so that all four safety rods must be withdrawn sequentially before any control rods may be moved and all control rods and the neutron source are fully inserted before the electromagnets may be energized.

An air dashpot-type shock absorber, accomplishes deceleration of each scrammed safety rod.

5.7.3 Scram-logic Circuitry

The NTR is equipped with a scram-logic safety system that receives signals from core instrumentation and other reactor parameters to initiate a scram by removing electrical power from the safety rod magnets. Specifically, the following parameters can initiate a reactor scram:

- ♦ High reactor power (2 out of 3 or 1 out of 2).
- ♦ Loss of positive high voltage to ion chambers (2 out of 3 or 1 out of 2).
- ♦ Fast reactor period from the log N channel.
- ♦ Log N amplifier switch not in the operate position.
- ♦ Loss of high voltage to the log N detector.
- ♦ High core outlet temperature.
- ♦ Low primary coolant flow.
- ♦ Operator/personnel manual scram.

5.7.4 Conclusions

The NTR reactor is equipped with a control and safety system typical of non-power reactors, incorporating multiple control and safety rods and multiple and redundant sensors that can initiate an automatic scram. The design incorporates sufficient safety rod redundancy to enable the reactor to be shut down safely from any operating condition, if the most reactive safety rod fails to insert and the control rods are fully withdrawn from the core.

In addition to the electromechanical safety controls, the negative moderator temperature coefficient above 124°F (51°C) and the negative void coefficient provide an inherent backup safety feature.

In accordance with the above, and the evaluations in Sections 8 and 14 of this SE, the NRC staff concludes that the reactivity control systems of the NTR are designed and function acceptably to ensure safe operation and shutdown of the reactor under all normal operating conditions.

5.8 Conclusions

The NRC staff concludes that the NTR is designed and built according to standard industrial practices. The reactor consists of standardized components representing many reactor-years

of operation, and it includes both diverse and redundant safety-related systems. The staff's review of the reactor facility included studying its specific design, installation and operational limits as identified in the licensee's safety analysis, and the TSs. On the basis of its review of the NTR operating experience since 1957 and its evaluation of possible degradation of components as discussed in Section 4.0 above, the NRC staff concludes that there is acceptable assurance that the reactor can continue to operate safely, as limited by its TSs for the proposed duration of the license.

6.0 REACTOR COOLING SYSTEM

The cooling systems for the NTR are composed of 5 subsystems:

- Primary coolant
- Secondary coolant
- Primary coolant cleanup
- Primary coolant makeup water
- 500-gallon (1900-liter) hold-up tank

6.1 Primary Coolant System

The NTR core is in a core tank container. The primary coolant system is an un-pressurized light water system with a total volume of 28.5 gallons (108 liters). The main flow path has 19.5 gallons (74 liters), and 9 gallons (34 liters) are in the core tank. The components of the primary coolant system in contact with primary water are aluminum or stainless steel. The flow rate in the primary coolant system is 20 gallons per minute (gpm) (76 liters per second (l/sec)).

The primary coolant flow path is from the primary pump through a check valve and flow control valve, to the bottom of the reactor core tank. A baffle tube distributes the flow. The flow is then up around the fuel assemblies to the top of the core tank. The water then flows out of the tank, through a flow orifice, the shell-side of a U tube heat exchanger, an air trap and back to the pump.

The primary coolant system contains an atmospheric vent line. This line provides a continuous vent to the atmosphere for air and other gases from the primary system and prevents over-pressurization of the primary system.

The 500 gallons (1900 liters) hold up tank in the reactor cell can hold the primary water. This tank can retain the water or the water is transferred to other tanks and removed from the facility. This tank also receives the discharge from the atmospheric vent line, and from a sump in the reactor cell.

6.2 Secondary Coolant System

The secondary coolant system provides potable water from the Building 105 supply to the tube side of the heat exchanger. The coolant line passes through the reactor control room. A shut-off valve, a check valve and a flow indicator are in the control room. The coolant line ends at

the site retention basins. The flow rate in this once through system is about 35 gpm (2.2 l/second).

During operation, the primary system pressure is lower than that of the secondary system. In the unlikely event of a failure in the heat exchanger tubes, secondary water would leak into the primary water. Sampling of the primary water would detect the leakage from the attendant decrease in the quality of the water. A leak from the primary to the secondary system could take place when the secondary system is not operating. If such a leak occurred, contaminated water would drain to the site retention basins where it is sampled before release.

6.3 Primary Coolant Cleanup (Coolant Purification) System

The TSs establish limits regarding the quality of the primary coolant. The primary water is monitored for conductivity, pH and radioactive content. To maintain acceptable water quality, part of the coolant flow passes through a demineralizer system and a 5-micron after-filter. The flow through the system is about 16 gallons per hour (61 liters per hour).

6.4 Primary Coolant Makeup Water System

Primary coolant makeup water is provided from the raw water main supplied from the site's 500,000-gallon (1,900,000-liter) storage tank. A potable water line coming off the tank feeds a deionizer unit that provides deionized water to the entire facility including makeup to the reactor primary system.

The makeup system also supplies the 1800-gallon (6840 liters) fuel loading tank, which serves as a reservoir for the primary system.

Check valves are used to ensure that the primary water cannot enter the storage tank and the city's water system.

6.5 500 Gallon (1900 Liter) Hold-up Tank

A hold up tank is in the reactor cell. The tank holds water drained from the primary coolant system, discharged from the atmospheric vent line, and collected from the reactor cell sump. The tank is discharged to the site retention basins where it is sampled before release.

6.6 Conclusions

The NRC staff concludes that the NTR cooling systems are acceptable to remove the heat from the fuel and prevent loss of integrity under the full range of normal operating conditions. During operations leakage from the primary system to the secondary system in the heat exchanger is not likely. However, potential leakage would be detected and controlled acceptably. On the basis of its review of the NTR operating experience since 1957 and its evaluation of possible degradation of components as discussed in Section 4.0 above, the NRC staff concludes that there is acceptable assurance that the reactor can continue to operate safely, as limited by its TSs for the proposed duration of the license.

7.0 ENGINEERED SAFETY FEATURES

The licensee has shown that a loss of coolant would not result in fuel cladding temperatures that would lead to fuel cladding integrity failure or release of fission products (see Section 14.10 of this SE). Therefore, the reactor does not require an emergency core cooling system.

The licensee has analyzed an accident involving the release from an experimental device of fissile material and fission products (see Section 14.1 of this SE). In this analysis, credit was taken for an engineered safety feature (ESF) consisting of building confinement and a controlled, filtered, elevated release of airborne radioactivity.

7.1 Confinement and Ventilation

The NTR is installed in a concrete shield room (the reactor cell) in the northeast corner of a multi-use building. Although the cell is not designed to provide gas tight containment, controlled release of airborne radioactivity is possible through the operation of the cell ventilation system. During reactor operation, two doors in series close the cell entrance: a manually operated thick paraffin door and a motor-driven door composed of concrete and steel. Power for moving the concrete door is interlocked with a key switch so that the operator can control power to the motor and therefore cell entry.

The NTR is equipped with a ventilation system that includes an exhaust fan and a bank of absolute high-efficiency particulate air (HEPA) filters. The exhaust air is then discharged through a 45-foot (13.5-meter) stack. This system, which draws air from the reactor cell, south cell and the north room modular stone monument is designed to control airborne radioactivity that could be released. All penetrations through the concrete walls are controlled to ensure that the effectiveness of the cell to contain radiation and radioactive materials is not significantly reduced.

The ventilation system has a flow rate capability of 3000 cubic feet per minute (cfm) (1.5 cubic meters per second (m^3/s)) but is normally operated at a flow rate of 1800 cfm ($0.9 m^3/s$). An air monitoring system provides continuous indication of the concentration of radioactive materials in the ventilation effluent and energizes an alarm at the reactor console if the concentration reaches a set point. The setpoint has been selected to ensure that the airborne release does not exceed established limits. The basis of the set point is to allow action to be taken to maintain doses to members of the public from airborne releases to a maximum of 10 millirem (mrem) per year.

In the event of a significant increase in radiation level from any of several operating radiation detectors, the reactor will be scrammed and confinement is established using the already operating ventilation system. No further actions are required.

The licensee has analyzed the effectiveness of this system for the MHA (as addressed in Section 14.1 of this SE) and has established TSs based on this analysis.

7.2 Conclusions

The NRC staff reviewed the installation and design of the confinement and ventilation ESF employed at the NTR facility. As a result of this review, the NRC staff concludes that the system has been effectively planned and engineered and would acceptably mitigate the consequences of postulated accidents. Therefore, on the basis of its review of the NTR operating experience since 1957 and its evaluation of possible degradation of components as discussed in Section 4.0 above, the NRC staff concludes that there is acceptable assurance that the reactor can continue to operate safely, as limited by its TSs for the proposed duration of the license.

8.0 INSTRUMENTATION AND CONTROL SYSTEMS

The instrumentation and controls (I&C) used for the NTR are functionally similar to those widely used for research reactors in the United States. Control of the fission process is achieved using four scrammable safety rods, two non-scramming coarse control rods and one non-scramming fine control rod. The instrumentation system, which includes the reactor safety system, is composed of nuclear, non-nuclear and process instrumentation.

8.1 Instrumentation System

The instrumentation system used in the NTR is composed of both nuclear control and process instrumentation circuits. The electronics system provides annunciation and/or indication in the control room. In addition, an automatic scram function is provided through the scram logic units discussed below. Additional features of the instrumentation system include alarms, interlocks, rod-drive inhibits and reverse drive functions.

8.1.1 Nuclear Instrumentation

The nuclear instrumentation used in the NTR gives the operator the information needed to manipulate the nuclear controls, and provides automatic protective functions (scram, as follows:

- The three linear power level channels are sufficiently sensitive to be on scale with the flux provided by the neutron source and are therefore used as start up channels. Each power level channel has a down scale alarm. When 2 out of 3 channels alarm, the control and safety rod motors cannot be energized to withdraw the control rods. A separate source range monitor channel is available for use during special procedures. The three linear channels receive a signal from one of three compensated ionization chambers (CICs). Using a 20-range switch, each channel monitors the power level from source range to 125 percent of full power. In addition to providing power level display to the operator with a meter and recorder, these 3 channel on a 2 out of 3 (or 1 out of 2) coincidence logic provide a power level scram of the reactor at 125 kW. The high voltage on each of the CICs is monitored and loss of high voltage on 2 out of 3 or 1 out of 2 will cause a reactor scram.

- The log N and period amplifier receives a signal from a CIC and provides a display of reactor power level and reactor periods. Using non-coincidence logic, this channel provides

a reactor scram for periods of less than 5 seconds and for overpower. The high voltage on the CIC is monitored and loss of high voltage will initiate a scram.

The four neutron sensing CICs are positioned in thimbles in the fuel storage tank or at the face of the reflector. The exact location chosen depends on the intended use of the reactor, sensitivity of the system and the desired meter readings. The four CICs are adjustable over a limited distance to allow each channel to be calibrated to the reactor thermal power.

8.1.2 Reactor Safety Systems

The control and nuclear instrumentation systems are interconnected through scram logic units. These units, manual scrams and other monitored parameters control the magnet power supply that provides the current to the safety rod magnets. All scram functions cause a loss of current to the electromagnets, which then permits rapid insertion of the spring-loaded safety rods. Fail safe philosophy has been incorporated into the design as much as is practical. Magnet current cannot be returned until the scram condition clears and the reactor operator resets the scram logic unit.

8.1.3 Inhibits, Interlocks, Alarms and Annunciation

An inhibit signal that prevents control and safety rod withdrawal (reactor startup) is provided by a down scale monitor (reading less than a preset minimum level) on each linear power level channel. When 2 out of 3 indicate a down scale condition, the control and safety rod motors cannot be energized to withdraw the control rods.

An interlock prevents a particular action from occurring until all the prerequisites for that action to occur are satisfied. These interlocks are as follows:

♦ For initial startup and following a scram, magnets cannot be energized unless all safety and control rods and the neutron source are at their inner limits.

♦ Safety rods must be withdrawn in sequence one at a time to their outer limits before a control rod can be withdrawn.

♦ The Log N amplifier mode switch must be in the operate position to get electromagnet current.

Control console mounted annunciator lights and audible alarms provide the operator with information on conditions including all important variables related to reactor safety. Following annunciation of an event, the condition must be addressed in accordance with alarm response procedures. After the condition is corrected, the operator must reset to restore the annunciator to normal operating conditions. In some cases, it is possible to turn off the audible portion of the annunciation before correcting the condition. In this case, the visual annunciator cannot be reset until the condition is corrected.

8.2 Control System

The NTR control system is composed of both nuclear and process control equipment in which redundant or diverse safety-related components are designed for independent operation in case of a single failure or malfunction of components essential to the safe operation of the reactor.

8.2.1 Nuclear Control Systems

Control of the NTR is achieved by horizontally inserting and withdrawing neutron absorbing control rods using the control drive units, which are mounted on an aluminum plate on the north side of the reactor. These control rods have a solid coupling and cannot scram the reactor. In addition, four safety rods are coupled to the rod drives through electromagnets so that any electrical power interruption in the magnet power supplies causes uncoupling of the safety rod from its drive. A scram then occurs when compressed springs drive the safety rods into the core. The control rod drives are manipulated from the control room by the reactor operator in response to indications from various instruments. The control and safety rod systems were discussed in Section 5.1.2 of this SE.

The NTR also incorporates manually controlled poison sheets, which may not by moved during routine reactor operation. These poison sheets are set to limit the reactivity available to the operator or to increase the shut down margin. The poison sheets were discussed in Section 5.1.2 of this SE.

8.2.2 Supplementary Control Systems

The supplementary control systems used in the NTR include non-nuclear and process control systems. These systems are designed to control the various processes involved in reactor operation. Not all these processes are directly related to safety. This category includes circuits and devices that energize and monitor coolant pumps and coolant parameters such as flow rate, temperature, resistivity and amount of water in the core container.

8.3 Supplementary Instrumentation

Supplementary instrumentation consists of the facility's fixed radiation monitoring system. Section 12.1.6.2 of this SE considers those used as area radiation monitors. It is notable that these monitors can detect all types of airborne radioactivity or direct radiation that is credible at the NTR. The exhaust stack monitoring system is discussed in Section 12.1.6.2 of this SE.

8.4 Conclusions

The NRC staff concludes that the I&C systems at the NTR are acceptably designed and maintained. The I&C systems are designed so that the reactor is automatically and safely shut down if electrical power is lost. Redundancy in the important ranges of power measurements by nuclear instruments is ensured by overlapping ranges of the log and linear channels. Additionally, important nuclear and process variables are monitored and displayed at the control console. On the basis of its review of the NTR operating experience since 1957 and its evaluation of possible degradation of components as discussed in Section 4.0 above, the NRC

staff concludes that there is acceptable assurance that the reactor can continue to operate safely, as limited by its TSs for the proposed duration of the license.

9.0 ELECTRICAL POWER

The two types of electrical power at the NTR are the normally used main power and emergency power.

9.1 Main Power

A 480-volt load center in Building 105 is fed from the site's 12-kilovolt bus and, in turn, feeds power and lighting distribution panels for the NTR facility. The 12-kilovolt bus is supplied from an on site 5000 kVA transformer fed from parallel off site 60 kV Pacific Gas and Electric Company supplies. Two 120/240 volts circuit breakers in the control room feed individual breakers for the primary coolant pump, service outlets, facility lights and the reactor console. Power supplied to the console is used for the reactor instrumentation and the control and safety rod drive motors.

9.2 Emergency Power

Because the reactor will scram in case of a power failure and the decay heat generated in the core after scram will not cause fuel overheating, emergency power for and during reactor shutdown conditions is not necessary for reactor safety. Accordingly, emergency battery power is supplied only for semi-portable emergency lighting units that are installed at several locations in the facility. Each unit contains a battery that maintains its charge from a 115-volt ac system. On loss of normal power, the units are energized automatically to provide light for emergency actions and personnel exits.

9.3 Conclusions

On the basis of its review, the NRC staff concludes that the electrical power provisions at the NTR facility provide reasonable assurance of acceptable operation. In addition, the NRC staff concludes that loss of off site power will lead to safe shutdown of the reactor and that emergency power for other than lighting is unnecessary.

10.0 AUXILIARY SYSTEMS

The auxiliary systems discussed in this section include the fuel handling and storage systems, the compressed air system, the communication system and the fire protection provisions. The ventilation system is discussed in Section 7, "Engineered Safety Features," and radioactive waste storage is discussed in Section 12, "Radiation Protection Program and Radioactive Waste Management."

10.1 Fuel Handling and Storage

The NTR has no unirradiated fuel assemblies or fuel disks. The fuel assemblies were installed in the reactor in 1957 and have only been removed for maintenance of reactor components.

Reactor fuel is not handled routinely. If it becomes necessary to remove a fuel assembly, the fuel supporting reel mechanism is rotated until the desired element is at the entrance of the fuel-loading chute. The fuel handling tool (guided by slots in the sides of the chute) slides down the chute, attaches to the fuel assembly shaft, and pulls the assembly up the chute and into the fuel-loading tank. A fuel assembly is installed by reversing this process. The removed irradiated fuel assembly would remain in the fuel-loading tank.

If it is necessary to remove more than one fuel assembly from the core, special arrangements must be made to use a shielded transfer cask and storage facilities elsewhere on the site. Authorization would have to be obtained before such transfers are made, and procedures would have to be developed to ensure safe handling with consideration for radiation protection and criticality control. (The transfer of the entire core to another on-site facility in 1977 demonstrated the validity of this process.)

10.2 Compressed Air System

Compressed air for the facility is supplied from the Building 105 service air compressor in the second floor mechanical equipment room. The compressor can deliver 50 cfm (0.025 m^3/s) of free air at a pressure of 100 psig (6.8 atm). A relief valve maintains the system pressure at less than 120 psig (8.2 atm). A low-pressure switch provides an audible and visual alarm.

Compressed air is supplied to the air piston operator for the south cell door and to an air-operated shutter used for radiation shielding for the south radiation beam emitted from the horizontal facility.

Outlets are provided to supply compressed air for experiments and for service.

10.3 Warning and Communication Systems

In the event of a fire or radiological emergency, an alarm is sounded. These mechanisms are discussed in the NTR emergency plan.

Several communication systems are available in the facility. The south cell is entered through the control room and communication between the two is direct. An intercom system is used between the control room and the north room and the setup room. Standard telephones may also be used in these areas.

A local public address system, activated from the control room is available as is a site-wide public address system activated from the security building. An open line high-level conference circuit exists between the NTR and the security building. Two-way radios are available for use.

10.4 Fire Protection Provisions

The fire protection equipment and procedures for the NTR facility are part of the conventional industrial-plant fire protection equipment and procedures established for the VNC site. The equipment and procedures are in accordance with company-wide standards, state and local regulations and the recommendations of insurance agencies.

Six-inch (15.2 centimeters) fire mains, on the east and west sides of Building 105, supply outdoor fire hydrants at the northeast, southeast and southwest corners of the building. An extensive sprinkler system is within Building 105. Fire hoses and nozzles are permanently in the hallway and southeast corner of the building. An elevated 500,000-gallon (1.9×10^6 liters) water storage tank is the source of the water for the fire mains. One hundred thousand gallons (380,000 liters) are reserved for fire protection.

Conventional portable fire extinguishers are throughout the NTR facility and Building 105.

A trained site-wide fire brigade provides fire protection for the entire site.

A California Division of Forestry fire station is about 9.6 km west of the site and will respond to fires at VNC. The fire fighters periodically visit the site to familiarize themselves with the facilities.

10.5 Conclusions

The fuel handling system design and storage provisions are acceptable to ensure that reactor fuel can be moved, serviced and stored without danger to operating personnel or the public because of fuel radioactivity or a possible accidental criticality event.

The facility's compressed air system is designed to service the facility acceptably under normal operating conditions. The compressed air is not required to mitigate emergency situations.

The warning and communication systems are acceptable to ensure that sufficient warning can be given of abnormal events and that appropriate communications can be conducted.

The NTR fire protection provisions are consistent with provisions at NRC-licensed non-power reactor facilities.

On the basis of the above findings, the NRC staff concludes that the NTR auxiliary systems can provide the necessary service to the reactor facility to provide acceptable assurance of continued safe operation.

11.0 EXPERIMENTAL PROGRAMS

The NTR serves as a source of radiation for use in the research, development, analytical and commercial programs of GE and its clients. Typical experiments include reactivity worth measurements, radiation effect studies, small and large sample irradiations and neutron

radiography. The TSs provide limitations for the effect on reactivity of all experiments and means for technical and safety review of experiments

11.1 Experimental Facilities

Experimental facilities include:

- One large 5-inch (12.7 centimeters) horizontal (central tube) facility.
- The modular stone monument (MSM) dual neutron radiography facility.
- Miscellaneous horizontal facilities: Seven 4-inch (10.16 centimeters) and one 2-inch (5.1 centimeters) square horizontal penetrations.
- One 5-inch (12.7 centimeters) square vertical facility.
- A fuel loading chute.
- An external thermal column.
- Reactor face facilities.
- The Cable Held Retractable Irradiation System (CHRIS).

11.1.1 Large Horizontal Facility

The large horizontal or central tube facility penetrates the reactor graphite pack along its horizontal axis. This facility provides neutrons in the south cell, reactor cell and MSM and is accessible from the south cell or the reactor cell.

From the south cell, the facility is accessed through an 8-inch (20.3-centimeter) diameter opening passing through the south radiation shield into the thermal column. In the thermal column, the facility decreases to a 5-inch (12.7 centimeters) diameter cavity that continues along the centerline of the fuel core can and is in line with a 24-inch (61 centimeters) diameter beam penetration in the north reactor cell wall. The 5-inch hole contains a removable sleeve that makes it a 3-inch (7.6-centimeter) diameter hole. Within the 3-inch tube is an aluminum source 30.5 inches (77.5 centimeters) in length, which contains graphite, lead and plastic to make the neutron beam uniform. A pinhole is installed to focus the neutron beam for radiography.

In the reactor cell, objects may be irradiated in any position within the sample tube or in the external irradiation beams.

The south cell is provided with an air operated radiation shield shutter while an electrically operated shutter is used at the penetration in the north wall of the reactor cell to the MSM.

11.1.2 Modular Stone Monument Facility

The MSM is a dual neutron radiography facility that provides the capability for neutron radiography of unirradiated and irradiated objects. The MSM is in the north room and is made up of 6 concrete blocks that are the structural and shielding components of the facility. A 12-inch (30.48 centimeters) inside diameter pipe runs vertically through the facility and extends 20 feet (600 centimeters) into the ground beneath the MSM. This pipe is used for neutron radiography of long objects such as fuel elements. A recess on the top of the MSM provides

access to the pipe from large shielding casks for neutron radiography of irradiated objects. A facility on the north end of the MSM is used for neutron radiography of unirradiated objects.

11.1.3 Miscellaneous Horizontal Facilities

Two graphite layers in the main graphite pack have been modified to accommodate up to seven 4-inch (10.2 centimeters) square by 4 feet (120 centimeters) long horizontal penetrations. In addition, a section of graphite 2 square inches (5 square centimeters) by 36 inches (91 centimeters) long below the core can be removed and used for an equipment penetration. Each of the horizontal penetrations is accessible from inside the reactor cell through holes in the north shield wall, which are plugged when the facilities are not being used.

11.1.4 Vertical Facility

The vertical facility is a 5-inch (12.7-centimeter) square aluminum can that penetrates the graphite reflector and is essentially tangential to the east side of the fuel container. When not in use, the facility is filled with a piece of reflector graphite. The facility is accessible only from inside the reactor cell, and irradiations can be performed within the tube or in the external radiation beam emerging from the top of the facility.

11.1.5 Fuel Loading Chute Facility

Removal of the fuel loading chute aluminum clad graphite plug provides access to the inside of the fuel container for irradiations. Irradiations inside the core tank are permitted only at power levels up to 100 watts. The experimental objects are secured to prevent their entry into the core region during operation. This facility is used for experiments to determine the nuclear characteristics of the reactor.

11.1.6 Thermal Column

The thermal column is a 4-foot (1.2-meter) cube of graphite located against the south side of the reactor's graphite reflector. A centrally located plug of graphite 20 inches (50.8-centimeters) square by 4 feet (120 centimeters) long can be removed to accommodate experiments within the thermal column or to provide an external radiation beam. The thermal column south face is accessible from the south cell. Sections of the biological shield consisting of a Boral plate and concrete and lead walls can be removed to provide access to the thermal column face or for use of radiation beams. An air piston operated shutter is installed at the face of the lead brick wall to provide shielding from the horizontal cavity.

11.1.7 Reactor Face Facilities

Radiation from any of the faces of the 5 foot (1.5-meter) cube graphite reflector may be utilized for experimentation. The aluminum box that contains the reflector is provided with 4-foot by 4-foot (1.2-meter by 1.2-meter) removable sections on the top and east faces. Limited space between the reflector and the top shield slab can be used without removing the large concrete plug in the shield. Access to the reactor face facilities is from the reactor cell.

11.1.8 The Cable Held Retractable Irradiation System (CHRIS)

The CHRIS is a dry tube that allows access for a cable held sample carrier to an experiment irradiation position during reactor operation. The experiment position is a horizontal tube in the graphite pack in line with and parallel to the horizontal facility, approximately 6 inches (15 centimeters) above the top of the reactor core can. Samples to be irradiated are placed in an aluminum carrier that slides through the irradiation system. Entrance to the irradiation system is from the north room mezzanine.

11.2 Experiment Review

All new types of irradiation experiments require a written description and analysis of hazards. The experiments must be reviewed and approved by the NTR facility manager or his designated alternative and must have an independent review and approval by the Regulatory Compliance Group (RC) before the irradiation or experiment can be performed. The administrative structure and reporting responsibilities at the NTR and nuclear safety are discussed in Section 12 of this SE.

The NTR TSs limit the kinds and quantities of materials, the effects on reactivity, the physical locations and restraints, and the administrative procedures for review and approval of experiments allowed in the experimental facilities. The TSs cover both fueled and non-fueled experimental materials, and explosive and flammable limits. The TSs are generally consistent with guidance provided by the NRC staff in Reg Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," and Reg Guide 2.4, "Review of Experiments for Research Reactors."

11.3 Conclusions

The NRC staff has determined that the design of the NTR experimental facilities, combined with the review and administrative procedures applied to all of the licensee's research and development activities, is acceptable to ensure that experiments are unlikely to fail, release significant radioactivity to the environment or cause damage to the reactor systems or its fuel. On the basis of its review of the NTR operating experience since 1957 and its evaluation of possible degradation of components as discussed in Section 4.0 above, the NRC staff concludes that there is acceptable assurance that the reactor can continue to operate safely, as limited by its TSs for the proposed duration of the license.

12.0 RADIATION PROTECTION PROGRAM AND RADIOACTIVE WASTE MANAGEMENT

12.1 Radiation Protection Program

The General Electric Company has a structured radiation protection program at the VNC with a health physics staff equipped with radiation detection instrumentation to determine, control and document radiation exposures for the NTR.

NTR monitors both liquid and airborne effluents at the points of release to comply with applicable regulations. The licensee has an environmental monitoring program to verify that radiation exposures in the unrestricted areas surrounding the reactor facility are well within regulations and guidelines.

12.1.1 ALARA Commitment

GE has established and implemented the policy that all operations are to be planned and conducted to keep all radiation exposures as low as is reasonably achievable (ALARA). This policy is implemented by a set of guidelines and procedures. Reviews of the program and radiation exposures are regularly done. The licensee has committed that all proposed experiments and procedures at the reactor are reviewed for ways to minimize the potential exposures of personnel. All unanticipated or unusual reactor-related exposures are investigated by both the health physics and operations staffs to develop methods to prevent recurrences.

12.1.2 Health Physics Staffing

The radiation protection program for the NTR is the responsibility of the RC. Its staffing includes professional health physicists and technicians. Its staffing level is dependent on the activity at the site as a whole. In addition to health physics monitoring and nuclear safety engineering oversight, the health physics staff is available for consultation. The onsite staff has sufficient training and experience to direct the radiation protection program for the NTR. In addition, the health physics staff has been given the responsibility, the authority and acceptable lines of communication to implement and conduct an effective radiation protection program within the guidelines promulgated by the VNC Safety Standards (See Section 13.4 of this SE).

12.1.3 Health Physics Procedures

The licensee has prepared written procedures that address health physics activities and the support that the health physics staff is expected to provide for routine operation of the NTR. These procedures identify the interaction between the health physics staff and the operational personnel and experimenters. They also specify administrative limits and action points, as well as appropriate responses and corrective action for use when these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staffs, as well as the health physics and administrative personnel.

12.1.4 Health Physics Training

All NTR reactor facility personnel receive indoctrination in radiation safety before they assume their work responsibilities. Additional radiation safety instruction is provided to those who will work directly with radiation or radioactive materials. This training program is designed to identify the particular hazards of each specific type of work to be undertaken, as well as methods to be used to mitigate the consequences of these hazards. The licensee also provides retraining in radiation safety and periodically reviews health physics and personnel protection practices in the reactor operator requalification program.

12.1.5 Radiation Sources

The major radiation sources at the reactor facility that the radiation protection program has been established to guard against are as follows.

12.1.5.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, water purification system including the demineralizer system and after-filters, radiation from the primary piping containing coolant with nitrogen-16 and sodium-24 and radioactive gases, primarily argon-41.

The reactor fuel is contained within the aluminum cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water, graphite and concrete shielding. In addition, the water purification system is entirely enclosed in the reactor cell in a shielded area.

The primary coolant system is in the reactor cell that provides shielding against the nitrogen-16 that is generated during operation. The small amounts of nitrogen-16 produced do not require a nitrogen-16 control system. The sodium-24 in the primary coolant is also at a low enough level not to require special treatment.

The licensee has measured the radiation levels in the NTR facility while the reactor is operating at its maximum power level of 100 kW with shutters open and closed. All workers in the area are acceptably monitored to ensure compliance.

Personnel exposure to the radiation from chemically inert argon-41 is limited by dilution and prompt removal of this gas from the reactor cell. The gas is then discharged to the atmosphere from the elevated stack where it is further diluted and diffused before reaching occupied offsite areas. Section 12.2.1.3 of this SE further discusses gaseous radioactive waste.

12.1.5.2 Extraneous Sources

Sources of radiation that may be considered as incidental to normal reactor operation, but associated with reactor use, include radioactive isotopes produced for research, activated components of experiments and activated samples or specimens.

Personnel exposure to radiation from intentionally produced radioactive materials, as well as from required manipulation of activated experimental components, is controlled using reviewed and approved operating procedures that use the standard protective measures of time, distance and shielding.

12.1.6 Routine Monitoring

Aspects of the NTR radiation protection program concerning the routine monitoring of radiation are discussed in this section of the SE.

12.1.6.1 Health Physics Instrumentation

The NTR facility has a variety of detection and measurement instruments available to monitor potentially hazardous radiation. Established instrument calibration procedures and techniques ensure that any credible type of radiation and any significant radiation intensities will be promptly detected and measured.

12.1.6.2 Fixed-position Monitors

The NTR facility uses 5 fixed position radiation monitors in addition to portable monitors. These 5 monitors are gamma ray sensitive. These monitors are placed at strategic locations in the facility where radiation levels might be significant or where increases might indicate abnormal or hazardous conditions. These locations include the reactor cell, south cell, control room and two in the north room. Each channel is equipped with an audio and visual alarm in the control room and the affected area.

The exhaust stack monitoring system separately measures the activity of both radioactive gases and particulates in the effluent from the reactor cell. A continuous sample is drawn from the discharge of the stack. This exhaust stack monitoring system is used to initiate isolation of the confinement building as discussed in Section 7.1 of this SE.

A continuous air monitor (CAM) with readout in the control room samples the air in the reactor cell. This CAM and the area monitors can provide information for manual isolation of the confinement.

12.1.6.3 Experimental Support

The health physics staff is involved in the review and approval process for experiments at the reactor facility. The health physics staff participates in experiment planning by reviewing and approving all proposed new experiment types. The health physics approval specifies the type and degree of radiation safety support required for each experiment activity.

12.1.6.4 Non-routine Tasks

One-of-a-kind, short-term tasks (such as non-routine maintenance activities) are occasionally performed in potential radiation or contamination areas. Depending on the character of the task, a new standard operating procedure (SOP), engineering release (ER) or change authorization (CA) (See Section 13.4 of this SE) will be developed and subject to a health physics review by RC. The SOP, ER or CA will specify health physics coverage required for the work.

12.1.7 Occupational Radiation Exposure

The NTR personnel monitoring program is described in a VNC Safety Standard. To summarize the program, the licensee measures personnel exposures using film badges, neutron albedo dosimeters and self-reading pocket dosimeters assigned to individuals who might be exposed to radiation. In addition, thermoluminescent (TLD) finger rings, high-range self-reading pocket dosimeters and alarming dosimeters are prescribed for extremity exposure and high dose rate

exposure. Instrument dose rate and time measurements are used administratively to keep occupational exposures below the applicable limits of 10 CFR Part 20.

In addition to monitoring personnel exposure to external radiation, potential internal deposition of radionuclides is monitored by whole body counting.

Radiation exposure to NTR personnel varies from year to year. However, the annual exposure, based on records, is from 1.5 to 2.5 Rem per person. Non-NTR personnel, working in the same facility, based on records, are predicted to receive <100 mrem per year. Personnel providing service for the NTR are predicted to have a total annual exposure from all site sources of <1.0 Rem with <30percent attributable to NTR exposure. The result of applying the ALARA principle is evident in the licensee's control of personnel exposures.

12.1.8 Effluent Monitoring

Argon-41 is the primary airborne radioactive effluent during normal operation of the NTR. Gas is continuously swept from the reactor cell and discharged from the 45-foot (14-meter) high stack. The stack exhaust monitoring system measures the radioactivity in the effluent discharged from the north room, reactor cell and south cell. The monitoring system is discussed in Section 12.1.6.2. A periodic calibration of the monitoring system enables the licensee to maintain continuous direct measurements of the principle airborne radioactive effluent.

No radioactive liquids are directly released from the reactor facility during normal operation. All potentially radioactive liquids are collected and eventually placed in tanks along with other VNC liquid radioactive waste. The waste is subsequently disposed of in accordance with approved site practices and procedures. The liquid radioactive waste generated at the NTR is approximately 1 liter per year.

12.1.9 Environmental Monitoring

GE has an extensive environmental monitoring program and the results are transmitted to the NRC in the licensee's annual operating report. Areas outside the VNC are monitored by environmental air samplers and TLD stations. In addition, GE samples the water, vegetation, soil and stream bottom sediment and analyzes radiological and chemical content. The environmental monitoring program is described in the Nuclear Safety Manual, Volume II and in VNC Safety Standards (See Section 13.4 of this SE). Since initial reactor commissioning, the program has not detected any reactor-related radioactive or chemical contamination in the unrestricted area.

12.2 Radioactive Waste Management

Radioactive waste resulting from reactor operations is either discharged to the environment in gaseous form, transferred as a liquid to VNC tanks, or packaged as solids and released to licensed waste processors or disposal facilities all in accordance with applicable regulations. Further, the NTR administration and staff follow the principles of their ALARA policy in handling radioactive materials and in considering the release of such materials to the unrestricted environment.

12.2.1 Waste Generation and Handling Procedures

Operation of the NTR and conduct of the facility experimental program generates liquid, solid and airborne radioactive waste.

12.2.1.1 Liquid Waste

Normal operation of the NTR produces only a small amount (about 1 liter) of liquid waste per year. However, some research activities also generate limited volumes of liquid waste.

Radioactive liquids generated by the NTR and related operations are collected in the reactor cell sump and are then pumped to a 500-gallon (1900-liter) holdup tank. Portable waste tanks are available if larger quantities of radioactive liquids are handled.

Radioactive liquids are transferred to the Waste Evaporator Plant where they are disposed of in accordance with approved site practices and procedures.

12.2.1.2 Solid Waste

Occasionally, low-level solid waste results from reactor operations and the experimental program. This may consist of experimental residues, purification system resins and filters, potentially contaminated paper and gloves and occasional small, activated components. The annual amount of solid waste produced by reactor operations and utilization is very small and estimated to be 1-3 ft^3 (0.03-0.09 m^3). The radioactive content is measured in millicuries. Solid waste is collected and initially packaged by reactor personnel. It is then transferred to the Remote Handling Operation where it is inspected and prepared for shipment to commercial facilities for volume reduction or disposal in accordance with approved procedures and applicable regulations. Disposal of spent NTR fuel assemblies has not been required.

12.2.1.3 Airborne Waste

Radioactive waste is principally produced by the neutron irradiation of the coolant water and air dissolved in the coolant water as well as the air and airborne particulates in the thermal column, graphite pack and experimental facilities. The potential airborne radioactive wastes during normal operation include gaseous nitrogen-16, neutron-activated dust particles, and argon-41. The nitrogen-16 that escapes the primary coolant system is very small, as is the activation of dust particles. The predominant airborne waste from the reactor is argon-41.

The reactor is not allowed to operate, except at low power to search for a leaking element, if fission products escape from the fuel cladding during normal operations.

Occupational exposure of personnel in the restricted area as a result of airborne radionuclides is limited by a ventilation system that constantly sweeps the air from the north room, reactor cell and south cell. The air is discharged to the stack through a pre-filter and a bank of absolute filters. Most of the airborne particulates are removed. Because of its short half-life, nitrogen-16 decays to insignificant levels before it can be released. The exhaust is monitored for residual radioactivity before discharge through the stack. The exhaust system is designed to operate during normal operation and accident conditions.

On the basis of calibrated radioactivity monitors, the licensee states that 1996 was a typical year and that the annual releases from the NTR stack were as follows:

- Argon-41 — 70 Ci (2.5 x 10^{12} Bq)
- Iodine-131 — 31.1 Ci (1.13 x 10^{6} Bq)
- Gross beta activity — 1.6 ci (5.8 x 10^{4} Bq)
- Gross alpha activity — 0.056 ci (2.04 x 10^{3} Bq)

Argon-41 is clearly the predominant airborne waste released to the environment.

The committed effective dose equivalent due to exposure to the releases from the entire VNC site is routinely very low, less than 1 mrem per year with a thyroid dose due to iodine less than 1 mrem per year.

12.2.2 Potential Dose Assessments

Natural background radiation levels in the San Francisco Bay Area result in an exposure of about 50-100 mrem/year to each individual residing there. An additional 8 percent (about 8 mrem/year) will be received by those living in a brick or masonry structure. Medical diagnosis and therapy x-ray or other radiation will add to these natural background radiations, increasing the total cumulative annual exposure.

As noted above, argon-41 and nitrogen-16 are the two principal airborne radionuclides formed during routine operation of the NTR. Nitrogen-16 decays with a half-life of 7 seconds. Therefore, no significant quantities of nitrogen-16 escape or are released from the reactor cell. The licensee has estimated (using applicable methods) the formation of both nuclides in various reactor operations. The licensee also monitors the release of argon-41 and other possible radioisotopes from the exhaust stack (see Section 12.2.1.3).

12.2.2.1 Unrestricted Area

The licensee has established a total effective dose equivalent to a member of the public of 10 mrem in any one year for all releases from the VNC. Because of releases from other VNC facilities, the NTR release is further divided by a factor of two. Using the airborne radioactive material concentration limits provided by 10 CFR 20, Appendix B, Table 2, Column 1 and calculated dilution-dispersion factors, the licensee has determined the annual averaged release rates for the allowable boundary concentrations. The associated stack release action levels have been developed by the licensee for noble gases, halogens, alpha emitters and beta emitters.

The NRC staff concludes that the licensee has developed procedures for monitoring argon-41 and other potential releases, as well as methods for evaluating potential doses from finite clouds in the unrestricted area that are acceptable. The projected annual doses for continued operation of the reactor are well within the regulatory limits of 10 CFR Part 20 and acceptably protect the public or the environment.

12.2.2.2 Restricted Area

Radiation exposure control is achieved at the NTR by several methods including shielding and the ventilation system.

Based on dosimetry and survey records, the licensee has determined that it is unlikely that any individual would receive, in one year, an intake in excess of 10 percent of an Annual Limit on Intake of Table 1, Columns 1 and 2 of Appendix B to 10 CFR Part 20. Therefore, committed effective dose equivalent is not typically added to the external dose for determination of total effective dose equivalent. The intake of radioactive materials during work in restricted areas is limited using a ventilation system, access control, working time limitations and other measures.

The licensee provided information concerning the sources of argon-41 and nitrogen-16 in the restricted area during normal operation of the reactor. The nitrogen-16 is produced by a fast-neutron activation from the oxygen-16 in the primary coolant water. The reactor is designed so that the coolant is pumped up through the core in a "closed" loop that is vented to the atmosphere through a hold up tank. Nitrogen-16 does not escape into the restricted area and is not an exposure problem in either forced or natural convection operating modes.

The licensee has determined that the dose rate in the facility due to argon-41 is extremely small because of the control measures discussed above and would not be expected to lead to significant occupational doses to the facility's staff or users of the reactor. Dosimetry and monitoring records confirm this.

12.3 Conclusions

The NRC staff concludes that radiation protection receives appropriate support from the NTR and VNC administration. In addition to other guidance, the NRC staff considered the guidance of ANSI/ANS 15.11, 1993, "Radiation Protection at Research Reactor Facilities." On the basis of this review, the NRC staff reached the following conclusions:

- The NTR radiation protection program is acceptably staffed and equipped.
- The NTR reactor health physics staff has acceptable authority and lines of communication.
- The radiation protection procedures are integrated into the research plans.
- Surveys verify that operations and procedures achieve ALARA goals.
- The effluent monitoring programs and environmental monitoring programs, conducted by personnel from RC and the NTR staff, are acceptable to identify significant releases of radioactivity promptly and to predict maximum exposures to individuals in the unrestricted area. (These measured and predicted maximum levels are a very small fraction of the levels permitted by the applicable regulations and guidelines specified in 10 CFR Part 20.)
- There is acceptable assurance that NTR personnel and procedures will continue to protect the health and safety of the public, the facility staff and the environment from significant radiation exposures related to normal reactor operations.
- Waste management activities at the NTR facility have been conducted and can be expected to continue to be conducted consistent with both 10 CFR Part 20 and ALARA principles.
- The NTR systems and procedures limit the production of argon-41 and nitrogen-16 and control potential exposures of facility staff. Measurements, conservative computations and analysis of the quantities of these gases released beyond the limits of the reactor facility

give acceptable assurances that the potential doses to the public would not be significant, even if there were a major increase in the operating schedule of the reactor.

13.0 CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operation and the facility emergency and security plans. The administrative aspects of facility operations are the facility organization, training, operational review and audits and procedures.

13.1 Overall Organization

Responsibility for the safe operation of the NTR is vested within the chain of command. The Manager, NTR, is the Facility Manager and has overall responsibility for the safe, reliable and efficient operation of the NTR. The Facility Manager is also responsible for development, maintenance of the facility, implementation of procedures, training and retraining of operating personnel and the safety of personnel and visitors.

The facility manager reports to the Manager, Vallecitos and Morris Operation.

13.2 Training

The training of reactor operators is conducted by GE personnel. The Facility Manager administers the program. Technical Specification 6.1.4 requires that the selection, training and requalification of operations personnel shall meet or exceed the requirements of:

- ANSI/ANS 15.4-1977, "Selection and Training of Personnel for Research Reactors," Sections 4 through 6.
- 10 CFR Part 55.
- The latest revision of the Facility Operator Requalification Program.

13.3 Operational Review and Audits

The RC has responsibility to ensure that the use of radioactive materials and radiation producing devices, including the reactor, is conducted safely with minimum impact on site personnel and the public. This same group has responsibility for ensuring that the reactor is operated in compliance with the facility's license and applicable regulations. The RC conducts independent appraisals of reactor operations and performance of the reactor program.

The Vallecitos Technological Safety Council (VTSC), an organization external to the NTR, performs independent reviews required by the TSs. This group, when utilized, advises the Facility Manager on safety matters affecting any of the operations or activities on the VNC site.

The Safety Analysis Report and the TSs outline the qualifications that RC and VTSC members must possess and discuss operational aspects of the two groups.

The RC reviews the following aspects of reactor operations:

- All proposed procedures required by the TSs and proposed changes to such procedures.
- Proposed types of experiments, facility modifications, and facility procedures.
- Proposed changes to the facility operating license including the TSs.
- Violations of federal regulations, TSs, facility-license requirements and safety significant internal procedures.
- Unusual or abnormal occurrences, which are reportable to the NRC, as required by federal regulations or the TSs.
- Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affects nuclear safety.
- Periodic audits of facility operation, maintenance and administration to include:
  (1) The conformance of facility operation to the federal regulations, TSs and facility license requirements.
  (2) The results of all actions taken to correct deficiencies or increase effectiveness in facility equipment, structures, systems or methods of operation that effect nuclear safety.
  (3) The facility emergency procedures, security plan, requalification program and their implementation.

The VTSC regularly reviews the following:

- The results and actions for all reportable incidents.
- Non-reportable incidents at the discretion of the VTSC.
- Proposed new facilities or changes to facilities that contain unreviewed safety questions.
- Change authorizations when requested by management or nuclear and industrial safety groups.
- Any other matter that it determines to be of safety importance.

Identified deficiencies that affect reactor safety are immediately reported to the Facility Manager who is responsible for the reports to the RC, VTSC and the NRC.

13.4 Procedures

The licensee has developed a comprehensive set of written operating procedures for all aspects of reactor facility operations. These procedures are in four categories as follows.

13.4.1 VNC Safety Standards

These standards, which are reviewed and accepted by the Facility Manager, are established for protection against hazards arising from activities under licenses issued by appropriate regulatory authorities and provide guidance for complying with the several licenses and regulations governing the facilities, activities and materials at the VNC. Many of the standards govern the general radiation protection practices at the site.

13.4.2 Standard Operating Procedures (SOPs)

SOPs have been established to delineate administrative and operational requirements to comply with NRC regulations and the facility license.

The SOPs address:

(1) Normal startup, operation and shutdown of the reactor and all pertinent systems and components as specified by the Facility Manager involving nuclear safety of the facility.
(2) Defueling, refueling and fuel transfer operations.
(3) Safety related preventive or corrective maintenance.
(4) Abnormal occurrences for which an alarm is received.
(5) Response to abnormal reactivity change.
(6) Surveillance, testing and calibrations required by the TSs.
(7) Emergency conditions involving potential or actual release of radioactive material.
(8) Radiation protection consistent with 10 CFR Part 20 requirements.
(9) The review and approval of changes to all required procedures.
(10) Security, operator requalification, emergency plan, and others as required by the Facility Manager.

The SOPS are reviewed by the RC and reviewed and approved by the Facility Manager. Minor modifications to SOPs that do not change their original intent may be made and documented by assigning a minor review number to the change. Temporary changes to SOPs may be made by an Engineering Release (see next section), but for no longer than six months. A senior reactor operator may authorize deviations from SOPs during an emergency to prevent injury to personnel or damage to the facility as allowed by 10CFR50.54(x) and (y). Such deviations must be documented and reported to the Facility Manager.

13.4.3 Engineering Release (ER)

An ER is issued to request work, establish temporary procedures or instructions, distribute information, document actions and other reasons to ensure safe, efficient operation of the NTR. The ERs are reviewed and approved in accordance with the SOP covering ERs. Independent review by the RC is required for activities listed in Section 13.4.2 of this SE.

13.4.4 Change Authorization (CA)

CAs are required for changes to the facility or to the SAR. The CA provides the documented description and safety evaluation required by 10 CFR Part 50.59. CAs are required for changes, activities and projects that involve significant safety considerations or unreviewed safety questions. CAs are controlled by a SOP and are independently reviewed by the RC. The Facility Manager approves CAs.

13.5 Emergency Planning

The NRC staff reviewed the "Emergency Response Plan for the VNC Site" dated March 1996. The NRC staff concluded that this plan maintains acceptable compliance with applicable portions of Appendix E to 10 CFR Part 50.

13.6 Physical Security Plan

VNC has established and maintains a program designed to protect the reactor and its fuel and to ensure its security. Accordingly, the NRC staff reviewed the plan submitted in June 1996.

The NRC staff concludes that the plan acceptably meets the applicable requirements of 10 CFR Part 73. The NTR Physical Security Plan is withheld from public disclosure under 10 CFR Part 2.790(d)(1). The amendment renewing Facility Operating License R-33 incorporates the physical security plan as a condition of the license.

13.7 Quality Assurance Program

Design and construction of new (and modification of existing) structures, systems and components important to safety are subject to a comprehensive quality assurance program. The objective of the program is to maintain an assurance of quality of the scram systems and safety-related systems of the NTR. The program includes the RC that is organizationally independent of reactor operating functions. It has full authority and responsibility to identify, evaluate, and recommend solutions to quality and safety-related problems.

13.8 Conclusions

On the basis of the above discussions, the NRC staff concludes that the licensee has sufficient experience, management structure and procedures to provide acceptable assurance that the NTR will continue to be managed in a way that will not cause any significant radiological risk to the health and safety of the public.

14.0 ACCIDENT ANALYSIS

To help establish safety limits, limiting safety system settings (LSSSs), and limiting conditions for operation (LCO) of the NTR, the licensee analyzed potential reactor transients and other hypothetical accidents. Specifically, the licensee analyzed the potential effects of such events on the reactor fuel and the health and safety of the public. The NRC staff then evaluated the licensee’s analytical assumptions, methods and results.

None of the credible accidents postulated would lead to the failure of the cladding of any fuel assembly or the uncontrolled release of fission products. However, the licensee postulated an enveloping event involving a failure in a fueled experiment. This event would lead to the maximum potential radiation hazard to personnel. The licensee makes no assumptions as to the cause of the failure. The licensee evaluated only the potential consequences of this event and not the likelihood or mechanisms for occurrence. The NRC staff has designated this event to be the maximum hypothetical accident (MHA). In addition, because the initiation and planned operation of the confinement system are relied upon to mitigate the consequences, this event may also be considered an experiment design basis accident.

The licensee and the NRC staff considered the MHA and, in addition, considered the following types of potential initiating events:

- MHA (Experiment Design Basis Accident).
- Loss of electric power.
- Loss of secondary cooling.
- Loss of facility air supply.
- Inadvertent start of primary pump.

- Fuel handling errors.
- Uncontrolled reactivity increases.
- Loss of primary coolant flow.
- Rod withdrawal.
- Loss of primary coolant.
- Flammable or explosive device.
- Criticality accidents in storage.

14.1 MHA (Experiment Design Basis Accident)

The licensee analyzed the postulated failure of a fueled experiment in the reactor. The licensee established limits on the maximum mass of plutonium (Pu) and uranium (U) an experiment can contain and the maximum fluence for the experiment. The limits on these parameters are chosen so that the failure would result in acceptable doses in restricted and unrestricted areas. Based on the demonstrated calculation technique, the licensee has proposed TSs on the radioactive material content, including fission products, of all experiments.

The licensee has developed the material quantity limits, clad requirements, operating limits and required safety equipment for irradiation experiments involving single and double clad, fueled capsules at the NTR. The development is based on the radiological criteria given in Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," and the capabilities of the facility and the site to accommodate a radioactive material release from the experiment. The limits are dependent on the physical form of the material and the performance of the confinement system at the reactor.

14.1.1 Accident Description

Based on Regulatory Guide 2.2, Part c.2.a, the event consists of ". . . a single mode nonviolent failure of the encapsulation boundary that releases all radioactive material into the immediate environment of the experiment or to the reactor building as appropriate . . ." In addition it states that "the analysis should establish the most probable trajectory of the material, if any, into restricted and unrestricted areas. Credit for natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms."

The licensee has described the accident as follows:

(1) Experiment material is Pu-239 or U-235. Limits are established for both single and double encapsulation and for both loose powder form and sintered oxide pellet form resulting in four different cases with independent limits.

(2) The most probable trajectory of the released material is from the experiment location to the reactor cell area. Since the event is a single-mode nonviolent failure, the NTR ventilation system will be considered to be operational. The airborne material will be exhausted from the reactor cell area through the confinement system consisting of the ventilation system with a HEPA filter bank and out the NTR stack. Experiments with a potential for release that would not be released via the reactor cell will be provided with

a close-capture system to ensure release is through the HEPA filters. For purposes of this evaluation, the installed HEPA filter system filtration efficiency for 0.3 micrometer diameter particles is conservatively assumed to be 99 percent.

(3) Release fractions of the Pu fuel and fission products to the reactor cell and to the environment are assigned for powder and pellet form (Table 14-1).

(4) Dose limits for single and double encapsulation samples, whole body and critical organ, and restricted and unrestricted areas have been assigned (Table 14-2 and 14-3).

(5) The 2-hour unrestricted area exposure at the nearest site boundary is calculated for a NTR stack release under Type F meteorological conditions for a wind velocity of 1 m/second.

(6) The restricted area exposure will result from the submersion in and inhalation of 3.06 x 10^{-3} percent of the isotopes released from the NTR stack for 5 minutes assuming that:
   (a) The total release will be uniformly distributed over the 2 hours following the experiment failure;
   (b) The fission products from this release will cause high activity alarms on the stack monitor at preset level that will limit occupational exposure to 520 Maximum Permissible Concentration (MPC) hours, the maximum allowable quarterly average; and
   (c) The operator will respond to the alarm and announce an area evacuation to an upwind location.

The NRC staff found the description of the accident proposed by the licensee acceptably conservative for the MHA analysis.

Table 14-1
RELEASE FRACTIONS OF THE PU FUEL AND FISSION PRODUCTS TO THE ENVIRONMENT

Release from capsule to the reactor cell:	**Powder (%)**	**Pellet (%)**
Pu-239	100	0
Noble Gas	100	100
Iodine	100	25
All remaining fission products	100	0
Release from the reactor cell to stack:		
Pu-239	1	1
Noble Gas	100	100

Table 14-1 RELEASE FRACTIONS OF THE PU FUEL AND FISSION PRODUCTS TO THE ENVIRONMENT		
Iodine	100	100
All remaining fission products	1	1

Table 14-2 DOSE LIMITS		
	Single Encapsulation	**Double Encapsulation**
2-hour Fence-Post Man	0.05 Rem	0.5 Rem or 1.5 Rem to Thyroid
Operator, During Evacuation	0.5 Rem	5 Rem or 30 Rem to Thyroid

Table 14-3
MAXIMUM ALLOWABLE ORGAN DOSE FOR SINGLY CLAD EXPERIMENTS

Organ	Unrestricted Area (Boundary) (Rem)	Restricted Area (NTR Stack) (Rem)
Total Body	0.05	0.5
Kidneys	0.15	1.5
Liver	0.15	1.5
Bone	0.3	3.0
Lungs	0.15	1.5
Thyroid	0.15	3.0
Stomach	0.15	1.5
Small Intestines	0.15	1.5
Upper Larger Intestines	.015	1.5
Lower Large Intestines	0.15	1.5
Skin of Body	0.3	3.0

14.1.2 Calculation Method

Using the above data and assumptions, the licensee used computer code DOSE77 to calculate organ doses resulting from the inhalation and submersion in a cloud of radioactive materials. RIBD, an optional part of DOSE77 calculates the fission product inventory that results from irradiation of fissionable material. The input to the Code to calculate the fission product inventory and organ doses resulting from exposure to the released isotopes from a 1 gram Pu

experiment in a thermal flux of 10^{12} neutrons per square centimeter-second (n/cm^2-sec) for one day is presented on pages 13-39 and 13-40 of the licensee's Safety Analysis Report.

The NRC staff found the assumptions proposed by the licensee acceptably conservative. Because the postulated accident could lead to consequences to the public, the NRC staff reviewed and evaluated the analytical methods of the licensee. The NRC staff concludes that the licensee has acceptable techniques to analyze such events.

14.1.3 Results

The licensee ran DOSE77 to determine doses, using a unit quantity of Pu, for the cases of single and double encapsulation, powder and pellet form, the 510-meter boundary (unrestricted area) and within the building (restricted area). These were compared with the maximum allowable doses given by Regulatory Guide 2.2 and the International Commission on Radiological Protection No 9. Based on this comparison, maximum quantities of Pu as a function of neutron fluence were determined for each condition based on the critical organ dose.

The licensee performed similar calculations for U-235.

The results are presented in Table 14-4 and 14-5. As a result of these analyses, the licensee has established TS 3.5.3.12 and TS 3.5.3.13 concerning the radioactive material content for single and double encapsulated experiments.

The NRC staff concludes that the computational approach and the projected quantities and fluences for Pu and U are acceptable for this very unlikely event. In addition, the NRC staff concludes that the licensee could analyze similar experiments containing radioactive materials and fission products other than Pu and U to assure compliance with the above TSs.

Table 14-4 SINGLE CLAD NTR EXPERIMENT Pu-239 LIMITS BASED ON CALCULATED ORGAN DOSES AND ALLOWABLE DOSES				
Organ	Calculated Limits, Grams Pu-239			
	Based on Pellet-form		Based on Powder-form	
	Boundary	At Stack	Boundary	At Stack
Total Body	10.35	1.98	1.09	0.26
Kidneys	104.17	25.64	0.98	0.24
Liver	173.81	42.86	0.31	0.078[a]
Bone	686.5	170.45	0.40	0.10
Lungs	42.37	10.71	0.59	0.15
Thyroid	0.93	0.46[b]	0.23	0.11
Skin,	39.53	6.33	19.48	3.69

Note: The gastrointestinal doses were all generally small compared with the limits.

[a] Powder-form capsule (single clad) limit=0.078 g (~5 mCi) Pu-239.

[b] Pellet-form capsule (single clad) limit=0.46 g (~30 mCi) Pu-239.

Table 14-5 PU-239 AND U-238 EXPERIMENT LIMITS				
Capsule Form	Pu-239 Limits		U-238 Limits	
	Quantity (g)	24-hour Power (watts)	Quantity (g)	24-hour Power (watts)
Single-Clad Solid Pellet	0.46	28	0.60	28
Powder	0.078	5	0.15	7
Double-Clad Solid Pellet	4.6	280	6.0	280
Powder	0.78	50	1.5	70

Note: Assumes a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec.

14.2 Loss of Electric Power

The licensee analyzed the consequences of a loss of electrical power at the NTR facility and found no unacceptable consequences.

The battery powered emergency lights are the only emergency power supplies at the NTR facility. Loss of electrical power will result in a scram of the reactor based on several processes including loss of power to the safety rod electromagnets. While the control rod and safety rod drives would not be inserted automatically because of the lack of ac power, the safety rods would be inserted by the action of their springs to shut down the reactor and maintain the shutdown condition.

The NRC staff has reviewed the NTR analysis of a loss of electrical power and considers the results and analysis to be acceptable. On the basis of these considerations, the NRC staff concludes that a loss of electrical power will not lead to unacceptable results and will lead to the safe shutdown of the reactor.

14.3 Loss of Secondary Cooling

The licensee has analyzed the consequences of a loss of secondary cooling and has found no unacceptable results.

Secondary coolant flows by gravity through the tube side of the primary heat exchanger. Loss of secondary cooling when the reactor power level is high enough to produce an appreciable heating rate will cause the reactor to scram from high primary coolant temperature. If the heating rate is not high enough to cause a scram quickly, the loss of secondary coolant will be evident to the operator from control room indicators including a reactivity change.

The NRC staff has reviewed the NTR analysis of the loss of secondary coolant and considers the results and analysis to be acceptable. The NRC staff concludes that a loss of secondary coolant will have minimal effects on the reactor and will lead to a reactor shutdown by its effect on the primary coolant temperature or by operator action. It will not lead to fuel temperatures that would cause loss of integrity of the fuel cladding.

14.4 Loss of Facility Air Supply

The licensee has analyzed the consequences of a loss of facility air supply and has found no unacceptable results.

The facility compressed air supply is used to operate the air piston for the south cell door and the radiation shield shutter for the horizontal facility in the south cell. One person can manually move the south cell door. The shutter would remain in the position it was in at the time of the air-supply failure.

The NRC staff has reviewed the NTR analysis of the loss of facility air supply and considers the results and analysis to be acceptable. The NRC staff concludes that a loss of the facility air supply will have minimal effect on the NTR facility.

14.5 Inadvertent Start of Primary Pump

The licensee has analyzed the consequences of an inadvertent start of the primary pump and has found no unacceptable results.

If the primary pump were inadvertently started, the effect would be to change the reactor inlet temperature. A decrease in the temperature will cause the reactor power to drop if it is below 124°F (51°C) because of the positive temperature coefficient. For the same reason, an increase in the temperature will cause the reactor power level to rise until 124°F (51°C) is reached. The amount of positive reactivity that could be added is less than 0.10$ from room temperature to turnover temperature. After this temperature, the power level will begin to drop.

The licensee has shown (see Section 14.7, below) that a step insertion of 0.76$ of reactivity would not cause fuel damage even if the reactor failed to scram on high power level. Therefore, the licensee concludes that a transient caused by a small amount of reactivity (<0.10$) from the temperature increase would also not cause fuel damage.

The NRC staff has reviewed the NTR analysis of the inadvertent start of the primary pump and considers the results and analysis to be acceptable. The NRC staff concludes that this transient is bound by the reactivity insertion accident and will not lead to fuel temperatures that would cause loss of integrity of the fuel cladding.

14.6 Fuel Handling Errors

The licensee analyzed a fuel handling error based on the following considerations. Sixteen fuel assemblies are available for the operation of the NTR. These 16 assemblies completely fill the core reel assembly and are used for operation. The only other available space for a fuel assembly is in the core-loading chute. An element in the loading chute results in a less reactive core configuration than the cylinder formed by having all elements in the core support reel. Dropping a fuel element could only cause an accident if the control and safety rods were withdrawn during loading so that the reactor was almost critical before the element fell into the reel assembly. Such a condition is contrary to operating procedures and requires errors by the console operator and fuel loaders.

In addition to having all existing NTR fuel assemblies in their most reactive configuration in the core, additional safety features ensure safety during all phases of fuel handling. These are:

- Reactor design, fuel handling equipment and administrative controls are such that not more than two assemblies can be handled at once.
- All fuel movement must be performed in accordance with written procedures.
- By using the manually positioned poison sheets, the core can be made subcritical by 6.1$. Removal of the graphite plug from the fuel-loading chute provides negative reactivity of about 1.25 percent.
- Movement of the source and special nuclear material within the NTR facility must have the approval of the licensed operator on duty.
- Any storage arrangements used will be analyzed to ensure a subcritical configuration.

The licensee concludes that a fuel handling error resulting in a large increase in reactivity is unlikely because of administrative procedures, operating procedures and physical characteristics of the NTR design.

The NRC staff has reviewed the licensee's analysis of fuel handling errors and considers the results and analysis to be acceptable. On the basis of these considerations, the NRC staff concludes that a fuel handling error would not lead to fuel temperatures that would cause loss of integrity of the cladding. In addition, the NRC staff finds acceptable assurance that the health and safety of the public would not be endangered by a fuel handling error.

14.7 Uncontrolled Reactivity Increases

The licensee analyzed transients resulting from step reactivity insertions up to 2.0$ and finite ramp reactivity insertions up to 4.0$ using a range of initial reactor power levels and primary coolant flow rates. The NRC staff concludes that if the reactivity addition caused by control rod and experiment movement is sufficiently large, the resulting power excursion, if not terminated by a scram, could result in fuel melting. Therefore, the NTR will be operated in a way that limits the potential excess reactivity to less than that required to cause fuel damage, assuming failure to scram.

The calculations performed for step and ramp increases produced similar results. The licensee calculated that for a 0.76$ reactivity insertion during operation at 100 kW with forced cooling, the reactor power level peaked at 4000 kW and that the fuel temperature peaked at 255°F (124°C), assuming failure to scram. The licensee also calculated the results of a 0.76$ reactivity insertion during operation at source level with an initial inlet water temperature as low as 65°F (18°C) and a failure to scram. For these assumed conditions, the reactor power level peaked slightly above 4000 kW and the fuel temperature peaked around 255°F (124°C). For these calculations, the reactivity addition from the initially positive temperature coefficient was included.

The licensee believes that these transient calculations are extremely conservative since no credit is taken for the negative reactivity feedback from subcooled voids during nucleate boiling. Based on this, the licensee estimates that a realistic limit for reactivity insertions would be about 0.90-1.0$. The licensee has conservatively established TS 3.1.3.1 limiting the potential excess reactivity for all reactor core configurations to 0.76$.

It is not considered credible that the poison sheets would fail. The licensee has provided information on the poison sheets to show that they are securely held in place and will not result in a reactivity insertion event even for a seismic event.

The NRC staff has reviewed the uncontrolled reactivity increase accident and considers the methods, results and analysis to be acceptable. On the basis of these considerations, the NRC staff concludes that by limiting the potential excess reactivity of the reactor to 0.76$, a power excursion will not lead to fuel temperatures that would cause loss of integrity of the fuel cladding. There is acceptable assurance that the health and safety of the public would not be endangered by such an event.

14.8 Rod Withdrawal

The licensee analyzed the transient caused by the simultaneous withdrawal of all control rods. For this to occur, interlocks must fail and the operator must deviate from normal operating procedures. In addition, the reactor period and power level scrams must malfunction.

The licensee concludes that since this reactivity addition will be limited to 0.76$ based on TS 3.1.3.1, the analysis of Section 14.7 of this SE applies and fuel will not melt.

The NRC staff concludes that the rod withdrawal accident is within the constraints of the uncontrolled reactivity increase-accident and is acceptable.

14.9 Loss of Primary Coolant Flow

The licensee analyzed the sudden loss of primary coolant flow for the following conditions:

- The reactor is operating at full power.
- The pump flow will coast down to natural convection in 0.1 second.
- The low flow scram fails.
- The initial core average coolant temperature is 110.6^oF (44^oC).
- The initial core excessive reactivity is 0.

Based on the initially positive temperature coefficient, the excess reactivity increases because of the temperature increase. Reactor power level will rise, but will begin to slow as the temperature coefficient goes negative. The final steady state operating point will correspond to a power and flow combination that gives the same reactivity contribution from temperature as for initial steady state operation. This final coolant temperature is 138^oF (59^oC).

The licensee concludes that the peak reactor power during the loss of flow is 101.2 kW with a maximum fuel temperature of 238^oF (114^oC). The transient is terminated by bulk boiling in the core. Equilibrium reactor power level will be 16 kW with a maximum equilibrium fuel temperature of 226^oF (108^oC).

The NRC staff has reviewed the methods and assumptions of the licensee's analysis and concludes that they are conservative resulting in calculated temperatures more than those reasonably expected as a consequence of any loss of coolant flow accident. Therefore, the NRC staff further concludes that there is acceptable assurance that such an event would not lead to loss of integrity of fuel or release of fission products.

14.10 Loss of Primary Coolant

The licensee analyzed the postulated loss of all coolant from the core as a result of a rupture in the primary system. The assumptions are as follows:

(1) The reactor is operating at 100 kW.
(2) Primary system ruptures at some point below the core entrance so that gross removal of the core coolant supply occurs.
(3) All scrams fail.

(4) Uncovering of the fuel acts to shutdown the reactor.
(5) The rupture is large enough to cause rapid coolant loss.
(6) The power peaking factor is 1.3.

The shutdown decay heat removal can only occur by natural circulation of air currents and by radiation heat transfer from the core to the graphite stack. It is assumed that no heat escapes from the graphite stack to the outside environment and axial heat transfer is neglected.

The calculation was performed using a version of the Transient Heat-Transfer (THT) computer program.

Based on this calculation, the licensee concludes that the fuel temperature reaches a maximum of 570^{o}F (299^{o}C) 100 minutes after coolant loss and then begins to decline. The rise in the graphite stack temperature is only 15^{o}F (8^{o}C) in about 3 hours time.

In addition, the licensee analyzed a loss of coolant accident where the reactor was crushed and the graphite pack was fractured so that its thermal conductivity was reduced by 67 percent. No initiating event was identified for this accident. Using the THT computer code, it was found that the peak fuel temperature would be 645^{o}F (358^{o}C), which is below the melting points of the fuel clad and fuel meat.

Compaction of the fuel would not cause the reactor to become critical because of the water loss, increased self-shielding in the fuel and the geometry change (flattening the cylindrical core). These are all negative reactivity changes. Therefore, in spite of the mechanical damage to the reactor or the reactor fuel, the licensee concludes that there would be no danger of fuel melting.

The NRC staff has reviewed the NTR analysis of the loss of primary coolant accident and considers the methods, results and analysis to be acceptable. On the basis of these considerations, the NRC staff concludes that a sudden loss of all coolant from the NTR, even if accompanying or caused by a core crushing accident, would not lead to fuel temperatures that would cause loss of integrity of the fuel cladding. The licensee has acceptably addressed direct exposure of personnel and there is acceptable assurance that the health and safety of the public would not be endangered by such an event.

14.11 Flammable or Explosive Device

Some experimental programs at the NTR facility utilize neutron radiography involving flammable or explosive materials. The licensee has analyzed potential accidents involving fissile materials (see Section 14.1 of this SE) and flammable or explosive materials and has established criteria for experimental activities involving these materials. These criteria and their bases are discussed below.

The licensee’s TSs permit experimental activities, primarily neutron radiography, involving flammable and explosive material in the form of finished or test sample devices only. An analysis of the consequences of accidental explosions at the NTR facility is presented in the SAR, Section 13.5. This analysis was used to establish limits on distances from specific points and equivalent TNT mass for the south cell, north room and set-up room. An explosive storage

magazine for storage of 10 pounds (4.5 kg) of class A and B explosives with a total maximum of 100 lbs. (45 kg), including class C materials, is provided at a location that is separate from the NTR facility. The various weight and distance limits, as well as other limitations on explosive materials irradiated, are incorporated into Section 3.7 of the TSs and summarized in Table 14-6 of this SE. For flammable materials, TSs limit the potential chemical energy and provide controls to ensure no damage to the reactor.

The NRC staff has reviewed the computational procedures and the TS limits and controls used by the licensee and concludes that they are acceptable, thus ensuring that the licensee's activities involving flammable and explosives do not represent a hazard to the facility, the staff or the public.

Table 14-6
Explosive Material Limitations

Maximum cumulative radiation exposures:

Neutron	3×10^{12} n/cm^2
Gamma	1×10^{4} R

Mass and distance limits:*

South Cell	$W \leq (D/2)^2$	$W \leq 9$ lb, $D \geq 3$ ft
North Room		
Without MSM	$W \leq D^2$	$W \leq 16$ lb, $D \geq 1$ ft
With MSM	$W \leq 2$ lb	
Setup Room	$W \leq 25$ lb	

where W = TNT equivalent mass, and
D = distance from south cell blast wall or north room wall

Radioactivity and Fissile Material:

10 Ci maximum and $\leq$ 50 g uranium may be in storage in the South Cell or the North Room when explosive materials are present provided the storage location is $\geq$ 5 ft from explosive material. No radioactive materials are allowed in setup room other than those produced by the neutron radiograph exposure when explosive materials are present.

High-frequency Generating Equipment:

Must not be operated $\geq$ 50 ft from an explosive device.

*An assembly or accumulation of fissile material is one wherein the parts are separated from each other by less than 12 inches.

14.12 Fissionable Material Storage

The TS 5.4 establishes criticality limits for fissionable material storage as $k_{eff} \leq 0.9$ for all conditions of optimum moderation and full reflection using light water. These limits apply to all special nuclear material (SNM) at the NTR facility with the exception of SNM in the reactor core.

The NRC staff finds that the criticality limits contained in the TSs are conservative and provide assurance that the licensee's activities involving SNM external to the reactor core will not lead to an inadvertent nuclear excursion.

14.13 Conclusions

The NRC staff concludes that the licensee has postulated and analyzed sufficient accident initiating events and scenarios to demonstrate that the reactor is designed acceptably to avoid inadvertent reactor damage that could prevent a safe shutdown. There is assurance that no credible accident would cause unacceptable radiological risk to the facility staff, the environment or the public.

On the basis of its review, the NRC staff finds that the license and TS provide acceptable assurance that the assumptions and conditions of the licensee’s safety analysis will be met. Facility operation within the limits of the license and TSs will not result in offsite radiation exposures in excess of 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will limit the likelihood of a malfunction and mitigate the consequences to the public in regard to accident events.

15.0 LICENSE AND TECHNICAL SPECIFICATIONS

In the course of this licensing action, the NRC staff reviewed and evaluated the license, and the TSs submitted by the licensee. These documents define certain features, characteristics and conditions governing the continued operation of the NTR facility. The TS are specifically included in the renewal license as Appendix A to the license. In addition, the NRC staff reviewed the format and content of the TSs using guidance from ANSI/ANS 15.1-1990, "The Development of Technical Specifications for Research Reactors."

By letter dated November 7, 2000, the NRC staff requested verification of the accuracy of the proposed license and Technical Specifications to ensure the accuracy of these documents. The licensee was requested to provide verification and, if need be, corrections to these documents. By telephone conversation, the licensee provided two corrections. The spelling of “regulatory” in Figure 6-1 was incorrect and the reference to “Title 10 of the Code of Federal Regulations, Part 55” in TS 6.1.4 should delete the reference to “Appendix A.” These minor corrections were made.

Also, the NRC staff added two conditions to the license. Specifically, paragraph 2.B(2)(f) was added as follows: “Such special nuclear material as may be produced by the operation of the reactor. The licensee is not authorized to separate this special nuclear material.” Also, the parenthetical phrase, “(except for byproduct material produced as allowed for experiments),” was added to 2.B(3)(d). These conditions were added to ensure acceptable authorization and use of materials in accordance with research reactor licensing practices. The licensee had no

objection to these conditions. The NRC staff also changed TS 6.6.1, so that the obsolete requirement to send the annual report to the Region IV Administrator was deleted.

16.0 FINANCIAL QUALIFICATIONS

General Electric Company, a multi-billion dollar diversified corporation, operates the NTR as part of the GE Vallecitos Nuclear Center.

The NRC staff has reviewed the financial status of the licensee and concludes that funds will be made available to support continued operations and, when necessary, to shut down the facility and carry out decommissioning activities. The licensee's financial status is in accordance with the requirements of 10 CFR 50.33(f). Therefore, the NRC staff concludes that the licensee's financial qualifications are acceptable.

17.0 CONCLUSIONS

On the basis if its evaluation of the application as set forth in the previous sections, the NRC staff has reached the following conclusions:

- The application filed by GE for renewal of Facility Operating License No. R-33 for the General Electric Nuclear Test Reactor complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), as well as the Commission's regulations set forth in the NRC’s regulations.

- The facility will operate in conformity with the application (as amended), as well as the provisions of the Act and the rules and regulations of the Commission.

- There is acceptable assurance that (a) the activities authorized by the operating license can be conducted without endangering the health and safety of the public and (b) such activities will be conducted in compliance with the Commission's regulations as set forth in 10 CFR Chapter I.

- The licensee is technically and financially qualified to engage in the activities authorized by the license in accordance with the Commissions regulations as set forth in 10 CFR Chapter I.

- The renewal of this license will not be inimical to the common defense and security or to the health and safety of the public.

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