November 7, 2000

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

SUBJECT: VERIFICATION OF LICENSE AND TECHNICAL SPECIFICATIONS FOR RENEWAL OF FACILITY OPERATING LICENSE NO. R-33 - GENERAL ELECTRIC NUCLEAR TEST REACTOR (TAC NO. MA0226)

Dear Mr. Stimmell:

Enclosed is a proposed License and Technical Specifications for Facility License No. R-33 for the General Electric Nuclear Test Reactor 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. To ensure the accuracy of these documents, you are requested to provide verification and, if need be, corrections within 60 days of the date of this letter.

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

Enclosure:
Proposed License and Technical Specifications

cc w/enclosures:
Please see next page

cc:

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

November 7, 2000

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

SUBJECT: VERIFICATION OF LICENSE AND TECHNICAL SPECIFICATIONS FOR RENEWAL OF FACILITY OPERATING LICENSE NO. R-33 - GENERAL ELECTRIC NUCLEAR TEST REACTOR (TAC NO. MA0226)

Dear Mr. Stimmell:

Enclosed is a proposed License and Technical Specifications for Facility License No. R-33 for the General Electric Nuclear Test Reactor 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. To ensure the accuracy of these documents, you are requested to provide verification and, if need be, corrections within 60 days of the date of this letter.

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

Enclosure:
Proposed License and Technical Specifications

cc w/enclosures:
Please see next page

DISTRIBUTION:

PUBLIC	REXB r/f	EHylton	GHill (2)
DMatthews	SNewberry	OGC	TDragoun
MMendonca	AAdams	TMichaels	LMarsh
SHolmes	CBassett	WEresian	PIsaac

ACCESSION NO.: ML00 **TEMPLATE #: NRR-056**

*Please see previous concurrence

OFFICE	REXB:LA	REXB:PM	REXB:BC
NAME	EHylton:rdr	MMendonca	LMarsh
DATE	11/ 02 /2000	11/ 02 /000	11/ 05 /2000

C = COVER **E = COVER & ENCLOSURE** **N = NO COPY**

OFFICIAL RECORD COPY

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-73

GENERAL ELECTRIC COMPANY

RENEWAL OF THE FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-33

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

   A. The application for amendment 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 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 that (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 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) which is owned by the General Electric Company and located at ins 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 licensee, 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.

      (3) Pursuant to the Act and Title 10, Chapter I Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” (1) 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; (2) any byproduct materials necessary for purposes of instrument calibration and startup sources; (3) 10 curies of tritium for pulsed neutron sources; and (4) to possess, but not to

separate, 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 Specification

The Technical Specifications contained in Appendix A, as revised through Amendment No. , 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

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:

NEDO-32765
Class 1
August 1997

TECHNICAL SPECIFICATIONS

FOR THE

GENERAL ELECTRIC NUCLEAR TEST REACTOR FACILITY LICENSE R-33

CONTENTS

Section			Page
1.0	INTRODUCTION		1-1
	1.1	Purpose	1-1
	1.2	Definitions	1-1
2.0	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS		2-1
	2.1	Safety Limits (SL)	2-1
	2.2	Limiting Safety System Settings (LSSS)	2-3
3.0	LIMITING CONDITIONS FOR OPERATION (LCO)		3-1
	3.1	Reactor Core Parameters	3-1
	3.2	Reactor Control and Safety System	3-4
	3.3	Reactor Coolant System	3-10
	3.4	Reactor Cell and Ventilation System	3-12
	3.5	Experiments	3-15
4.0	SURVEILLANCE REQUIREMENTS		4-1
	4.1	Reactivity Limits	4-1
	4.2	Reactor Control and Safety System	4-3
	4.3	Reactor Coolant System	4-6
	4.4	Reactor Cell and Ventilation System	4-7
	4.5	Experiments	4-8
	4.6	Frequency of Testing	4-9
5.0	DESIGN FEATURES		5-1
	5.1	Site and Facility Description	5-1
	5.2	Reactor Primary Coolant System	5-2
	5.3	Reactor Core and Fuel	5-3
	5.4	Fissionable Material Storage	5-3
6.0	ADMINISTRATIVE CONTROLS		6-1
	6.1	Organization and Staffing	6-1
	6.2	Independent Reviews	6-5
	6.3	Radiation Safety	6-7
	6.4	Procedures	6-8
	6.5	Required Actions	6-10
	6.6	Report	6-11
	6.7	Records	6-14

TABLES

Tables	Title	Page
3-1	Reactor Safety System – Scram	3-6
3-2	Reactor Safety System – Information	3-7
3-3	Stack Release Rate Limits	3-13
4-1	Surveillance Requirements of Reactor Safety System Scram Instruments	4-4
4.2	Surveillance Requirements of Reactor Safety System Information Instruments	4-5

ILLUSTRATIONS

Figure	Title	Page
6-1	Facility Organization	6-2

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 P	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	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 Ci/wk
Alpha	8.7 Ci/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 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)	Daily
		Channel Test (high level trip test)	Daily
		Channel Check (comparison against a heat balance)	Semi-annual
		Channel Calibration	Annually
2.	Log N System	Channel Test	Daily
		Channel Check	Monthly
		Channel Calibration	Annually
3.	Primary Coolant Temperature	Channel Test	Daily
		Channel Calibration	Annually
4.	Primary Coolant Flow	Channel Check	Daily
		Channel Test	Daily
		Channel Calibration	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.l.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 l Reactor Supervisor (if utilized) is the individual responsible for supervising the daily operations. In the absence of this position, the Level l manager or the Facility Manager is responsible for supervising the daily operations.

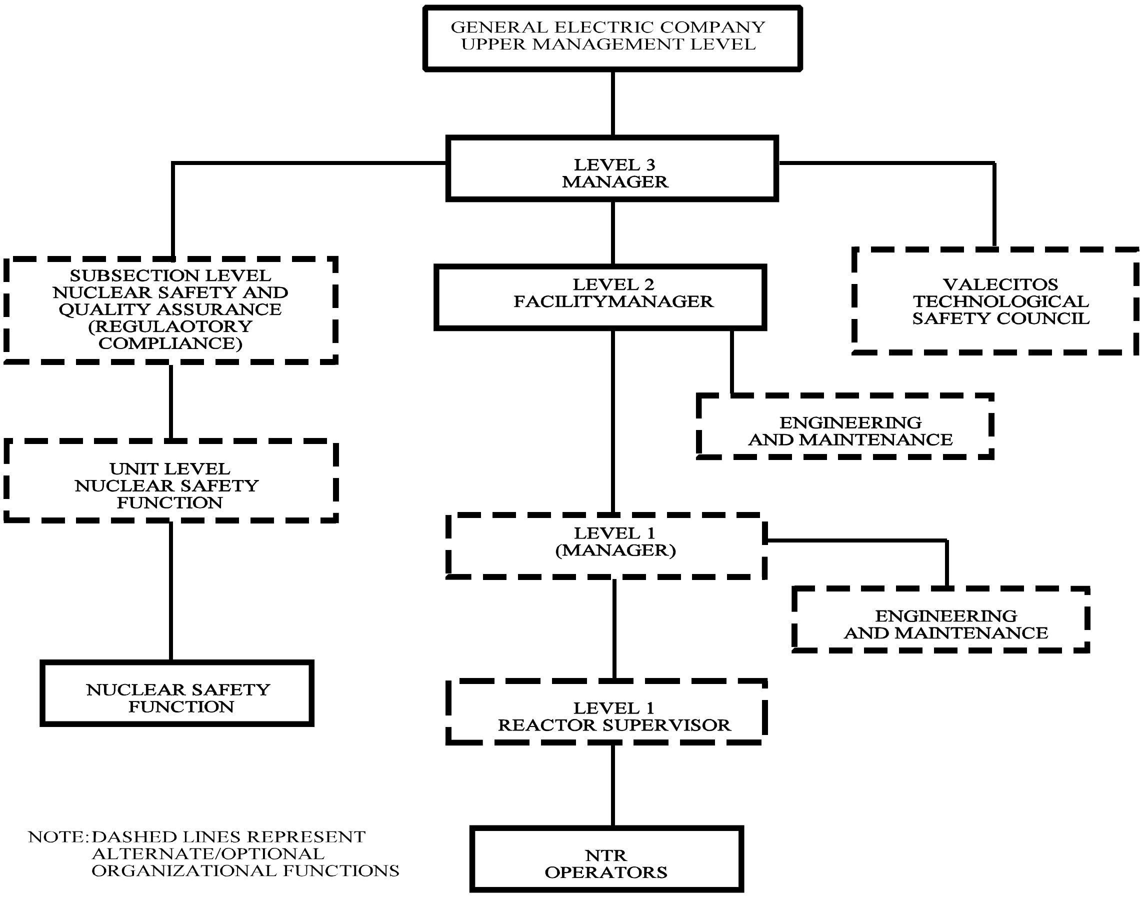

Figure 6-1. Facility Organization

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, Appendix A; 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 with a copy sent to the Region IV administrator. 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.