GENERAL ELECTRIC
NUCLEAR TEST REACTOR
LICENSE NO. R-33
DOCKET NO. 50-73

LICENSE RENEWAL APPLICATION
SAFETY ANALYSIS REPORT
AND TECHNICAL SPECIFICATIONS
NOVEMBER 20, 1997

REDACTED VERSION*

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GE Nuclear Energy

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November 20, 1997

U.S. Nuclear Regulatory Commission
Attention: Document Control Center
Washington, D.C. 20555

Subject: Nuclear Test Reactor License Renewal

References: 1) License R-33, Docket 50-73
2) Application for Renewal of License R-33; September 30, 1997

Gentlemen:

Enclosed are one copy each of the GE Nuclear Test Reactor (NTR) Safety Analysis Report and the Technical Specifications for the GE NTR, in support of the R-33 License Renewal Application of September 30, 1997.

Sincerely,

B. M. Murray
Senior Licensing Engineer

Enclosures

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NEDC-32765
Class I
August 1997

Technical Specifications for The General Electric Nuclear Test Reactor

Facility License R-33

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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 10CFR55 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 (1/2 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 1.5 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 1/2 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 further 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 125kW	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 222°F	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 – INFORMATION

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	<200°F	Visible and audible alarm; audible alarm may be bypassed after recognition.
4.	Primary Coolant Δ 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 0.1 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 conducitivity 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 13.4.5). Natural convection cooling is sufficient. Requiring forced coolant flow above 0.1 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	174 mCi/wk
Particulate, > 8d T1/2	
Beta-Gamma	8.69 μCi/wk
Alpha	8.69 μCi/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 1800 cu ft/min equals approximately 10,000. That is, the concentration at the site boundary of any release from the NTR stack will be ≤1/10,000 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 $D \geq$ i 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 10CFR, 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 3.1 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 Table 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 Δ 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 4.4.3.1 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 10CFR20 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	[redacted]
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 6-1 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 1 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 1 Reactor Supervisor or the Facility Manager shall assume the Level 1 manager responsibilities.

6.1.2.4

The Level 1 Reactor Supervisor (if utilized) is the individual responsible for supervising the daily operations. In the absence of this position, the Level 1 manager or the Facility Manager is responsible for supervising the daily operations.

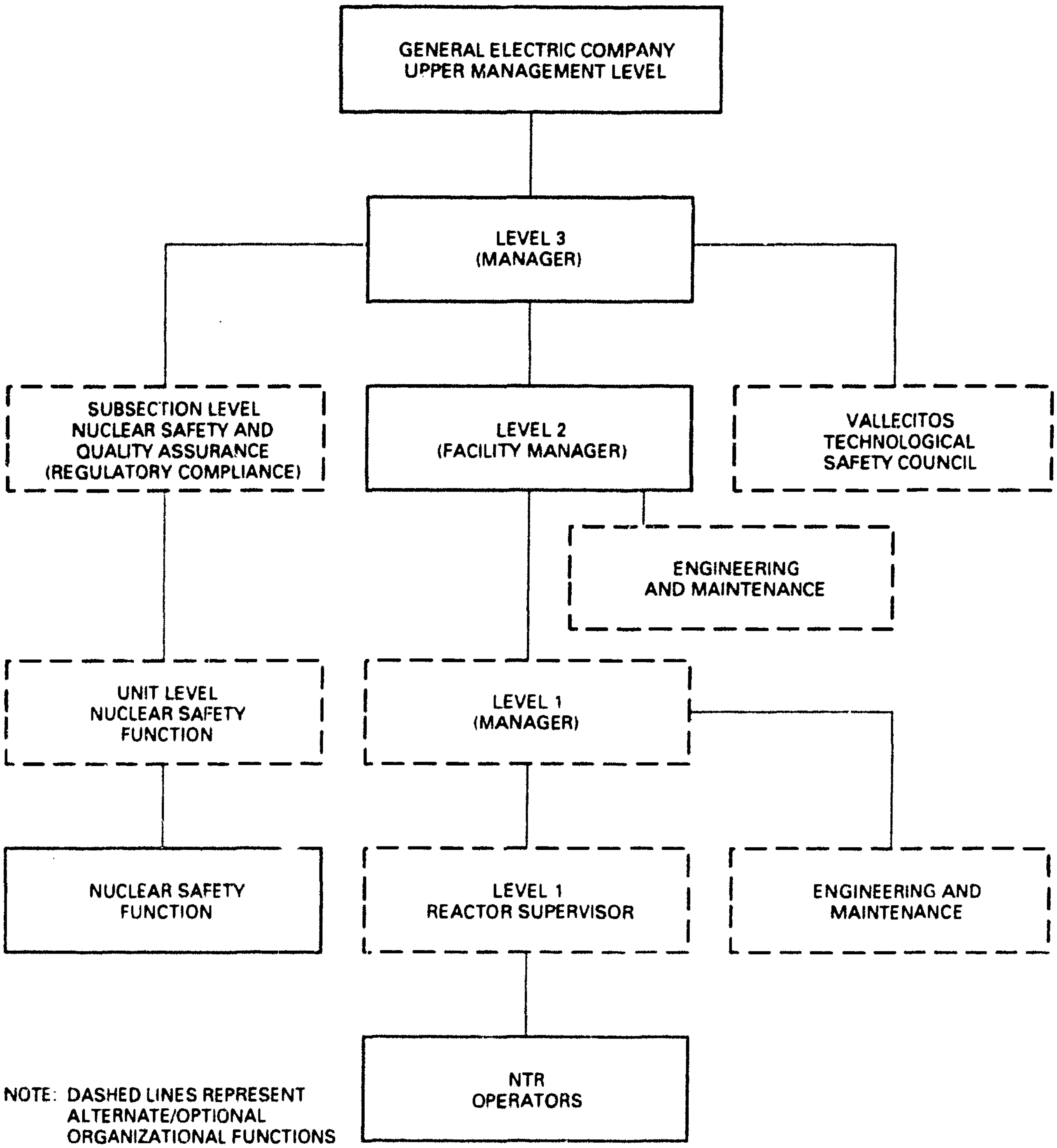

Figure 6-1. Facility Organization

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 10CFR50.59, 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 10CFR50.59, 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 10CFR20.

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 10CFR20 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 1 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 10CFR 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 10CFR 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.

GE Nuclear Energy

NTR

NUCLEAR TEST REACTOR

Safety Analysis Report & Technical Specifications

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NTR
NUCLEAR TEST REACTOR

Safety Analysis Report & Technical Specifications

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GENERAL ELECTRIC
NUCLEAR TEST REACTOR
SAFETY ANALYSIS REPORT

Prepared by the Technical Staff
of the
Vallecitos and Morris Operation

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NOTICE

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ABSTRACT

The GE Nuclear Test Reactor is described and a summary of the facility safety evaluation is presented. The description includes the GE Nuclear Test Reactor history; the Vallecitos Nuclear Center Site and area characteristics; a detailed facility description; descriptions of Irradiation Facilities, instrumentation and control systems; and facility administration, including the Quality Assurance programs and shielding around the facility. The safety evaluation contains a summary of the analyses performed and the consequences of normal and off-normal conditions, and postulated reactor accident conditions.

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1.0 THE FACILITY

1.1 INTRODUCTION

GE – Nuclear Energy (GE) designed, constructed, and is operating the GE – Nuclear Test Reactor (NTR) as a part of the experimental facilities at its Vallecitos Nuclear Center (VNC) Site in Alameda County, California. The reactor was originally designed as an experimental tool to: (1) advance the Company's progress in the nuclear energy program, (2) provide a source of neutrons for sample irradiation or exponential experiments, and (3) provide a sensitive device for measuring reactivity.

Current use of the NTR facility is for: (1) neutron radiography of radioactive and nonradioactive objects, (2) small sample irradiation and activation, (3) sensitive reactivity measurements, (4) training, and (5) calibrations and other testing utilizing a neutron flux.

The NTR is a heterogeneous, highly enriched-uranium, graphite-moderated and reflected, light-water-cooled, thermal reactor, licensed to operate at power levels not in excess of 100 kW (thermal). [redacted] It has a confinement building to restrict the release of radioactivity to the environment, diversity and redundancy of instruments and controls and extremely low operating heat flux and temperatures.

Over 39 years of operation in the performance of a variety of experiments and testing for GE and its customers has demonstrated the safety and effectiveness inherent in the reactor's design and the Company's operating methods.

1.2 SUMMARY AND CONCLUSIONS OF PRINCIPAL SAFETY CONSIDERATIONS

The impact of the operation and use of the NTR is inconsequential. The reactor is designed to contain radioactivity and monitor radioactive releases. The facility is operated in accordance with approved procedures which limit radiation exposures and off-normal operation of the reactor. In addition, built-in design features and automatic shutdown features prevent temperatures from exceeding heat flux limits.

The VNC site is not adjacent to a large population center and the weather is not prone to damaging extremes. The NTR core consists of an aluminum can filled with water in a graphite pack. The fuel is a stable aluminum-uranium alloy operated at low heat flux and thermal temperatures. The reactor is in a confinement building, which is maintained at a negative pressure for inward air flow. Air for the confinement building is exhausted through a stack and is monitored for radioactive releases.

The NTR has a negative void coefficient of reactivity and a negative temperature coefficient of reactivity above 124°F, which is approximately the steady state operating temperature. Additionally, because of low stored heat content, the NTR fuel will not melt when the fuel

coolant water is lost. These features greatly contribute to the protection of occupational workers, general public and the environment in case of an accident.

The NTR has a scram system which automatically inserts enough negative reactivity to shut down the reactor and maintain it shut down. The system is activated by both manual and automatic switches when predetermined parameters approach preestablished limits

The facility response to certain postulated credible events and less probable accidents which have potential safety significance has been evaluated. These events include the following:

1. Loss of electrical power
2. Loss of secondary cooling
3. Loss of facility air supply
4. Inadvertent start of primary pump
5. Fuel handling error
6. Uncontrolled reactivity increases
7. Loss of primary coolant flow (pump shaft seizure)
8. Rod withdrawal
9. Loss of primary coolant
10. Experiment failure

The three acceptance criteria for anticipated operational occurrences are the following:

1. Release of radioactive material to the environs does not exceed the limits of 10CFR20.

2. Radiation exposure of any individual does not exceed the limits of 10CFR20.

3. An established safety limit is not exceeded.

The acceptance criteria for postulated accidents are as follows:

1. Release of radioactive material does not exceed the limits of 10CFR100.

2. An established safety limit is not exceeded.

Because of the many safety features provided and the strong administrative control applied to operation of the facility, the possibility of an accident involving high radiation exposure or the dispersion of substantial quantities of radioactivity is considered extremely remote. However, the protection of the health and safety of the public is ensured further by housing the reactor in a thick-walled concrete cell that provides radiation shielding and permits controlled release of airborne contamination. On the basis of the descriptive and analytical information provided in this report and the proven performance of the facility over an extended operating period, it is concluded that the design and operating methods of the NTR facility provide the reasonable assurance required by the regulations that the health and safety of the public will not be endangered by continued operation of the facility.

1.3 GENERAL DESCRIPTION OF THE FACILITY

The NTR is located at the Vallecitos Nuclear Center (VNC). VNC is situated on the north side of Vallecitos Valley in Southern Alameda County within five miles of Livermore and Pleasanton and ~35 air miles east-southeast of San Francisco and ~20 air miles north of San Jose.

Vallecitos Valley is approximately two miles long and 1 mile wide. The valley is at an elevation of 400 to 500 feet above sea level and is surrounded by barren mountains and rolling hills. There is very little commercial and residential development in the valley. The VNC site is relatively flat on the southern end and slopes upward to a ridge on the north.

The NTR is a heterogeneous, enriched-uranium, graphite-moderated and -reflected, light-water-cooled, thermal reactor, licensed to operate at power levels not in excess of 100 kW (thermal). The fuel consists of highly enriched uranium-aluminum alloy disks, clad with aluminum. The core is cooled either by natural or forced circulation of deionized lightwater circulated in a primary system constructed primarily of aluminum. The reactor operates at very low temperature and low heat flux. Reactivity is controlled by up to six manually positioned cadmium sheets, four boron-carbide-filled safety rods (spring-actuated for reactor scram), and three electric-motor-driven boron-carbide-filled control rods. Conventional instrumentation is provided to indicate, record, and control important variables, and shut down the reactor automatically if assigned operating limits are exceeded. The reactor's irradiation facilities include a central sample tube, penetrations through and into the reflector, the reflector faces, and the beams from any of these facilities

When used as a neutron source, the reactor can provide unperturbed neutron fluxes (at 100 kW) of about 2×10^{12} thermal n/cm^2-sec and an epicadmium flux of about 1×10^{12} n/cm^2-sec. When used as a detector, reactivity effects can be measured with a precision of 10^{-6} $\Delta k/k$ without the use of a pile oscillator.

The reactor is located in a thick-walled concrete cell which, with the control room, north room, setup room and the south cell, houses the NTR facility. An overall view of the facility, except the north room and set-up room, is shown in Figure 1-1. Principal equipment in the concrete reactor cell includes the reactor, the reactor control mechanisms, the coolant system, and a fuel loading tank which provides radiation shielding and the primary water system reservoir. The control room contains the control console and provides space for experiment equipment, preparation, and an operator work area. The south cell is a concrete-shielded room which provides access to the thermal column, the horizontal facility and the horizontal facility south beam. The north room provides space for performing experiments utilizing the horizontal facility north beam and the Cable Held Retractable Irradiation Facility (CHRIS). The set-up room is used for storage and setup of experiments prior to irradiation or testing.

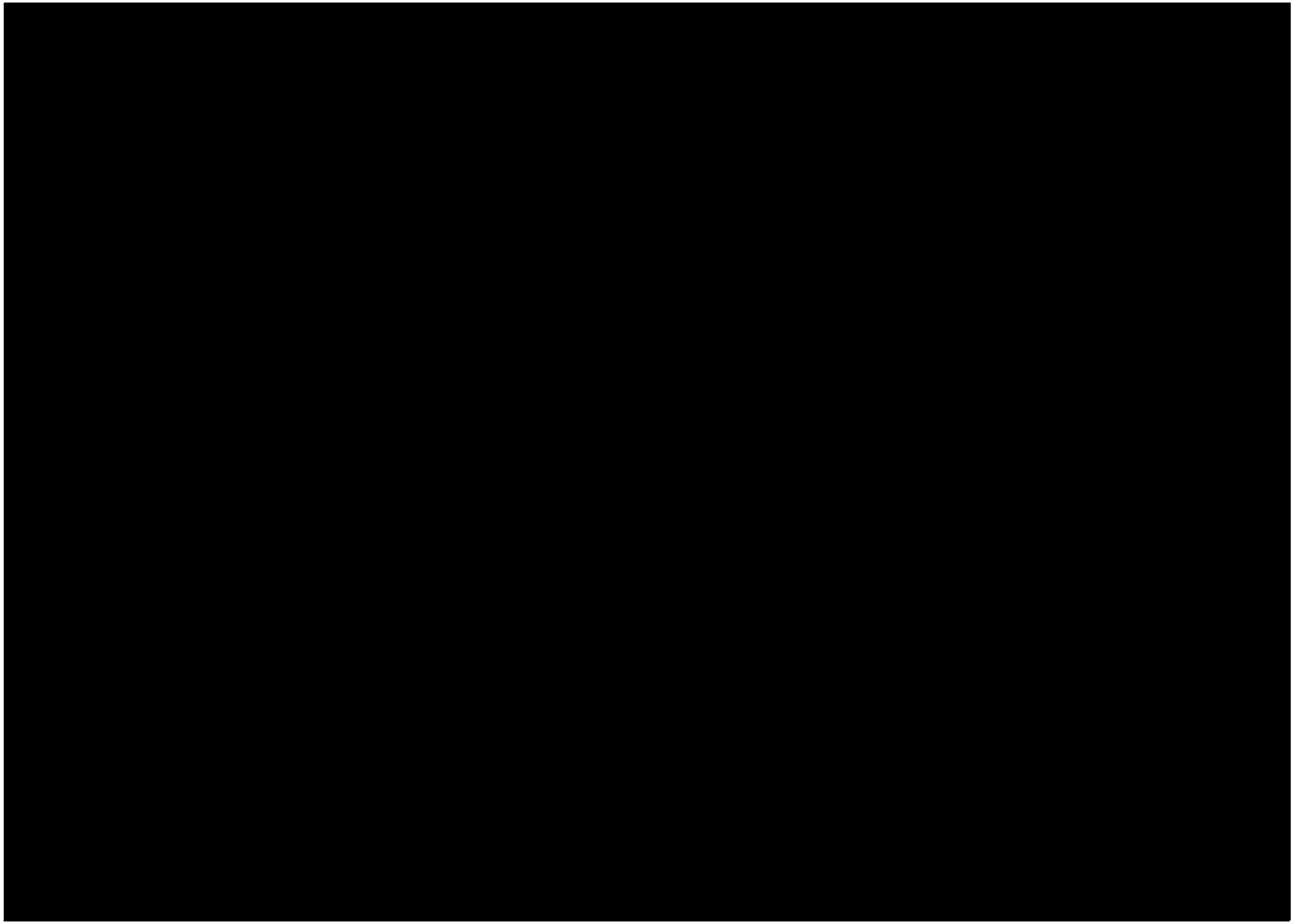


Figure 1-1. Nuclear Test Reactor Facility

Release of radioactive materials is strictly regulated. Radioactive gas and particulates released from the reactor cell are monitored continuously and the reactor is shut down if required to reduce emissions below release limits. Solid and liquid radioactive wastes are collected by trained individuals in accordance with approved procedures and disposed of in accordance with applicable regulations.

Radiation protection of individuals is controlled by a variety of means. A radiation protection program has established posted zones to contain radioactive materials. Routine and special surveys assure that radioactive materials are controlled and that there is no unplanned exposure or movement. Also, there are ion chambers and filter sample stations strategically located in the facility to warn of unusual increases or releases of radioactive materials. An ALARA (as low as reasonably achievable) program also requires review of facility changes and new experiments to design for reduced radiation exposure. In addition, routine audits and reviews are conducted by personnel independent of reactor management and reactor operations personnel who are highly trained in radiation protection and work to approved written procedures.

1.4 SHARED FACILITIES AND EQUIPMENT

The NTR shares many facilities and equipment in Building 105 with other laboratory facilities. These shared things include potable water supply, fire protection, emergency supplies and support, HVAC System, AC electrical distribution, compressed air system and the occupied spaces of Building 105.

Some of the laboratories in Building 105 handle small amounts of byproduct material licensed under California Radiactive Material License 0017-59. However, there are no other licensed equipment or facilities in Building 105 such as hot cells, critical or subcritical assemblies, neutron sources, irradiation facilities or radioactive material storage areas.

The NTR shared building spaces are adequately separated by walls to delineate the NTR facility from the other offices and laboratories. There is only one shared work space, a shop, and this is not a posted Radiation Area nor a posted Radioactive Materials Area. Other separations have been installed to adequately isolate the shared facilities and equipment. For instance, the potable water supply to the NTR contains an approved reduced pressure backflow preventer. Although there are shared load centers, reactor safety equipment are connected to electrical circuits which are not shared with other facilities and equipment outside NTR.

Other shared facilities and equipment have been established at NTR in order to increase the convenience and the strength of resources available to a small facility. These include the fire protection system (building sprinkler system, fire hoses, and portable fire extinguisheers), building emergency response teams, an emergency supply cabinet, HVAC system, and a compressed air system all of which does not supply any reactor safety system.

1.5 COMPARISON WITH SIMILAR FACILITIES

The design of the NTR resulted from the evolution of a series of reactors designed by scientists at the GE Knolls Atomic Power Laboratory (KAPL) in Schnectady, New York. The earlier reactors were known as thermal test reactors (TTR). Three models were built and operated successfully. The GE TTR operated from 1954 to the mid-eighties at KAPL. The TTR No. 2 operated from 1955 until 1972 at the Battelle Memorial Institute Pacific Northwest Laboratory. The third TTR, the Savannah River National Laboratory Standard Pile operated from 1953 to 1979.

The logical evolution which led to the design of the NTR produced a versatile and safe reactor. Features which contribute to the safety of the reactor and which were incorporated into the design and construction of NTR include:

1. Negative void coefficient of reactivity.

2. Small positive coolant temperature coefficient of reactivity which becomes negative at a water temperature slightly above the operating temperature.

3. A control system extremely sensitive to changes in reactivity so that minute changes are detectable.

4. Safety and control functions that are completely separate, except for an interlock which requires all safety rods to be fully withdrawn prior to withdrawing any control rod. This ensures that negative reactivity is available if needed for scram before a control rod can be moved.

5. Manually positioned cadmium sheets that can be used to limit reactivity controllable from the console and to provide sufficient negative reactivity to preclude any possible danger or criticality during fuel loading.

6. An instrumentation system which includes fail-safe and redundant features as well as proven reliable components.

7. A system constructed from materials having properties compatible with their intended service.

Safety measures which have been incorporated into the operation of the facility include:

1. Very low heat flux, even at the maximum operating power.

2. Temperatures and pressures only a little above ambient.

3. Low operating power, resulting in a low fission-product inventory.

4. Rigid control by operations management of all experiments performed in the reactor facility.

5. Performance of all activities that can affect nuclear safety under the direction of an NRC-licensed reactor operator or NRC-licensed senior reactor operator, as required.

1.6 SUMMARY OF OPERATIONS

The NTR was originally built as an experimental tool for diverse applications. In the first 5 years of operation it was used for pile-oscillator measurements of nuclear cross sections of materials, calibrations of foils and nuclear sensors, neutron activation analysis, studies of radiation damage in semiconductors, nuclear fuel enrichment measurements, and cryonuclear investigations.

Over the years the reactor has been used for a variety of purposes from neutron absorption measurements of material at a reactor power level of 10 watts to 24 hour/day irradiation of filter tape. Current use of the NTR is for: (1) neutron radiography of radioactive and nonradioactive objects, (2) small sample irradiations, (3) sensitive reactivity measurements, (4) training, and (5) calibrations and other testing utilizing a neutron flux.

The reactor is capable of operating at extremely low power levels not in excess of 100 kW (thermal).

1.7 COMPLIANCE WITH THE NUCLEAR WASTE POLICY ACT OF 1982

The NTR has entered into an agreement with the Department of Energy (DOE) whereby the DOE will accept for reprocessing the NTR spent nuclear fuel. This satisfies the requirements of Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982.

1.8 FACILITY MODIFICATIONS AND HISTORY

GE entered the nuclear power industry in the early 1950's and has become a leader in the industry with the Boiling Water Reactor. Earlier nuclear industry experience was gained while operating the Hanford reactors for the Department of Energy in the 1940s and 1950s and in the U.S. Navy Nuclear Power Plant Program.

The Vallecitos Nuclear Center (originally called the Vallecitos Atomic Laboratory) was established in 1956. Experience at the site with NRC licensed activities include the following:

Radioactive Materials Laboratory (License SNM-960), June 1956-Present.

Vallecitos Boiling Water Reactor (License DPR-1), August 1957 – December 1963.

Critical Experiment Facility, November 1957 – mid 1966.

General Electric Test Reactor (License TR-1), December 1958 – October 1977.

Empire State Atomic Development Associates Vallecitos Experimental Superheat Reactor (License DR-10), January 1964 – February 1967.

In addition, some activities are performed under a State of California source and byproduct material license 0017-59 beginning December 1962.

The NTR was constructed under construction permit CPRR-19, issued October 24, 1957, as requested by General Electric's application, dated June 5, 1957. Operation of the reactor up to powers of 30 kW was authorized by facility license R-33, issued on October 31, 1957. Initial loading of the reactor began on November 7, 1957, and criticality was first achieved on November 15, 1957. Operation since that time was in accordance with license R-33, as amended. During that period of authorized operation, the reactor was operated at powers up to 30 kW for more than 5000 hours. On July 22, 1969, the license was amended, which revised the license in its entirety, authorizing operation of the reactor at power levels of up to a maximum of 100 kW steady-state power (later amended to power level not in excess of 100 kW). Since then, the reactor has operated at power levels up to 100 kW for more than 30,000 hours while performing a wide variety of experiments. In 1976, the reactor core can developed a leak in a weld area, necessitating replacement. The reactor fuel was removed and inspected and a major portion of the reactor was dismantled. The core can was replaced, as well as some of the graphite in the central area. Some modification of the irradiation facilities occurred at this time. The reactor was reassembled, utilizing the original fuel, and routine operation resumed.

Prior to 1985, many original instruments were replaced, including the picoammeters (linear wide range neutron monitors), log N (log wide range neutron monitor), remote area gamma monitors, and the stack effluent gas and particulate monitors.

Descriptive information for the NTR facility was originally contained in GEAP-1005, *Safeguards Report, Nuclear Test Reactor*. This document was part of the application, dated June 5, 1957, for a construction permit and facility license; pursuant to this application, as amended, construction permit CPRR-19 and facility license R-33 (Docket 50-73) was issued. In 1958, General Electric amended the license application to incorporate changes in procedures and equipment. At that time, the information is document GEAP-3068, *Summary Safeguards for the Nuclear Test Reactor* (October 7, 1958), was substituted for the related information in the original application.

A new summary safeguards report for the nuclear test reactor (APED-4444) was submitted February 1, 1965, for updating and providing new information about design, operation and safety analysis. It accompanied a license application amendment requesting separate technical specifications and provided a documentation mechanism for simplifying amendatory actions.

A revision of APED-4444 (APED-4444A) was submitted November 21, 1968 and amended March 31, 1969 and May 28, 1969 as part of a license application amendment requesting an increase in authorized maximum steady-state power level from 30 to 100 kW.

The SAR, APED-444A was again revised and reissued as NEDO-12727 in April 1977. The purpose of NEDO-12727 was to update the description of the facility and its organization and to summarize additional safety evaluation of the facility. NEDO-12727 was submitted as supporting material for the renewal of the operating license.

Since the facility license was renewed on December 28, 1986, there has been no significant changes made to the facility, its procedures or administration. One license change approved August 19, 1992, increased the 200-cruie byproduct material limit to 2000 curies. One Technical Specification change (amendment No. 19, dated June 2, 1989) allowed one exception to Tech Spec 3.5.3.5 which requires separation of explosive and radioactive materials.

This document is a revision to the current SAR, NEDO-12727, dated April 1981 and is intended to accomplish the following:

1. Update the description and organization of the facility since April 1977.

2. Support an application for the NTR license renewal to permit continued operation of the reactor to power levels not in excess of 100 kW (thermal).

The safe and efficient operation of the NTR, now completing its fortieth year of operation, is evidenced by the 37 annual reports submitted to the NRC on operating experience pertinent to safety.

Department of Energy
Washington, D.C. 20585

JUL 13 1983

Mr. R. W. Darmitzel
Manager, Irradiation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Darmitzel:

Subject: Contract Number DE-CR01-83NE44426

Enclosed herewith is one fully executed copy of subject contract.

Please acknowledge receipt of this contract by completing the "Acknowledgement" below and returning it to:

U.S. Department of Energy
Office of Procurement Operations
Attn: Thomas S. Keefe, MA-453.1
Room Number 1J-027
1000 Independence Avenue, SW
Washington, DC 20585

Sincerely,

Thomas S. Keefe

Thomas S. Keefe
Contracting Officer
Office of Procurement Operations

Enclosure

ACKNOWLEDGEMENT:

R. W. Darmitzel, Manager
Irradiation Processing Operation

Contractor's Authorized Representative
Name and Title (Type/Print)

RW Darmitzel

Signature

July 18, 1983

Date

APPENDIX A

NUCLEAR POWER REACTOR(S) OR OTHER FACILITIES COVERED

Purchaser General Electric Company

Contract Number/Date DE-CR01-83NE44426 / 6/29/83

Reactor/Facility Name Vallecitos Nuclear Center

Location:

Street Vallecitos Road, P.O. Box 460

City Pleasanton

County/State Alameda / California

Zip Code 94566

Capacity (MWE) - Gross None

Reactor Type:

BWR ☐ *See below.

PWR ☐

Other (Identify)

Facility Description Nuclear research and development facility including research reactors, hot cells, and various laboratories.

Date of Commencement of Operation 1956
(actual or estimated)

NRC License #: SNM-960, TR-1, TR-33

By Purchaser:

Signature	Title	Date
R W Darmitzel	Manager, Irradiation Processing Operation	June 15, 1983

*The SNF to be disposed of by the Vallecitos Nuclear Center under this contract has been irradiated in one or more of the following reactors: General Electric Test Reactor, Nuclear Test Reactor, Monticello, Quad Cities-1, Big Rock Point, Vermont Yankee, Millstone-1, Humboldt Bay,3 and Brunswick-1.

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Site Location and Description

Site Location

The NTR is situated on the 1590 acre (6.4-km^2) Vallecitos Nuclear Center (VNC) near Pleasanton, California (Figure 2-1) The VNC site is owned by GE and is used for nuclear research and development.

The VNC is located on the north side of the Vallecitos Valley. The valley is approximately 3.2-km long and 1.6-km wide; its major axis is east-northeast and west-southwest. The valley is at an elevation of 120 to 150-m above sea level and is surrounded by barren mountains and rolling hills.

The Site consists of a quadrilateral, (Figure 2-2) bounded on the west, north, and east by hilly terrain; in some places, the hills are about 220-m above the general Site elevation. Vallecitos Road (State Highway 84) forms the southern boundary of the Site, from which an expanse of gently rolling grassland extends for about 5-km. Beyond 5-km mountain ranges form a southern barrier which completes the encirclement of the Site.

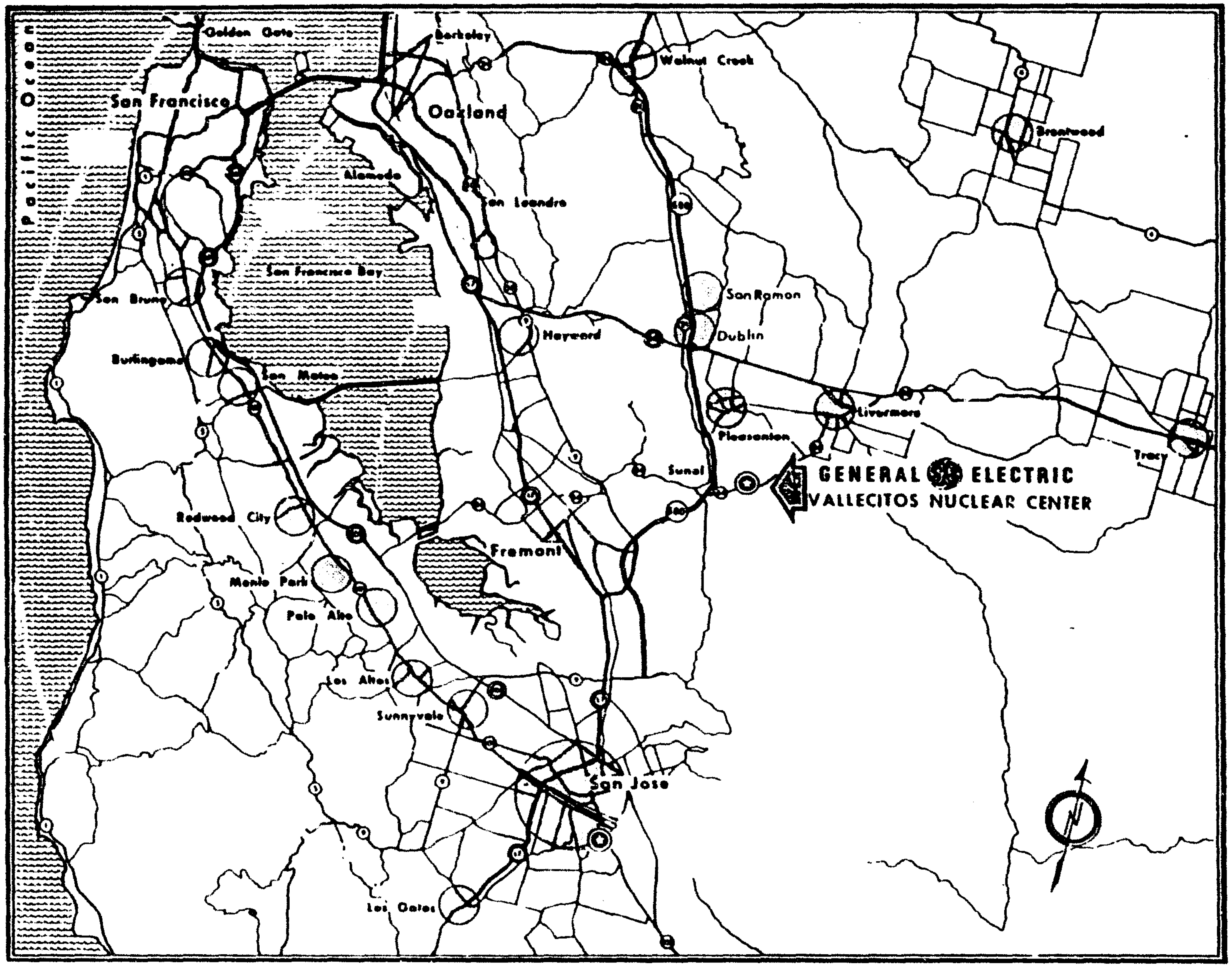
Site Description

Approximately one-third of the Site is gently sloping or rolling terrain (Figure 2-3). The remainder consists primarily of the southwestern slope of a ridge serrated by several small draws. The southern part of the Site, adjacent to the Vallecitos Road, is relatively flat and accommodates the NTR, laboratories, and administrative facilities.

Approximately 1500-acres (6.1-km^2) of the Site is normally leased for grazing and for cattle-feed crops. The land surrounding the Site is devoted to agriculture and cattle raising.

Security

Since its inception, VNC has operated under a controlled-access security plan. A perimeter fence maintains the Site as a restricted area to the general public. The entrance gate approached over VNC property from the Vallecitos Road is guarded at all times to control the entrance and exit of personnel. Additional security and control within the Site and its facilities it extensive. The plan conforms to the NRC requirements delineated in 10 CFR, Section 73.40. Since the northeast portion of the Site is mountainous, the general security of the Site is increased.



BAY AREA MAP

Figure 2-1. Bay Area Map

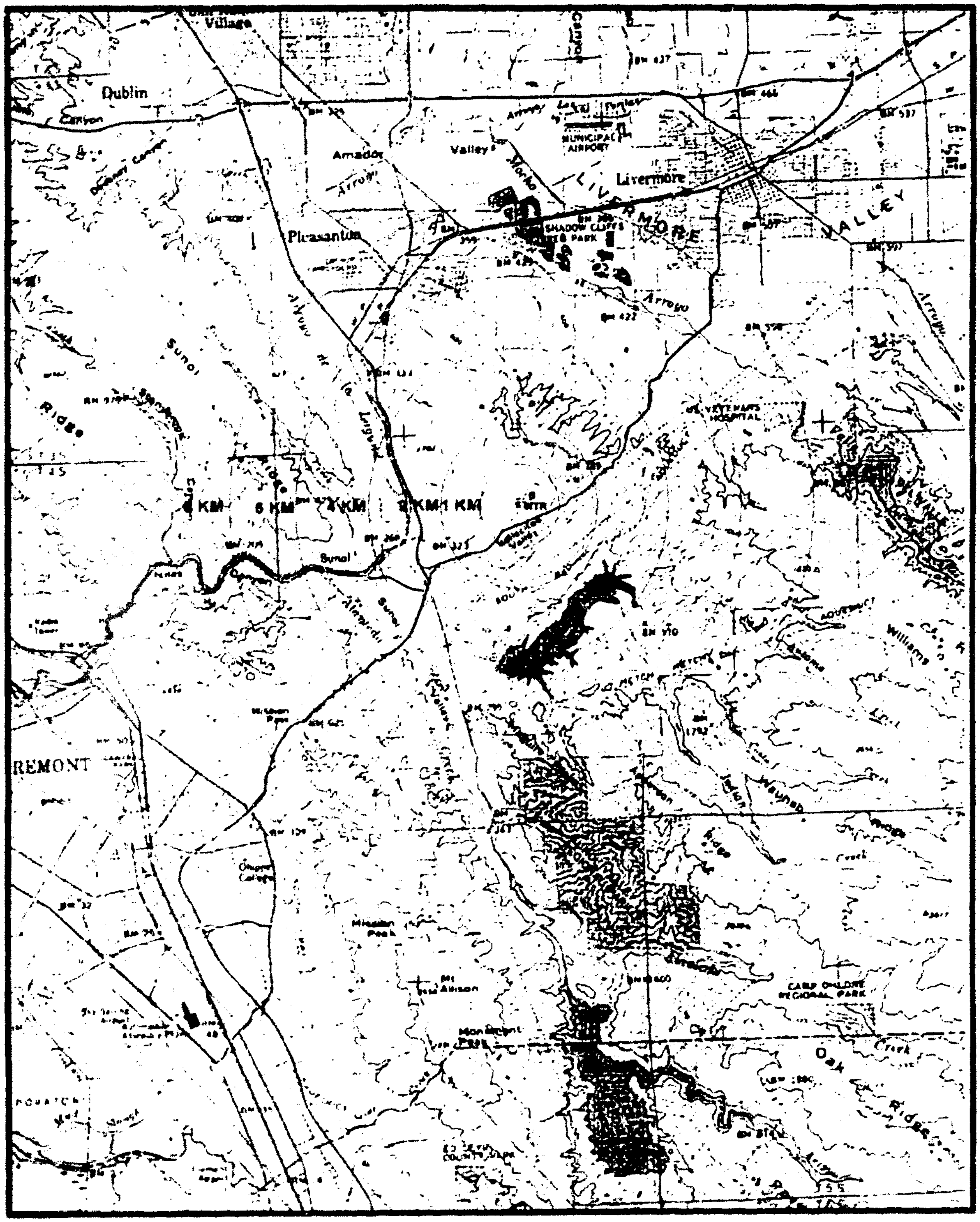


Figure 2-2. Population Distribution Map

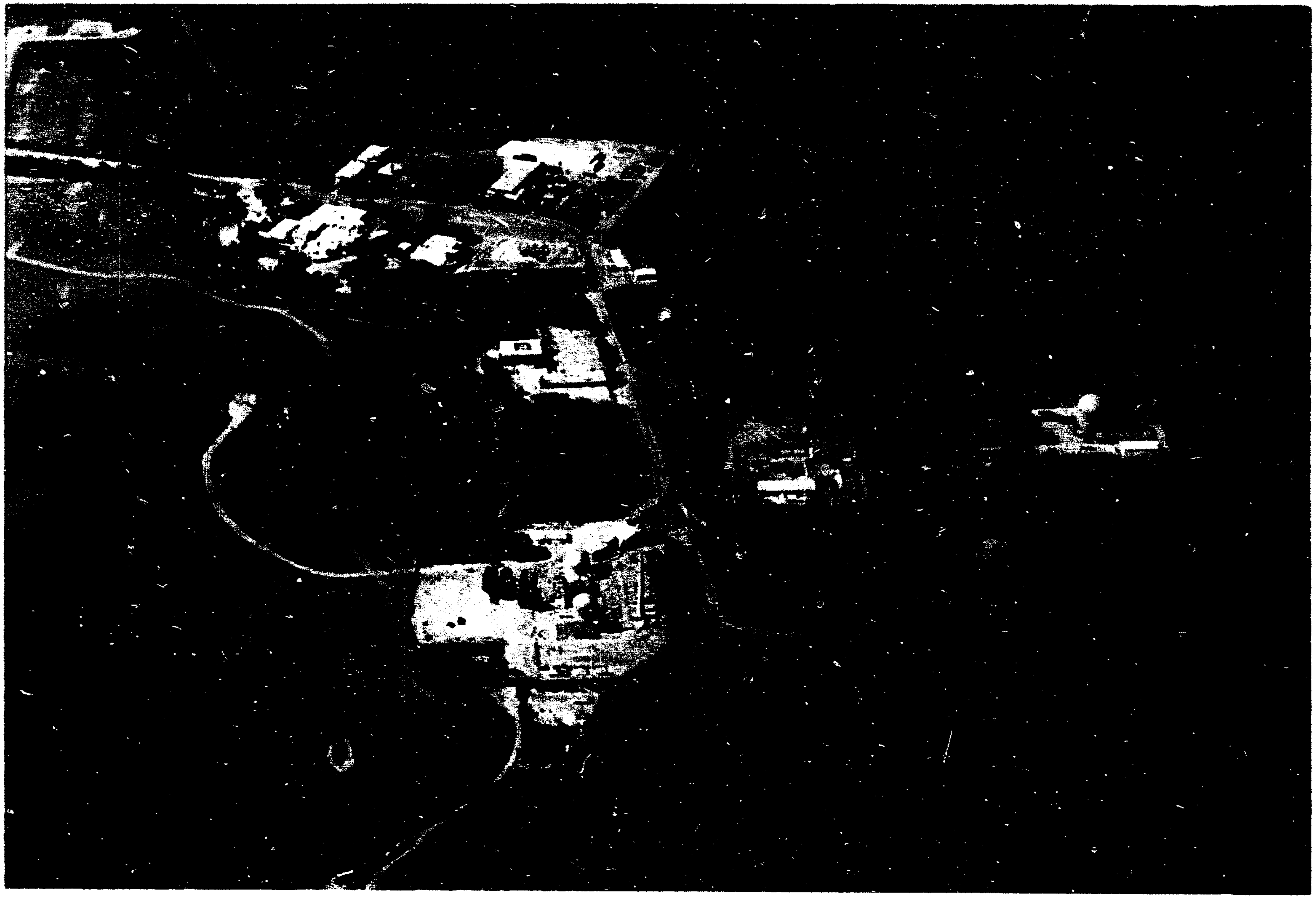


Figure 2-3. Vallecitos Nuclear Center and Surrounding Area

Safety

Violent storms are infrequent in this area. The main consequence of such storms would be the possible interruption of service from the Pacific Gas and Electric (PG&E) system (described in the Electrical Power section). Reactor shutdown is automatic in the event that normal electrical service is lost. There is a remote likelihood of major flooding at VNC. However, there is a possibility of substantial sheet flow caused by heavy rainfall and resultant runoff from the surrounding hills. All roadways and facilities are constructed with drainage to preclude damage caused by such an occurrence. Surface water drains away from the Site facilities to several natural ravines and man-made channels which empty into Vallecitos Creek.

Fire Protection

The design of the building containing the reactor and the reactor itself makes maximum use of noncombustible structural material. A 500,000 gallon water storage tank, located on a hillside above the Site, provides gravity flow to the Site. 100,000-gal of this stored water is reserved for fire protection. The Site fire truck (pumper) is garaged at the Building 104 area and auxiliary firefighting equipment is also available at strategic locations. A fire alert system for personnel instruction is provided. Fire teams for potential fire emergencies at the Site have been organized and are maintained in a trained status.

Water Supply

Water is normally supplied to the Site from the Hetch-Hetchy Aqueduct, which provides water to the city of San Francisco. A 14-in., 4600 m-long pipe has been installed from the aqueduct to the Site. The pumps installed have a capacity of 1,000,000 gpd. The pipe line capacity is over 3,000,000 gpd. The Calaveras Reservoir, located about 13-km south of the VNC, provides backup for Hetch-Hetchy.

Electrical Power

Electric service is supplied by PG&E to the Site substation where it is distributed to the Site facilities. The PG&E transmission line, passing just south of the Site, is fed from two directions in an electrical loop to ensure a most reliable and continuous parallel 60-kV supply to the substation. This substation feeds electrical power to the NTR.

VNC Facilities

Facilities located at the VNC are shown in Figure 2-4. The main laboratory buildings are located approximately 500-m north of the Vallecitos Road.

Building 102 contains the Radioactive Materials Laboratory and administrative offices. In the laboratory, post-irradiation studies and research and development activities are performed.

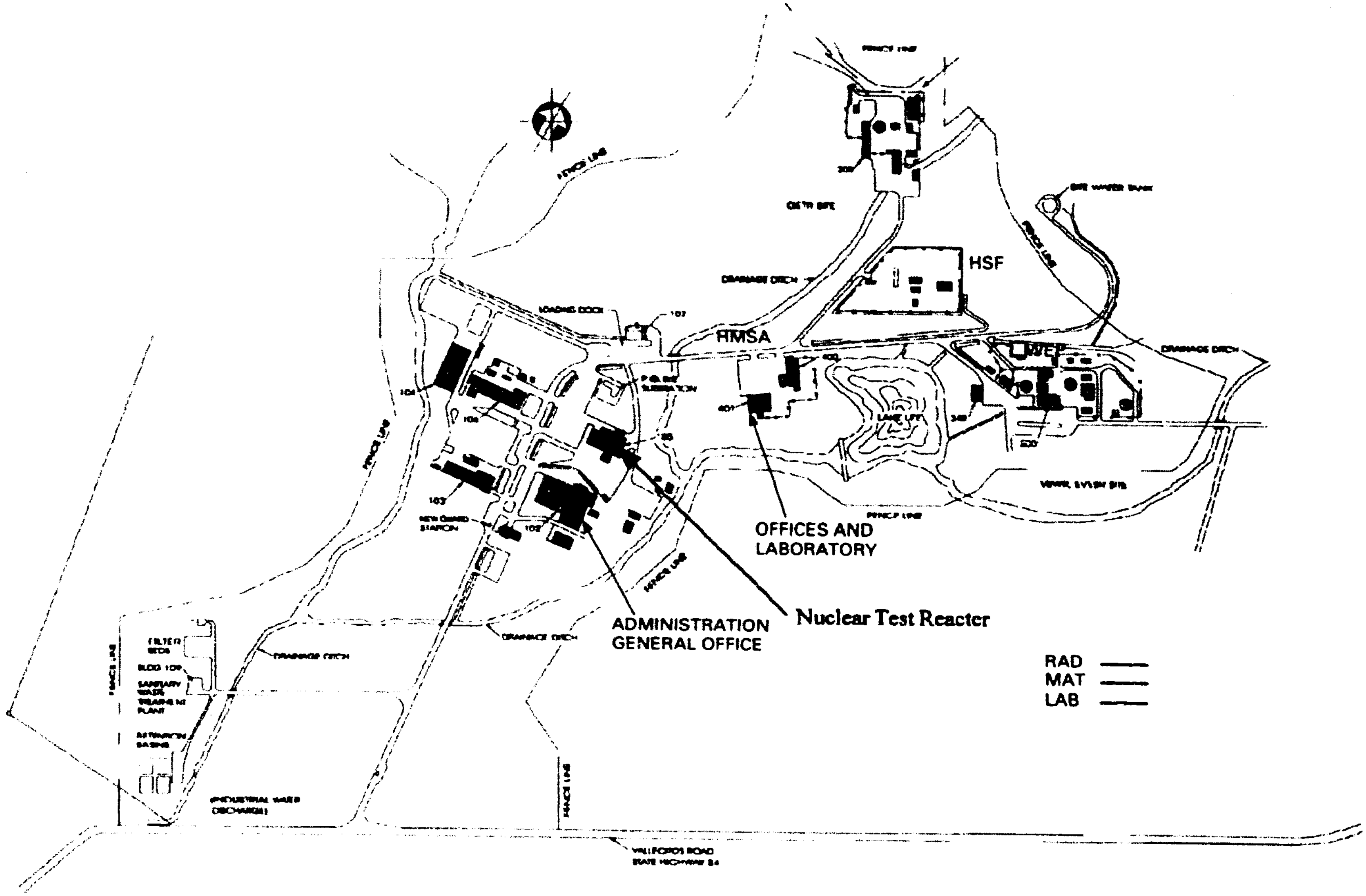


Figure 2-4. Vallecitos Nuclear Center Sitemap

Building 103 houses chemistry, metallurgy and ceramics research and development activities as well as extensive analytical chemistry laboratories.

Building 104 contains Site storage and warehouse facilities.

Building 105 contains offices and laboratories and houses the NTR and another shielded cell that was formerly used as a critical experiment facility.

Building 106 contains machine, sheet metal and facilities maintenance shops.

Building 107 is the Hazardous Material Storage Building.

Building 400 and 401 have been assigned to Chemical Engineering and Materials Development.

The General Electric Test Reactor (GETR), the Vallecitos Boiling Water Reactor (VBWR), and the ESADA Vallecitos Experimental Superheat Reactor (EVESR), developmental reactors located on the Site, are now in a deactivated status.

Temporary storage of solid radioactive waste is accommodated at the hillside storage facility. Facilities are available on the Site for handling, sorting, and processing liquid and solid radioactive wastes generated at all VNC nuclear facilities. A liquid waste evaporator facility is located near the hillside storage facility. A nonradioactive liquid waste chemical treatment plant and sewage treatment plant are located in the southwest corner of the Site.

2.1.1.1 Specification and Location

The VNC is located east of the San Francisco Bay in Alameda County, California. The VNC Site is on highway 84 approximately 4-km east of where highway 84 crosses highway 680.

On the USGS map, NJ 10-9, the Universal Transverse Mercartor location for NTR is 10SFS0263. The reactor is centered at LAT: 37.6, Lon: -121.8

2.1.1.2 Boundary and Zone Area Maps

The NTR is located at the east end of building 105. The operations boundary is defined as the emergency prepardness zone (EPZ) for the reactor facility. The operation boundary for NTR is the boundary defined by the portions of Building 105 occupied by NTR facilities.

Building 105, with reference to the other VNC buildings, is shown on Figures 2-4 and 2-5. The entire VNC Site boundary is also shown on Figure 2-5.

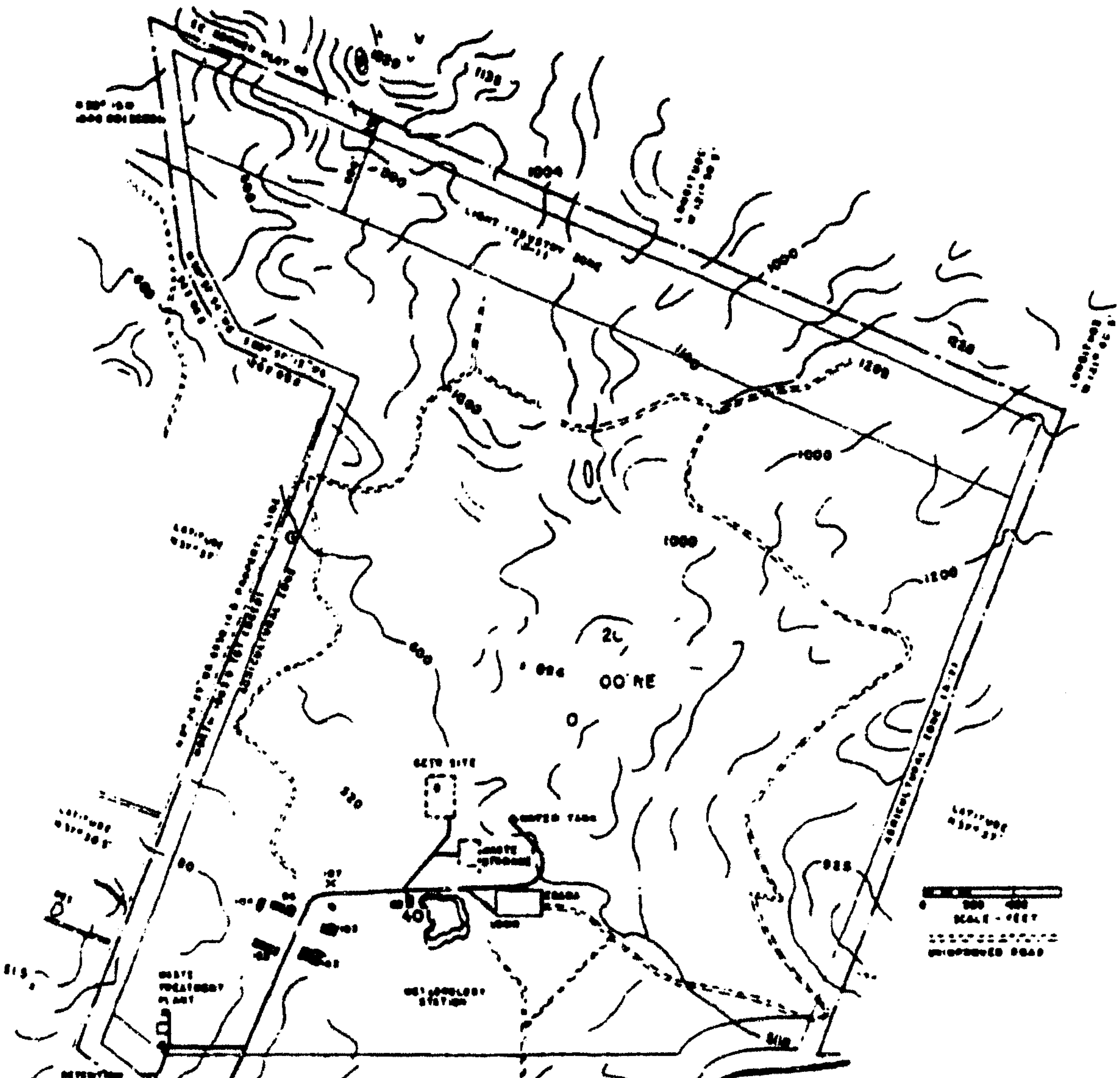


Figure 2-5. Topography Contour of Vallecitos Nuclear Center

2.1.2 Population Distribution

Population around VNC is shown on Figure 2-2, Population Distribution Map. On this map are one, two, four, six and eight kilometer radii with NTR at the center.

Inside the 1-km radius are four houses and one BMX race track (Sunol BMX) which are located on the south side of Vallecitos Road. There are approximately ten houses immediately west of the site. Eigh. more houses are west of the Site, within the 2-km radius.

The City of San Francisco's San Antonio Reservoir lies within the 4-km radius. The rate of population growth within this area is expected to be slow in the coming decades. The growth rate has been negligible over the last century. Excluding the GE property, a considerable portion of the land within this areas is owned by the City of San Francisco and it is believed the city will not sell the land. The major part of the remaining land is rugged terrain which does not attract industrial or residential buildings. Substantial parcels of privately owned land have been placed into the Alameda County Land Preserve Program under the California Land Conservation Act of 1965.

Minor land development has occurred within the 6-km radius. One location is 4 to 5-km to the west and northwest, associated with the expansion of the town of Pleasanton and the unincorporated areas of Happy Valley and Sunol. The other location is approximately 5-km to the east associated with the expansion of the town of Livermore. The Hetch-Hetchy underground aqueduct is in this area running approximately in the east west direction.

The 8-km radius contains two towns. Pleasanton (population 54,347) is approximately 7-km to the northwest. Over a 365-meter mountain range is the "outskirts" of Livermore at 8-km to the northeast. Livermore, the largest population center (population 60,000), is largely a bedroom community with some light industry and agricultural activities. The Livermore Division of the Veterans Administration Palo Alto Health Care System (population 400) is also located approximately 7-km to the east.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

Though both towns of Livermore and Pleasanton do contain some light industry, they are not considered industrial centers.

Highway 680 runs within the 4-km area in approximately the north south direction.

A Southern-Pacific and a Western-Pacific rail line also runs in the 4-km area, about half way in the north south direction, and then east west.

There are no military bases within 8-km of the VNC.

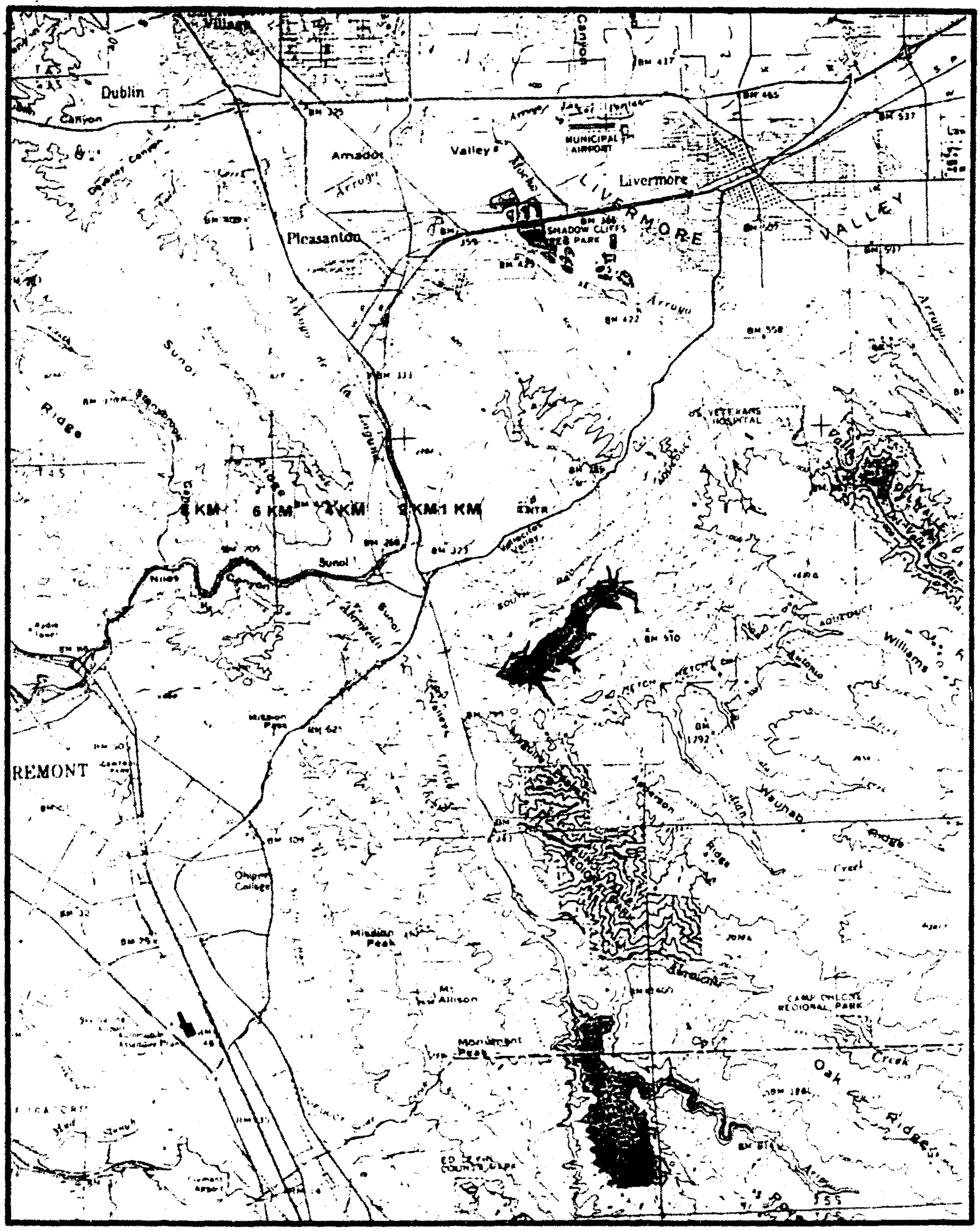


Figure 2-6. Population Distribution Map

2.3 METEOROLOGY

The VNC meteorology was studied in 1976 by Nuclear Services of Campbell, California. The summary of the results of their investigations is published in a separate document (1). Further studies were performed by the U.S. ological Survey. Results of these studies are published in a separate document (1a).

2.4 HYDROLOGY

The hydrology of VNC was studied in 1976 by Nuclear Services of Campbell, California. A summary of the results of their investigations is published in a separate document (1). Further studies were performed by the U.S. Geological Survey. Results of these studies are published in a separate document (1a).

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

Comprehensive geological and seismological studies have been conducted at and near VNC in the years 1977-1980. The results of these studies (References 2 through 6) have been reported and submitted to the Nuclear Regulatory Commission in relation to the General Electric Test Reactor, also located at VNC. The Nuclear Test Reactor design and the excess reactivity limit under which it operates makes off-normal condition evaluations (see section 11) insensitive to geological conditions and seismological parameter values.

3.0 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 DESIGN CRITERIA

The NTR is designed to operate at steady power levels up to 100 kW (thermal). The rate of power change is controlled by limiting the positive reactivity insertion. This is accomplished by limiting the control rod drive speed so that an operator may safely change power levels. A fast-period-scram will prevent the reactor power increase from being uncontrollably fast.

Facility shielding has been provided and beam shutters have been interlocked to maintain personnel radiation exposure less than 10CFR20 limits.

The NTR has an extended life core. While irradiated fuel is not stored at the NTR, the original fuel installed in 1957 is expected to last until 2007 before replacement is required. Potential excess reactivity is maintained within limits by periodically removing Manual Poison Sheet cadmium to compensate for positive excess reactivity loss from fuel burnup. About 2007, all Manual Poison Sheet cadmium will be removed and further fuel burnup will eventually prevent the reactor form attaining criticality.

The limiting accident at NTR is a step-positive reactivity insertion of 0.76$ while the reactor is operating at full power. Assuming that the reactor fails to scram from a fast-period-trip or an overpower trip, fuel damage does not occur. Also, assuming that the primary coolant system failed at the time of the positive reactivity insertion, fuel damage would still not occur even if the reactor failed to scram on a low flow condition.

Building 105 contains an automatic fire sprinkler system to suppress fire in the operating areas. Fire extinguishers and a trained on-site fire team is available for use. The fire team is dispatched rapidly by a two-way radio or site all-call system by a telephone call to the Emergency Control Center, via an emergency number. The reactor cell and the north room does not contain fire sprinklers. However, combustible materials in these areas is administratively minimized. The reactor graphite has been analyzed and found not to contain enough stored energy for combustion.

The NTR does handle pyrotechnic devices for neutron radiography. These devices are prohibited in some areas and are strictly controlled in other areas to prevent damage to the reactor and the control rods and to prevent the spread of radioactive materials outside posted radiation areas.

3.2 METEOROLOGICAL DAMAGE

The meteorological conditions at the site are generally very mild and any adverse conditions are not extreme. The reactor and the control room are contained in reinforced concrete structures which are extremely unlikely to fail. External cooling water is not required to maintain fuel temperature below the fuel and the fuel cladding melting points, so the secondary cooling water system is not required.

Under extreme conditions, the ventilation system may be damaged. This system, however, is not required since there is no credible reactor event that could cause fuel melt or a significant release of fission products from the fuel. In addition, the reactor confinement building would effectively limit small radioactive releases.

There has been no damage to the reactor systems caused by the wind in the last 40 years.

3.3 WATER DAMAGE

Major flooding at the NTR is extremely unlikely. The historical data show that flooding does not occur in this area. The NTR is situated on an area which slopes so that the whole site can accommodate significant runoff. Drainage ditches have been added when the site was built to prevent runoff from entering buildings. About 20 years ago, measures were added to Lake Lee to prevent the immediate release of all water from the Lake from affecting other site buildings.

Any flood water intrusion into the reactor cell or the control room would not cause a hazard. The Safety Rods and Control Rods and the reactor core can are located two feet above the floor and any shorting or grounding of electrical systems would immediately scram the reactor.

3.4 SEISMIC DAMAGE

California is a seismically active state. The San Francisco Bay Area contains many active faults, some near the Vallecitos Nuclear Center. Seismic studies have shown the possibility of a 0.8g vibratory ground motion at VNC. Large seismic events leading to collapse of the NTR and building structure have not been analyzed in detail. Nevertheless, even if catastrophic failure of the NTR facilities is assumed, there are no potential consequences resulting in fuel melt or gross dispersal of radioactive material. Compaction of the fuel, while essentially impossible mechanistically, would not cause the reactor to go critical since water loss, increased self-shielding in the fuel and the geometry change due to flattening of the cylindrical core are all negative reactivity effects. Also, deformation of the core causing fuel to contact the core can structure would improve heat-transfer and result in lower LOCA temperatures.

If a large seismic event occurs, it may be hypothesized that certain structures used to support the control and safety rod mechanisms as well as experiments might fail or move in such a manner as to withdraw the control rods and experiments from the core region and prevent operation of the safety rods. The cadmium poison sheets are manually positioned entirely within the graphite reflector, have no drive mechanisms, and will not move relative to the core during a seismic event. In order to preclude any mechanism which could lead to fuel melting, operation of the NTR is conducted in such a manner as to limit the reactivity available from control rods and experiments to less than or equal to \$0.76. The analysis of an \$0.76 step reactivity addition without scram has been analyzed, and the power excursion terminates without fuel melting or release of fission products.

In spite of the inconsequence of a catastrophic failure of structures and systems at the NTR, major structures and systems have been structurally analyzed and determined capable of surviving a 0.8g vibratory ground motion. These structures and systems include the following:

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.

3.5 SYSTEMS AND COMPONENTS

The NTR is a low temperature, low heat reactor which is very forgiving of off-normal and accident conditions. The NTR is operated in such a manner as to limit the potential excess reactivity to less than that required to cause fuel damage, assuming a failure to scram. This is accomplished by manually positioned cadmium poison sheets. The Manual Poison Sheets (MPS) are positioned entirely within the graphite reflector, have no drive mechanisms, and are mechanically restrained so they will not move relative to the core during a seismic event. The MPS are verified to be latched each time they are installed and the potential excess reactivity is verified to be below limits at each reactor startup.

The worst loss-of-flow accident (instantaneous seizure of the rotor in the single recirculation pump in the system) does not result in fuel damage or release of fission products even with a failure to scram and the reactor operating at 100 kW.

The total loss-of-coolant inventory in the core as the result of a rupture in the primary system with the reactor operating at 100 kW combined with a failure to scram does not result in fuel damage or release of fission products.

The reactor fuel was fabricated in accordance with GE Specification AP-RG-56-8-1.1, dated October 23, 1956. This specification includes material type, dimensional tolerances, a helium leak test, and a corrosion test.

The reactor cell and stack ventilation system are required in the event of a fueled experiment failure. It should be noted that fueled experiment irradiations are rare and have not been performed in the last 16 years. Nevertheless, the reactor cell would serve to contain the radioactive release while it is exhausted through the ventilation system and out the stack. A fueled experiment rupture is not likely during a seismic event. The reactor is not operated above 0.1 kW unless the ventilation system is operating. Operability is determined by the reactor cell pressure differential with the control room. The reactor power is reduced to 0.1 kW or less if the ΔP alarm is received during reactor operation. This maximizes the operability of the ventilation system whenever a fission product inventory is present in a fueled experiment.

Additionally, the ventilation system contains absolute filters prior to discharge from the stack. These filters must be at least 99% efficiency. The ventilation flow rate is required to be 1000 cfm to 3000 cfm.

4.0 REACTOR DESCRIPTION

4.1 SUMMARY DESCRIPTION

The NTR is a light-water-cooled, high-enriched-uranium, graphite-moderated and -reflected, thermal reactor with a nominal power rating of 100 kW. The reactor is 14 feet long (including thermal column and control rod drives) by 6 feet wide by 6 feet high. The reactor consists of a core in the form of an annular cylinder centered in a 5-foot cube of AGOT-grade graphite. The core container is a hroizontal aluminum cylinder 20 inches long and 18 inches in diameter, with an inner annulus diameter of 11.5 inches. The 11.5-inch-diameter cylindrical space is filled with graphite and traversed by the horizontal facility. The annulus thus formed is fitted with 16 fuel assemblies, each consisting of 40 aluminum-clad 2.75-inch-outside diameter (o.d.) uranium-aluminum fuel disks. The disks are spaced on aluminum rods to give an active length of about 15.25 inches. Up to six manual poison sheets, four safety rods, and three control rods are arrayed around the outside of the fuel container (Figure 4-1).

4.2 REACTOR CORE

Figure 4-2 is an assembly drawing of the present reactor fuel container. This container was put into service in 1976 after the previous container, which had been in service for approximately 18 years, sprung a leak in a weld area. The annular ends of the container, 0.5-inch aluminum plates, are welded to the inner and outer cylindrical skins, which are rolled aluminum sheets 0.25 and 0.0625 inch thick, respectively. The outer cylinder is made from two pieces welded together opposite the loading chute. Attached to the inside surface of each end plate is an aluminum circular raceway which supports and guides the core reel assembly. The reel assembly, in turn, supports the fuel assemblies. The core reel assembly is described in more detail in Subsection 4.2.5.

Openings are provided in the north end plate for the 1.5-inch primary coolant inlet and outlet lines. The inlet pipe is connected to a flow-distributor tube located inside the container below the core. A row of 25 0.25-inch holes is drilled into the lower side of the 1.375-inch flow-distributor tube, with the holes near the core midplane closer together than those at the ends to distribute water flow to correspond to power distribution along the core. The center-to-center distance between the 10 holes nearest the midplane is 0.4375 inch; the next three holes, toward each end, are on 0.5-inch centers; the next two holes on 0.75-inch centers; and the last two holes on 1-inch centers. An identical baffle tube located above the core (with holes on the top side of the tube) is connected to the outlet line.

A 3.25-inch opening in each end plate accommodates the drive mechanism for the reel assembly. These openings are in the area just below the junction between the loading chute and the fuel container. This drive mechanism is discussed in Subsection 4.2.5.

[redacted]

Figure 4-1. Vertical Section Through the NTR

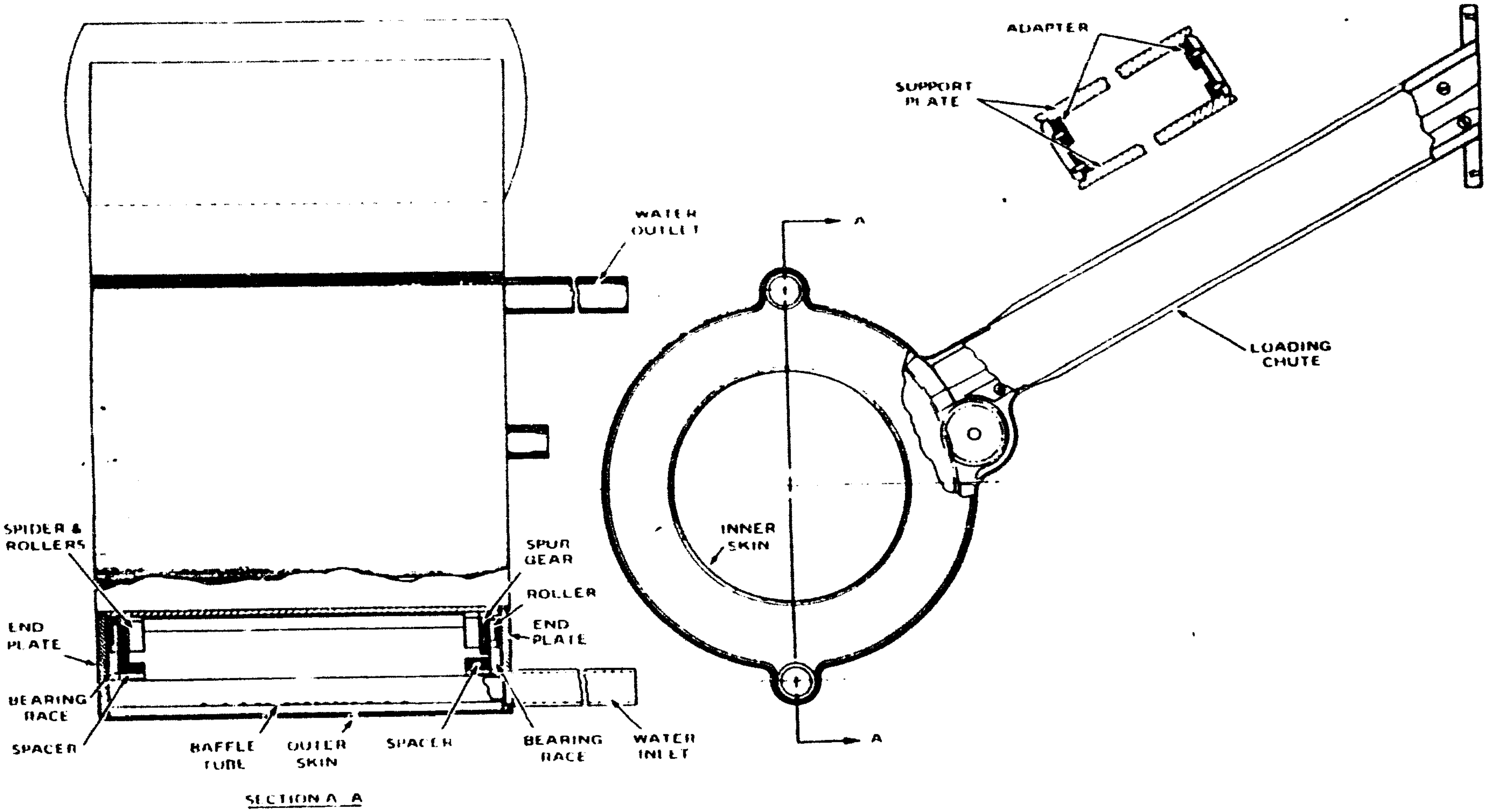


Figure 4-2. Fuel Container Assembly

Attached to the outer wall of the can, inclined upward at an angle of about 30 degrees with the horizontal, is a rectangular aluminum loading chute about 30 inches long, 20 inches wide, and 4 inches high. The chute is connected to the fuel loading tank. Slotted adapters fastened inside the chute provide a guide for the chute plug. The slots in the adapters line up with radial slots in the two circular raceways to guide the fuel loading tool to the core reel during refueling operations. When not in use, the loading chute is filled with an aluminum-clad graphite plug, and the chute opening is at least partially covered with an aluminum gate located in the storage tank.

Eight 0.75-inch aluminum tubes supported from brackets attached to the end plates run horizontally along the outside surface of the fuel container to the north face of the reactor. These tubes are guides for the control, safety, and neutron source rods. Six slotted graphite ways attached to the north end plate, oriented parallel to these guide tubes, serve as guides for the manually positioned poison sheets. The only other attachment to the container is a bracket fastened to the south end plate to help support the 5-inch horizontal facility.

4.2.1 Reactor Fuel

There are 16 fuel assemblies in the NTR core. Each fuel assembly consists of 40 fuel disks and spacers skewered on a shaft to form a shishkabob-type assembly. Lateral motion of the disks and spacers on the shaft is prevented by lock nuts placed on both ends of the shaft. All available spaces in the core support reel are filled by the 16 assemblies. [redacted] All the fuel disks in the core are from the original fuel load fabricated in 1957. The fuel was fabricated in accordance with GE Specification AP-RG-56-8-1.1, which included corrosion and helium leak testing.

When the fuel container was replaced in 1976, the fuel was removed, inspected, and leak checked. No cleaning, replacement, or repair was necessary.

Each space between the disks contains a 0.180-inch-thick aluminum spacer, and an additional 0.031-inch-thick aluminum washer is located in every other space. This arrangement produces an assembly with an active length of approximately 15.25 inches with the face-to-face distance between disks alternating between 0.24 and 0.27 inch.

Each fuel disk (Figure 4-3) is composed of a fuel bearing, flat, doughnut-shaped sandwich and an inner and outer edge ring. The three pieces were brazed together to clad the uranium-aluminum alloy meat. The sandwich consists of the uranium-aluminum alloy meat, which contains, on the average, [redacted] after 4.54% depletion as a 23.5 wt % alloy, plus 0.027 inch of 1100-series aluminum cladding on each face. This type of fuel was the industry standard for many years. The finished, flat, doughnut-shaped sandwich is 0.142 inch thick and has a 2.68-inch o.d., with a 0.58-inch-diameter center hole. The inner edge ring (0.516-in. inside diameter (i.d.), 0.033 inch thick, and 0.20 inch wide) fits into the center hole of the disk and is brazed to the faces of the sandwich cladding. A 2.75-inch o.d. outer edge ring, with the same width and thickness as the inner ring, fits around the circumference of the disk and is also brazed to the faces of the sandwich.

Figure 4-3. Fuel Disk

A 0.75-inch length of each end of the 0.5-inch aluminum support shaft is machined to provide a tip suitable for supporting and positioning the fuel assembly accurately in the core reel. Tolerances on the shafts, reel, and fuel container were set so that the maximum radial and circumferential movements of the shaft, and hence a fuel assembly, are less than 0.125 and 0.016 inch, respectively. The support tip extends past the ends of the core reel about 0.375 inch into the raceway; it is this section of the fuel assembly that is engaged by a tool during fuel handling.

4.2.2 Control Rods

Movable neutron absorbers located about the periphery of the fuel container include:

(1) Two boron-carbide-filled motor-driven coarse control rods;

(2) One boron-carbide-filled motor-driven fine control rod;

(3) Four boron-carbide-filled safety rods, motor-driven carriage with an electromagnet that attaches to the poison section, and scram force by cocked springs.

Orientation of these rods about the core is shown in Figure 4-1, and all run 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 poisons on a 9.5-inch radius or about 0.6 inch from the outside edge of the active core. The control and safety rods have horizontally mounted drive mechanisms that are supported from the north face of the reactor on a 5-foot-high aluminum support plate located about 4-1/2 feet in front of the north face. Both types of poisons were designed to perform a specific function.

The four safety rods were designed for rapid insertion to scram the reactor. The control rods (two coarse and one fine) were designed for the precise position control and indication required for analytical work during which the reactor is used as a detector.

Relative positions of the four boron-carbide-filled safety rods are shown in Figure 4-1. The poison section of each rod is 20 inches long and consists of a solid core of 1/2-inch-diameter boron carbide cylinders contained in a stainless steel tube. A plug in the north end of the stainless steel tube connects to an extension rod which has a rod stop armature assembly pinned to the other end. Two constant force spiral springs are attached to the extension rod so that withdrawal of the safety rod cocks the springs to store energy. Housings for the springs are secured to a support bracket attached to the north shield face. Also attached to this support bracket is a long piece of steel angle which passes through and is attached to the aluminum support plate to protect and support most of the hardware for the drive mechanism.

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. The electric motor is a 1/12 hp, capacitor start and run, 57-rpm output, gear motor connected for instantaneous reversing and is provided with automatic reset overload protection. Power to the motors is from the 115-Vac supply. 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.

The sequence of operations to withdraw a safety rod is as follows;

- Run carriage in to engage armature;
- Energize the electromagnet; and
- Run carriage full out to withdraw safety rod and cock the constant force spiral springs.

Upon initiation of a scram signal, the following sequence takes place:

- Scram signal deenergizes the electromagnets;
- Constant force spiral springs cause the armature to separate from the magnet and rapidly insert the safety rods; and
- Automatic signal runs all rod carriages to the fully inserted positions.

Deceleration of each scrammed safety rod is accomplished by an air dashpot-type shock absorber. The rod-stop armature begins to compress the air in the shock absorber housing about 4 inches from the full-in position. An orifice is provided to control the release rate of the compressed air and the deceleration rate of the rod.

Four microswitches are associated with each safety rod and drive mechanism. Listed below are the actions initiated by each switch:

(1) Drive-Out Limit Switch

- Interrupts motor circuit at outer limit of rod stroke.
- Energizes yellow light at console.
- Interlocked so that all safety rods must be withdrawn sequentially before any control rods may be moved, except when rod test panel is utilized.

(2) Drive-In Limit Switch

- Interrupts motor circuit at fully inserted limit of stroke.
- Energizes green light at console.
- Interlocked to prevent energizing the electromagnets unless all control rods and neutron source are fully inserted.

(3) Safety-Rod-In Position Switch

Energizes green light at console.

(4) Separation Switch

Operates in series with carriage-out limit switch to energize yellow light, indicating that rods and carriage are out. This interrupts voltage in the safety system and deenergizes the safety rod electromagnets.

The scram mechanisms for the NTR are essentially the same as those that operated satisfactorily on the TTR reactors for many years. The present mechanisms have operated without showing appreciable wear. As required by the administrative procedures, flight time for the rods is measured periodically, and, if it is found that a rod does not meet the required insertion time, the rod will be considered inoperable until repairs are made.

As a result of the lack of symmetry in the arrangement of the nuclear poisons around the core and the possibility of strong shadowing effects, the reactivity worth of the individual safety rods vary. In the normal core, the reactivity worth of the most effective safety rod is about 1$. The minimum worth of all four safety rods is 2$. If it is assumed the entire worth of the rod (for conservatism, use 1.5$) is realized in the first 18 inches of rod movement at the withdrawal speed of 1 inch/second, the average reactivity addition rate by withdrawal of the most effective safety rod is only 8.3 ¢/sec, which is a reasonable addition rate for manual control. Actually, the full stroke of the safety rod is approximately 28 inches and the rods are interlocked so that each rod

must be fully withdrawn before the next one can be started out. Thus, the reactivity in two safety rods cannot be added to the reactor in less than approximately 1 minute by normal withdrawal.

Figure 4-1 shows the location of the two coarse control rods with respect to the core and other neutron poisons. The poison section of each rod is 16 inches long and consists of a solid core of 1/2-inch-diameter by 2-inch boron carbide cylinders contained in a stainless steel tube. A plug in the north end of the tube is connected to an extension rod which 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. The gear motor is identical to the one described above for the safety rods. Power to these motors is supplied by the 115-Vac supply. Pushbutton switches at the console permit manual control. As with the safety rods, scram provides an automatic signal which takes control away from the operator and runs the rods to their fully inserted position, provided ac power to the console is maintained. Position indicators are provided to indicate the position of each rod, to the nearest one hundredth inch, over the full stroke for rod movements in either direction.

Two limit switches on each control rod drive mechanism perform the following functions:

(1) Rod-In Limit Switch

- Energizes green light at the console.

- Interlocked to prevent energizing the electromagnets unless all control rods (in addition to the safety rods and neutron source) are fully inserted.

- Interrupts the motor circuit at the full in position.

2. Rod-Out Limit Switch

- Energizes yellow light at the console.
- Interrupts motor circuit at the outer limit of the stroke.

Location of the fine control rod with respect to the other poisons is shown in Figure 4-1. The poison section is 18 inches long and consists of a solid core of 0.365-inch-diameter by 2-inch boron carbide cylinders contained in a stainless steel tube that extends through the north face of the reactor to the drive mechanism. An aluminum rod is used to fill the remainder of the tube between the boron carbide cylinders and the drive mechanism.

The stainless steel tube containing the poison connects to a nut block that travels on a lead screw. The lead screw is rotated through a right angle gear box by a gear motor. The motor is 1/2-hp, 1725-rpm, (173-rpm output) with an electrically operated brake. Power to the motor is supplied by the 115-Vac supply. Pushbutton switches on the console are used for remote manual control.

The fine control rod is automatically driven to the fully inserted position following a scram, provided ac power to the console is maintained.

Two limit switches actuated by the traveling nut block perform the identical functions discussed above for the control rods. Position indicators are provided to indicate fine rod position to the nearest one hundredth inch over the entire stroke for rod movement in either direction.

The control rods were calibrated using the rising period technique, with the rods in an essentially unshadowed condition. The results indicate the total worth of all the rods is approximately 2.3$. The speed of withdrawal of each coarse control rod drive is 0.140 inch/second. The speed of withdrawal of the fine control rod drive is 0.145 inch/second. If it were assumed that all three rods were withdrawn simultaneously, the average reactivity addition rate would be approximately 2¢/sec, which is an amount that is easily manageable.

There is a third type of control designed to change reactivity, but the change does not take place during reactor operation. These are manually positioned poison sheets and are used to limit the reactivity available to the operator or to increase the shutdown margin. The manual poison sheets are inserted or removed manually, during shutdown, in the graphite around the fuel container (Figure 4-1.), through access holes provided in the north shield.

The sheets consist of 0.032-inch-thick cadmium, 19 inches long laminated between two sheets of 0.08-inch-thick 6061-T6 aluminum 3-inch wide by 40-1/2-inch long. The width of the cadmium in each sheet is as follows:

(1) Full sheet: 2.75 inches
(2) 3/4 sheet: 2.06 inches
(3) 1/2 Sheet: 1.38 inches
(4) 3/8 sheet: 1.03 inches
(5) 1/4 sheet: 0.69 inches
(6) 1/4 sheet: 0.69 inches

All manual poison sheets used are equipped with a spring-loaded latc[illegible]andle that latches to a special latch plate on the north face of the aluminum box that contai[illegible] graphite reflector assembly. This latching assembly provides positive restraint of the m[illegible]al poison sheets with respect to the reactor assembly. The reactor shall be operated using only those positions which have the latches installed – currently positions 1, 2 and 5.

The manual poison sheets do not have a drive mechanism or any automatic functions associated with them. A sheet can only be inserted or withdrawn by entering the reactor cell to remove a shield plug in the shield face. The sheet is then positioned by engaging the sheet handle with a special latching tool and physically unlatching or latching it prior to removal or full insertion. The removed sheets are stored in a rack in the reactor cell and are accounted for before reactor startup.

Reactivity worth of individual manual poison sheets in the core with all safety rods inserted were obtained by utilizing a pulsed neutron source. The worth of a full sheet was approximately 1$, and a half-sheet was worth about 0.5$. The worth of all six manual poison sheets is approximately 3.0$. In a typical core configuration (3/8 sheet in slot #5), the worth of the manual poison sheet is approximately 0.7$ (subject to change based on experiment worth, etc.). Excluding the transient temperature worth (reactivity addition from the primary coolant temperature change) and the experiment transient worth, the typical excess reactivity available from the control rods at the console is 0.3$ (subject to change based on experiment worth, etc.).

4.2.3 Neutron Moderator and Reflector

The graphite reflector-moderator is a 5-foot cube of reactor-grade (AGOT) graphite which not only serves as a neutron reflector and moderator, but also physically supports the fuel container. The fuel container is centered in the reflector with the core cylindrical axis horizontal. Many small pieces (primarily 4 inches by 4 inches by varying lengths) were machined carefully and stacked together to form the 5-foot cube. The reflector is contained and supported by the aluminum box and base discussed in Subsection 4.2.

Among the numerous items penetrating the reflector are 1) the fuel loading chute through the west face, 2) the control rod, safety rod and neutron source guide tubes, 3) the manually positioned poison sheet slots, 4) the cable held retractable irradiation system and 5) the core reel drive shaft through the north face. The horizontal facility tube and the experiment tube traverse the reflector from the north to the south face (the horizontal facility tube continues through the thermal column); the vertical facility extends from the top to the bottom of the reflector. Section 10 contains a discussion of the vertical and horizontal facilities and the thermal column. Several modifications were made to the main graphite pack during the major outage of 1976 and are also described in Section 10. The modifications added new irradiation facilities capabilities.

Two special sections of the reflector were designed to be removable: the set of blocks situated between the fuel container and the north face; and, the group of blocks that fill the 11.5-inch-diameter hole formed by the inner skin of the fuel container. These sections make it possible to inspect the fuel container without disturbing the rest of the reflector.

4.2.4 Neutron Startup Source

A reactor start-up neutron source is installed on an electric motor drive mechanism similar to the control rod drives. The source drive has the same controls and indications as a control rod drive, with the exception that continuous position indication is not provided. The same interlocks as those on the control rods are provided (the safety rod magnets cannot be reenergized until the source is full in), except that it is not necessary to pull any safety rods to withdraw the source. Following a scram, the source automatically runs to the fully inserted position. The source travels in a guide tube identical to that used for the control rods, and the limit switches are adjusted so that the source moves about 30 inches from the full-in to full-out positions. A 0.2-Ci radium-beryllium source emitting about 10^6 n/sec is used for a startup source. It is an R-Monel encapsulation approximately 1/2-inch in diameter and 3-1/2 inches long, attached to an aluminum

extension rod that connects to the source drive mechanism. The source-detector arrangement provides at least the minimum neutron flux signal required for the nuclear instrumentation for startup and also gives good indication of subcritical multiplication.

4.2.5 Core Support Structure

The fuel container rests on the sections of the 5-foot graphite cube pack beneath it. The graphite was machined to close tolerance, then fitted around the container and loading chute to provide maximum support. An aluminum box constructed of 0.375-inch plate and 2-inch angle aluminum contains the 5-foot graphite cube on all faces except the bottom and the south face. The south face is joined to the 4-foot graphite cube thermal column. A 0.031-inch cadmium liner is provided for the north and east sides of the box. The box containing the graphite cube rests on a base consisting of a 0.625-inch-thick aluminum plate fastened to a framework of 5-inch aluminum I-beams. The I-beam base is clamped to steel support plates anchored to the reactor cell floor. At the time the reactor was installed, the base was shimmed level and grouted.

A 0.75-inch length of each end of the 0.5-inch aluminum support shaft is machined to provide a tip suitable for supporting and positioning the fuel assembly accurately in the core reel. Tolerances on the shafts, reel, and fuel container were set so that the maximum radial and circumferential movements of the shaft, and hence a fuel assembly, are less than 0.125 and 0.016 inch, respectively. The support tip extends past the ends of the core reel about 0.375 inch into the raceway; it is this section of the fuel assembly that is engaged by a tool during fuel handling.

Located within the fuel container can is the core reel assembly, which consists of a pair of spur gears tied together with eight separator bars. Radial slots in these spur gears receive the machined tips of the fuel assembly shafts to support and position each fuel assembly. Stainless steel rollers attached to the outer face of each gear guide and support the reel in the radial raceways attached to the fuel container end plates (Figure 4-2). A reel drive mechanism is provided to rotate the entire reel assembly to any desired position with respect to the loading chute.

The two large spur gears are almost identical. These gears, made of 0.34-inch thick aluminum, have a 16.3-inch o.d. and a 12.9-inch i.d. Eight stainless steel rollers are bolted to the outer face, and eight triangular-shaped separator bars are bolted to the inner face (through a 0.75-inch-thick spacer ring) of each gear. The roller and raceway on the north end are V-shaped to prevent lateral motion of the entire reel. Sixteen equally spaced slots, 0.189 inch wide, are cut into each gear to receive the machined tips of the fuel assembly shafts. These radial slots terminate at an inner radius that places the center of fully inserted fuel elements on a 7.48-inch radius.

The reel drive assembly consists of two pinion gears (3-inch o.d. by 0.34 inch thick) keyed to a single 0.625-inch shaft. The shaft seal is a double O-ring seal with a tattle-tale petcock. Outside the reactor shield the shaft extends through a right-angle gear box, to the top of the fuel loading tank, to a hand-operated drive wheel with a dial indicator. The dial indicates the orientation of the reel and the position of any fuel assembly. The reel may be rotated to any desired position for core work; once the work is complete, the reel is no longer moved. Movement of the reel

assembly is permitted only when the reactor is shut down. Since the reel can be rotated only from within the reactor cell and is locked in position, unauthorized or unintentional movement during reactor operation is not considered credible.

4.3 BIOLOGICAL SHIELD

Reactor shielding is such that personnel radiation exposures throughout the building can easily be maintained within established limits with the reactor at full power. A list of typical levels in and around the facility, with the reactor at 100 kW, is given at the end of this section. During initial operations under new conditions, radiation dose rates and personnel exposures are closely monitored. If required, modifications are made to the shielding or the procedures to ensure continued compliance to established limits and consistence with As Low As Reasonably Achievable (ALARA) practices.

Reactor Shield

At present, radiation shielding for the reactor includes the graphite and the cadmium-lined aluminum frame which were discussed previously, local shielding in the vicinity of the reactor, and the thick concrete walls of the reactor cell and the south cell. The arrangement of most of these materials can be seen in Figures 1-1 and 10-1.

The reactor is situated in a high-density concrete alcove (10 feet, 8 inches wide by 10 feet high) in the reactor cell so that the 5-feet-thick alcove walls provide adequate shielding on the east and south sides. On the west side, the fuel storage tank provides 4 feet of water shielding in addition to the 3-foot-thick alcove wall. A lead shield wall, 8 inches thick in front of the reactor with 6-inch-thick extensions on each side, provides shutdown gamma shielding for the control rod drive area. A removable shield slab of reinforced heavy concrete covers the top of the reactor. The 1-foot-thick shield is a four-sided, irregularly shaped, reinforced, heavy concrete slab with a 48-inch-diameter hole centered above the reactor core and an 8-inch-diameter hole that may be used to gain limited access to the east face of the reactor. A 16-inch-thick stepped concrete plug is available for closing the 48-inch hole when that facility is not in use. The large plug contains a 6-inch-i.d. hole directly above the vertical facility to permit access to this facility without removing the large plug. This slab and plug are sometimes removed to perform tests and experiments during operation, install equipment, or perform maintenance when the reactor is shutdown. During periods when this slab or large plug is removed, radiation monitoring is performed and working time or reactor power is limited, if necessary.

Reactor Cell

Chapter 6 contains a detailed description of the reactor cell, including a list of the cell penetrations; the high-density concrete alcove was discussed at the beginning of this section.

The remainder of the cell is constructed of ordinary reinforced concrete and provides the following shielding:

(1) 5-foot-thick north and east walls;

(2) 4-foot-thick west wall between the NTR and adjacent laboratory cell;

(3) 3-foot-thick south wall between the control room and reactor cell, containing the shield door discussed in Chapter 6; and

(4) 3-foot-thick roof.

Radiation levels listed in the last part of this section demonstrate the effectiveness of the cell as a radiation shield. Whenever new operating or maintenance conditions are encountered, radiation surveys are made to determine that existing shielding is adequate and consistent with ALARA practices. Either temporary or permanent improvements in the shield are made if the results of the survey indicate they are necessary.

South Cell

The main source of radiation in the south cell is that which comes directly from the reactor and that which is induced in experiments and experimental equipment. As shown in Figure 10-1, high-density concrete walls completely surround the experiment area. The wall between the cell and the control room is 2 feet thick by 7.5 feet high and contains a shield door consisting of about 8 inches of high-density aggregate, 5.25 inches of paraffin, and two 0.125-inch sheets of boral. The portion of the east wall of the cell separating the south cell and the shop area is 3 feet thick. A false ceiling is installed above the south cell and consists of 1/4-inch lead sheet and staggered layers of 4-feet by 8-feet by 0.25-inch Masonite to a thickness of 2 inches. Additional shielding has been added in the form of a shield wall extension of the south wall of the south cell. The wall is 50 inches by 59 inches by 8 inches, with a 15-inch by 24-inch by 8-inch hole for movement of experiments, and is constructed of high-density concrete blocks enclosed in a support structure. Shielding material has also been added above the south cell entry ceiling and on the back porch of the NTR adjacent to the penetration through the east wall of the south cell.

In 1988 two-inch thick 7% boron-polyurethane sheets were added to the south cell ceiling and an 8-inch thick high density concrete shield wall 38-inches wide was built in the south cell doorway to reduce the dose rate in the control room.

Radiation coming from the reactor is reduced by the presence of a 4-foot-thick graphite thermal column. In front of that is approximately a 3-1/2-foot-thick x 7-foot wide density concrete block wall that stairsteps from 4-feet to 8-feet high and a 4-inch lead brick wall. A thick shutter

consisting primarily of lead and borated polyethylene is operated from the control room and shields radiation from the horizontal cavity. An interlock alarm system is provided which:

- Prevents opening the door if higher than normal levels of radiation are present.
- Initiates automatic closure of the shutter to reduce radiation levels.
- Initiates audible and visual alarms to warn personnel of higher than normal radiation levels.

In addition, a photo-cell alarm is provided at the south cell door access point which will sound whenever the light beam is broken when it is armed.

North Room Modular Stone Monument (MSM)

The MSM, which is discussed in Chapter 10 (Figure 10-4) provides shielding in the north room from:

(1) The north neutron radiography beam; and
(2) Radioactive objects during the neutron radiographic process.

The MSM is made of high-density concrete modular blocks (so design changes may be made easily in the future) and houses a borated lead polyethylene beam catcher. Additional lead shield closures may be utilized, as required, to further reduce the radiation from two of the penetrations, as shown in Figure 10-4.

Radiation Levels

A list of typical radiation levels in the areas of the NTR facility, while the reactor is operating at 100 kW, is given below. Unless shielding changes are made, the listed radiation levels are all proportional to reactor power. The values listed include contributions from fast, intermediate, and slow neutrons and gamma rays.

Location	Shutters* Open mRem/h	Shutters* Closed mRem/h
At reactor console	3.0	≤ 1
Hallway south of control room	≤ 1	≤ 1
Building 105 equipment room (2nd floor)	≤ 1	≤ 1
Reactor cell roof directly above reactor (top shield slabs in place)	65	65
South cell roof	57	2.5
Setup Room	≤ 1	≤ 1
North Room (center of room)	≤5.5	≤ 1

*North and south cell horizontal cavity shield shutters

4.4 NUCLEAR DESIGN

4.4.1 Normal Operating Conditions

The reactor consists of a core in the form of an annular cylinder that contains 16 fuel assemblies as discussed in Sections 4.1 and 4.2. The core is centered in a 5-foot cube of AGOT-grade graphite. Arrayed around the outside of the fuel container are four safety rods, three control rods and up to six manual poison sheets. All fuel assembles, control rods and safety rods are in fixed positions that are not changed.

Normal operation of the NTR is at powers no greater than 100 kW, with maximum temperature and pressure in the core at 150 degrees F and 20 psia, respectively.

As a result of these very conservative operating conditions, none of the nuclear characteristics (except the water moderator temperature coefficient of reactivity) varies significantly with normal temperatures.

The reactor configuration is controlled to ensure that the potential excess reactivity is less than or equal to 0.76$. NTR burns ~0.06$ positive excess reactivity per year. The planned core configurations during the life of the reactor are to remove enough cadmium from the remaining manual poison sheet to maintain normal operation and still ensure that the potential excess reactivity is less than or equal to 0.76$.

Low power generation of the NTR makes reactivity changes from fuel burnup and fission product poisoning small. During the past 39 years, the reactor has accumulated approximately 133 MWD of operation. Based on this history, the total reactivity losses are estimated to be 0.95$ from the fuel burnup, 0.71$ from aggregate fission product poisoning, and 1.03$ (at equilibrium) from samarium-149 poisoning. Plutonium buildup in the NTR is negligibly small.

Selected reactivity worth of reactor components are listed in Section 4.5.2 Reactor Core Physics Parameters.

The following are administrative and physical constraints that prevent inadvertent addition of positive reactivity.

During reactor operation the reactor cell door is locked so that core changes are not possible. The manual poison sheets are physically latched and cannot move during operation. During operation, then, the only positive reactivity additions possible are from moveable experiments, coolant flow changes and movement of control and safety rods. Control and safety rods are manipulated by licensed operators in accordance with written procedures. These reactivity additions are limited physically (water coefficient of reactivity) and by design (control and safety rod drive speeds and experiment reactivity worth).

Entry into the reactor cell, when the reactor is critical, is authorized by special procedure (Engineering Release) describing the operation to be performed. This procedure must be approved by the Manager, NTR, and reviewed by the Manager, Nuclear Safety, prior to entry.

When the reactor is operating, there is a south cell door/shutter interlock to prevent inadvertent entry into the south cell with the south cell shutter open. An electric photocell light mechanism causes an audible alarm to actuate when an entry into the south cell is made.

Entry into the south cell is permitted when the reactor is critical if the power is stable, the entry does not distract the operator and no more that the minimum number to safely perform the task is permitted.

During shutdown, positive reactivity changes are possible by safety and control rod movement, manual poison sheet movement and horizontal facility changes.

Safety and control rod movement defines the reactor as not secured. When the reactor is not secured the minimum staffing is composed of the following: A licensed operator in the control room. A second person at the site familiar with NTR emergency procedures and capable of carrying out facility written procedures. A licensed SRO shall be present at the facility (Bldg 105) or readily available on call.

A licensed SRO shall be present at the NTR facility during manual poison sheet changes. Each individual irradiation, in the horizontal facility, shall be reviewed to determine that the irradiation satisfies the approved irradiation criteria. This review is documented by the signature of an SRO.

4.4.2 Reactor Core Physics Parameters

Several important features of the NTR that affect the nuclear characteristics result from an effort to enhance the performance of the reactor as a sensitive detector of reactivity changes. Among these features are the low critical mass, the fuel-to-sample geometry, and sensitive control system. The reactor is constructed so that samples placed in the horizontal facility are in a neutron flux that is higher than the flux in the fuel lattice. The sensitivity of the reactor as a detector is proportional to the ratio of the thermal flux at the sample to that in the fuel lattice. A number of nuclear parameters are listed in Table 4-1.

Figures 4-4 and 4-5 illustrate the thermal neutron flux profiles in the horizontal facility and in three of the manual poison sheet slots. The profiles in the manual poison sheet slots are the profiles of the three upper slots on the east side of the core and are expected to correspond very closely to the axial neutron flux and thermal power distribution in the adjacent section of the core.

Discussions of temperature coefficients of reactivity usually separate the total coefficient into a nuclear cross section effect and an effect caused by density and volume changes in the system. These two major effects are subdivided further according to the location of material that is affected (i.e., fuel, moderator, or coolant) and the speed with which the effect occurs. For an NTR-type reactor, such a complete breakdown is not necessary. By far, the dominant effect for accident analysis is that of density changes, including displacement of cooling water by expansion of fuel within the fuel annulus. Although the results of earlier studies indicate that a positive effect may result from heating the reflector graphite, this temperature change would be too slow (on the order of minutes) to affect a nuclear excursion significantly. The effect from a

temperature change in the fuel annulus is observed in fractions of seconds. The over-all temperature coefficient of the fuel annulus was measured and found to be positive up to 124°F. As temperature is increased above 124°F (the turnover point), the coefficient becomes negative. The coefficient was measured between 65 and 156°F, and, for analyzing accidents, it is assumed the data can be extrapolated to boiling. The measured coefficient is given by:

$$dp/dt = -5.7 \times 10^{-3} (T-124)\ \text{¢}/°F \qquad (4\text{-}1)$$

where T is the primary coolant temperature in (°F) and ρ is the reactivity of the system (¢). This coefficient is not affected by fuel burnup and is not expected to vary significantly with core life.

An experiment was performed to check the sign of the void coefficient of reactivity. In this experiment, the reactivity effect of moving pieces of aluminum from the core was positive; therefore, the void coefficient was negative, as required. The magnitude of the void coefficient was not measured directly, but was determined from the results of the temperature coefficient experiment. In this determination, the source of reactivity change in the temperature coefficient is considered to be caused by the density changes only and is interpreted as an effect from void buildup. Extrapolation of the temperature coefficient data yields a void coefficient of -5.7 ¢/% void above the temperature coefficient turning point of 124°F.

Changes in reactivity caused by inserting materials during experiments are largest for experiments in the horizontal facility. Several measured reactivity effects in the horizontal facility and the vertical facility are given in the table at the end of this section. As indicated by the fact that the thermal column increases the flux at the south face of the reflector, experiments at the face of the 5-foot graphite cube, which contain large quantities of reflector materials, could have a small reactivity effect. However, during experiments performed to date, such an effect has never been observed.

Table 4-1

NUCLEAR PARAMETERS

Fuel Loading	
Critical mass (cold, 0.28 inch between disks)	[redacted] U-235 (512 disks)
Actual initial loading	[redacted] U-235 (640 disks)
Actual loading after 133 MWD of operation	[redacted] U-235
Reactivity Worth of Movable Nuclear Poisons	
All three control rods (typical[a] operational core)	0.016 Δk/k (2.3$)
All four safety rods (minimum)	0.014 Δk/k (2.0$)
All six Manual Poison Sheets (MPS)	0.021 Δk/k (3.0$)
Reactivity (Console Excess with Typical[a] Operational Core)	
All four safety rods and three control rods withdrawn	+0.002 Δk/k (+0.3$)
All four safety rods withdrawn and all three control rods inserted	-0.014 Δk/k (-2.0$)
All four safety rods inserted and all three control rods withdrawn (minimum)	-0.012 Δk/k (-1.7$)
All four safety rods inserted and all three control rods inserted (minimum)	-0.028 Δk/k (-4.0$)
All four safety rods inserted and all three control rods inserted and all six manual poison sheets inserted (minimum)	-0.043 Δk/k (-6.1$)
Reactivity (Console Excess)	
All four safety rods and all three control rods inserted and all manual poison sheets withdrawn	-0.022 Δk/k (-3.1$)
Reactivity Addition from Primary Coolant Temperature change (from 75 to 124°F).	+0.00048 Δk/k (+0.07$)

[a] 3/8 MPS in slot #5, neutrography source log in horizontal cavity, graphite in vertical and other experiment facilities (or similar arrangement); excludes temperature and experiment transient worth.

Table 4-1

NUCLEAR PARAMETERS (Continued)

Miscellaneous Reactivity Effects	
Removing all graphite from central sample tube (3-in. cavity)	-0.004 Δk/k (-0.6$)
Filling central sample tube with water (3-in. cavity)	-0.02 Δk/k (-3$)
Removing all graphite from vertical facility	-0.008 Δk/k (-1.1$)
Removing the fuel loading chute plug	-0.009 Δk/k (-1.25$)
Equilibrium xenon at 100 kW	-2.3×10^{-3} Δk/k (-0.3$)
Yearly fuel burnup (typical use)	-4.6×10^{-4} Δk/k (-0.06$)
1 g U-235 in small (25 g) low-enriched fuel samples in central sample tube	~+2 x 10^{-4} Δk/k (-0.03$)
1 cm^2 of absorber in small (50 g) nonfuel sample in central sample tube	~-1.8 x 10^{-4} Δk/k (-0.025$)
Coefficients of Reactivity	
Temperature coefficient in	
Water coolant (measured)	-5.7×10^{-3} (T-124)¢/°F
Inner graphite (calculated)	$+0.17 \times 10^{-3}$ ¢/°F
Outer graphite (calculated)	$+4.1 \times 10^{-3}$ ¢/°F
Average void coefficient	-5.7 ¢/% void
Doppler coefficient	Negligible
Mean Lifetime of Prompt Neutrons	2×10^{-4} sec
Neutron Fluxes at 100 kW	
Average thermal flux in fuel	7×10^{11} nv
Peak thermal flux in central sample tube	2.5×10^{12} nv
Peak thermal flux in CHRIS	8.0×10^{11} nv
Thermal flux at face of thermal column	7×10^{8} nv
Thermal flux at face of 5-ft graphite cube	5×10^{10} nv
Miscellaneous Parameters After 133 MWd of Operation	
Reactivity lost due to fuel burnup	0.95$
Reactivity lost due to aggregate fission product poisoning	0.71$
Reactivity lost due to samarium-149	1.03$ (at equilibrium)

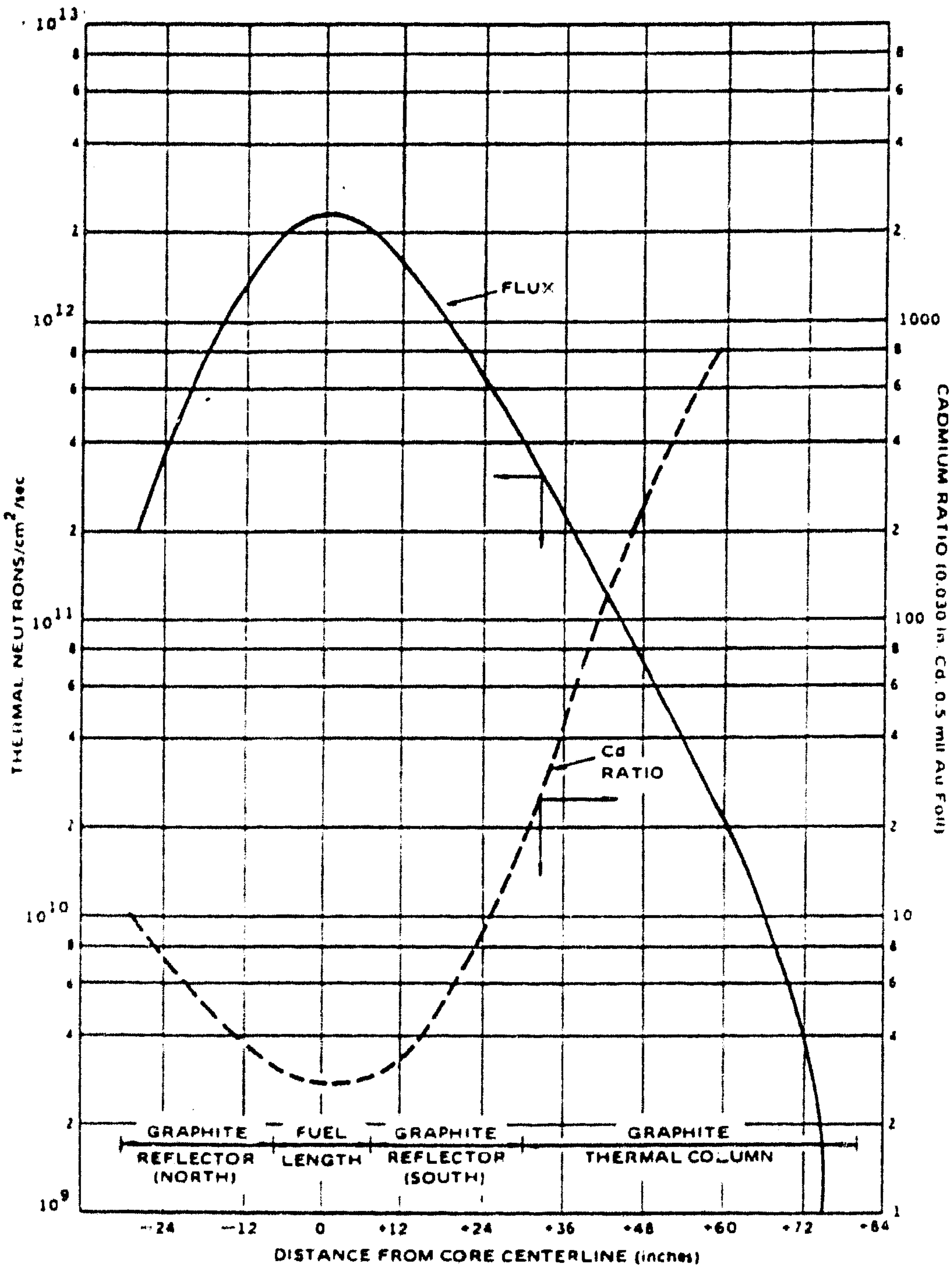


Figure 4-4. Thermal Neutron Flux and Cadmium Ratio Traverse of Horizontal Facility Reactor Power (100 kW)

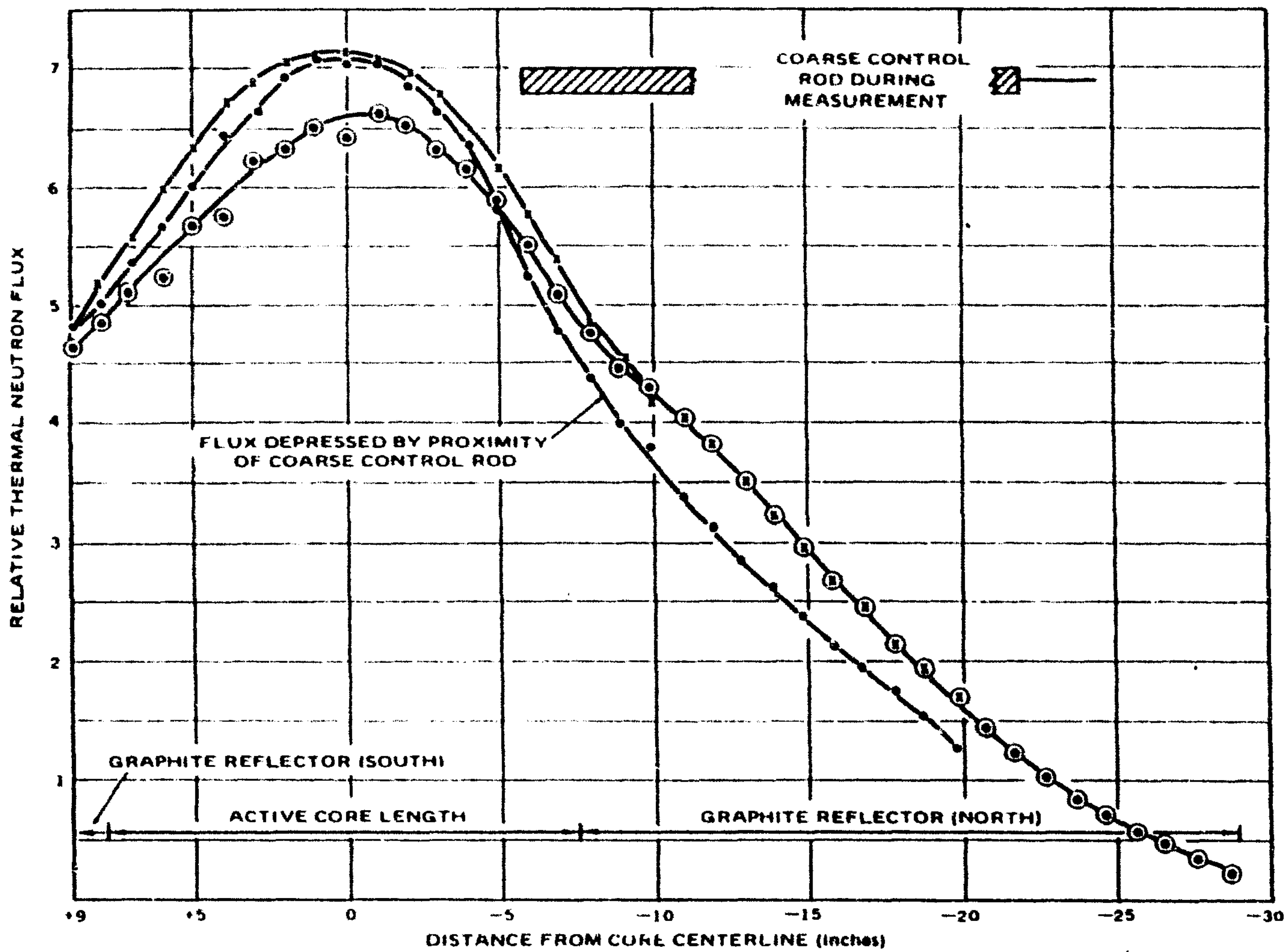


Figure 4-5. Thermal Neutron Flux Traverses in Three Manual Poison Sheet Slots

4.4.3 Operating Limits

The NTR operates with a single fixed core configuration. Reactor fuel is not reconfigured in any way. Outside the core, experiments and manual poison sheets may be altered. However, the potential excess reactivity, at the NTR, is always to limited to 0.76$. Since a 0.76$ step reactivity insertion will not cause fuel damage, even with a failure to scram, operation of the reactor will not pose a threat to the health and safety of the public.

Even if an instrument malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region, the reactor period and neutron flux monitors would scram the reactor. If the reactor did not scram, and reactivity is introduced in either step or relatively long ramp (with the potential excess reactivity being 0.76$ or less), a total reactivity addition of the control rods, experiments, and temperature effect will not result in fuel damage.

The shutdown margin for NTR is 2$. This is calculated with: the strongest safety rod stuck in the full out position, all control rods full out and a xenon free core. The total safety rod worth of 3.86$ minus the maximum potential excess reactivity of 0.76$ is 3.1$. This value minus the strongest rod of 1.1$ gives the shutdown margin of 2$.

The Technical Specifications for NTR state that, the minimum shutdown margin with the maximum worth safety rod stuck out shall be 1$. Operation in accordance with this specification ensures that the reactor can be brought and maintained subcritical without further operator action under any permissible operating condition even with the most reactive safety rod stuck in its most reactive position.

Safety Limit

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.

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 1.5 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.

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 steady- and quasi steady-state. The reactor thermal power level when the DNBR=1.5 is 190 kW.

Another Safety Limit under reactor transient conditions is not required. Conservative transient analyses show that 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, the Safety Limit of 190 kW was conservatively selected to apply to the transient condition.

Limiting Safety System Setting

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

Transient analyses 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 (125% of nominal operating power is the currently preferred value for research reactors) is extremely conservative for the NTR.

Each linear neutron power monitor channel set point is set to trip at 120%. Full power of 100 kW is verified to indicate 100% on the linear channel.. Therefore 120% trip point is within the 125 kW requirement. The trip points are verified on the Daily Surveillance Check sheet prior to each day's operation.

Limiting Condition for Operation

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\$. This 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 Section 13. The analyses show that a reactivity step addition of 0.76\$ will not cause significant fuel degradation.

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

The potential excess reactivity is verified to be ≤0.76\$ power to every startup. The calculation is recorded on the Startup-Shutdown Report. The calculation considers changes in reactivity with regard to: temperature, manual poison sheets, horizontal facility and any other possible reactivity changes.

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

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

Each manual poison sheet used to satisfy the 0.76$ limit shall be restrained in its respective graphite reflector slot in a manner which will prevent movement by more than 1/2 inch relative to the reactor core.

This ensures that the manual poison sheets will not be removed from the reactor core during the maximum postulated seismic event.

Any time a manual poison sheet is changed, it is verified to be properly latched in the new position.

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

This ensures there is no significant positive reactivity feedback from coolant temperature change during reactor power transients.

The over-all temperature coefficient of the fuel annulus was measured in earlier studies and found to be positive up to 124°F. As temperature is increased above 124°F (the turnover point), the coefficient becomes negative. The coefficient was measured between 65 and 156°F, and, for analyzing accidents, it is assumed the data can be extrapolated to boiling. This coefficient is not affected by fuel burnup and is not expected to vary significantly with core life.

4.5 THERMAL-HYDRAULIC DESIGN

Maximum authorized power for the NTR is 100 kW. High-power trips are routinely set at powers no higher than 125 kW and a core outlet high-temperature scram is set to ensure that the core outlet temperature is less than 222°F. For powers above 0.1 kW, forced circulation of deionized water is used to transfer the heat from the core to a heat exchanger, as described in Section 5. When forced circulation is required, the reactor shall scram if flow is less than 15 gpm. At powers less than 0.1 kW, operation is permitted without forced circulation (i.e., the primary recirculation pump need not be operating and the low-flow scram is bypassed). The 0.1-kW limitation for natural circulation operation is extremely conservative (established in the past) but will continue to be used even though more recent analysis for the loss-of-flow accident described in Section 13 shows the core can be adequately cooled by natural circulation at much higher powers. Under both operating conditions, natural or forced circulation, the performance of the core is good with regard to the avoidance of natural thermal limits. These thermal limits include melting of the fuel and cladding, and burnout of the fuel cladding.

The maximum authorized operating power, 100 kW thermal with a rated recirculation flow of 20 gpm, has been used for the reactor to establish values for the thermal and hydraulic characteristics of the reactor core. A summary of these characteristics given in Table 4-2 shows that the thermal loading on the core is quite modest. The core inlet coolant temperature is typically 90°F; the core average exit temperature is 120°F, and, in the hottest channel, the exit temperature is only 150°F. The saturation temperature of the coolant corresponding to the average reactor pressure, 20 psia, is 228°F. Thus, the state of the coolant is far removed from boiling at the design operating condition.

The cladding surface temperatures were established on the basis of known coolant temperatures and the heat flux distribution in the core. The flow through the core is laminar, and the surface film heat-transfer coefficients were calculated from a known laminar correlation. Fuel-plate temperatures increase with power up to a certain point; however, when the surface temperature is elevated to a value that will support local boiling of the coolant, the heat-transfer mechanism undergoes a marked change. There is substantial increase in the heat-transfer coefficient, and, consequently, the plate surface temperature is practically "held" at a maximum value, corresponding to the value needed to establish local boiling. The Jens-Lottes correlation[7] was used to predict the local value of wall superheat necessary to establish local boiling. This phenomenon is important because metal temperatures are limited to values well below melting, which is particularly evident during certain accidental transients discussed in Section 13.

The core flow distribution out of the inlet header (described in Subsection 4.2) is such that adequate cooling of all portions of the core is achieved. The pipe is orificed to give higher-than-average flow rates in the horizontally central region of increased power generation, and lower-than-average flow to the end regions.

The peaking factors used in this evaluation were maximum expected values that result from operation of the reactor with neutron flux peaked on one side of the core. The circumferential power distribution used resulted in a circumferential power peaking factor of 1.25. The longitudinal shape is symmetrical, with a total axial peaking of 1.15. The total over-all power peaking in the core is 1.58, which includes a local peaking factor of 1.1.

Of considerable importance is the ability of the recirculation system to maintain a mode of natural circulation flow when the primary pump is not operating and core power is up. In the absence of pump head, the driving pressure difference around the recirculation loop is the net elevation head of the coolant. This is directly proportional to the density differences between the water in the core and riser section and the water leaving the heat exchanger. Again, this density difference is a function of core power. The length of piping over which this density difference exists is slightly more than 5 feet. System response to loss or recirculation pumping is discussed in Section 13.

Table 4-2

TYPICAL NTR CORE THERMAL AND HYDRAULIC CHARACTERISTICS

Maximum thermal power level (scram)	125 kW
Maximum thermal power level	100 kW
Average fuel disk surface heat flux	6600 Btu/h-ft^2
Maximum fuel disk surface heat flux	10600 Btu/h-ft^2
Total fuel to coolant heat-transfer area	52.7 ft^2
Total core power peaking factor	1.58
Core average pressure level	20 psia
Coolant flow characteristic	
Total core flow area	0.39 ft^2
Channel flow area	0.70 in.^{2}
Channel hydraulic diameter	0.51 in.
Total recirculation flow rate	20 gpm (9800 lb/h)
Inlet velocity, average channel	0.14 ft/sec
Inlet velocity, hottest channel	0.07 ft/sec
Mass flow rate, average channel	122 lb/h
Mass flow rate, hottest channel	64 lb/h
Coolant inlet temperature	90°F
Coolant exit temperature, average channel	120°F
Coolant exit temperature, hottest channel	150°F
Coolant saturation temperature	228°F
Fuel disk cladding temperature	
Average channel	170°F
Hottest channel	195°F
Maximum temperature difference, fuel-to-cladding surface	1°F

Fuel plate steam-blanketing is a condition that may occur even in a pressurized water system and can be of considerable concern. This condition is caused by going from local surface boiling into film boiling upon reaching very high surface heat fluxes. This could be of concern because the steam film degrades the heat-transfer, and the fuel plate temperature increases greatly as a result. However, during steady-state operation, this is of no real concern in the NTR for these reasons: Heat fluxes at maximum power in the reactor are quite small because of the low power rating, and the burnout heat fluxes, or the heat flux necessary to cause steam-blanketing, are very high for the coolant conditions existing in the reactor, as evidenced by experimental data. For instance, in the hottest channel in the core, the data indicates a burnout heat flux of 227,000 Btu/h-ft^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/h-ft^2. Thus, the burnout ratio, or the ratio of burnout heat flux to maximum operating heat flux, is 22. This is a considerable margin and represents a highly safe condition.

5.0 REACTOR COOLANT SYSTEMS

5.1 SUMMARY DESCRIPTION

The reactor primary coolant system is an unpressurized light-water system which provides forced circulation coolant to the reactor core at powers above 0.1 kW; below 0.1 kW, no coolant circulation is required for typical operating periods, although it may be utilized if desired. The primary water system removes the heat from the core and transfers it through a single, two-pass, fixed tube and shell heat exchanger to the secondary water system. Primary water flows through the shell side of the heat exchanger. The secondary cooling water, to the heat exchanger, comes from the Building 105 potable water supply, which is fed from the site raw water main supplied from the site's 500,000-gal storage tank. Upon leaving the heat exchanger, the water goes to the facility drain, which discharges to the site retention basins. The heat exchanger is designed to eliminate strains from temperature differentials.

Figure 5-1 shows the NTR coolant systems. Typical conditions for reactor power of 100 kW are: 35 gpm secondary water, 20 gpm primary water, 90°F core inlet temperature, and 124°F core outlet.

5.2 PRIMARY COOLANT SYSTEM

The primary system is an unpressurized light-water system which provides the coolant to the reactor core. The cooling system contains a volume of about 28.5 gallons, of which 19.5 gallons are contained in the main flow path piping and 9 gallons are contained in the core tank.

The major portion of the primary system is constructed of 1-1/2-inch Schedule 40 aluminum pipe. The internal parts of in-line equipment such as the pump and heat exchanger are stainless steel. The primary coolant flow path is from the primary pump (1 hp, 25 gpm at 30 ft. head) through a check valve and flow control valve, to the bottom of the reactor core tank. Water is distributed by a baffle tube and flows up around the fuel assemblies to the top of the core tank. The water then flows out of the graphite pack, through a flow orifice, heat exchanger (two-pass U-type, 36 inches long, 3.4×10^5 Btu/hr) an air trap and back to the primary pump. Refer to the primary P&ID Figure 5-1 and the primary isometric Figure 5-2.

Primary water may be drained to the 500-gal holdup tank in the northeast corner of the cell where it can be retained or transferred to the other tanks for transfer from the facility. The holdup tank also receives the discharge from the primary system atmospheric vent line, which is connected to the inlet of the heat exchanger. This line is a design feature and is the highest point in the system. It provides a continuous vent to atmosphere for air and other gasses and prevents over-pressurizing the primary system. An overflow line from the fuel storage tank to the holdup tank connects into the primary system atmospheric vent line. A sump pump is located in a sump which is in the northwest corner of the cell. Any water collected in this sump is automatically pumped into the 500-gal holdup tank.

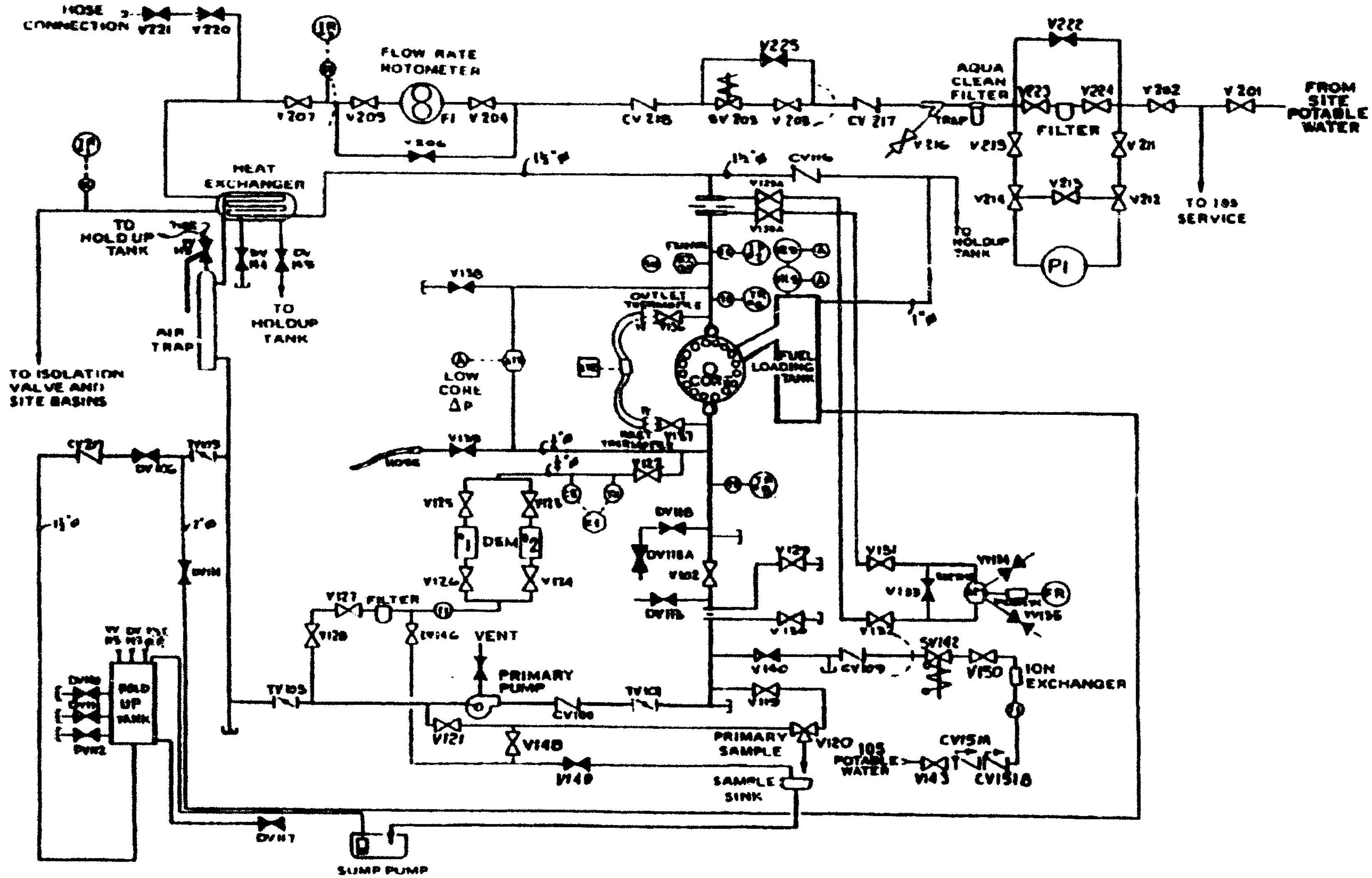


Figure 5-1. Nuclear Test Reactor Coolant Systems

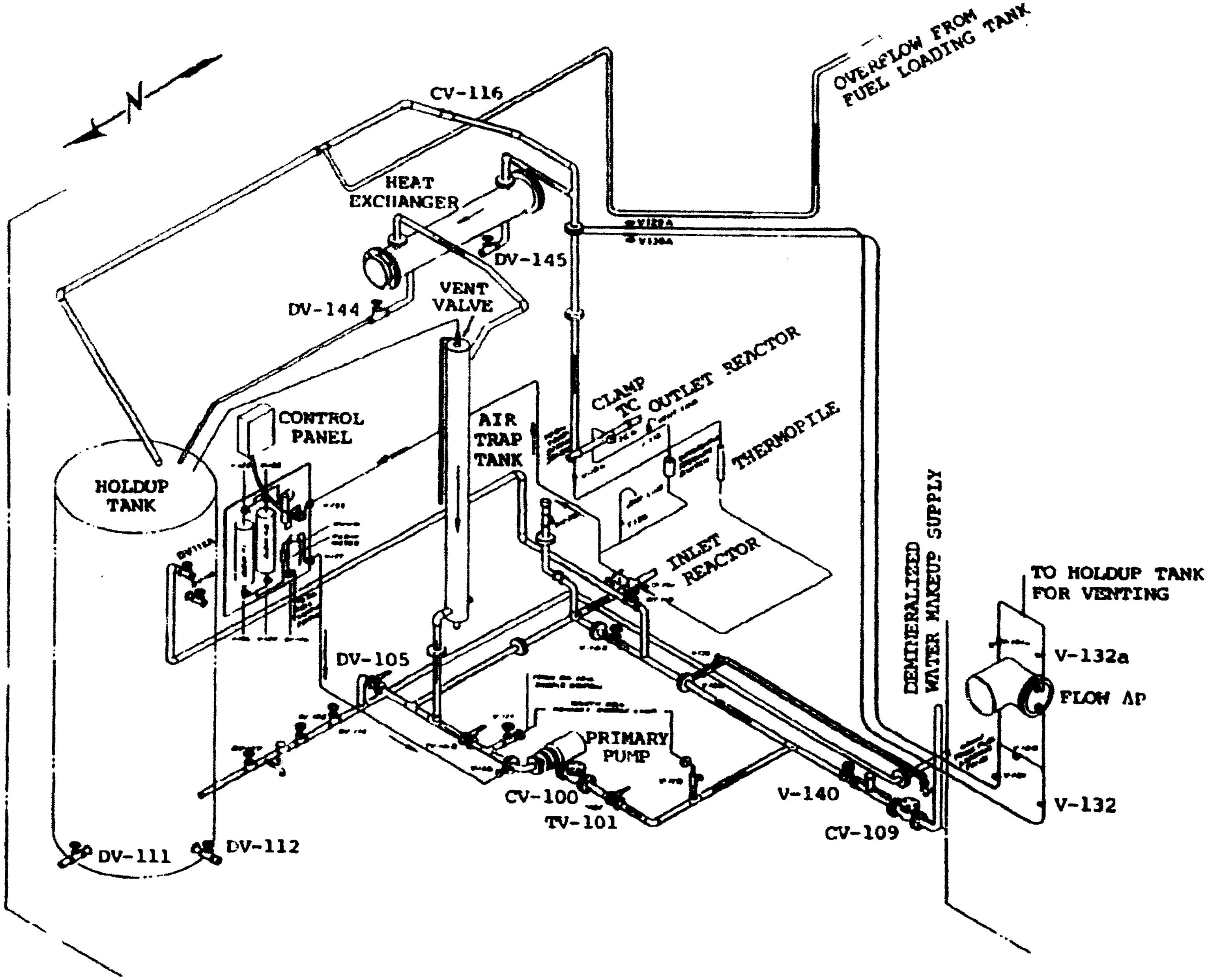


Figure 5-2. Primary Coolant System

The air trap is a 5-foot length of 4-inch aluminum pipe that originally contained Cal-Rod-type heaters rated at 5 kW. The heaters have been removed from the system but the tank remains and is utilized as a system air trap.

The allowable primary system leakage rate is established at 10 gallons/day. Leakage at this rate would cause the water level in the fuel storage tank to drop from the high-level alarm point to the low-level alarm point in 10 days. This leak rate allows ample time for manual makeup of coolant water.

In the event of a primary pump failure, or a significant leak or break in the primary cooling system, the reactor shall be shut down immediately. If the event causes a reactor scram, personnel should verify that all safety systems functioned as intended. If the reactor fails to scram, the console operator shall manually scram the reactor.

An instantaneous seizure of the primary pump rotor with a simultaneous failure to scram at full power would result in a peak power of 101.2 kW and a reactor shutdown to an equilibrium 16 kW about 2-1/2 minutes after seizure. Power reduction is caused by the negative temperature coefficient above 124°F. No damage to the fuel will result, so the consequences of this accident are minimal. For more details, refer to Chapter 13.

Primary water samples can be taken at the sample station located in the northwest comer of the south cell.

Typical cooling system conditions at reactor full power are as follows:

Flow Rate, gpm	20
Core Inlet Temp., °F	90
Core Outlet Temp., °F	124
Conductivity, µmhos	< 1.0
pH	5.5 to 6.5

The objectives of the following limits are to minimize the adverse effects on reactor components and to ensure the proper conditions of the coolant system for reactor operation. Many of these limits are extremely conservative.

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

During a complete loss off primary coolant flow without a reactor scram, fuel damage does not occur. Natural convection cooling is sufficient. Requiring forced coolant flow above 0.1 kW, then, is extremely conservative.

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. Maintaining water in the core tank 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.

A complete loss of coolant in the core tank with a simultaneous failure to scram the reactor at full power would result in a reactor shutdown because of moderator voiding. Peak fuel temperature would reach a maximum of 600°F about 1-1/2 hours after coolant loss. No damage to the fuel will result, so the consequences of this accident are minimal.

The maximum Primary Coolant High core outlet temperature scram set point is 222°F. This provides assurance that the reactor fuel temperature will not attain a temperature which will cause damage to the fuel.

The Primary Coolant Low Flow scram setpoint is no less than 15 gpm when the reactor power is >0 1 kW. This, ensures (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 primary coolant high core outlet temperature is <200°F.

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

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.

This ensures that the temperature coefficient will remain negative above 124°F. The coefficient 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).

The temperature coefficient at temperature T (Fahrenheit) is -5.7×10^{-3} (T-124) ¢/°F.

5.3 SECONDARY COOLANT SYSTEM

Secondary cooling water (Figure 5-3), for the NTR, comes from the building 105 potable water supply, which is fed from the site raw water main supplied from the site's 500,000-gal storage tank. The 1.5-inch supply line to the NTR facility supplies the one heat exchanger. It passes through a filter, in the Building 105 equipment space, across the roof of the building, and then enters through the ceiling of the control room. In the control room it passes through a shut-off valve, a check valve, and a flow indicator and then enters the reactor cell. Inside the cell, the line goes directly to the tube side of the heat exchanger and then through a manual valve to the Site retention basins.

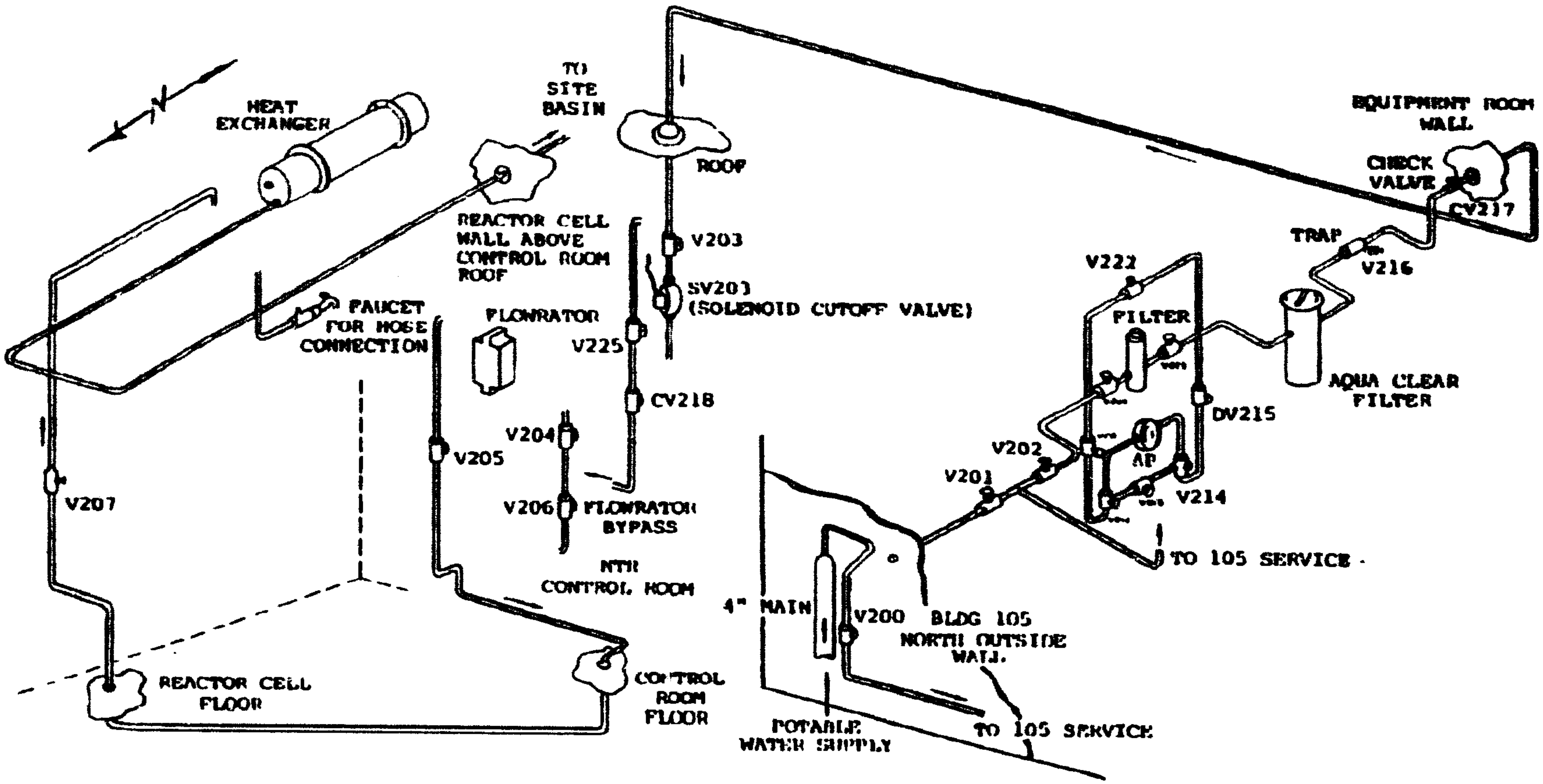


Figure 5-3. Secondary Cooling System

Adequate heat-transfer capability is provided by the single two-pass fixed tube and shell heat exchanger. The heat exchanger is capable of transferring 3.4 x 10^5 Btu/h. The tube bundle consists of 0.25-inch-diameter, 36-inch long stainless steel tubes, and provides a heat-transfer surface of approximately 36 ft^2. The heat exchanger is located on the east wall of the reactor cell about 6 feet above the reactor core. Pressure at the inlet of the heat exchanger is normally about 70 psig. The design specifications pertinent to maintaining system integrity are given in Table 5-1.

Primary water flows through the shell side of the heat exchanger. The probability and consequences of a leak between the two systems in the common heat exchanger have been evaluated. The primary side of the heat exchanger is <40 psig, and the secondary is, as mentioned, (~70 psig), therefore a heat exchanger leak would result in a secondary-to-primary leak. The evaluation showed that the probability of leaking contaminated water from the primary to secondary system is extremely low; furthermore, should such a leakage occur, the contaminated water would drain to the Site retention basins. The basin water is analyzed for radioactive material content before it is released.

Secondary coolant flows by gravity through the tube side of the primary heat exchanger. Loss of secondary coolant flow would result in a primary coolant temperature increase to the alarm set point. A reactor scram could occur if rapid remedial action is not taken. Loss of secondary coolant flow must be corrected before the reactor can be operated at a reactor power level which would generate an appreciable amount of heat. Safety Analyses are discussed in Chapter 13. A primary coolant system change from forced convection flow to natural convection, with or without the secondary system, will not result in fuel damage.

Table 5-1

HEAT EXCHANGER SPECIFICATIONS

	Shell Side	**Tube Side**
Fluid (coolant)	Primary	Secondary
Fluid Flow rate (gpm)	20	35
Fluid velocity (ft/sec)	1.48	3.43
Temperature in (°F)	124	60
Temperature out (°F)	90	80
Pressure Drop (psig)	1.6	4.0
Design Pressure (psig)	300	150
Test pressure (psig)	500	300
Design temperature (°F)	400	400
Material (stainless steel)	316	316

If there is a break in any of the coolant piping, in the reactor cell, the coolant will flow to the reactor cell sump pump where it is automatically pumped to the waste water hold up tank. High water level in the reactor sump will activate an alarm in the control room and also in the security building. This water can be held for evaporation inside the cell or transferred to the waste evaporator for processing. It is never used to add to other systems.

The NTR control room temperature recorder records the heat exchanger inlet and outlet temperatures of the secondary coolant water, along with other reactor temperatures. Any deviations, from normal, of secondary temperatures, as well as the other temperatures, would be noted by the operator.

A "flowrater", that indicates secondary coolant flow, is located on the wall in the reactor control room. Any deviations, from normal flow, would be noted by the operator. The heat exchanger is regularly inspected per the NTR Preventive Maintenance Procedures.

Although both the secondary coolant and the primary system makeup water come from the same potable water supply, they are completely separate systems. Each has their own check valve, manual valves and solenoid valve (operated by keylock at the reactor console) so that one coolant system cannot flow to the other system. Addition to the primary system can only be accomplished from inside of the reactor cell; otherwise the makeup valve remains closed at all other times.

As mentioned, the secondary coolant system is a single pass system as opposed to a closed loop system. Because the coolant is used only once and is initially drinking water quality, there are no radiation monitors or detectors incorporated into the secondary system. However, as mentioned above, if the secondary should become contaminated, the contaminated water would drain to the Site retention basins and the basin water is analyzed for radioactive material content before it is released.

The secondary system is used only to remove heat from the primary system. There are no emergency core cooling, experiment cooling, biological or thermal shield cooling systems at the NTR.

There are no limitations required for the secondary coolant system.

5.4 PRIMARY COOLANT CLEANUP SYSTEM

The purity of the primary coolant system is maintained by two Barnstead Model BD-2 Pressure Bantam Demineralizers installed in parallel (see Figures 5-1 and 5-2). The cleanup system normally operates with both demineralizers on line. If both units are used, the system will service 32 gph. The demineralizers contain replaceable cartridges. A conductivity monitor is located upstream of the demineralizers. To further ensure the purity of the system, a 5-micron cartridge-type filter is installed in the discharge line of the demineralizer system. The entire primary coolant cleanup system is located inside the reactor cell, on the east wall. The system

input is from the primary system, just before the reactor inlet, and the output is to the primary system upstream of the primary pump.

The primary system pH is controlled by controlling the water conductivity. The conductivity is maintained at or below 2 μmho/cm. The pH then, will be between 5.6 and 9.0, which is compatible with aluminum/stainless steel systems. The conductivity of the primary coolant is checked prior to the first startup of the day in accordance with NTR Standard Operating Procedures and is checked annually by Analytical Chemistry in accordance with NTR Preventive Maintenance Procedures. The pH is also measured annually by Analytical Chemistry in accordance with NTR Preventive Maintenance Procedures. A contact radiation reading on the demineralizers is checked prior to the first startup of the week in accordance with NTR Standard Operating Procedures.

Normal radiation readings on the demineralizers are up to 2 R/hr. Resins should be scheduled for replacement when their radiation level reaches a consistent 3 R/hr level. High radiation and contamination levels may be expected during the performance of resin replacement work. Prior to replacement, working radiation dose rates and personal protection are established. The resin cartridges are changed and stored shielded in the reactor sump until they are properly dried at which time they are transferred for disposal. The filter is also changed as required.

As mentioned, resins are replaced when their radiation level reaches a consistent 3 R/hr level and a contact radiation reading on the demineralizers is checked prior to the first startup of the week. Also, the entire primary coolant cleanup system is located inside the reactor cell, on the east wall, where personnel infrequently go. Radiation shielding is mounted in front of the high radiation areas. Routine conductivity readings are taken from a low dose rate area on the other side of the reactor cell.

An inadvertent release of excess radioactivity in the primary coolant system, of high enough level, would cause the reactor cell remote area monitor to alarm. The area monitor detector is located on the reactor cell near the primary flow transmitter and is set to alarm at 10^6 mR/hr. Primary samples may be taken from the primary sample station. The Senior Reactor Operator will determine the cause and initiate corrective action.

The entire primary cleanup system is enclosed in the reactor cell. The piping for the system is 1/4-inch stainless steel tubing. The flow is preset by design to ~16 gph. If there is a break in any of the pip ıg the loss of water would be noted by a low level alarm on the fuel tank. The coolant will flow to the reactor cell sump pump where it is automatically pumped to the waste water hold up tank. This water can be held for evaporation inside the cell or transferred to the waste evaporator for processing. It is never used to add to other systems.

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.

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 these limitations ensures aluminum corrosion is within acceptable levels and that activation of impurities in the primary water remain below hazardous levels.

Specifications regarding surveillance requirements of the reactor coolant system are included in the reactor safety system, Section 4.2, (Tables 4-1 and 4-2). Table 4-2 lists the surveillance requirements of the reactor safety system information instruments. The list requires the primary coolant conductivity channel check to be quarterly and the primary coolant conductivity channel calibration to be annually.

This ensures that all required channels are operational.

5.5 PRIMARY COOLANT MAKEUP WATER SYSTEM

Primary coolant makeup water (Figure 5-3), for the NTR, comes from the Building 105 potable supply, which is fed from the site raw water main supplied from the site's 500,000 gal. storage tank. A potable water line, coming off of the supply, feeds the Building 105 deionizer unit located in the Building 105 equipment space. The deionizer provides all of the Building 105 deionized water needs as well as NTR's primary coolant makeup water. The makeup line enters the reactor cell from a line located above the control room ceiling (see Figure 5-1), through a penetration in the south wall of the reactor cell, and connects into the primary system, between the primary pump and the inlet side of the reactor, through a solenoid valve energized by the reactor console key lock switch and a manual valve in the reactor cell. Makeup to the primary system can only be done from inside of the reactor cell.

Through the reactor fuel loading chute, the makeup system also supplies the 1800-gal. fuel loading tank, which serves as a reservoir for the primary system. The fuel loading tank is discussed in Section 5.7 and Chapter 6. High and low water level, in the fuel loading tank, is indicated by level switches. The switches actuate annunciators and alarms in the control room. They also actuate alarms in the security building that is occupied around the clock. High and low tank level alarms are always investigated when they are received.

The primary system and the fuel loading tank are located in the reactor cell. Regardless of whether the fuel loading tank overflows or there is a break in the primary line, the coolant will flow to the reactor cell sump pump where it is automatically pumped to the waste water holdup tank. This protects against leakage of contaminated coolant to the potable water supply. This water can be held for evaporation inside of the cell or transferred to the waste evaporator for processing.

5.6 NITROGEN-16 CONTROL SYSTEM

There is no nitrogen-16 control system at the NTR.

5.7 AUXILIARY SYSTEMS USING PRIMARY COOLANT

The fuel storage tank is connected to the reactor core tank by a 3-inch by 20-inch by 30-inch-long chute inclined on a 30° angle. When not being used, the loading chute is filled with an aluminum clad graphite plug and the aluminum access gate in the tank is closed. The fuel storage tank is located on the west side of the reactor graphite pack and provides biological shielding for fuel which is removed from the core. The tank is 4 x 5 x 12 feet high and is constructed from 1/4-inch aluminum. There are two 4-inch diameter tubes and one 2-inch-diameter aluminum tube mounted on the east side of the tank. These tubes contain neutron detection chambers for the reactor nuclear instruments. Access to the tank is provided from the mezzanine on the west side of the tank.

The tank water level is monitored by high- and low-level float-actuated switches. An overflow drains water to the hold tank.

The fuel loading tank water low level set point is <3-ft below the overflow. This alarm gives assurance that there is adequate water in the primary system for operation of the reactor.

6.0 DESIGN BASES AND ENGINEERED SAFETY FEATURES

6.1 CONFINEMENT

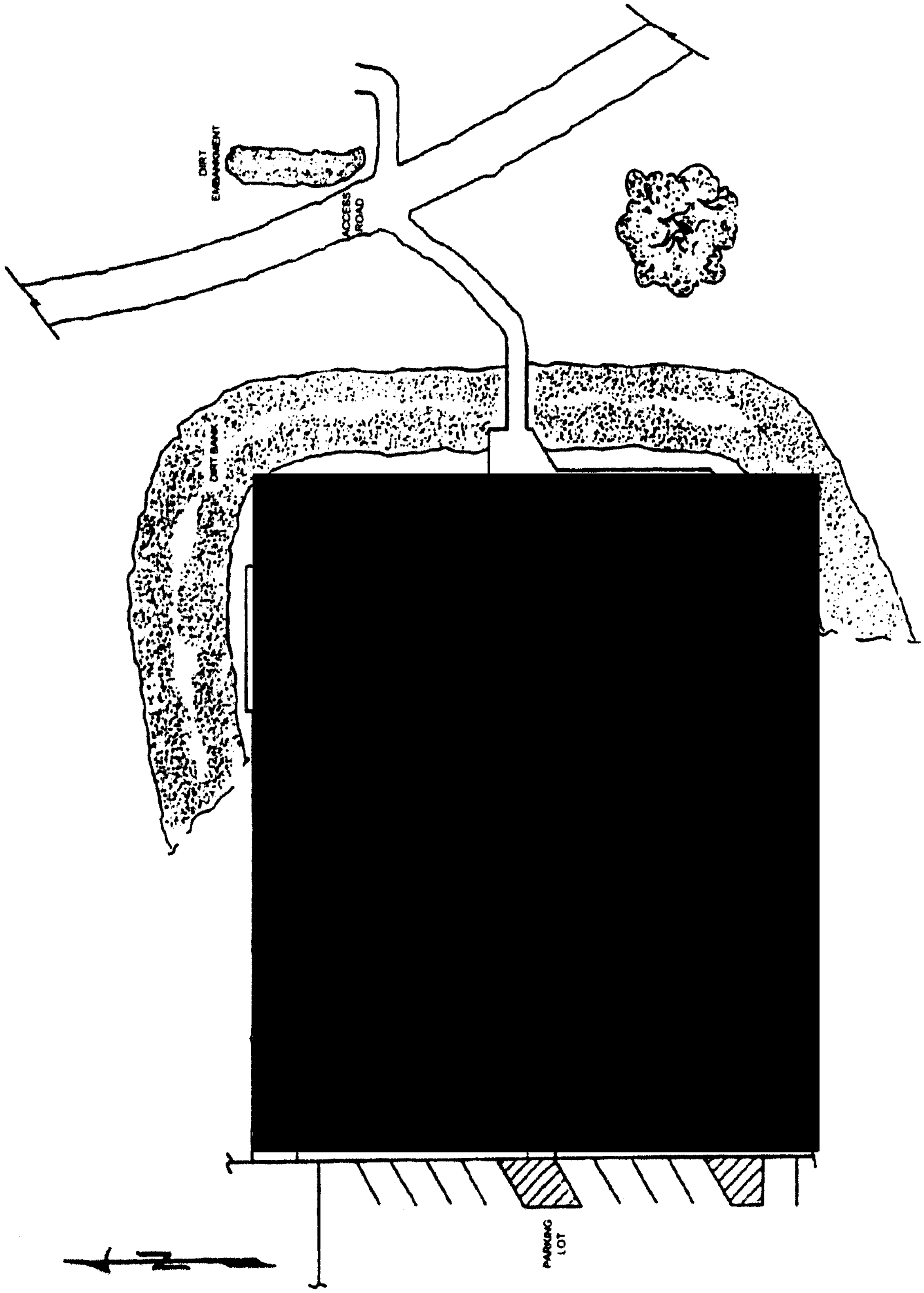
The reactor is installed in a concrete-shielded room (the reactor cell) located in the northeast corner of Building 105. Although the cell is not designed to provide gas tight containment, controlled release of airborne reactivity is possible through the operation of the cell ventilation system. Figure 6-1 shows the floor plan of Building 105, and Figure 6-2 shows plan and elevation views of the area that contains the NTR facility, including the NTR and reactor cell, and the north room with its shielded Neutron Radiography Facility. These illustrations are not necessarily current with respect to the arrangement of the office and laboratory areas. Other details that are not pertinent to safety considerations for the NTR facility may not be as shown.

The reactor cell is a rectangular-shaped room with approximate internal dimensions of 22 feet wide by 23 feet long by 24 feet high. A heavy-concrete-shielded alcove surrounds the reactor in the southeast corner and a mezzanine is located above the cell door in the southwest corner. Equipment of appreciable size located within the cell includes the reactor, reactor cooling system, fuel loading tank, holdup tank, 5-ton bridge crane, and storage shelves. Approximate gross volume of the cell is 11,300 ft^3 and, with the above-mentioned equipment installed, the net air volume is approximately 10,500 ft^3.

Normal access to the cell is through the large doorway in the south wall. During reactor operation, the doorway is normally closed by a 1-foot-thick, motor-driven, sliding concrete door lined with 1.25 inches of steel. Power for moving the cell door in either direction is interlocked with a key-operated switch so that it is possible for the reactor operator to control all cell entries. A manually operated, 1-foot-thick paraffin door covered with aluminum and located just inside the reactor cell is normally closed to reduce further the radiation dose rates outside the reactor cell door.

A large removable equipment hatch is provided in the cell roof. Use of this hatch is limited to very special occasions when use of the cell door is impossible or impractical.

The refueling of the reactor and the maintenance of equipment in the cell are performed only with the reactor shut down (i.e., sufficient manual poison sheets and safety rods inserted to satisfy minimum shut-down margin requirements). Normally, the radiation and contamination levels are quite low; therefore, these activities can be performed with the cell door open. If expected radiation levels, results of radiation monitoring, or some other nonroutine nature of these activities makes closing the door desirable, either maintenance or refueling may be done with the door closed.



[redacted]

Figure 6-1. Building 105 Floor Plan

[redacted]

Figure 6-2. Part Floor Plan and Elevation

6.2 PENETRATIONS

For discussion, the penetrations into the reactor cell have been divided into two types--reactor and experiment services. Reactor service penetrations normally are used for passing water, electric power, and air into and out of the cell; their use generally does not change from day to day. Experiment service penetrations normally are used to move materials and equipment for experiment programs into the reactor cell. The list of penetrations below is the present arrangement. Future changes in the number, location, or type designation of any of the penetrations will be made in a manner to retain the effectiveness of the cell to control radiation, contamination and security.

In addition to the cell door and roof hatch, the existing reactor service penetrations include:

- One 1.5-inch secondary cooling water supply line;
- One 1.5-inch deionized water supply line;
- One 2-inch drain line to the Site retention basins. This drain has been disconnected and plugged. It has not been used in the last 20 years.
- Twelve 3-inch electrical conduits for wiring between the cell and control room;
- Eight small-diameter (≤1.5 inches) electrical power and lighting conduits, including spares;
- One 0.75-inch compressed air supply line;
- Two 8 by 8-inch ventilation exhaust ducts through the cell roof;
- Four 1.5-inch pipes to the adjoining laboratory; and
- Two 4-inch holes approximately 10 feet above the cell floor, one of which penetrates into the control room and the other of which penetrates over the control room roof.

Existing experiment service penetrations are listed below. Use of these penetrations is controlled carefully to ensure that the effectiveness of the cell to contain radiation and radioactive materials is not significantly reduced.

1. South thermal column into south cell.
2. Horizontal facility tube into south cell.

3. A 24-inch-diameter hole through the north wall at approximately core centerline height (blocked on the south end by an aluminum plate and a motor-driven shutter. Blocked on the north end by the Modular Stone Monument).

4. Stepped hole (6-8 inches in diameter) through the east wall approximately 10 feet above the cell floor.

5. Hole for future thermal column through the east wall (presently filled with unmortared concrete bricks).

6. Two holes (3 and 4 inches, respectively) in the north wall, penetrating into the north room. The 3-inch hole contains radiation area monitor cables and the 4-inch hole is used for the CHRIS experiment facility.

6.3 VENTILATION

The NTR exhaust system includes a 3000-ft^3/min fan located on the reactor cell roof. The fan draws air from the reactor cell, south cell and the north room modular stone monument. The air goes through a prefilter and a bank of absolute filters and is then discharged through a stack of sufficient height to disperse the exhaust upward.

An air-monitoring system provides continuous indication of the concentration of radioactive material in the ventilation effluent and energizes an alarm at the reactor console if the concentration reaches a set point which has been selected to ensure that the airborne release does not exceed established limits (Table 6-1). Separate detection channels and alarms are used for particulate material and for nonfilterable radioactive gases. A continuous sample is drawn from the discharge of the NTR ventilation stack and passes through the particulate detector, a charcoal cartridge, the nonfilterable radioactive gas detector, flow control valve, and a central blower (Hoffman). It is then released through the Building 105 NTR Furnace Exhaust. Particulate materials are collected on a high-efficiency filter paper and their emissions measured with a

Table 6-1

STACK RELEASE ACTION LEVELS

Stack	Nominal Flow Rate, cfm	Noble Gas Ci/wk µCi/cc	Halogen mCi/wk µCi/cc	Alpha µCi/wk µCi/cc	Beta µCi/wk µCi/cc
105, NTR	1.80E+03	18 *1.9E-04	1.74E+02 3.4E-07	8.69E+00 1.7E-11	8.69E-02 1.7E-09

*The NTR noble gas concentration limit during non-operating time, i.e., when the reactor is shut down and the cell can be open, is set at 2E-6 µCi/cc

shielded Geiger-Müller detector. Nonfilterable radioactive gases are detected by an internal gas flow ionization chamber with a relatively high sensitivity for beta emitters. Current from the chamber is measured by a picoammeter. Each channel is recorded on a multipoint recorder. The charcoal cartridge and particulate filter are changed periodically (normally weekly) and counted by the VNC Counting Lab for I-131 and gross β-γ and α, respectively. The stack sampling/monitoring system is discussed in detail in Appendix A. Figure 6-3 presents a line diagram of system.

6.4 BASES FOR THE STACK ACTION LEVELS

The stack release action levels are defined as the release rates for each radionuclide group (noble gas, I-131, beta particulate, or alpha particulate) at which action should be taken to reduce the release rate. The design basis for setting the action levels is the requirement to maintain doses to members of the public from airborne releases to a maximum of 10 mRem per year. The method for establishing these action limits is described below.

10CFR20, Appendix B, Table 2, Column 1 gives airborne radioactive material concentration limits for releases to the general environment. Inhalation of one of these radioisotopes at that concentration continuously over the course of a year would produce a total effective dose equivalent of 50 millirem. Therefore, the release rates from the effluent stacks at VNC must be controlled to a level which will not exceed 20% of the 10CFR20 effluent concentrations at the site boundaries. Annual average release rates are converted to boundary concentrations by a dilution-dispersion factor. Dilution-dispersion factors are calculated from the measured meteorological conditions for a year's period (or more). Consideration also is given to concurrent releases from the other stacks on site.

The action level for the noble gas releases from the NTR stack is selected as the rate which would give an annual average concentration of Ar-41 at the site boundary of 20% of the concentration limit, further divided by a factor of two for other stack releases. Ar-41 has been shown to be the predominant noble gas in the stack effluent (Climent, 1969). Fission-produced noble gases are a minor fraction unless fuel material is exposed to the effluent air. Ar-41 is produced by the neutron irradiation of the air passing through the reactor.

The action levels for all other isotope groups are selected as 10% of the concentration limit for the restrictive, credible isotopes of each of the isotope groups: I-131, Sr-90, and Pu-239. These, too, are reduced further by a factor of two for other stack releases. The release limits are specified as release rates (μCi/sec); this makes the limit independent of the stack flow rate. A limit expressed as a concentration (μCi/ml) is dependent on the stack flow rate. However, radioactive concentrations are easily determined and therefore commonly used in reporting effluent releases.

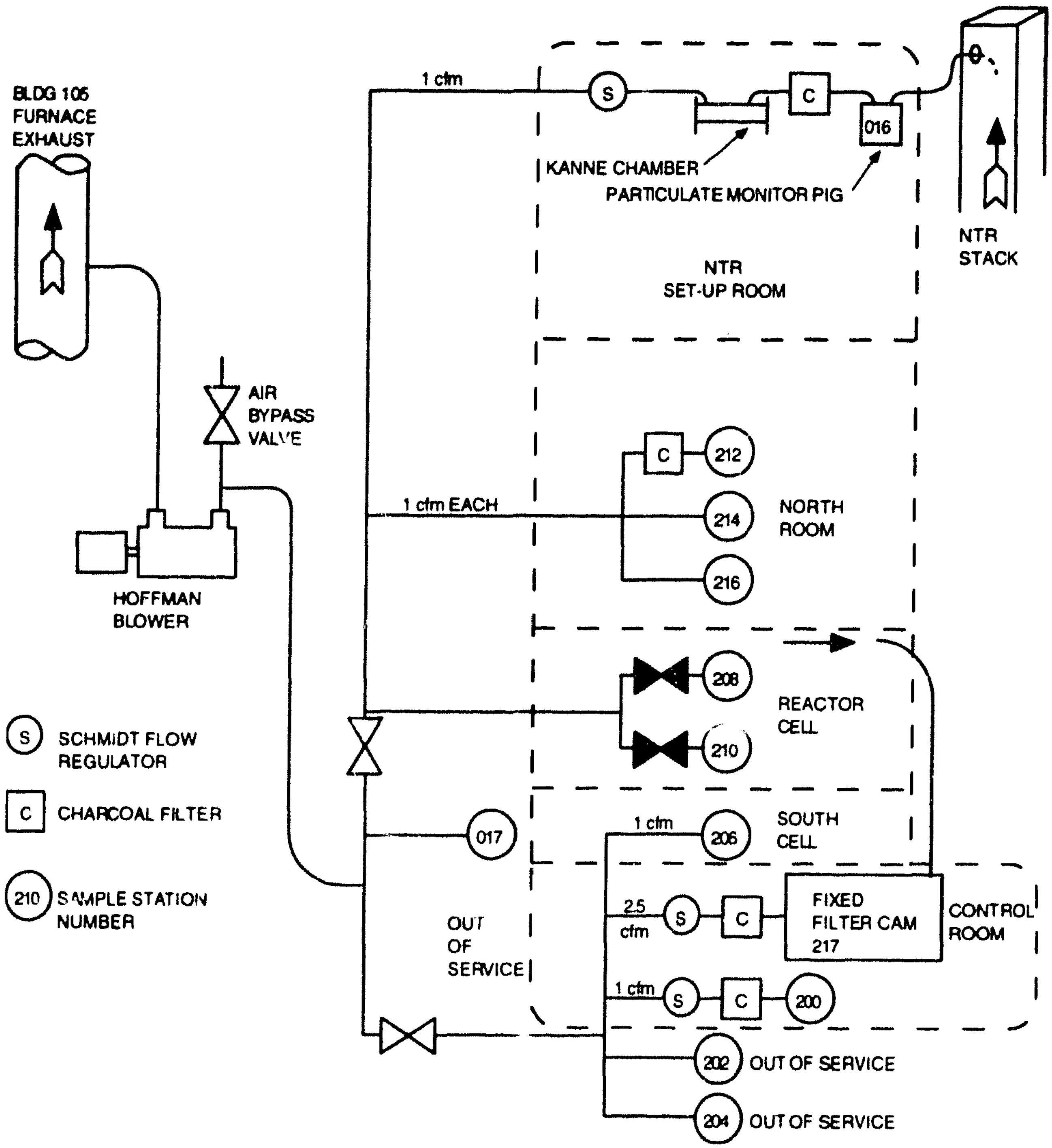


Figure 6-3. Line Diagram of System

The stack flow rates fluctuate, sometimes by design and sometimes randomly. For example, the NTR flow depends on the position of the cell door. The flow in all filtered systems varies as the dust loading on filters increases and as containment systems are changed. The following stack flow rate is the anticipated highest nominal flow rates or weighted average flow rate (where applicable) used for limiting concentrations and calculating releases:

Stack Location	Flow Rate, cfm
Building 105, NTR	1,800

The applicable effluent concentration limit values from Appendix B, Table 2, Column 1 of 10CFR20 are given below:

Release Category	Limiting Isotope	10CFR20 Effluent Concentration Limit, µCi/ml
Noble Gas:*		
NTR	Ar-41	1.00E-08
Halogen	I-131	2.00E-10
Alpha Particulate	Np-237	1.00E-14**
Beta-Gamma Particulate	***	1.00E-12

The dilution-dispersion (χ/Q) factors and reduction factor to account for releases from "other stacks" are given below:

Stack Location	χ/Q, sec/ml	"Other stack", reduction factor
Building 102A	8.25E-11	2
Building 103	8.25E-11	2
Building 105, NTR	3.48E-11	2
GETR	8.25E-11	2
Waste Evaporator	8.25E-11	2
Hillside Storage, Sandblast	8.25E-11	2
Hillside Storage, Bunker	8.25E-11	2

*The NTR noble gas inventory available to the boundary has been found to be primarily Ar-41, which is an activation product of air. Fission products would be of concern in the event of fuel failure, an abnormal condition.

**There are several isotopes with more restrictive limits, but they can be shown to be insignificant fractions of the typical mix of alpha emitters found at VNC.

***Unidentified isotopes, where several natural, transuranic, and other rare elements are known to be absent. These are mainly alpha emitters which would be accounted for in the alpha analysis.

The dilution-dispersion factors were calculated for two stacks, the NTR and Building 102A. The NTR factor was calculated from measured meteorological histories for a two-year period in 1976 and 1977. The Building 102A factor was later calculated for measured metorological histories over a five-year period. For convenience, the other stacks on site were assigned the same χ/Q value as the 102A stack. This is a conservative action since the 102A stack is nearer the site boundary than any other stack.

Using the above information, the 50 mRem/yr annual average release rate limits for the site stacks can be calculated as the concentration limit divided by the χ/Q and divided by the "other stack" reduction factor.

Annual Average Release Limit, µCi/sec (50 mRem/yr)

Stack	Noble Gas	Halogen	Alpha	Beta
105, NTR	1.44E+02	2.87E+00	1.44E-04	1.44E-02

The NTR noble gas concentration limit is based on a typical operating week of 30 hours at 1,800 cfm. The other release limits are divided by 10 and converted to the weekly release rate action levels and, for convenience, concentration limits for Table 6-1.

7.0 INSTRUMENTATION AND CONTROL

7.1 SUMMARY DESCRIPTION

The reactor is equipped with sufficient instrumentation to control operation of the facility, measure operating parameters, warn of abnormal conditions, and scram the reactor automatically if an abnormal condition occurs (Figure 7-1 and Table 7-1). All reactor scram functions cause a loss of energizing currents to electromagnets, which, when deenergized, permit rapid insertion of the spring-loaded safety rods. The energizing currents can be disrupted by contacts in the power switches, scram relays, or by a manual scram switch. The power switches are controlled by logic units which monitor the trip circuits, on a two-out-of-three coincidence basis, for high reactor power from 3 picoammeters and for loss of high voltage for the three Compensated Ion Chambers (CIC) in the picoammeter channels. Another logic unit monitors singly (noncoincidence) the fast reactor period trip, and high log N power trip. All other scrams, except the console manual scram, operate through the scram relays and are initiated by the following signals:

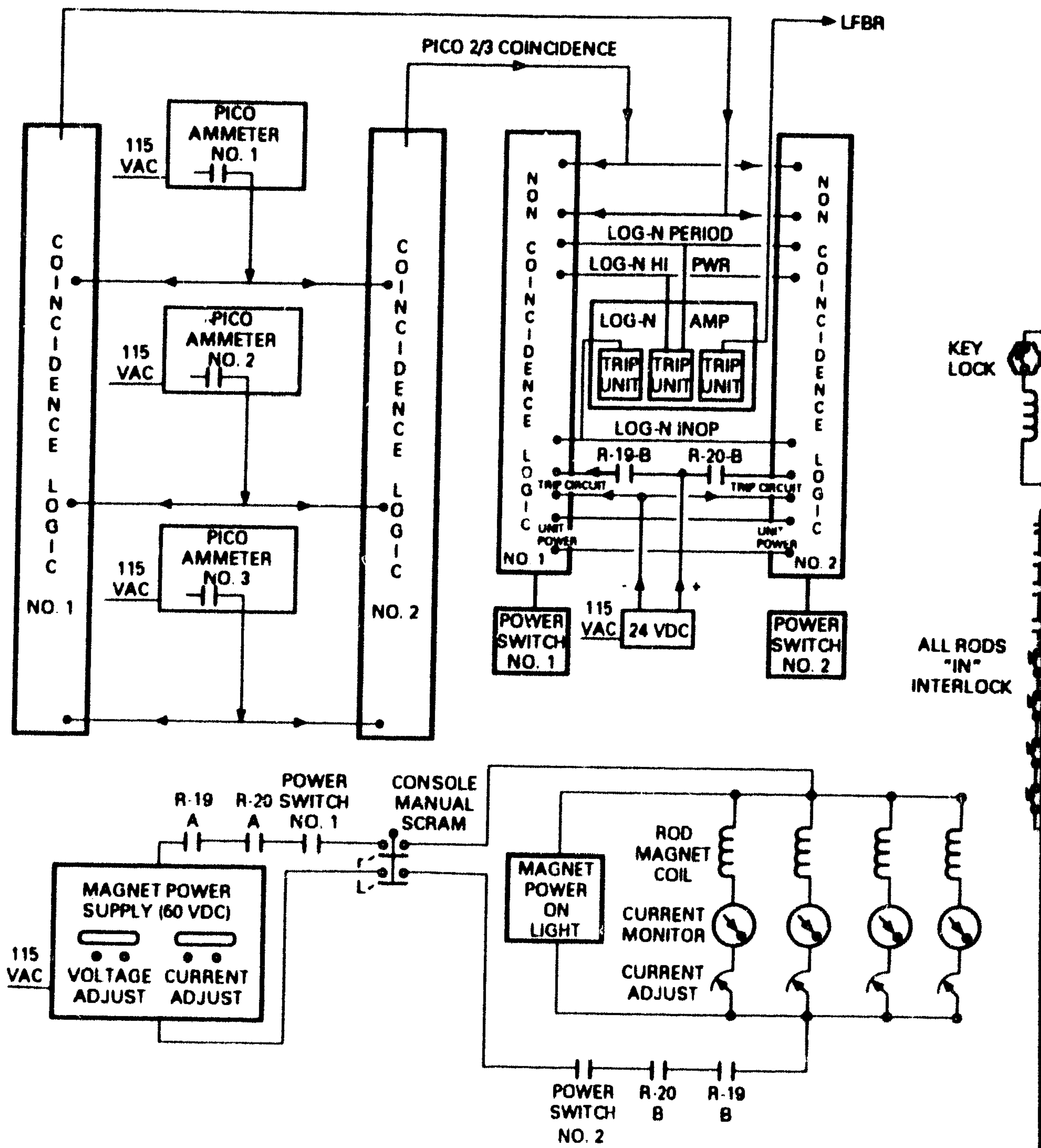
- Log N amplifier mode switch position
- Log N CIC loss of "positive" high voltage
- Primary coolant high core outlet temperature
- Primary coolant flow low
- Loss of ac power
- Reactor cell manual scram.

Safety is also provided by having each scram (except loss of ac power) cause the control rod drives to run to their fully inserted positions. Also, a rod withdrawal permissive interlock is provided that blocks control rod and safety rod withdrawal if a picoammeter (in a two out of three coincidence logic) is not indicating above a preset minimum level. The rod withdrawal permissive circuit ensures that instrumentation is seeing the neutron source for reactor startups. Additional interlocks associated with the rod drive system include the following:

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

2. Safety rods must be drawn one at a time to their outer limits before more than one control rod can be withdrawn.

3. The rod test panel consists of a key-lock arm switch and a seven-position selector switch. The seven positions on the selector switch are OFF, Safety Rod #2, Safety Rod #3, Safety Rod #4, Coarse Rod #1, Coarse Rod #2, and Fine Rod. This panel bypasses the sequential withdrawal interlocks and permits the selected rod to be withdrawn out of sequence. All other rods, however, must be fully inserted.



LFBR
PICO 2/3 COINCIDENCE
PICO AMMETER NO. 1
115 VAC
PICO AMMETER NO. 2
115 VAC
PICO AMMETER NO. 3
115 VAC
COINCIDENCE LOGIC NO. 1
COINCIDENCE LOGIC NO. 2
NON COINCIDENCE LOGIC NO. 1
NON COINCIDENCE LOGIC NO. 2
LOG-N PERIOD
LOG-N HI
PWR
LOG-N
AMP
TRIP UNIT
TRIP UNIT
TRIP UNIT
LOG-N INOP
R-19-B
R-20-B
TRIP CIRCUIT
TRIP CIRCUIT
UNIT POWER
UNIT POWER
115 VAC
24 VDC
POWER SWITCH NO. 1
POWER SWITCH NO. 2
KEY LOCK
ALL RODS "IN" INTERLOCK
R-19 A
R-20 A
POWER SWITCH NO. 1
CONSOLE MANUAL SCRAM
MAGNET POWER SUPPLY (60 VDC)
VOLTAGE ADJUST
CURRENT ADJUST
115 VAC
MAGNET POWER ON LIGHT
ROD MAGNET COIL
CURRENT MONITOR
CURRENT ADJUST
POWER SWITCH NO. 2
R-20 B
R-19 B

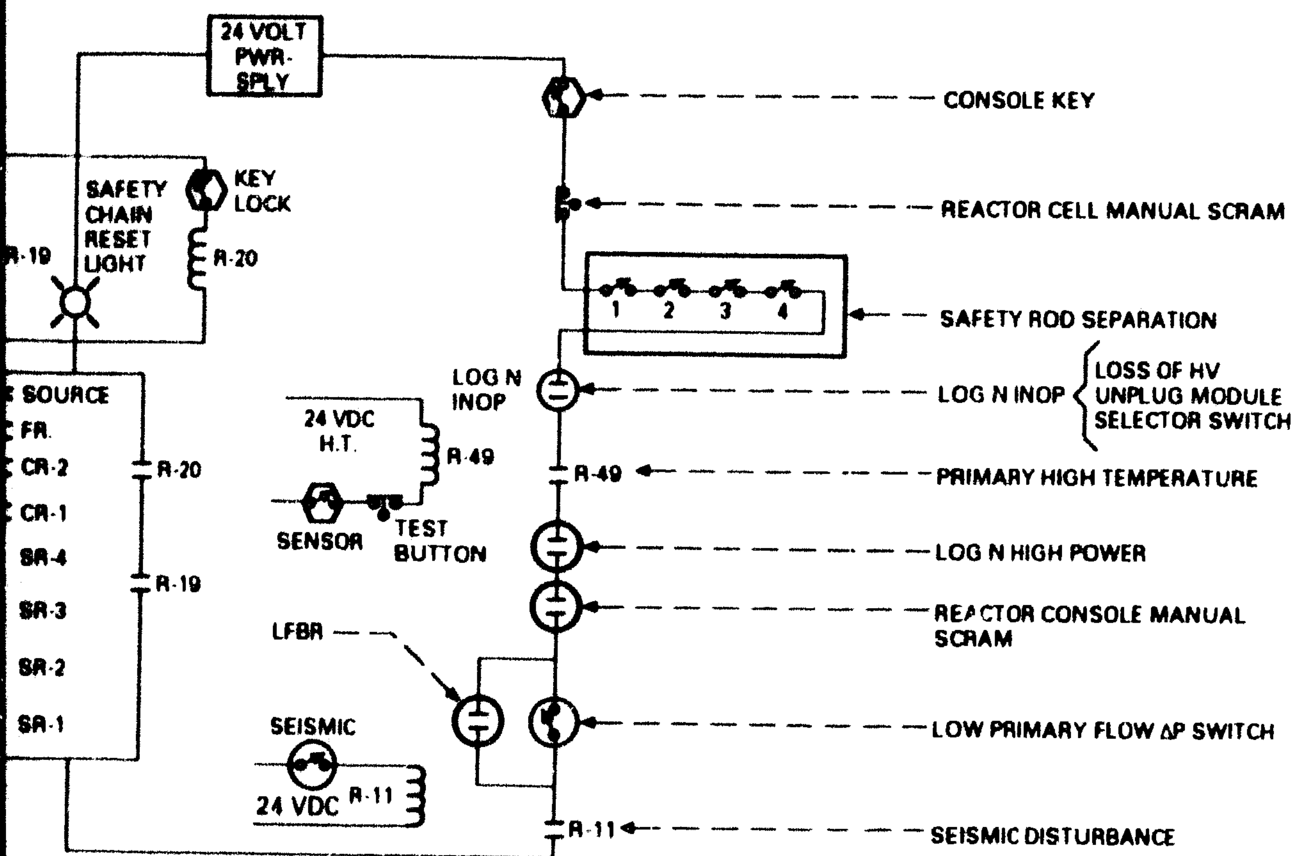


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Figure 7-1. NTR Scram Systems

Table 7-1

SCRAM SYSTEMS

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	Short reactor period	No less than +5 seconds	Scram
		Amplifier mode switch not in operate position	NA	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 222°F	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	NA	Scram
		Reactor cell button depressed	NA	Scram
6	Electrical Power	Reactor console key in off position	NA	Scram
		Loss of ac power to console	NA	Scram

4. The safety rod timer panel consists of an electronic timer, a key-lock arm switch and a five-position selector switch. Selector positions on the latter switch are OFF, Safety Rod #1, Safety Rod #2, Safety Rod #3, and Safety Rod #4. The timer measures the time lapse between a trip signal from the nuclear instruments and the safety rod-in limit switch closure.

To keep the system as simple as possible, bypasses are not provided in most of the scram circuits. It is felt that simplicity and ease of operation are more important than continuity of operation. If

important components become defective, the reactor is shut down until repair or replacement is completed. However, some bypasses are necessary; for example, an automatic bypass has been provided for low primary coolant flow while at powers less than 0.1 kW.

The fail-safe philosophy has been incorporated into the design as much as is practical. In most instances, circuits are completed by energized relays or actuated microswitches to give protection against loss of voltage, poor contacts, or broken wires. Manually operated switches with a spring-return to open circuit (or a more safe position) are used where practical, and include places such as the control rod drive circuits.

7.2 REACTOR CONTROL ROOM

The location of the reactor control room is shown in Chapter 1, Figures 1-1 and Chapter 3, Figure 3-1. The reactor cell is immediately to the north, the south cell to the east, Building 105 hallway on the south, and other laboratories to the west. The shielded doors between the control room and the two cell areas are the only personnel entries to these areas. A flashing warning light at the south cell doorway is actuated if the door is opened; however, the radiation level in the south cell must be above a preset minimum for this warning device to actuate. Since the south cell door will be open to perform some experiments, it also has an audible alarm which may be actuated by breaking the beam of light from an electric eye across the doorway. This alarm system alerts the reactor operator to traffic to or from the cell as required.

The reactor control console is a vertical metal structure approximately 6 feet high by 13 feet wide, designed to accommodate racks of standard 19-inch instrument chassis. Attached to the front of the vertical panel is a small sloping benchboard that contains the controls and indicating devices for the rod drives and most of the lights and switches for the alarm system. Also attached to the front of the panel is a horizontal work surface for the convenience of the operator. The vertical panels contain visual readout, power supplies, and recording devices for nuclear process and stack effluent release parameters.

An intercommunication system is installed with stations in the control room, set-up room, office areas, shop, north room, and others, as required. A loudspeaker page system is also available, which can be heard in all NTR areas. Communication between the control room and reactor cell is by a microphone and speaker system or by direct voice contact.

7.3 SCRAM SYSTEM

The scram system, which is shown in Figure 7-1 and Table 7-1, consists of manual, process, and nuclear scrams.

7.3.1 Manual Scram

The manual scram system consists of a manual button/switch located on the reactor console. When depressed, it directly opens the circuit supplying power to the safety rod magnets, providing 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. A manual button/switch is also located in the reactor cell.

7.3.2 Process Scrams

The process scram chain consists of relay contacts and switches connected in series between the +24V direct current bus and the coils of scram relays R-19 and R-20. Normally open contacts of these relays are in the circuit supplying power to the safety rod electromagnets and in the circuits for the rod drive motors. Two additional normally open contacts of these relays are used in the process scram chain parallel to the rod-in limit switches; this parallel circuit requires all motor-driven rods to be fully inserted before the scram relays can be energized. Any off-standard condition of any component supplying action to either the switches or to the relay contacts in the scram chain will disrupt power to scram relays R-19 and R-20 and cause them to deenergize. Process scrams are as follows:

1. The log N channel has an amplifier position mode switch which is used for checkout and testing of the instrument. Not having the amplifier mode switch in the operate position will prevent the safety chain from being made up, or, if moved from the operate position during operation, will scram the reactor.

2. Loss of positive high voltage below a predetermined value to the log N Compensated Ion Chamber (CIC) will scram the reactor. Loss of positive high voltage provides assurance that the ion chamber is capable of detecting neutrons.

3. A thermally actuated switch in the core outlet line senses the primary water core outlet temperature. A high outlet temperature will cause the switch to deenergize the scram relays and scram the reactor.

4. The primary flow is measured with a differential pressure transducer sensing the pressure drop across an orifice in the primary water coolant loop. An electric signal from the transducer is indicated at the control console. The reactor will scram when reactor power is greater than 0.1 kW and the primary coolant flow drops below a predetermined value.

5. Loss of ac power to the console will cause the scram relays and the magnet power supply to deenergize and scram the reactor.

6. A manual scram button is located in the reactor cell to scram the reactor from this area, if required. Actuation of this button also deenergizes the scram relays and will scram the reactor.

When the scram relays (R-19 and R-20) are deenergized, the following actions take place:

- The power being supplied to the safety rod electromagnets is interrupted to allow the spring-loaded safety rods to be inserted.

- All rod and motor circuits are closed to cause the rod carriages to drive in.
- The process scram chain is blocked open until the scram condition is corrected and all rods are fully inserted.

In addition to the above actions, the scramming condition will cause an annunciator to actuate, give an audible alarm, and illuminate a pushbutton lamp on the console which indicates the source of the scram. The audible alarm will continue until an ACKNOWLEDGE push-button switch is actuated; the pushbutton lamp remains illuminated until the scram condition is corrected and the pushbutton lamp is depressed. Some conditions cause indicator lamps to illuminate, but do not cause audible alarm; these conditions are not scrams.

7.3.3 Nuclear Scrams

The nuclear scrams consist of four power range channels. A block diagram of the system is shown in Figure 7-1. The power range instrumentation is used to monitor neutron flux (reactor power) and to protect the reactor against excessive power levels or rates of power rise. This instrumentation is required to be operable and connected to the safety system during each startup and the subsequent operating period. The system consists of four independent neutron detection channels; three are monitored by picoammeters and the fourth by a log N and period amplifier. The three picoammeters have trip circuits which operate into a two-out-of three (or one-out-of-two if one channel is inoperative) coincidence logic circuit capable of causing reactor scram. The log N and period amplifier is capable of causing reactor scram on fast period. The picoammeters have 20 ranges covering 10 decades of power from 10^{-9} and 10^{0}. Two ranges are available for each decade of 0 to 40 and 0 to 125 percent of that decade. The Log N channel normally covers the power range from 15 milliwatts to 150 kilowatts.

A gamma-compensated ion chamber is used as a detector in each power range channel. The detectors are positioned in thimbles in the fuel storage tank or at one of the faces of the reflector. The exact location selected for a particular chamber is determined by the intended use of the reactor, sensitivity of the system, and the desired meter reading. The desirability for seeing the neutron source for startup, the minimization of shadowing effects, and the provision of physical protection for the chamber are the primary factors considered when positioning the chambers. The CIC output currents can be interpreted in terms of reactor thermal power through calibrations based on measurement of thermal power as determined by a heat-balance measurement which utilizes the coolant flow through the core and the differential temperature across the core.

High voltage for the three picoammeters CICs is internal. When in a two-out-of-three situation, loss of positive high voltage on any two picoammeters will cause a reactor scram. When in a one-out-of-two situation, loss of positive high voltage on any one picoammeter will cause a reactor scram.

High voltage for the log N period amplifier CIC is supplied by a power supply in the log N period amplifier. Loss of positive high voltage from this power supply will initiate a scram through the process scram relays.

Three multirange picoammeters are normally used (although operation is permitted with two) to monitor the signals from three (or two) of the CICs. Sensitivity and range of the systems are such that the flux (with the reactor shut down) from the reactor neutron source will bring all channels well on scale, and maximum reactor power does not exceed the range of the instrument. Each picoammeter amplifier output signal, in addition to driving the picoammeter meter, and remote meter is connected to an internally mounted trip circuit and externally through a selector switch to a linear power recorder. Each trip circuit is set to trip when the meter reads 125 kW or less. When the instrument reading is less than the trip point, the trip circuit supplies 12 Vdc to a coincident logic circuit, wired to cause reactor scram if 2 of 3 (or 1 of 2) inputs are tripped. Each picoammeter has a downscale alarm. When two indicate an alarm, the control and safety rod motors cannot be energized to withdraw. Reactor operation may continue with one picoammeter out of service, provided the trip circuit is set up so that a trip signal from either of the remaining picoammeters will cause reactor scram. If one picoammeter is out of service, the interlock at the low end of the scale for that picoammeter which prevents rod withdrawal may be bypassed. These automatic actions ensure that the picoammeters have the proper start-up sensitivity and the high power scram trip point is always within a decade of operating power during operations which increase power.

Reset/bypass switches allow remote resetting of picoammeter trips and bypassing the pico trip when changing ranges due to spurious noise generation. Use of bypass is restricted to range changing operation only, and the bypass switches must be used one at a time.

The log N and period amplifier receives its signal from the fourth CIC and displays the reactor period and reactor power on front panel meters. This system may be set up to cover the power range from source or reactor critical level, depending on CIC position, to 150% of power. Relay outputs from the period amplifier trip circuit and the log N amplifier trip circuits are connected through the noncoincident logic circuit to initiate a scram for reactor periods of less than 5 seconds. At powers of less than 0.1 kW, a signal from the log N recorder actuates automatic bypass of the primary coolant low-flow scram. At powers greater than 0.1 kW, a signal from the log N amplifier actuates a relay which automatically switches the signal to a single function recorder from the start-up channel (source range monitor) if utilized to the thermocouple pile (millivolts) signal, which is required in the heat balance calculation. The log N power signal is also recorded on a strip chart recorder. The mode (multiposition calibration) switch for the log N amplifier is interlocked to scram the reactor when the switch is not in the OPERATE position.

The diode logic element system consists of two units. One unit performs coincidence logic functions and the other performs noncoincidence logic functions on signals from the nuclear instrumentation system.

A coincidence logic unit contains five independently functioning component boards which can accommodate a total of 16 signals. Four of the component boards are identical, and provide circuits for performing two-out-of-three coincidence logic. The fifth circuit component board (not used) is designed to perform selective two-out-of-four coincidence.

The 12-V trip output from each of the three picoammeters passes through contacts in the meter relays monitoring positive high voltage to the CIC in that channel. The trip outputs from the three channels are converted parallel to the 2/3 coincidence logic unit. A trip on any two picoammeters or loss of high voltage to any two CICs or a loss of voltage on one channel plus a high power trip in another will cause trip outputs from the coincidence logic unit to be sent to the noncoincidence logic unit which causes deenergization of the power switches.

The noncoincidence logic unit contains two independent noncoincidence logic component boards, each of which accommodates nine input signals and provides one output signal. Depending on the input signal levels, each noncoincidence logic component board provides either 16 volts dc or less than 1 volt output to a power switch. For the output level to be 16 volts, all of the inputs must be normal. If any one or more inputs drop to zero, the output signal drops to less than 1 volt.

Input signals to each nine-channel noncoincidence logic board consist of the following:

- Two 24-V signals that pass through scram relay R19B;
- Two 24-V signals that pass through scram relay R20B;
- One 12-V signal from log N high power trip;
- One 12-V signal from log N fast period trip; and
- Three 12-V trip signals from one coincidence logic unit.

Output signals to each noncoincidence logic board go to a power switch which controls the electromagnet excitation current. If either power switch trips, current to all magnets will be interrupted.

Power for the safety rod electromagnets is supplied from a direct current power supply with the capacity of supplying all four electromagnets. Power to each electromagnet is routed through individual power-adjust modules so that minor variations in the electromagnets can be compensated.

7.4 SAFETY-RELATED SYSTEMS

Safety-related systems are listed in Table 7-2. They consists of instrumentation and systems to assist in the operation of the facility, measure operating parameters, or warn of abnormal conditions. The safety-related instrumentation or systems provided include the following:

1. A differential pressure switch measures the pressure difference between the reactor cell and control room. This switch actuates a visual and audible alarm if cell negative pressure drop below a preset level (not less than 0.5 inches of water). The reactor

Table 7-2

SAFETY-RELATED SYSTEMS

Item No.	System	Condition	Set Point	Function
1	Reactor Cell Pressure	Low differential pressure	≥0.5 inch of water	Visible and audible alarm; audible alarm may be bypassed after recognition
2	Fuel Loading Tank Water Level	Low water level	8-inches below tank overflow	Visible and audible alarm; audible alarm may be bypassed after recognition
3	Primary Coolant Temperature	High core outlet temperature	≤140°F	Visible and audible alarm; audible alarm may be bypassed after recognition
4	Primary Cooling Core Temperature Differential	Core delta temperature	NA	Provide information for the heat balance determination
5	Radiation Monitors	North room high level	100 mr/h	
		South cell high level	100 mr/h	
		Reactor cell high level (reactor shutdown)	100 mr/h	Visible and audible alarm; audible alarm may be bypassed after recognition. May be temporarily out of service if portable instruments are used during personnel entry and occupancy
		Reactor cell high level (reactor operating)	10^6 mr/hr	
		Control room high level	5 mr/h	
6	Stack Radioactivity	Beta-gamma particulate high level	$\leq 1 \times 10^4$ cpm	Visible and audible alarm; audible alarm may be reset after recognition
		Noble gas high level	$\leq 2 \times 10^{-11}$ amps	
7	Linear Power	Low power indication	≥5% of full scale	Safety or control rods cannot be withdrawn (1-out-of-3 or 1-out-of-2)
8	Control Rod	Rods not in	NA	Safety rod magnets cannot be reenergized, may be bypassed to allow withdrawal of one control rod or one safety rod or one safety rod drive for purposes of inspection, maintenance, and testing
9	Safety Rod	Rods not out	NA	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 or one safety rod drive for purposes of inspection, maintenance, and testing

power must not be increased above 0.1 kW unless the cell negative pressure is as noted above. If the cell negative pressure drops below the preset level and the reactor power is above 0.1 kW, then the reactor power shall be lowered to ≤0.1 kW immediately and corrective action taken, as required. This ensures that the direction of air flow is from the control room into the reactor cell and that potentially contaminated reactor cell air due to reactor operation is released through the ventilation system.

2. Liquid level switches are provided on the fuel loading tank and actuate an alarm circuit when the tank is either low or too high. As long as the level is above the low-level alarm, it can be assured that the core tank is filled with water. The high-level alarm assures that adequate indication is given to the operator during the filling of the fuel loading tank that it will not overflow, or denotes possible secondary system to primary system leak.

3. Thermocouple in the primary water core outlet line senses the primary water temperature and reads out on a panel meter in the control room. A high-temperature warning alarm is actuated when the set point is reached to indicate a high primary water temperature.

4. A thermocouple pile is provided which indicates the primary coolant core temperature differential. This is utilized in combination with the primary coolant flow rate to provide information for a heat balance determination.

5. An area radiation monitor system is provided. The use of this system will assure that the area(s) of the facility in which a potential high-radiation area exists are monitored to assure protection of personnel.

6. An NTR stack radioactivity air monitoring system is utilized. Separate detection channels and alarms are used for particulate material and nonfilterable radioactive gases to assure that the releases are acceptable.

7. A low power level rod block and alarm is provided on the linear power system. This rod block and alarm assures that the operator has a linear power channel operating and indicating neutron flux levels during rod withdrawal.

8. Interlocks are provided on the control rods to prevent outward movement unless the safety rods are all in a full-out position. This condition assures that the reactor will be started up by withdrawing the four safety rods prior to withdrawing the control rods. A bypass is provided for testing purposes.

9. Interlocks are also provided on the safety rods. Each safety rod must be withdrawn in sequence to assure the normal method of reactivity control. A bypass is provided for testing purposes which will allow any one safety rod or safety rod drive to be withdrawn.

Other systems provided include the following:

1. A variable area flow meter (Rotometer type), mounted in the control room, to indicate the secondary coolant flow to the tube side of the heat exchanger;

2. A recorder to monitor thermocouples placed at several locations throughout the primary and secondary systems and the graphite pack.

3. A constant air monitor located in the control room to monitor the air activity in the reactor cell. The monitor is checked prior to each initial entry into the reactor cell.

4. A differential pressure switch senses the pressure difference between the core inlet and outlet lines. When core differential pressure, which is an indication of flow, falls below a preset value, an alarm circuit is actuated and indicates a low differential pressure in the core.

7.5 REACTOR REACTIVITY CONTROL SYSTEMS

Three types of movable neutron poisons are provided to control core reactivity: safety rods, control rods, and manual poison sheets. All these poisons are located about the periphery of the fuel container, and all run 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 poisons on a 9.5-inches radius or about 0.6-inch from the outside edge of the active core. The control and safety rods have horizontally mounted drive mechanisms that are supported from the north face of the reactor on a 5-foot-high aluminum support plate located about 4-1/2 feet in front of the north face. The manual poison sheets are inserted or removed manually through access holes provided in the north shield. Each type poison was designed to perform a specific function. The four safety rods were designed for rapid insertion to scram the reactor. The control rods (two coarse and one fine) were designed for the precise position control and indication required for analytical work during which the reactor is used as a detector. The manually positioned poison sheets are used to limit the reactivity available to the operator or to increase the shutdown margin. Figure 7-2 shows the control circuits for the safety rod and control rod drives. Refer to Section 4.2.2 for more details.

7.6 RADIATION MONITORING SYSTEMS

Radiation levels (gamma) are monitored by a five-station remote area monitor. Areas monitored are the reactor cell, south cell, control room, and north room (two stations). Radiation levels are indicated on the control console. Each channel is equipped with an alarm which will actuate visual and audible alarms in the control room and the affected area. The set points of the detectors are as specified in Table 7-2. In addition, the south cell monitor is interlocked with the south cell shutter and door controls to prevent inadvertent exposure to the radiation beam from the reactor.

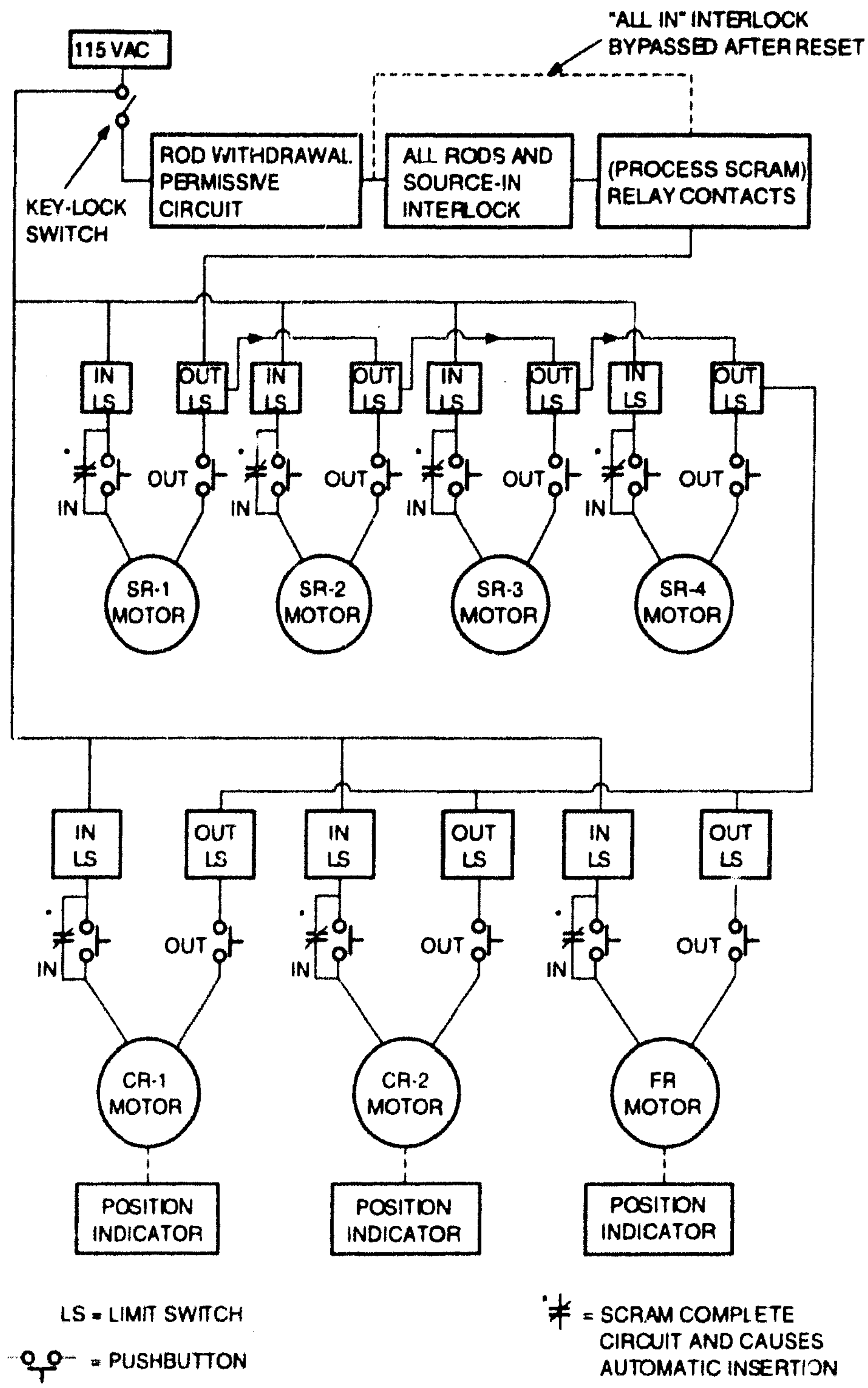

Figure 7-2. Simplified Block Diagram of Rod Drives

7.7 NEUTRON SOURCE

A reactor start-up neutron source is installed on an electric motor drive mechanism similar to the control rod drives. The source drive has the same controls and indications as a control rod drive, with the exception that continuous position indication is not provided. The same interlocks as those on the control rods are provided (the safety rod magnets cannot be energized until the source is full in), except that it is not necessary to pull any safety rods to withdraw the source. Following a process scram (de-energize R19 and R20), the source automatically runs to the fully inserted position. The source travels in a guide tube identical to that used for the control rods, and the limit switches are adjusted so that the source moves about 30 inches from the in to out positions. A 0.2-Ci radium-beryllium source emitting about 10^6 n/sec is used for a startup source. It is an R-Monel encapsulation approximately 1/2-inch in diameter and 3-1/2 inches log, attached to an aluminum extension rod that connects to the source drive mechanism. The source provides at least the minimum neutron flux signal required for the nuclear instrumentation for startup and also gives good indication of subcritical multiplication.

8.0 ELECTRICAL POWER SYSTEMS

8.1 NORMAL ELECTRICAL POWER SYSTEMS

A 480-V load center in Building 105 is fed from the site's 12-kV bus, and in turn, feeds power and lighting distribution panels for the NTR facility. The 12-kV bus is supplied from an on-site 5000 KVA transformer fed from parallel off-site 60 KVA (Figure 8-1) utility supplies. Historically, the NTR suffers one electrical outage a year, but with the reactor on-line only a percentage of an 8-hour 5 day work week, most outages occur when the reactor is secured.

Four breakers in the control room console (two regulated and two unregulated) feed individual switches and breakers for the primary coolant pump, service outlets, facility lights, and the reactor console. Power supplied to the console is used for reactor instrumentation and control rod and safety rod drive motors.

Upon loss of electrical power to the facility, the four safety rods will scram and the three control rods will remain as-is. All non-operations and non-nuclear safety personnel are evacuated from the control room, north room and south cell. Procedures are in place to ensure nothing is done to increase the reactivity of the reactor. Radiation readings are to be taken to verify the reactor is shutdown. The "Rod insert bus" breaker remains closed so that the control rods will insert automatically when power is restored.

While the control rods are withdrawn, the reactor is not secured. A licensed operator will remain in the control room with a second trained individual on site. A licensed SRO will be present or readily available on call. If the power outage occurs during reactor shutdown, a barrier is placed across reactor cell doorway if it is open. If the power outage occurs during reactor shutdown and the reactor cell door is open and the reactor has been operating less than one-hour before, a plastic sheet is taped over the cell doorway. These actions ensure that there is no uncontrolled release of radioactive material and, that qualified individuals are at the facility as necessary.

Safety Rod Magnet Power Supply

This power supply is a regulated, constant voltage/constant current DC Hewlett Packard Model 4633A. The ranges are 0-150 VDC and 0-3 amps. The unit is operated at 60 VDC constant voltage resulting in a current of 0.75 amp to the safety rod magnets. This current will decrease approximately 50 ma as the magnet coils heat up.

The power supply is regulated to less than 18 mV change for 105 to 125 VAC input variation and less than 36 mV variation for a 0 to 10-amp load change. The power supply has an external jack located on the front panel.

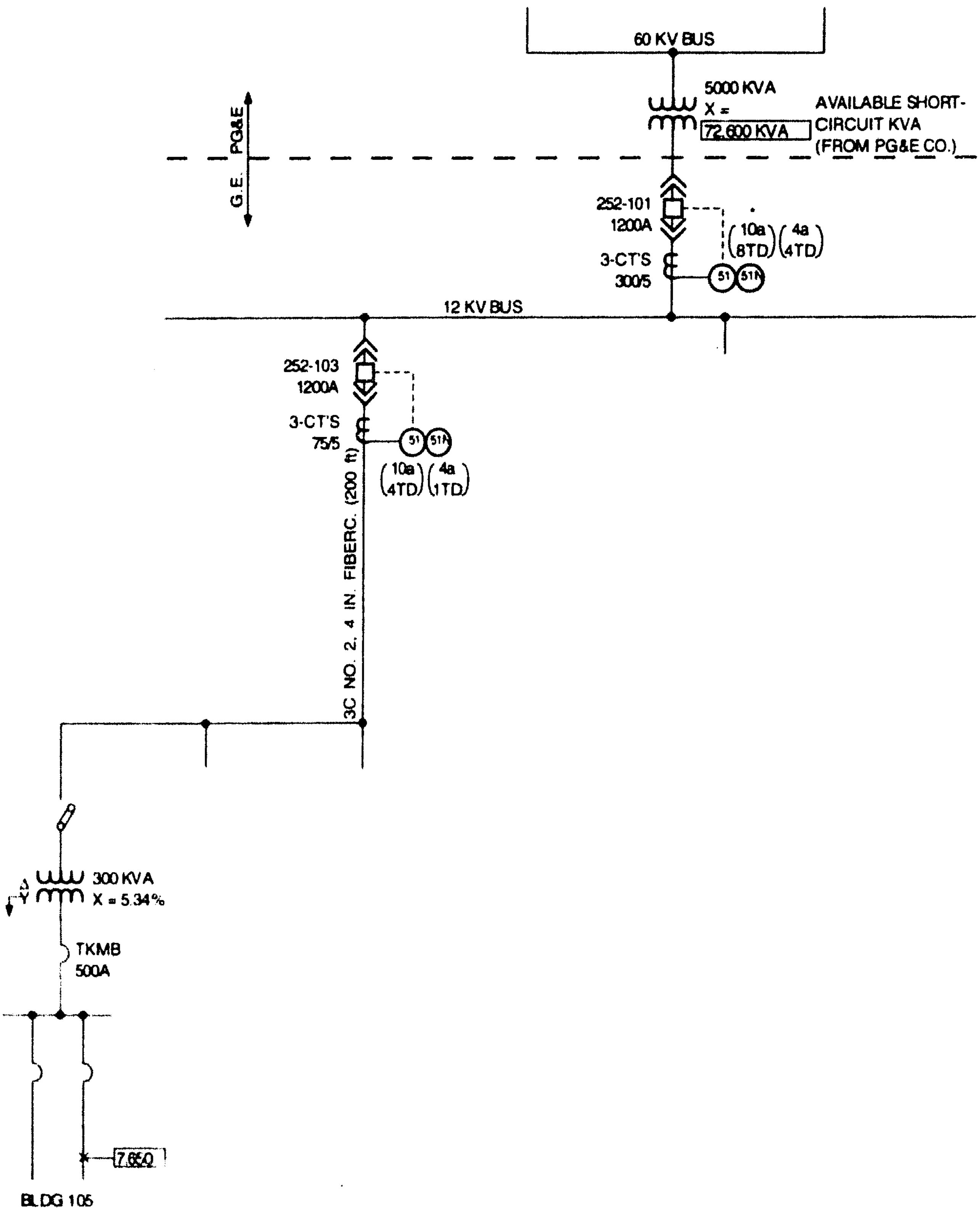

Figure 8-1. Parallel Offsite 60 KVA

Safety System Power Supply

This power supply is a Hewlett-Packard 6433B power supply. It is the same as the safety rod magnet power supply except the rating is 0 to 36 VDC and 0 to 10 amps. The unit is operated at 24 VDC, and the system draws approximately 1 amp during reactor operation. This power supply provides power to the scram system relays, the console annunciator lights and relays, and the rod drive motor controls.

Safety System Power Supply

This power supply is a General Electric NDO4 power supply. It receives 120-VAC input and supplies 26 VDC at 50 mA. It is regulated to provide ±0.5% output for line variations of 100 to 130 VAC and 50 to 60 cps. The power supply has one toggle switch on the front panel to switch the unit ON and OFF. An indicator light on the front panel indicates ON-OFF status. This power supply provides power to the power switches, the logic elements, and the CAS alarm relays and lights.

Log N Power Supply

This power supply is a General Electric INMAC 194X606G1. It receives 120-VAC input and supplies positive and negative 24-VDC, 10-amp output for the Log N amplifier. It is temperature compensated to operate at a constant output between 5 and 50°C. The power supply is regulated to ±5% for a 10% change in line voltage and to ±5% for a 0-5 amp change in load.

Picoammeter Power Supply

This power supply is a General Electric INMAC 194X606G1. It receives 120-VAC input and supplies positive and negative 24-VDC, 10-amp output for the three picoammeters. This power supply is identical to the Log N power supply described above.

8.2 EMERGENCY ELECTRICAL POWER SYSTEMS

The NTR has no emergency power system for the physical plant. [redacted]

Semiportable emergency lighting units are installed at several locations in the facility. A battery maintains its charge from the regular 115 VAC circuits and upon loss of AC power is energized automatically to provide light for emergency action and personnel exit.

9.0 AUXILIARY SYSTEMS

9.1 HEATING, VENTILATION AND AIR CONDITIONING SYSTEMS

The Heating, Ventilation and Air Condition (HVAC) Systems at the NTR are a combination of central and wall units to control temperatures for personnel comfort. The setup room uses a central HVAC unit while the shop utilizes two wall units. The control room and office spaces share a central HVAC unit on the roof of Building 105 with other non-NTR work spaces in Building 105. There are no safety features or controls associated with these HVAC systems.

Potentially contaminated air and other gas s are collected in the reactor cell ventilation system and passed through absolute filters before being discharged from the NTR stack. Ducts draw air from the reactor cell, the ceiling of the south cell, and from the forward position (for neutron radiography of radioactive objects) in the Modular Stone Monument in the north room. This ventilation system does not have heating or air conditioning capabilities.

The reactor is not operated above 100 watts unless the Reactor Cell Ventilation System is operating. For more details refer to section 6.2.1 and 7.7.

9.2 HANDLING AND STORAGE OF REACTOR FUEL

The NTR core is designed to last a lifetime. The fuel assemblies were installed in 1957 and have operated satisfactorily since then. There are no unirradiated fuel assemblies or discs stored on site. Refer to Section 4.2.1 for more information about NTR fuel.

Facilities are available to perform any fuel handling operations that might become necessary in conjunction with operation of the reactor. A handling tool is provided for remote underwater transfer of the assemblies between the core and the fuel loading tank. If it is necessary to remove more than one fuel assembly, special arrangements may be made to use a shielded transfer cask and storage facilities elsewhere on the Site. Proper authorization would be obtained before such transfers were made, and procedures would be developed to ensure safe handling with adequate consideration for radiation protection and criticality control. The transfer of the entire core to another on-site facility in 1977 demonstrates the validity of this process.

The fuel loading tank, approximately 12 feet high by 5 feet long by 4 feet wide, is located adjacent to the west face of the graphite reflector. An expansion joint connects the west end of the reactor fuel loading chute to the east side of the tank. An aluminum gate for the loading chute is attached to the inside of the east wall of the tank and is normally in a partially closed position. A loading platform which essentially extends the loading chute into the tank may be attached to the inside of the east wall of the tank, as required. Affixed to the opposite tank wall is a storage rack for the fuel loading chute's aluminum-covered graphite plug. A pulley for the plug cable-lift is attached to the storage rack. Two 4-in.-diameter and one 2-in.-diameter aluminum thimbles installed in the tank are used to hold detectors for reactor nuclear instrumentation or samples for irradiation in a low-flux region. Level switches indicate high and

low water level in the tank by energizing annunciator lights at the console. Access to the loading tank is from the reactor cell mezzanine. With a normal level in the tank, there is about 5 feet of water for shielding above the loading platform.

9.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

Fire protection equipment and procedures for the NTR are similar to conventional industrial plant fire protection equipment and procedures established for the existing Site. Equipment, buildings, procedures, etc., are in accordance with company-wide standards, state and local regulations, and the recommendations of insurance agencies.

Six-inch fire mains, which are legs of a loop surrounding Buildings 102 and 105, are located on the east and west sides of Building 105. These mains supply outdoor fire hydrants located at the northeast, southeast, and southwest corners of Building 105, and an extensive sprinkler system located within the building. Fire hoses and nozzles are kept permanently located in the Building 105 hallway and the southeast corner of the building. The 500,000-gal Site raw water storage tank, located approximately 130 feet higher than Building 105, is the source of water for the fire mains; 100,000 gallons of this water is reserved for fire protection. In addition to the water system, conventional portable fire extinguishers are located throughout the NTR facility and Building 105.

A Site-wide fire brigade provides fire protection for the entire Site. The brigade members are trained in the use of the Site fire truck and other fire fighting equipment. All members are familiar with work under radiation conditions and requirements of the Site instructions for radiation protection.

In addition, a California Division of Forestry fire station is located about six miles west of the Site and other fire fighting facilities are able to respond through a mutual aid system. The local CDF fire fighters periodically visit the Site to familiarize themselves with the facilities.

Flammable liquids and combustible materials are limited and controlled in all areas. These are regularly checked by the building safety inspections and by insurance carrier audits.

The reactor shall be shut down immediately if a fire occurs in the control room, the south cell or the adjacent hallway or laboratory rooms. In the event of a fire in another part of Building 105, reactor operation may continue if the fire is small and contained. However, if the fire is uncontained or is not small, the reactor will be shut down and secured.

9.4 COMMUNICATION SYSTEMS

The NTR is a small, simple, compact facility. Most work is performed in the south cell which is entered from the control room. Coordination with experimenters is minimal and is accomplished with an intercom from the control room to the north room and the setup room. Standard telephones in these areas may also be used.

Numerous other communication means are available for normal operating, maintenance and emergency situations. These include two-way radios (which are limited to areas where radio frequency sensitive pyrotechnic material is not present), a local public address system activated from the control room, a site-wide public address system [redacted] and a High-level Conference Circuit (HICON) which is an open line to the security building.

9.5 POSSESSION AND USE OF BYPRODUCT, SOURCE AND SPECIAL NUCLEAR MATERIAL

The NTR facility is described in Chapter 1. The NTR facility license, R-33, applies only to Byproduct, Source and Special Nuclear Material needed for operation of the reactor and its experimental programs. All Byproduct, Source and Special Nuclear Material used in other laboratory areas of Building 105, and other laboratory areas in other buildings will be possessed and used in accordance with another license issued by the NRC or the State of California.

9.6 COMPRESSED AIR

Compressed air for the facility is supplied from the Building 105 service air compressor located in the second-floor mechanical equipment room. The compressor will deliver 50 scfm of free air and is capable of a discharge pressure of 100 psig. A relief valve at the air compressor maintains system pressure at less than 120 psig. 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, which can be emitted from the horizontal facility. Conveniently located outlets are provided to supply experiment equipment or for service air. A loss of compressed air would cause the south cell door and the south shutter to fail in the as-is position. This failure would have no safety significance. Compressed air from this system has not been used for experiment equipment for decades. If an experiment were to use compressed air from this system in the future, a 10CFR50.59 review would analyze the consequences of compressed air failure prior to conducting the experiment.

9.7 RADIOGRAPHY VACUUM SYSTEM

Two identical Duo-Seal, belt-driven vacuum pumps provide vacuum to the neutron radiography areas. One pump provides vacuum to the dark room, south cell, and control room. The second pump provides vacuum to the north room.

The neutron radiography vacuum film cassettes are connected to these vacuum sources so that the radiography film maintains intimate contact with the conversion screen in the cassette for a quality radiography image.

The vacuum system is not used for any reactor system and has no safety significance.

10.0 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 SUMMARY DESCRIPTION

Though the NTR is used primarily for nondestructive, test of materials, it is also used for irradiation of various types of materials. The experiment facilities at the NTR are defined as locations for experiments on or against the external surfaces of the main graphite pack and thermal column including the horizontal facility, vertical facility, fuel storage tank, CHRIS facility, and the fuel loading chute. These facilities, according to types, are as follows:

Incore Facilities

The Horizontal facility is used for three different types of experiments depending upon its configuration. It can be configured with a source log and pinhole unit and used for neutron radiography (nondestructive testing), or filled with graphite and used for reactivity testing of materials (nondestructive testing), or if completely empty (or pinhole removed) it is used for an irradiation facility.

In-reflector Facility

The vertical facility extends vertically from the top of the graphite pack, through the graphite reflector, approximately tangent to the east side of the fuel container.

Another experiment location is the fuel loading chute which extends diagonally through the graphite pack from the fuel loading tank to the reactor core.

In-reflector and Automatic Transfer Facility

The Cable-Held Retractable Irradiation System (CHRIS) is a dry tube that allows access for a cable-held carrier to an experiment 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 above the top of the reactor core can.

Thermal Columns

There are three experimental areas that utilize the radiation coming from the external surfaces of the main graphite pack and thermal column. Two of them, the Top and East Face, are located in the reactor cell on the top and to the east side of the main graphite pack. The third, the Thermal Column, is located on the south side of the reflector.

Entry into the reactor cell is necessary for access to the Face facilities; therefore, objects to be irradiated are generally positioned before reactor startup or are provided with remote positioning devices.

Access to the Thermal Column is from the south cell. Sections of the biological shield may be removed to provide access to the graphite reflector or to permit use of the radiation beams.

10.2 EXPERIMENTAL FACILITIES

The Horizontal Facility

The horizontal facility is a 5-inch-diameter hole traversing the horizontal axis of the reactor. It contains a removable sleeve which makes it a 3-inch-diameter hole for current use. A source log which is 30.5 inches long and 2.96 inches in diameter is centered with the core can in the horizontal facility. The source log is an aluminum pipe containing pieces of graphite, lead and plastic to make the neutron beam more uniform. The thermal column end is stepped to 3.125, 5.0 and 8.0 inches diameter. A pinhole is installed in the 3.125-inch-diameter area for proper focus of the neutron beam for neutron radiography.

The horizontal facility is accessible from either the south cell or reactor cell. The south cell is provided with an air-operated radiation shield shutter that is also used for neutron radiography. An electrically operated shutter is used at the penetration to the modular stone monument, in the north wall of the reactor cell. Both shutters have their own timers and can be controlled from either the reactor control console or at the entrances to the south cell or the north room.

The modular stone monument (MSM) is a dual neutron radiography facility, located in the north room, designed to allow neutron radiography on unirradiated or irradiated objects. The design involves six concrete blocks that make up the shield and structural unit. A 12-inch-i.d. stainless steel pipe, capped off at the bottom, penetrates into the ground beneath the MSM for 20 feet. This penetration allows neutron radiography of long objects to be performed by lowering them into the pipe.

Mechanisms for changing neutron radiography imaging foils without returning irradiated objects to their casks have been incorporated into the design.

Irradiated objects normally arrive at the NTR in large casks which are placed on the MSM, using the overhead crane. The objects can be then lowered down into the MSM in front of an imaging foil and the neutron radiography is then performed.

Unirradiated objects are moved into a facility on the north end of the MSM, usually by a trolley arrangement. The imaging system is placed behind the object and neutron radiography or irradiation is performed.

Irradiations may be performed anywhere in the horizontal facility or in the radiation beams streaming from the tube ends. The pinhole or the pinhole and source log may be removed to install the sample. Material or equipment to be irradiated in the horizontal facility may be fastened to an extension rod and positioned manually from the south cell. Specially machined graphite logs suitable for use as sample holders for specific irradiations are available.

Electrical leads, cooling lines, and associated equipment for instrumented devices can be brought out either end of the horizontal facility for connection to equipment in the reactor cell, south cell, or through available penetrations to or above the control room, set up room or north room.

During irradiation, reactor instrumentation, as well as any instrumentation associated with the irradiation device, is observed carefully. In the event of an unexpected change in neutron multiplication, critical rod position, radiation levels, or reactor irradiation device behavior, the operator will take whatever immediate action is required to ensure the safety of personnel, the reactor, and the irradiation device. He will then notify a Licensed Senior Reactor Operator, as required, who will evaluate the situation and initiate whatever further action is necessary.

Unloading of the horizontal facility is usually done with the reactor shutdown; however, if the reactivity effect of the sample and the radiation level permits, the sample may be removed with the reactor operating. If the radiation level from the sample is such that conventional tweezers or pliers that are normally used to handle samples are inadequate to control exposure properly, temporary shielding and remote handling tools will be utilized. To minimize radiation exposure to the operators, irradiated samples are normally allowed to decay as long as practical before handling them.

The Vertical Facility

The vertical facility is defined by a 5-inch^2 by 5-foot-long aluminum can, which extends vertically through the graphite reflector approximately tangent to the east side of the fuel container. A piece of reflector graphite normally fills the facility when it is not in use. The facility is accessible only from the top of the reactor inside the reactor cell. Irradiations may be performed at any position within the aluminum can or in the beam streaming from the top.

Devices to be irradiated in the vertical facility are usually attached to a wire or extension rod supported from the top shield of the reactor. Since entry into the reactor cell during critical operation is normally forbidden, samples must be positioned manually before startup or provided with a means for remote positioning. Space not occupied by the device may be filled with graphite blocks. Leads from the device can be brought out of the top of the facility for connection to equipment in the reactor cell or through cell penetrations to or above the control room, set up room, or north room.

The precautions and procedures during irradiation are the same as those discussed for the horizontal facility. The only significant difference is that the shut-down radiation level of the reactor core may contribute appreciably to the radiation exposure received during the unloading. However, radiation monitoring is required for sample unloading at all times, which ensures that the operator is aware of radiation from all sources. In addition, the cell remote area radiation monitor (with readout in the control room) would indicate unexpected high radiation levels.

Fuel Loading Chute Facility

The aluminum-covered graphite plug for the fuel loading chute may be removed to provide access to the inside of the fuel container for irradiation. Use of this facility is necessary for performing experiments, such as some of those required to determine the nuclear characteristics of the reactor. No experimental objects are permitted inside of the core tank when reactor power is greater than 100 watts. Experimental objects in the fuel loading chute will be secured to prevent their entry into the core region during normal operating conditions.

CHRIS Facility

The CHRIS (Cable Held Retractable Irradiation System) is a dry tube that allows access for a cable-held carrier to an experiment position during reactor operation. Entrance to the irradiation system is from the NTR North Room mezzanine. The tube runs through the NTR reactor cell wall, across the reactor cell, and connects to the experiment position.

The peak thermal neutron flux in the experiment position (approximately at the core centerline) is approximately ~8×10^{11} nv.

Samples to be irradiated are placed in a carrier. The carrier is a capped aluminum tube that slides through the irradiation system. One cap is removable for sample loading and unloading. A flexible cable is attached to the carrier to insert it into and retract it from the irradiation system.

As a minimum, a portable dose rate instrument is used for monitoring the cable and carrier during withdrawal. Finger rings shall be worn when handling the irradiated cable.

At least 18 hours of decay is allowed after irradiation before handling the irradiated portion of the cable.

Top Face and East Face

Radiation escaping through any one of the faces of the 5-ft^3 graphite reflector may be utilized for experimentation. The aluminum box (partially cadmium lined) that contains the reflector is provided with 4-foot by 4-foot removable sections on the top and east faces. The removable section for the east face contains a 20-inch by 18-inch hinged section, which can be opened to eliminate the cadmium from this area. Limited space between the reflector and the top shield slab can be used without removing the 48-inch o.d. concrete plug in the shield. Entry into the reactor cell is necessary for access to the Face facilities. Therefore, objects to be irradiated are generally positioned before reactor startup or are provided with remote positioning devices. Irradiation devices utilizing the Face facilities have a negligible effect on core reactivity.

Thermal Column

The thermal column is a 4-ft cube of high purity graphite located against the south side of the reactor's 5-ft cube graphite reflector. The thermal column is traversed by the horizontal facility.

A 20-inch wide by 20-inch-high by 4-foot-long centrally located section, made of 4-inch by 4-inch by 4-ft logs was designed to be removed partially or entirely to accommodate experiments inside the thermal column or to provide external radiation beams. The south face of the thermal column is accessible from the south cell. Radiation shielding on this face consists of a 0.375-inch boral plate, a 26-inch-thick wall of concrete bricks, and a 4-inch wall of lead bricks. Sections of the biological shield may be removed to provide access to the graphite face of the thermal column or to permit 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.

Irradiations using the thermal column normally have a negligible effect on core reactivity and may be loaded or unloaded with the reactor operating or shutdown. Radiation exposure to the operator is usually of more concern during such activities than the effects of the irradiation device on the reactor.

10.3 EXPERIMENT REVIEW

Safety-oriented limits and restrictions applicable to experiment facilities and experiment programs follow. The limits and restrictions presented are derived from the reactor and experiment safety analyses, nearly 40 years of experience in conducting experiments at the NTR, and sound engineering practice. The majority of these limits are contained in the Technical Specifications. Adherence to the limits and restrictions below is mandatory and provides assurance that:

1. There is no anticipated mode of experiment operation that will endanger the health or safety of the general public or plant personnel.

2. No experiment will be performed that involves a technical specification change or an unreviewed safety question (as defined in 10 CFR, Section 50.59).

3. A proposed experiment type will be evaluated in detail and its execution controlled so as to reduce any radiation exposure to the public and plant personnel to the lowest practicable level.

A. General Experiment Requirements

1. 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. Records of such evaluation and approval shall be maintained.

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

3. Experimental capsules to be utilized in the experimental facilities shall be designed or tested to ensure that the pressure transients, if any, produced by any possible chemical

reaction of their contents and leakage of corrosive or flammable materials will not damage the reactor.

4. No experimental objects shall be inside the core tank when the reactor is operating at a power greater than 0.1 kW.

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

B. Reactivity Limits

1. Requirements pertaining to the reactivity worth of experiments are as follows:

   a. The sum of the potential reactivity worths of all experiments which coexist plus the reactivity available from control rods and coolant temperature shall not exceed 0.76$.

   b. No experimental object shall be moved during reactor operation unless its potential reactivity worth is known to be less than 0.5$ and the operation is performed with the knowledge of the licensed operator at the console. 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.

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

2. The potential reactivity worth of experiments shall be assessed before irradiation. If the assessment warrants, the reactivity worth of the experiment shall be measured and determined acceptable before reactor full-power operation.

C. Explosive and Flammable Material Lists

1. The maximum amounts of explosives permitted in the NTR facilities are as follows:

   a. South cell: $W = (D/2)^2$ with $W \leq 9$ pounds and $D \geq 3$ feet;

   b. North room (without MSM): $W = D^2$ with $W \leq 16$ pounds and $D \geq 1$ foot;

c. North room (with MSM): W = 2 pounds in the MSM and a maximum 16 pounds in the north room.

d. Set-up room: W = 25 pounds

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

D = Distance in feet from the south cell blast shield or the north room wall.

2. A maximum of 10 Curies of radioactive material and up to 50 grams 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 5 feet from any explosive device. Radioactive materials other than those produced by neutron radiography of the explosive devices and imaging systems are not permitted in the set-up room if explosive devices are present.

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

4. The cumulative radiation exposure for any explosive device shall not exceed 3×10^{12} n/cm^2 from thermal neutrons and 1×10^4 roentgens from gamma.

5. 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 watt. Experimental facilities containing flammable materials shall be vented external to the reactor graphite pack.

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 RADIATION PROTECTION

11.1.1 Radiation Sources

11.1.1.1 Airborne Radioactive Sources

The most significant source of airborne radionuclides during the normal operation of the NTR is the thermal neutron activation of the naturally occurring argon gas in the air and dissolved in the cooling water in and around the reactor core. A small contribution to the total airborne radionuclides can occur from the gaseous fission products emitted from the trace quantities of uranium which may have contaminated the aluminum skin of the reactor fuel elements during fabrication. The radioactive noble gas isotope Ar-41 is the predominant radionuclide emitted from the NTR reactor cell through the exhaust ventilation system.

It can be demonstrated using fission product generation and decay codes that the activity inventory of fission produced halogens and fission produced noble gasses are approximately equal at shutdown (zero decay time). Further it can be shown by the same method that the ratio of the total fission produced noble gasses is approximately 46 times the I-131 activity at shutdown. Therefore if the measured I-131 release rate is known, and the measured total noble gas release rate is known, the quantity of the total noble gas release due to fission product noble gasses can be estimated. By this process, the fraction of the total noble gas release due to fission produced noble gas turns out to be a very small fraction of the total noble gas release, and therefore it is confirmed that most of the measured noble gas release during normal operation is due to Ar-41. For example:

> During 1996, which is typical of other years of operation, the total measured release of I-131 from the NTR stack was calculated to be 31.1 μCi.
>
> The total measured release of noble gas during the same period was approximately 70 curies.
>
> A radioisotope inventory of the NTR core calculated with the Radioisotope Buildup and Decay,[1] RIBD, computer code showed that the ratio of fission produced total noble gas activity to the I-131 activity at shutdown was approximately 46:1.
>
> Therefore the total activity of fission produced noble gas released during that year was approximately 1,400 μCi(46 times 31.1 μCi), assuming equal fractions of I-131 and noble gasses were released from the core to the exhaust system.

1 RIBD, Radio Isotope Buildup and Decay Code and Library, RSIC Computer Code Collection, CCC-137

The difference of 70 curies minus 0.0014 curies is the activity of Ar-41 released during the year.

The other general groups of isotopes which are measured in the stack effluent consist of gross beta particulate and gross alpha particulate. Both of these releases are equivalently low like the I-131 releases. For example:

The total particulate activities released from the NTR stack during 1996 were measured to be:

Gross Beta Activity = 1.564 μCi

Gross Alpha Activity = 0.05631 μCi

The committed effective dose equivalent due to exposure to these releases can be demonstrated to be very low. For instance, the boundary doses calculated for the annual releases from the Vallecitos site, using the [1]COMPLY code are routinely found to be less than 1 mrem/yr., with a thyroid dose due to Iodine less than 1 mrem/yr. The on-site doses can likewise be demonstrated to be a very small fraction of the occupational dose limits.

11.1.1.2 Liquid Radioactive Sources

The only liquid radioactive source at the NTR is the primary coolant. Primary contributors to the radioactivity of the primary coolant are N-16, produced during reactor operation, and activated sodium from the aluminum primary coolant piping, also produced during reactor operation. The primary coolant is sampled annually to determine that there is no fuel leakage into the primary system, and to determine Na-24 content. The curie content of Sr-91, Sr-91, and Na-24 measured during the annual sample, are typically less than 1μCi/ml. The primary coolant system is vented to a holdup tank prior to startup, but the amount vented is small enough that the water in the holdup tank evaporates, and the tank does not fill. Dose rate measurements of the holdup tank indicate that no long-lived radioactive material accumulates in the tank. The only liquid radioactive waste generated is a result of the annual sampling, approximately one liter. This waste is placed in tanks with other laboratory generated liquid radioactive waste and subsequently disposed of in accordance with approved site practices and procedures. No liquid radioactive waste is released directly to the unrestricted environment.

11.1.1.3 Solid Radioactive Sources

Solid Radioactive Sources at the NTR consist of the reactor itself during operation, the reactor fuel, a Radium-Beryllium neutron source used for pre-startup instrument checks, an ion exchange demineralizer and filter system for the primary coolant system, spent power reactor fuel rod sections which are received from the hot cell facilities, neutrographed as part of a NDE process and returned to the hot cell facilities, activated experiments, and neutrographed parts and fixtures

[1] COMPLY code, A Guide for Determining Compliance with the Clean Air Act Standards for Radionuclide Emissions from NRC-Licensed and Non-DOE Federal Facilities. EPA 520/1-89-002, October 1989

which are exposed to the neutron beam or irradiated by placement in the horizontal cavity, byproduct material in experiments, instrument check sources, and solid radioactive waste. The following tables list appropriate parameters for some of the above sources which are located at the NTR.

Standard, Check and Startup Sources at the Nuclear Test Reactor							
Source	Isotopic Content	Radiation Type and Energy	Activity	Neutron Characteristic	Location	Form	Sealed/Unsealed
1	Cl-36	β - 0.709 Mev	2μCi	N/A	Victoreen, North Rm Mezzanine	solid	sealed
2	Cl-36	β - 0.709 Mev	2μCi	N/A	Victoreen, South Cell	solid	sealed
3	Cl-36	β - 0.709 Mev	2μCi	N/A	Victoreen, Rx Cell	solid	sealed
4	Cl-36	β - 0.709 Mev	2μCi	N/A	Victoreen, Rx Cell	solid	sealed
5	Cl-36	β - 0.709 Mev	μCi	N/A	Victoreen, Control Room	solid	sealed
6	Cl-36	β - 0.709 Mev	2μCi	N/A	Victoreen, North Rm MSM	solid	sealed
7	Cs-137	β - 0.514 Mev γ - 0.662 Mev	8μCi	N/A	CAM Cs7A2	solid	sealed
8	Cs-137	β - 0.514 Mev γ - 0.662 Mev	8μCi	N/A	Stack Monitor - Setup Room	solid	sealed
9	Cl-36	β - 0.709 Mev	0.02μCi	N/A	Stack Monitor - Setup Room	solid	sealed
10	Co-60	β - 0.318 Mev γ - 1.33, 1.17 Mev	40μCi	N/A	Stack Monitor - Setup Room	solid	sealed
11	Radium-Beryllium	Neutrons	0.2 Ci	1 E6 n/sec	Graphite Pack	solid	sealed
12	Sr-90	β - 0.546 Mev	2.24mCi	N/A	Control Room	solid	sealed

[redacted]

Solid radioactive waste consists mainly of latex gloves, shoe covers, masslin and other cleaning materials and swipe materials used to determine smearable contamination during routine and non-routine surveys. Lessor contributors include the clean-up resins and small pieces of equipment or tools that cannot be decontaminated economically. Occasionally, there may be a large piece of equipment or volume of construction material that requires disposal. The quantity of solid waste generated by NTR activities is very small, estimated to be one to three cubic feet annually with the radioactive content measured in millicuries. This waste is collected, classified, handled, compacted, packaged and transported for disposal by another component on the site, under separate State and NRC license regulatory oversight. All radioactive waste is handled and disposed of in accordance with written standards, procedures, and checksheets which assure compliance with the applicable sections of 10 CFR Parts 20 and 71, California Code of Regulations for byproduct material, and Department of Transportation regulations (49 CFR) for transporting radioactive material.

Solid radioactive sources in the form of experiments vary depending of the nature of the material and the experiment to be performed. R-33 license Section 2.B.(3)(1) allows a maximum of 2000 curies of contained byproduct material, for use at the facility. Experiments are reviewed in accordance with Section 10.3 of this SAR.

Typical radiation levels for occupied or accessible areas of the facilities under license R-33 are shown in Section 4.3 of this SAR. The radiation shielding of the facility is also described in Section 4.3. Based on the last five years of dosimetry for personnel at the facility, the estimated maximum annual dose is estimated to be 2.5 Rem, with the cumulative annual doses to the workers estimated to be a maximum of 4.5 Rem. This is well within the 10 CFR 20 requirements for occupational exposure. Exposure to individual members of the public as a result of NTR operations is essentially zero due to the existence of a restricted area, a controlled area, and the distance to the site boundary.

11.1.2 Radiation Protection Program

The radiation protection program for the site, including the NTR, is the responsibility of the Regulatory Compliance (RC) organization. The Manager, Regulatory Compliance, reports to the site manager, Manager, Vallecitos and Morris Operations (V&MO), independent of the Manager, NTR as shown in Figures 11-1 and 12-1. Additional non-radiological Environmental Health and Safety (EMS) support is available through the GE Nuclear Energy EHS organization.

The staffing level for the RC organization is dependent on the level of activity at the site as a whole. Minimum staffing for the RC activities for the NTR are health physics monitoring and Nuclear Safety Engineering oversight. The Manager, RC, shall have at least a bachelor's degree in engineering or one of the sciences with five years experience in assignments involving radiation protection. Technical personnel within the RC component shall have at least an associate degree or equivalent technical experience in the nuclear industry, and sufficient professional experience to provide authoritative and competent discharge of assigned responsibilities. Radiation monitoring technicians are trained and qualified in accordance with a comprehensive VNC certification program.

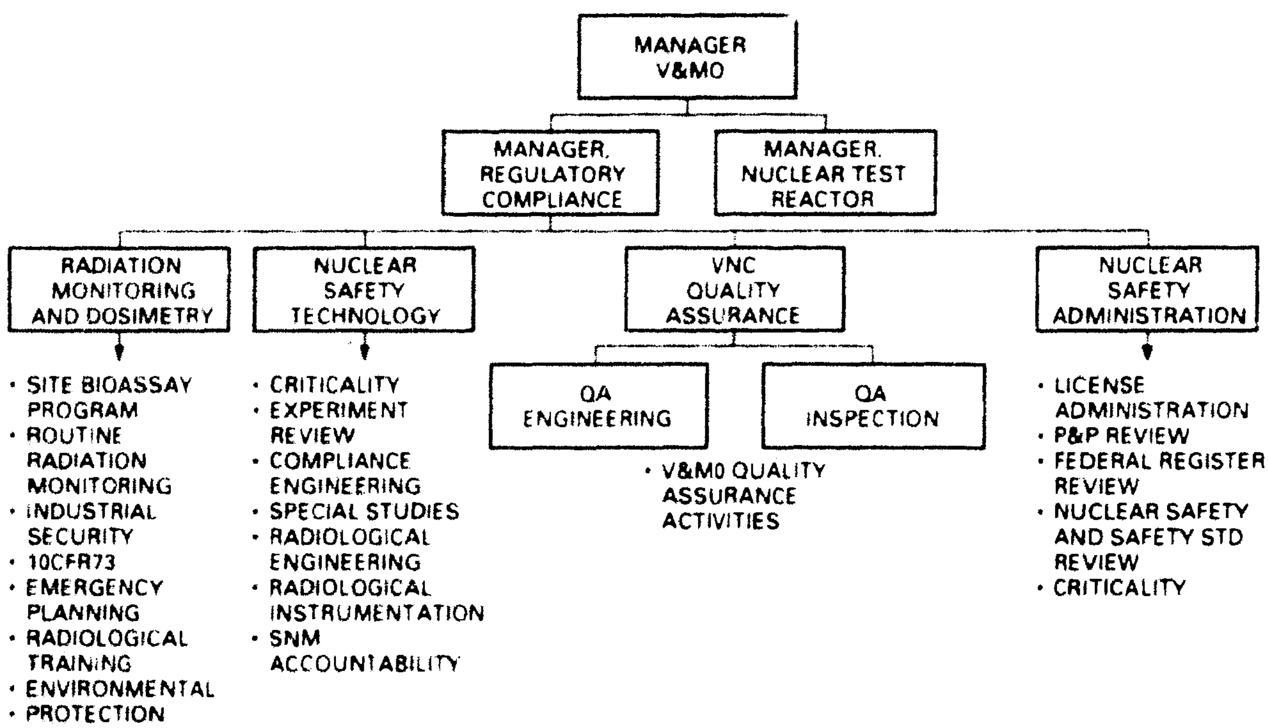

Figure 11-1. NTR Organization

Vallecitos Safety Standards (VSS) set forth the policy, responsibilities and charter of the RC organization, including the Nuclear Safety function. Nuclear Safety Procedures (NSP) implement the radiation protection requirements of the VSS's. NTR Standard Operating Procedures (SOP) specify the application and implementation of the VSS and NSP as they pertain to the NTR. The NTR SOP 9.1 describes the Safety Responsibility and Authority for the NTR by the various site organizations. The use of Change Authorizations (CA) and Engineering Releases (ER) for NTR activities provide documentation of changes and work and includes the determination whether the proposed change involves an unreviewed safety question as defined in Section 50.59 of 10 CFR Part 50. All of the above documents require RC review and approval.

All VSS, NSP, and Site Emergency Procedure documents are controlled by VSS 25.1. The NTR SOP documents are controlled by NTR SOP 9.2. Both of these documents provide assurance that procedures, including changes, relative to the radiation protection program are reviewed for adequacy, approved by authorized personnel, and distributed to the applicable staff.

Radiation safety training for VNC employees is per VSS 8.1, "Radiation Training for VNC Employees", which assigns primary responsibility for determining the appropriate training to the Area Manager. It also describes various radiation safety courses and includes a responsibility Matrix to assure training is conducted, courses are developed or changed, and that documentation of the training is maintained. Radiation Monitors are trained and certified in accordance with a comprehensive Health Physics program developed by VNC covering all site operations, including those at the NTR. NTR personnel receive radiological safety training per the Reactor Operator Requalification program described in NTR SOP 9.14. In addition, they receive annual Radiation Safety Refresher Training developed by the Nuclear Safety function for site radiation workers and personnel assigned to site emergency teams. Visitors to the site are trained per the requirements of VSS 8.2, "Radiation Training for Visitors", which provides the same guidance and requirements for training as VSS 8.1, as applicable to visitors.

The Vallecitos Technological Safety Council (VTSC) is the review and audit committee associated with the activities of the site as a whole and the NTR. It is discussed in Chapter 12.

Radiation safety audits of site activities are conducted per the requirements of NSP 6000, "Requirements for Reviewing Nuclear Safety Programs At Vallecitos Nuclear Center" and NSP 6100, "Nuclear Safety Reviews". The scope of the audits/reviews is sufficient to ensure that a nuclear safety program has been developed and documented for those activities being reviewed, that the program is being implemented to assess the effectiveness of the program, to identify deficiencies, and to verify corrective action has occurred where required. The audits are scheduled to cover audited programs and activities over a two-year period. The NTR reviews are typically done four times per year covering various aspects of Technical Specifications and License Conditions. The reviews are conducted by RC staff who are independent of the NTR organization and who have experience or training commensurate with the scope, complexity, or special nature of the activities to be reviewed. The written reports resulting from the reviews sent to the Facility Manager and/or Manager, RC, as appropriate. Follow-up action is performed by the reviewer and tracked to completion, to assure corrective action is accomplished.

The radiation protection program is reviewed each year in a report to the site manager. The VTSC reviews the report, annually and makes a determination as to the effectiveness of the program. The VTSC also receives and reviews incident investigation reports and countable event reports and uses all the information to implement program improvement and to ensure root causes are determined and effective corrective action is taken.

Personnel working in posted radiation areas are authorized for work in such areas by a Radiation Work Permit (RWP) or are specifically exempted from the requirements for an RWP, either by VSS 5.5, "Radiation Work Permit", or by the Area Manager or designated alternate responsible for the posted radiation area. Prior to issuing exemptions, the Area Manager or designated alternate will ensure that the exempted individual has had adequate radiation safety training for the particular job and area in which the work will be done. VSS 5.5 allows an Area Manger to exclude certain posted radiation areas under specified radiological conditions. Also, operations personnel and lab personnel working in their normally assigned work areas will not be required to work under an RWP, however, the Area Manager is responsible to ensure such personnel are adequately trained in radiation safety for those particular assigned work areas.

Radiation program safety records are generated per various safety standards, nuclear safety procedures and SOP's. VSS 5.9, "Retention of Safety-Related Documents" specifies that records shall be kept indefinitely, be accurate, legible, and permanent. It requires components generating the records to periodically survey their records to assure accuracy and uniformity. These records are required to be stored in a safe location and be easily retrievable. Radiation program safety records are used to develop trends, inform management, develop ALARA actions, and reporting to regulatory agencies. Radiation safety records include, employee exposure records, visitor and contractor exposure records, training records, medical records, radiation monitoring records, RWP's and VTSC meeting minutes.

11.1.3 ALARA Program

The radiation program at NTR includes a commitment to maintain radiation exposure as low as reasonably achievable, ALARA. The Manager V&MO, who has overall responsibility for the NTR license, and the Manager, RC, who has responsibility for radiological safety, are fully committed to the ALARA principle. VSS 5.8 and 5.8.1 describe site-wide ALARA requirements. NTR SOP. 7.1, 'Radiation Protection Program", Section 5 describes the NTR ALARA program.

The Manager, NTR, is fully committed to the ALARA principle. The implementation of the site-wide ALARA program, with additional NTR specific elements in NTR SOP 7.1, Section 5 comprise the NTR ALARA program.

During the first Quarter of each calendar year, the Manager, NTR, reviews the NTR ALARA program and prior year accomplishments and establish goals for the current year. All licensed NTR operators review each radiation exposure report. The Manager, NTR, also reviews radiation exposure records for all NTR personnel. Any unusual exposure is discussed and a probable cause determined. Individual workers, both NTR, and non-NTR personnel, are

adequately trained for the job and periodically reminded of ALARA principles. ALARA is considered when facility changes are made, via the Change Authorization (CA) process, when new or changed experiments are reviewed, and when major maintenance is planned.

11.1.4 Radiation Monitoring and Surveying

Radiation monitoring and surveying is accomplished at the NTR via the following methods:

1. Monitoring and Surveying routines by a VNC Radiation Monitoring Technician (RMT)/Health Physicist (HP)
2. Special monitoring and surveying by RMT/ HP
3. Fixed air sampling system
4. Stack Monitoring system
5. Continuous air monitor (CAM)
6. Remote area monitors (RAMs)
7. SOP by NTR operations personnel
8. Personal dosimeters
9. Sampling and counting of Industrial Wastewater prior to release.

The following summarizes the radiation monitoring equipment at the NTR:

Description	Location	Function
Portable survey instruments	Control Room North Room Setup Room Hallway outside Control Room	Dose rate (radiation fields); β, γ, n Contamination; α, β, γ
Continuous Air Monitor (CAM)	Control Room	Reactor Cell Air monitor
Hand and Foot Counter	Hallway near building exit	Contamination detector

Description	Location	Function
Remote Area Monitors (RAMs)	**Readout:** Reactor panel, control room **Detectors:** North Room (adj to CHRIS) South Cell Reactor Cell Control Room North Room (MSM)	Radiation field detection and alarm
Fixed Air Filters	Control Room South Cell Reactor Cell (2) North Room (3)	Airborne contamination detection
Stack Monitor System	Setup Room	Stack particulate and noble gas Monitoring and Alarm
Personal Dosimetry	MTR Personnel	Exposure monitoring

Records of monitoring and surveys conducted by RMT/HPs are maintained, reviewed and archived by independent RC personnel. Dosimetry and fixed air filter records are also maintained, reviewed and archived by RC personnel. Stack monitoring records and CAM records are maintained and archived by NTR personnel and reviewed by RC.

The policy for calibration of radiation protection instruments is provided by VSS 5.6, "Control of Radiation Monitoring Instrumentation". Nuclear Safety Procedures and Facilities Maintenance Procedures implement VSS 5.6. Calibration of radiation protection instruments used at the NTR is required upon initial acquisition, after major maintenance, and at least annually. Most radiation monitoring instruments are calibrated on-site by VNC Instrument Maintenance. Calibration sources used for calibration are traceable to NIST standards. Radiation monitoring instruments are controlled, and timely calibration is assured by RC maintaining a database system with inputs from the RMT/HP assigned to the NTR.

Monitoring and Surveying routines by a VNC RMT/HP for the NTR are directed by and performed in accordance with NSP 3550, "Building 105/NTR Work Routines". These routines provide reasonable assurance that radiation exposures to the public and workers or material releases can be detected. The routine frequency is based on potential for changes to affect the radiological situation, and run from daily routines to weekly, monthly, semi-annually, annually, and every five years.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control is achieved at the NTR by shielding, the ventilation system, security, entry control devices, an active ALARA program, the radiation protection program, environmental monitoring, equipment and materials, and through the VNC dosimetry program.

Shielding and typical radiation levels for occupied or accessible areas of the NTR facility are discussed in Section 4.3. Reactor cell ventilation is discussed in Chapter 9. The NTR Security Plan is discussed in Chapter 21. Entry control devices, consisting of alarms, locks, and interlocks are described in Chapter 6.

The ALARA program, the radiation protection program, and the environmental monitoring program, are described in Sections 11.1.3, 11.1.2 and 11.1.7 respectively, of this Chapter.

Equipment and materials used in radiation exposure control consist primarily of protective clothing and respiratory protection equipment. NTR SOP 7.6, "Protective Clothing", provides procedures for type and application of protective clothing including eye protection in high radiation and/or radioactive materials areas. VSS 5.3.3, "Respiratory Protection", describes the VNC Respiratory Protection Program. This program ensures proper respiratory equipment is obtained and maintained. It requires evaluation of work environment respiratory hazards including the establishment of a program for air sampling and analysis sufficient to identify hazards, evaluate individual exposures, and permit proper selection of respiratory protective equipment in accordance with 10 CFR Part 20 requirements. The respiratory protection program also describes minimum qualification requirements including initial and periodic training, refitting, and medical clearance requirements.

Radiation exposure to NTR personnel varies somewhat from year to year, however, expected annual exposure, based on historical records, is from 1.5 to 2.5 Rem per person. Non-NTR personnel, working in the same building, again based on historical exposure records, are expected to receive < 100 mRem per year per person, due to radiation associated with the NTR. Personnel providing service for the NTR, are expected to have a total annual exposure, from all site sources of < 1.0 Rem, with an estimated < 30% of that exposure attributable to NTR sources of exposure.

VSS 5.2, "Radiation Exposure Limits", states that based on dosimetry and survey records, it is unlikely that any individual would receive, in one year, and intake in excess of 10 percent of on ALI(s) of Table 1, Columns 1 and 2 of Appendix B to 10CFR20. Therefore, committed effective dose equivalent (CEDE) is not typically added to external dose for determination of total effective dose equivalent (TEDE). The philosophy at VNC and the NTR is to prevent, where possible, and when ALARA, the intake of radioactive materials during normal work in controlled areas. Air activity is controlled by the use of ventilation systems and contamination control. Intakes of radioactive materials is limited by the use of respiratory protection devices, control of access and limitation on the time of exposure when other controls are impractical. Whole body counts are done routinely, based on the Manager, NTR's input, to confirm the lack of intake.

Exposure limits and controls for occupational radiation exposure for adults (18 years of age or older), and minors, (no occupation exposure), are also described in VSS 5.2. Normal adult exposures are limited per the 10CFR20 requirements with an additional VNC limit for TEDE of 3.5 Rem per year. VSS 5.2.3, "Occupational Exposure of Embryo/Fetus", limits the exposure to the Embryo/Fetus of declared pregnant workers in accordance with the 10CFR20 requirements.

NTR SOP 7.5, "Radiation Exposure and Control", describes the exposure and dosimetry requirements for NTR Operations personnel. Exposure control limits are given, over the full range of operations, including normal operations, emergency conditions and planned special exposures. Administrative dose action levels are provided in the SOP of 750 mRem per quarter and 3.5 mRem per year, which require management approval to exceed.

Dosimetry used by NTR Operations personnel is also described in NTR SOP 7.5. Routine dosimetry includes a beta-gamma film dosimeter that is changed monthly, a neutron albedo dosimeter, changed monthly or quarterly depending on estimated neutron exposure, and a self-reading pocket dosimeters (SRPD) used to estimate exposure between monthly film badge processing. Special use dosimeters such as TLD finger rings, high-range SRPD, and electronic alarming dosimeters are prescribed for extremity exposure and high dose rate exposure per the SOP. Dosimetry records are kept indefinitely by RC for film badge, neutron badge and TLD finger ring exposures.

11.1.6 Contamination Control

Contamination control at the NTR is accomplished through some of the elements of the Radiation Protection Program discussed in the previous sections. These elements include the ALARA program, routine surveys and monitoring, the dosimetry program including evaluation and testing for internal depositions, the use of anti-contamination clothing, training programs for staff and visitors, and survey records. In addition, access control, area posting, and the use of RWP's are integral parts of contamination control at the NTR. VNC NSP's 3000 Series, "Radiation Surveying", and 4000 Series, "Air Sampling", are the bases for the contamination control program for the site. NTR SOP Section 7, "Radiological Safety", is the bases for contamination control at the NTR.

11.1.7 Environmental Monitoring

The site environmental surveillance program is described in NSP 9000, "Environmental Protection". The primary purpose of the environmental surveillance program is to obtain information essential to assessing and controlling the exposure of the neighboring population to industrial chemicals, radiation and/or radioactive materials. Secondary objectives include identifying the sources of specific contaminants that might be released, predicting trends in pollutant levels, and improving public relations by showing that the operations at VNC are not adversely affecting the health and safety of the public and surrounding areas. This program is responsive to several site licenses and permits and to VSS 7.0, "Environmental Protection".

At VNC the overall environmental program is separated into two distinct categories: (1) effluent monitoring and (2) environmental surveillance. Effluent monitoring includes all air samplers and radiation measurement systems on the ventilation exhaust stacks, including the NTR stack, for the various facilities. It also includes the sampling and analysis of water in the site retention basins prior to release. This program provides measurements of the amount of radioactivity that is released to the environment in gaseous effluents and provides quantitative data on the chemical and radioactive (mostly natural background) concentrations present in liquid effluents. These data are used to evaluate off-normal releases, should they occur, and to verify that the site release limits, per VSS 7.2, "Effluent Control", are not exceeded.

Environmental surveillance covers all measurements and observations made of the environment on and adjacent to the site. This includes environmental air samplers and TLD stations; the sampling of water, vegetation, soil and stream bottom sediment; and the resulting radiological and chemical analysis. This program provides assurance that there are no deleterious impacts on the environment from operations conducted at the site.

A complete description of the current VNC environmental program is contained in the Nuclear Safety Manual, Volume II (Environmental Monitoring). This manual is updated periodically as the environmental needs/requirements change. NEDO-12534 and NEDO-12731 (Statistical Evaluation of the Vallecitos Nuclear Center Environmental Sampling Program 1965-1973 and 1973-1979 are a compilation of the radiological data collected at the environmental sampling locations. The compiled data were analyzed statistically to obtain annual average background statistical levels. These results were used to provide a basis for the selection of the action levels against which to test the results obtained in subsequent sample analyses. If a sample for a particular group of samples is determined to be higher than the action level for that group, specific action will be undertaken to determine the possible sources of the activity.

Emergency environmental monitoring is described in the emergency implementing procedures contained in NSP 8550, "Implementing Procedure – Environmental Emergency Team".

The Specialist assigned to environmental protection is responsible to assure the requirements of the environmental protection program are met within the time frames established. This includes:

1. sample collection (method and frequency) and analysis (technique and sensitivity);

2. preparing required quarterly and annual summary reports;

3. assuring the proper installation, operation, and maintenance of environmental monitoring equipment.

The Specialist assigned to environmental protection has been granted the necessary authority by management to meet these responsibilities.

Regulatory Compliance (RC) is responsible for reviewing the environmental protection program for adequacy and for recommending changes as necessary. Further, RC prescribes equipment in

support of the environmental protection program and shall review periodically the activities of the Specialist assigned to environmental protection.

11.2 RADIOACTIVE WASTE MANAGEMENT

As stated in Sections 11.1.1.2 and 11.1.1.3 above, radioactive waste generated as a result of NTR activities is minimal. The management of radioactive waste is accomplished as a site-wide function with NTR SOP 9.7, "Radwaste Handling", as the implementing procedure for the NTR.

11.2.1 Radioactive Waste Management Program

VSS 7.3, "Radioactive Waste Handling" is the Standard for the Radioactive Waste Management Program at VNC. This Standard sets forth the management policy for the handling of low-level radioactive waste (radwaste) materials generated at VNC, including the NTR, and establishes a radwaste handling program designed to meet the objectives of this policy, and defines the responsibilities for carrying out the program.

VNC activities involved in the processing, packaging, transfer, receiving, interim storage, and shipment of radwastes shall be in compliance with applicable: (a) government regulatory requirements contained in CFR Title 10 and CFR Title 49; (b) requirements of VNC Safety Standards; (c) license and other special requirements of VNC and waste disposal sites to which the radwastes are shipped; (d) regulatory and license requirements of the State of California; (e) requirements of NRC IE Bulletin No. 79-19, "Packaging of Low Level Radioactive Waste for Transport and Burial"; and (f) requirements of 10CFR61, "Licensing Requirements for the Land Disposal of Radioactive Waste".

Low-level radwastes generated at the VNC site are ultimately packaged and transferred to licensed commercial radwaste disposal sites. Such radwastes will be reduced in volume to the maximum extent practicable prior to final packaging for disposal.

The responsibilities for the conduct of the Radioactive Waste Management Program falls under four primary categories: (1) Remote Handling Operation (RHO); (2) Regulatory Compliance; (3) Area Managers (the Manager, NTR, is the Area Manager for the NTR); and (4) personnel performing radwaste activities.

The RHO is responsible for radwaste inspection and the final preparation and shipment of radwaste to commercial volume reduction facilities or to commercial radwaste disposal sites. The Manager, RHO, is the author of the Safety Standard and is responsible for annual review and update of it.

Regulatory Compliance (RC) provides license and regulatory interpretations, radiation monitoring and surveying associated with radwaste activities and assistance, review, and approval of procedures generated by Area Managers. RC also performs periodic reviews and audits of site-wide radwaste activities.

Area Managers assure compliance with the Safety Standard in their area of responsibility and generate and update the necessary instructions and procedures for radwaste associated activities.

Personnel performing radwaste activities are responsible for doing radwaste work in compliance with all applicable regulations, Safety Standards, procedures and instruction, and for reducing the amount of radwaste generated as much as possible.

Personnel involved in radwaste generating or handling activities at VNC are trained and periodically retrained in these activities as appropriate for the specific tasks they perform. Instructions covering applicable DOT/NRC regulations, VNC radioactive material licenses, waste contractors' licenses (shipping and burial), and VNC Safety Standard requirements are provided to all personnel assigned to package radwaste for ultimate shipment off-site.

Records and checklists associated with radwaste activities are kept indefinitely by RHO and Area Managers. Such records consist of on-site transfers of radwaste to RHO, shipment to off-site disposal facilities, packaging checklists, and personnel training records. These records are reviewed by RC as part of the overall periodic radwaste program review.

11.2.2 Radioactive Waste Controls

NTR SOP 9.7 is the implementing procedure required by VSS 7.3 for radwaste activities at the NTR. This procedure provides radwaste controls for the NTR. It defines Radioactive Waste as: "Any material in which radioactivity is distributed or the surfaces of which are contaminated with radioactive material to levels that prevent release for unrestricted use or which is potentially contaminated and cannot be shown to be less than these levels and which has no further functional or monetary value to the user or owner." NTR SOP 9.7 describes the criteria, method and responsibilities to be used at the NTR for collection, interim storage, identification, characterization, and transfer of low-level radioactive waste to the site inspection/packaging area.

As described in Section 11.1.1.2, the only liquid radioactive waste generated is as a result of the annual sampling, approximately one liter. This waste is placed in tanks with other laboratory generated liquid radioactive waste and subsequently disposed of in accordance with approved site practices and procedures. No liquid radioactive waste is released directly to the unrestricted environment.

As described in Section 11.1.1.3, the quantity of solid waste generated by NTR activities is very small, estimated to be one to three cubic feet annually with the radioactive content measured in millicuries. VSS 7.3, NTR SOP 9.7, and periodic training emphasize radwaste reduction techniques, including planning, decontamination, use of reusable vs. disposable materials, depackaging of supplies and equipment prior to transfer into a radioactive materials area (RMA), and dedication of appropriate tools and equipment to RMA's for reuse as needed. The site also uses commercial facilities, where cost effective, to perform radwaste volume reduction and recycling of contaminated materials, e.g., metal melt technology.

11.2.3 Release of Radioactive Waste

As stated earlier in this Chapter, liquid waste is not released to the environment. Industrial waste water, single pass, non-contact cooling water is tested for radiological constituents as well as other potentially polluting parameters in accordance with a NPDES permit prior to release to the environment.

Release of routine gaseous effluents can be limited to Ar-41, which is generated by neutron activation of air. Airborne radioactive waste exiting through the NTR stack is well within the Technical Specification and 10 CFR 20 requirements. Monitoring and alarms associated with the NTR stack have been discussed previously in this Chapter and in other Chapters of the SAR. The annual average dilution factor from the NTR stack to the site boundary based on historical meteorological data, and a stack flow rate of 3000 cfm equals approximately 20,000; that is the concentration at the site boundary of any release from the NTR stack, will not be greater than 1/20,000 of the concentration at the stack when averaged over one year.

Solid radioactive waste is handled in accordance with VSS 7.3 and shipped to licensed waste processors and/or disposal facilities. VSS 7.3 and implementing procedures assure that solid radwaste is characterized, handled, packaged, surveyed and shipped in accordance with all applicable DOT and NRC regulations.

12.0 CONDUCT OF OPERATIONS

12.1 ORGANIZATION

12.1.1 Structure

The establishment of functional levels and assignment of responsibilities is the perogative of the organization authorized to operate the reactor facility.

The Nuclear Test Reactor (NTR) facility organization and interrelationships are shown in Figure 12-1. This figure shows the relationship between the operating organization and the primary supporting organizations. The organization may be modified from time to time to reflect changes in programs and objectives.

The NTR facility has been organized and charged with the responsibility and authority to discharge those assigned responsibilities, so that decisions are communicated via the proper levels and with adequate technical advice. Functions performed by one level may be performed by personnel at a higher level, provided they meet the minimum qualifications (i.e., Reactor Operator's license, etc.). The Manager, Vallecitos and Morris Operations (V&MO) has overall responsibility for the reactor license. Reporting directly to the Manager, V&MO, is the Manager, NTR, who is the Facility Manager. He is responsible for the safe and efficient operation, maintenance, and repair of the facility. Operation of the reactor may be performed under the direction of a reactor Supervisor. Contributing in a major way to the operating organization, but not reporting to the Facility Manager, is the Regulatory Compliance organization (RC). Within this organization are specialists in nuclear safety, health physics, licensing, safeguards, security, and criticality. This organization also contains health physics monitoring personnel and quality control technicians.

Also available to the Facility Manager are many other highly specialized technical individuals on and off the Vallecitos Nuclear Center in the GE Nuclear Energy organization.

12.1.2 Responsibility

The responsibilities of selected NTR facility positions are as follows:

1. Manager, Vallecitos and Morris Operation (V&MO)

   The Manager, V&MO, has the overall responsibility for the NTR facility's license.

2. Manager, Nuclear Test Reactor (NTR)

The Manager, NTR, reports to the Manager, V&MO. He is the Facility Manager and has the overall responsibility for the safe, reliable, and efficient operation of the NTR.

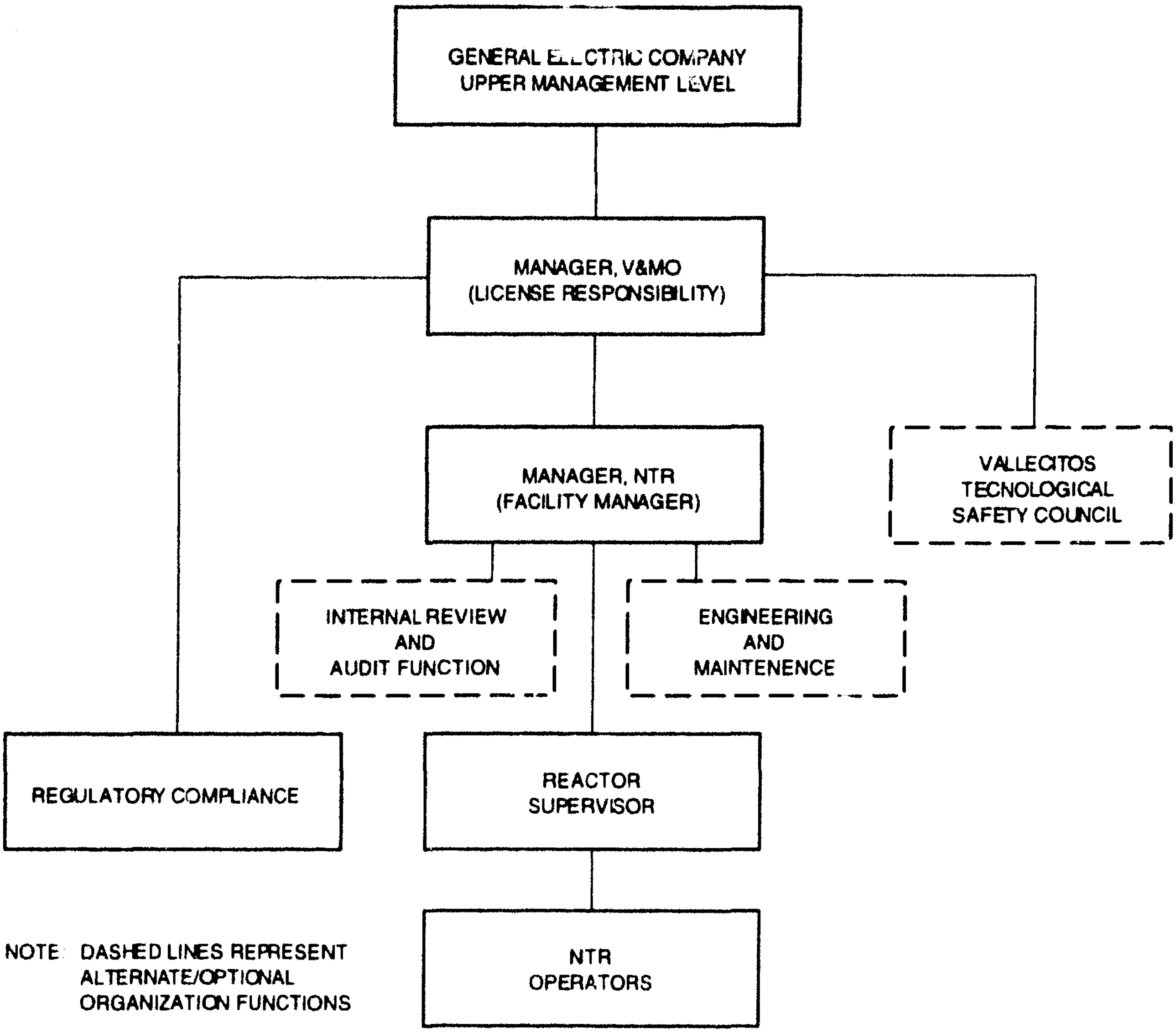

Figure 12-1. Organization Chart

He is responsible for maintaining a competent staff and an effective organization structure as measured by the overall safety performance. All changes to the facility or facility procedures and all new tests and experiments require the approval of the Manager, NTR, or his designated alternate as defined in the Technical Specifications. He is responsible for an adequate safety review and may utilize resources of other GE personnel (or outside consultants) not on his staff. When the positions exist, the Manager, NTR, directs the activities of the reactor supervisor, and the NTR engineer(s). He is also responsible for the development, maintenance, and implementation of written operating and maintenance procedures, the coordination of operation maintenance, repairs, modifications, the training and requalification of operating personnel and the safety of assigned and visitor personnel.

3. Reactor Supervisor

When the position exists, the Reactor Supervisor (shift supervisors, Operations Supervisor, etc.) reports to the Manager, NTR. He supervises the NTR operation in accordance with written operating procedures, administers planned work, and handles emergencies at the facility. He is responsible for the safety of personnel working at the facility, ensures the security of fissionable materials, regulates entry into radiation and restricted areas, and trains new personnel. In the absence of a separate Reactor Supervisor, the Manager, NTR, has these responsibilities.

4. NTR Engineer (engineering and maintenance personnel)

The NTR Engineer (specialist, etc.) may report to the Manager, NTR. He usually provides detailed direction and guidance in the installation and operation of experiment facilities and programs and overall plant maintenance, repairs, and modifications within the framework of operating procedures. In the absence of a separate Engineer, the Manager, NTR, has these responsibilities.

5. NTR Operators

The NTR operators consist of trainees, licensed Reactor Operators, and licensed Senior Reactor Operators, who operate the reactor and experiment facilities in accordance with the operating procedures under the supervision of the Manager, NTR, or the Operations Supervisor. Licensed Reactor Operators may direct the activities of trainees, and licensed Senior Reactor Operators may direct the activities of licensed Reactor Operators and trainees, in accordance with the operating procedures.

12.1.3 Staffing

The minimum staffing of the reactor when it is not secured shall be as follows:

1. A licensed Reactor Operator or Senior Reactor Operator shall be in the control room with access to the nuclear console.

2. A second person who is familiar with NTR Emergency Standard Operating Procedures shall be at the Site.

3. A Senior Reactor Operator shall be present at the NTR facility or readily available on call at all times the reactor is not secured.

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

1. The first startup each day.

2. The recovery from an unplanned or unscheduled shutdown.

3. Reactor fuel loading or reactor fuel relocation.

4. Manual poison sheet changes.

Additionally, a Licensed Reactor Operator or Senior Reactor Operator shall be present at the controls at all times during the operation of the facility.

Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of the reactor, shall be manipulated only with the knowledge and consent of a Licensed Reactor Operator or Senior Reactor Operator present at the controls.

12.1.4 Selection and Training of Personnel

12.1.4.1 Qualifications

Operations personnel shall have that combination of academic training, experience, health, and skills commensurate with their level of responsibility which provides reasonable assurance that decisions and actions during normal and abnormal conditions will be such that the plant is operated safety and efficiently in accordance with NRC license requirements and rules and regulations. Minimum qualifications shall include the following:

1. Manager, NTR

   At the time of appointment to the active position, the Manager, NTR, (Facility Manager) shall have at least 6 years of nuclear experience. Additionally, he shall have a baccalaureate or higher degree in an engineering or scientific field and have previous managerial experience. Equivalent education or experience may be substituted for a degree. The degree may fulfill 4 of the 6 years of nuclear experience required on a one-for-one basis. He shall have or immediately pursue steps for obtaining an NTR Senior Reactor Operator license.

2. Reactor Supervisor

At the time of appointment to the active position, the Reactor Supervisor, when utilized, shall have at least 3 years of nuclear-related experience. He shall have an NTR Senior Reactor Operator license or shall immediately pursue steps for obtaining an NTR Senior Reactor Operator license. A maximum of 2 years equivalent full-time academic training may be substituted for 2 of the 3 years of nuclear-related experience required.

3. NTR Engineer

At the time of appointment to the active position, the NTR Engineer shall have at least 1 year of nuclear-related experience. Additionally, he shall have a baccalaureate or higher degree. Equivalent education and experience may be substituted for a degree.

4. NTR Operators

a. Senior Reactor Operator

A Senior Reactor Operator shall have, as a minimum, the following qualifications, as determined by the Manager, NTR:

(1) A sufficient level of experience in NTR reactor operations, experiment setup and operation, and a high level of leadership.

(2) An NTR Senior Operator's license.

(3) Mature judgment and a capability for handling diverse problems under rapidly changing conditions.

(4) Reactor Operator qualifications.

b. Reactor Operator

A Reactor Operator shall meet, as a minimum, the following qualifications, as determined by the Manager, NTR:

(1) A high school diploma or equivalent, with a high degree of mechanical dexterity.

(2) NTR Operator's license.

(3) Sufficient training or experience in related nuclear fields.

c. Trainee

A Trainee shall meet, as a minimum, the following qualifications, as determined by the Manager, NTR:

(1) A high school diploma or equivalent, with a high degree of mechanical dexterity.

(2) Sufficient applicable training or experience.

Senior Reactor Operator and Reactor Operator candidates are required to obtain licenses issued by the U.S. Nuclear Regulatory Commission in accordance with the provisions of 10CFR55..

12.1.4.2 Initial Training

At the time of appointment to the position, all personnel listed on Figure 12-1 shall receive a briefing of the general operational, emergency and regulatory aspects of the NTR.

Initial training for the Manager, NTR, and the reactor operators shall be sufficient for the individuals to obtain a Reactor Operator or Senior Reactor Operator license issued by the NRC. Topics shall include the following:

Fundamentals of reactor theory and operation,
Facility design and operating characteristics,
Instrumentation and control,
Procedures and Technical Specifications,
Radioactive material handling and exposure control
Code of Federal Regulations,
Emergency response,
Security.

12.1.4.3 Operator Requalification

Licensed operators participate in a comprehensive Operator Requalification Program.

The program is designed to maintain the competence of the NTR operating personnel to handle abnormal events and to comply with the requirements and intent of 10CFR55.59. The NTR Requalification Plan approved by the NRC, is described in and is administratively controlled by the NTR Standard Operating Procedures.

12.1.4.4 Other Training

Personnel working in controlled areas are badged and receive basic radiation safety instruction when they receive their dosimeters. Work done at the reactor is performed under a Radioactive Work Permit (RWP) when additional instructions specific to the local work area are made in accordance with 10CFR19. The Manager, NTR, may exempt workers from the RWP requirement when the lack of a hazard exists. The exemption is documented in writing.

Experimental installation, movement and testing is normally conducted by NTR operations personnel. There is relatively little entrance into controlled areas by other workers.

12.1.5 Radiation Safety

Radiation protection is discussed in Chapter 11. Radiation protection at NTR is accomplished by highly trained and competent operators and independent radiation protection staff, appropriate dosimetry, effective monitoring, instrumentation, and written procedures.

The independent radiation safety function is performed by trained personnel in the Regulatory Compliance (RC) group. The Manager, RC, reports directly to the Manager, Vallecitos and Morris Operations, independent of the Manager, NTR.

12.2 REVIEW AND AUDIT ACTIVITIES

An effective independent review and audit process at NTR assures the following.

- Operations comply with the facility license, the Code of Federal Regulations, and established procedures;
- The operating organization discharges its responsibilities consistent with good safety practices; and
- The records accurately and adequately reflect actual operation.

12.2.1 Composition and Qualifications

Review and audit of the NTR is primarily conducted by members of the Regulatory Compliance organization. This organization is composed of highly specialized professionals in the areas of nuclear engineering, radiological engineering, health physics, safeguards, security and licensing. Most individuals have had nuclear industry experience prior to their appointment. These individuals are fully applied to nuclear industry work and support other nuclear facilities on site when not supporting the NTR.

Additionally, Vallecitos and Morris Operations currently has a Vallecitos Technological Safety Council (VTSC) composed of senior personnel from various site activities and technical disciplines. Membership is by appointment from the Manager, V&MO. The VTSC is

responsible to the Manager, V&MO, and is independent of both the Regulatory Compliance and NTR organizations.

12.2.2 Charter

The Regulatory Compliance organization assists operating groups in developing methods of implementing the regulations, licenses, permits and solutions to safety issues and performs independent reviews and audits. Reviews are performed as documents are submitted. Audits are performed quarterly. Comments and recommendations are made to the Facility Manager. Disputes are resolved by the Manager, V&MO.

The optional VTSC is a review body independent of all operating organizations. The VTSC has the authority to review: 1) reportable incident investigations; 2) unreviewed safety questions of facility changes; 3) operating standards, experiments, and receipt, possession, separation, use, processing and transfer of radioactive material; 4) proposed new criticality processes or methods of evaluation for operations at VNC; 5) new major facilities proposed for the VNC site; and 6) processes, operations and procedures which involve toxic, flammable, etc., materials. The VTSC shall evaluate the overall effectiveness and relevance of safety studies and Regulatory Compliance (RC) review activities as they collectively influence safety conditions at VNC. The VTSC provides expert advice and counsel, but it is not responsible for conducting routine inspections. The VTSC reports its deliberations and recommendations to the Manager, V&MO, with copies to Area Managers and affected operating managers. Affected operating managers reply to VTSC safety- or compliance-related recommendations in writing, addressed to the Chairman, VTSC. The VTSC maintains records of its safety- or compliance-related recommendations and follow-on action by operating components.

The VTSC meets quarterly unless there is no business to conduct. The council may meet as frequently as necessary. A quorum is 50% or more of its members, but VTSC recommendations are based on a majority vote of all members.

12.2.3 Review and Audit Functions

The Regulatory Compliance organization is responsible for reviewing the following:

- All proposed procedures required by the Technical Specifications and proposed changes to such procedures;
- Proposed types of experiments, facility modifications, and facility procedures; as described in this document;
- Proposed changes to the facility operating license, including Technical Specifications and revised bases;
- Violation of the Federal Regulations, Technical Specifications, facility license requirements, and internal procedures having safety significance;

- Unusual or abnormal occurrences which are reportable to the NRC, as required by the Federal Regulations or Technical Specifications;

- Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety; and

- Periodic audit of facility operation, maintenance, and administration, to include the following:

  a. The conformance of facility operation to the federal regulations, Technical Specifications, and facility license requirements.

  b. The results of all actions taken to correct deficiencies or increase effectiveness in facility equipment, structures, systems, or methods of operation that affect nuclear safety.

  c. The facility emergency procedures, security plan, requalification programs, and their implementing procedures.

The optional VTSC regularly reviews the following

- The results and actions resulting from all reportable incidents. Occurrences which are not reportable incidents shall be reviewed by the VTSC if, in their view, significant safety questions are involved.

- Proposed new facilities or changes to facilities which contain unreviewed safety questions.

- Change Authorizations when requested by operations management or the nuclear or industrial safety function.

Additionally, the VTSC has the authority to do the following:

- Consider and provide advice, as requested by management, on problems of nuclear safety, criticality control, and industrial safety as related to operations at the VNC site.

- Investigate problems of nuclear safety, criticality control and industrial safety.

- Evaluate the overall effectiveness and relevance of safety studies and Regulatory Compliance review activities as they collectively influence the safety conditions at the Vallecitos Nuclear Center.

- Review and recommend on "special topics" as requested by the nuclear or industrial safety function or operations management.

- Review any other matter which it conceives to be of safety importance.

12.3 PROCEDURES

12.3.1 Summary Description

The facility license, Technical Specifications, and Code of Federal Regulations establish the bounds within which the reactor must be operated. VNC Safety Standards as issued by the Manager, V&MO, and reviewed and accepted by the Facility Manager or designated alternate supplement the license and federal regulations to ensure further personnel and reactor safety.

In addition, Standard Operating Procedures (SOP) and Engineering Release (ER) have been established, as required, to delineate administrative and operational requirements to comply with NRC Regulations and the NTR License. Up-to-date copies of the SOPs and ERs, as applicable, are available to all personnel at the facility.

A Change Authorization (CA) Procedure has also been established to document and authorize all changes to the facility or facility procedures as they are described in this document.

12.3.2 Required Actions

In the event of an abnormal occurrence, action shall be taken to assure the safety of the plant and personnel and to take appropriate corrective measures. If required, the reactor shall be shut down. If the reactor is shut down because of an abnormal occurrence, the reactor operation shall not be resumed until the cause is determined and required corrective action is completed.

12.3.3 Vallecitos Nuclear Center Safety Standards

Criteria in the Vallecitos Safety Standards (VSS) have been established for protection against hazards arising from activities conducted under licenses issued by appropriate regulatory authorities and provide guidelines for complying with the several licenses and regulations governing the facility, activities, and materials at VNC. Many of the standards govern the general radiation protection practices for the Site.

12.3.4 Standard Operating Procedures

Standard Operating Procedures (SOPs) have been established to delineate administrative and operational requirements to comply with NRC Regulations and the NTR facility license. The SOPs are in place for the following activities, as required:

- Normal startup, operation, and shutdown of the reactor and all pertinent systems and components as specified by the Facility Manager or designated alternate involving nuclear safety of the facility;
- Defueling, refueling, and fuel transfer operations when required;
- Preventive or corrective maintenance which could have an effect on the safety of the reactor;
- Off-normal conditions relative to reactor safety for which an alarm is received;
- Response to abnormal reactivity changes;
- Surveillance, testing, and calibrations required by the Technical Specifications;
- Emergency conditions involving potential or actual release of radioactive materials;
- Radiation protection consistent with 10CFR20 requirements,
- Review and approval of changes to all required procedures; and
- Security, Operator requalification, emergency plan, and others, as required by the Facility Manager or his designated alternate.

The SOPs are approved by the Manager, NTR (Facility Manager), or his designated alternate. Independent review is in accordance with Subsection 12.2.

Minor changes to the SOPs may be made without independent review. These changes may be made by retyping the affected pages and making a normal distribution for revised SOPs. Minor changes are assigned a minor review number to document the change.

Temporary changes to an SOP may be made by an ER which would supersede the SOP for a time period. If a temporary change is to remain in effect longer than six months, revised SOP pages should be issued.

An SRO may authorize deviations from the SOPs during emergencies to prevent injury to personnel or damage to the facility. An SRO shall document the required emergency action in the log book and notify the Manager, NTR.

12.3.5 Engineering Release (ER)

An Engineering Release (ER) is issued, as required, to request work, establish temporary procedures or instructions, distribute information, document actions, and other items in an effort to ensure the safe, efficient operation of the NTR.

The ERs are written by NTR personnel and reviewed and approved in accordance with the SOP covering ERs. Independent review in accordance with Subsection 12.2 is required for those activities listed in Subsection 12.3.4.

12.3.6 Change Authorization (CA)

A Change Authorization (CA) is required for changes to the facility and changes to this document. The CA provides the documented description and safety evaluation required by 10 CFR, Section 50.59, and the review and approval of the change. A CA is required for changes, activities, or projects that are judged to involve significant safety considerations or a potential Technical Specification violation or "unreviewed safety question," warranting documented review and approval. A Change Authorization is also required for new types of experiments, or changes to types of experiments.

Change Authorizations involving experiments (experiment type approval, as discussed in Chapter 10) require the following as a minimum:

(1) All new types of experiments which could be postulated to affect reactivity or to result in unusual radiation exposure to personnel or an unusual release of radioactive materials, shall be reviewed for compliance with the facility license and the Technical Specifications.

(2) Changes to approved experiments shall receive appropriate review and approval.

(3) Approved experiments are implemented in accordance with written procedures (Standard Operating Procedures or ERs, as required by governing procedures).

Change Authorizations are administratively controlled by a Standard Operating Procedure.

The Change Authorization is reviewed independently by Regulatory Compliance to determine that the following criteria are satisfied:

- The proposed change can be made without prior NRC approval (10 CFR, Section 50.59).
- The change does not violate any license requirement or federal regulations.
- Special interim conditions which may exist during the period while the change is being made are analyzed to ensure that hazardous or unauthorized conditions do not exist during the modification or transition period.

More specific criteria and other review responsibilities are delineated in the Change Authorization Standard Operating Procedure.

Regulatory Compliance provides an independent review of the Change Authorization. These personnel may request or perform additional analyses to ensure the specified criteria are satisfied. Personnel in the RC will review the Change Authorization and:

- Recommend approval of the proposed change; or
- Recommend qualified approval of the proposed change; or
- Recommend disapproval of the proposed change.

The Manager, NTR, or his designated alternate has the responsibility of approving or disapproving proposed Change Authorizations.

12.4 REQUIRED ACTIONS

In the event that the true value of the reactor thermal power exceeds 190 kW, the reactor shall be shut down and secured immediately and notification made to the Manager, NTR, Manager Regulatory Compliance, and the Manager Vallecitos and Morris Operations. The NRC shall also be notified.

The reactor shall remain shut down and secured until authorized by the managers of NTR, RC and V&MO.

If routine or non-routine operation, maintenance, testing, or inspection reveals an unusual or unexpected result or situation which is potentially reportable, the individual noting the occurrence shall notify the Manager, NTR immediately. If the reactor is in operation, the condition or situation shall be returned to normal immediately or the reactor shut down. The Manager, NTR shall notify RC personnel. If the event is determined to be reportable, the NRC shall be notified in accordance with the SOPs and Federal Regulations.

12.5 REPORTS

Reports shall be submitted to the NRC as required by the applicable portions of Title 10 of the Federal Regulations, Parts 20, 40, 70, 71 and 73.

In addition, reports shall be submitted that describe the circumstances of any of the following events:

- Release of radioactivity from the site above allowed limits.

- 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 is 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.

- A violation of the Safety Limit.

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

- Permanent changes in the facility Management

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

An annual operating report shall be submitted to the NRC during the first quarter of each year. This report shall include the following:

- Changes to the NTR organization or personnel in the organization
- 10CFR50.59 changes to the facility
- 10CFR50.59 experiments performed
- 10CFR50.59 changes to the procedures
- A summary of major preventive or corrective maintenance performed
- A listing and explanation of scrams and unscheduled shutdowns
- A summary of radiation levels from monitoring stations
- NTR stack effluent release levels
- A summary of personnel radiation exposures.

12.6 RECORDS

In addition to those required by applicable government regulations, the following records shall be maintained, at a minimum, for the periods specified below. The records may be in the form of logs, data sheets, recorder charts, computer disk storage or other suitable forms. The NTR Standard Operating Procedure, "Reactor Records," containing specific retention requirement for records, serves as a guide. Records are maintained in file cabinets, binders computer disks or archive boxes.

Records required by 10CFR20.2103, 10CFR55, 10CFR30.51, 10CFR70.51, 10CFR50.71 and 10CFR40.61 are maintained in accordance with the Federal Regulations.

10CFR50.59 records are documented on a CA and retained for the life of the facility, (facility changes) or a minimum of 5 years (procedure changes and tests and experiment evaluations).

Records to be retained for the lifetime of the reactor facility include the following:

- Gaseous radioactive effluent released or discharged to the environment beyond the effective control of the licensee as monitored at or before the point of release or discharge.

- Off-site environmental monitoring surveys.

- Radiation exposure for all facility personnel monitored.

- Updated drawings, as required, of the facility and of facility changes.

Records to be retained for a period of at least 5 years or for the life of the component, whichever is smaller, include the following:

1. Normal reactor operation records, i.e., supporting documents, such as checklists and log sheets.

2. Unplanned or unscheduled shutdowns and scrams, including the reasons therefore.

3. Principal maintenance activities involving substitution or replacement of reactor equipment or components.

4. Occurrences reported to the NRC, as required by the Technical Specifications.

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

6. Experiments performed, including unusual events involved in their handling and performance.

7. Reviews performed for: a) changes made to procedures or equipment, and b) new tests and experiments not submitted for NRC approval pursuant to 10 CFR, Section 50.59.

8. Meetings and audit reports of the independent review and audit function.

9. Off-site radioactive shipments and receipts.

10. SOP revisions.

12.7 EMERGENCY PLANNING

In an emergency, resources are utilized from the entire site to protect the health and safety of the public and VNC employees. The emergency plan establishes the organizational and procedural steps to respond to emergencies, to evaluate the situation to collect hazardous data and to mitigate the consequences of the emergency.

An emergency is reported to Security who makes an announcement over a site-wide public address system. The Emergency Operations Coordinator (EOC) reports to an Emergency Control center (ECC). The EOC is responsible for the initiation and implementation of the emergency plan and procedures and for the coordination of all emergency response activities.

At the evacuation area for the affected area, an emergency team forms consisting of a Building Emergency Coordinator, radiation monitor technicians and a building reentry team. Other specialists such as Regulatory Compliance engineers, fire team, mechanics and electricians are immediately available.

An emergency is classified, an assessment is made and, with the concurrence of the EOC, corrective and protective actions taken.

The emergency plan is reviewed biennially and revised when necessary.

12.8 SECURITY PLANNING

A number of security measures are in effect at NTR to prevent unauthorized access to the facility, theft of materials and sabotage

Written procedures are available for incidents such as a bomb threat, breach of security, civil disorder, and a major power outage. The plan requires periodic testing of systems, recordkeeping and reports to the NRC. Authorized individuals may refer to the current Security Plan for more details.

12.9 QUALITY ASSURANCE

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.

12.9.1 Organization and Responsibilities

This section describes the organizational structure and functional responsibilities for the quality program (Figure 12-1).

1. NTR Operations

NTR Operations is responsible for operation of reactor and experiment systems in accordance with established Standard Operating Procedures. NTR operations is also responsible for those items in Section 12.9.2 as required.

2. Engineering

The NTR engineer (Manager NTR or SRO) or other personnel, as designated, is responsible for the performance of engineering on the NTR scram systems and safety-related systems. The engineer generates designs and design changes; prepares specifications, work instructions, and procedures; participates in design reviews; specifies which items in the scram systems and safety-related systems require quality assurance and the level of quality assurance required; and assures adequate proof of component and systems operability and other related engineering functions as required.

The engineer also evaluates system and structural performance and effects solutions, as appropriate, where operation is found to be inadequate.

3. Regulatory Compliance

Regulatory Compliance (RC) is organizationally independent of the reactor operating functions and has full authority and responsibility to identify, evaluate, and recommend solutions to quality and safety-related problems. RC performs an independent review of selected procedures; establishes overall safety and quality assurance programs, as required; acts as the primary interface between operating components and regulatory agencies; conducts audits, as required, to assure compliance with this plan, specifications, procedures, regulations, and company policy; and performs inspection services and conducts evaluations of nonconforming items, as required. The details of these responsibilities are defined in RC Procedures.

4. Drafting

Drafting is responsible for issuance of engineering definition documents, such as drawings and specifications, and changes to such documents.

5. Purchasing

Purchasing performs activities related to procurement of materials and services required from outside vendors in accordance with procedures issued by the Purchasing Function.

6. Transportation and Materials Distribution

The VNC Shipping and Receiving is responsible for receiving, shipping, and on-site movement of materials. The routine handling of materials is in accordance with procedures. Special handling instructions may be specified in implementing instructions issued by the organization requesting the service.

7. Instrument and Electrical Maintenance

The Instrument and Electrical Maintenance components perform installation, calibration, repair, and maintenance services on electrical and instrumentation systems for the NTR. Records of work performed and calibration standards traceability, as required, are maintained.

8. Shop Operations

Shop Operations (or an outside vendor) provides fabrication services as requested by the responsible engineer. This includes primarily machining and welding operations.

9. Mechanical Maintenance

Facilities Maintenance components perform installation, repair, and maintenance services on mechanical systems for the NTR. Records of work performed and calibration standards traceability, as required, are maintained.

12.9.2 Instructions, Procedures, and Specifications

Organizations responsible for work and/or performing work within the scope of this program are responsible for establishment and maintenance of documented systems and procedures for the performance of that work, unless provided for by NTR Operations or determined by NTR operations to be not required. Any changes of these documents are approved by the same function that authorized their issuance and use, unless otherwise specified within the document, or by governing Standard Operating Procedures.

Planning and/or implementing documents shall:

- Provide, when warranted, space for sign-off by the person who performs the work to show that he has followed the prescribed instructions.

- Call out essential controls and hold points, as required, which provide an independent assessment that the work was performed as prescribed and that the results meet specifications.

- Include, as necessary, special instructions for handling and transportation.

12.9.3 Design Control

1. Design Standards

The responsible engineer identifies in the design drawings and specifications, required codes and standards and practices that provide the basis for design methods, material evaluation and process controls.

2. Design Verification

Design verification is required for new systems or significant changes to existing systems for NTR safety-related items. This is accomplished by independent reviews (normally, RC review is sufficient), alternate calculations, or the execution of a test program. The verification is performed by individuals other than those who performed the original design. The normal method for documentation is the Change Authorization, which is discussed in Subsection 12.3

3. Engineering Change

Changes to engineering definition documents are implemented and recorded by means of the Engineering Change Notice (ECN). Field changes during installation, as determined by the responsible engineer, may be implemented by "redlining" the drawing or specification, provided the change is documented on an ECN and the change is evaluated by the same functions that approved the original prior to the operation of the component or system.

12.9.4 Procurement Control

1. Procurement Flow

Materials are ordered in accordance with the requirements of the engineering definition document, if applicable. Purchasing from outside vendors is performed by Purchasing in accordance with Purchasing procedures. Requests for Quotation (RFQs) and material to be purchased from outside vendors are documented on a Material Request form (MR). RC reviews MRs prior to submittal to Purchasing as required both for procurement or RFQ. Receiving inspection instructions, if required, are included on the requests.

2. Vendor Selection and Surveillance

Purchasing is responsible for soliciting quotes, negotiation of contracts, and procurement. Vendor evaluation from a technical standpoint is performed by the responsible engineer. Vendor quality capability evaluation, if required, is performed by RC. The quality of purchased materials is verified by supplier-furnished evidence, source inspection, receiving inspection, or a combination of these, as appropriate.

12.9.5 Document Control

Organizations performing work within the scope of this program generate documents such as Standard Operating Procedures, drawings, specifications, and work instructions. Procedures are established describing the document control system in each organization, as required. The document control system assures the proper review, approval, distribution, and control of documents and their revisions.

12.9.6 Material Control

Procedures are established, as required, to control the identification, handling, storage, shipping, cleaning and preservation of safety related material and equipment. The system provides measures to ensure the use of correct materials, to maintain traceability of components, and to clearly identify discrepant materials.

Storage areas are provided, if necessary, to shelter material from natural elements, and to protect material in special environments. Materials held in storage are properly identified, adequately protected to preclude damage, and segregated to prevent the use of incorrect or defective parts.

12.9.7 Process Controls

When required by engineering specifications or planning documents, production processes are accomplished under controlled conditions in accordance with applicable codes, standards, specifications, or other engineering criteria using appropriately qualified personnel and procedures.

1. Process Qualification

Qualification of a production process is achieved by performing the process under controlled conditions on samples and then analyzing the output to determine acceptability. When the process can be duplicated on a repetitive basis by holding essential variables constant, and meet the requirements, the process is considered qualified. Qualifications are performed to written instructions based upon engineering specifications and include essential variables.

2. Personnel Qualification

All personnel performing work activities have capabilities commensurate with their assigned functions, a thorough understanding of the operation they perform, the necessary training or experience, and adequate information concerning application of pertinent quality provisions to their respective functions. Supervisors responsible for directing work activities are responsible for assuring that personnel under their direction meet these qualification requirements.

12.9.8 Inspection

1. Inspection Planning

Inspections are performed to documented and approved plans for each work operation where it is necessary to measure quality. Inspection plans, as required, are incorporated into the detailed work instructions of the performing components.

2. Inspection Requirements

Inspections are performed, as required, to written instructions and the inspection results are documented. When requested, RC inspects raw materials, fabricated parts, assembly, and installation to the specifications provided by the requesting organization. For purchased material, the receiving clerk identifies and matches quantities received with the purchase order and then notifies the requesting organization that the material has arrived. The requesting organization is then responsible for making arrangements for receiving inspection, as required.

3. Hold Points – Approvals

Hold points are stages in the planned activity beyond which work cannot proceed until the preceding work has been evaluated and approved. Hold points are determined by specific job requirements. Hold points and approval requirements for each organization are specified, as required, in the appropriate work instruction or procedure.

12.9.9 Test Control

The responsible engineer identifies the need for development testing and/or for establishing test criteria for items not proven in design standard, mathematical analyses, or in state-of-the-art practices. Tests are aimed toward evaluation of performance capability under various conditions required by the design. Tests are conducted in accordance with written procedures; the test results are documented and evaluated to assure that the test requirements have been satisfied.

12.9.10 Control of Measuring and Test Equipment

Each component which performs work is responsible for the inventory, identification, and calibration of gages and instruments used for measuring quality parameters as required or as specified by the requesting engineer. Inspection gages and instruments are calibrated, as required, with traceability to certified standards. If no certified national standards exist, the basis for calibration is documented.

12.9.11 Nonconformances

1. Nonconforming Material Procedures

Procedures will be provided, as required for the control of materials or parts as specified by the responsible engineer, which do not conform to requirements, in order to prevent their inadvertent use.

2. Disposition of Scrap Materials

Disposition of nonconforming materials shall be accomplished after a review by responsible personnel or groups and will consist of acceptance, repair, rework, or rejection.

12.9.12 Corrective Action

Documentation of agreed-upon corrective action for conditions adverse to quality are governed by established NS&QA procedures. The procedure assures that corrective action commitments are implemented on a systematic and timely basis.

12.9.13 Experimental Equipment

This program provides, as applicable, controls over the fabrication and installation of experimental equipment to the extent that these relate to reactor safety.

12.9.14 Records

Records are retained in accordance with the requirements of Subsection 12.6.

12.9.15 Audits

RC conducts audits in accordance with established procedures to verify compliance with the various elements of this Quality Assurance program. Audits are conducted on a scheduled or random unscheduled basis, or both, as appropriate.

12.10 OPERATOR TRAINING AND REQUALIFICATION

All licensed operators participate in a comprehensive Operator Requalification Program. The program is designed to maintain the competence of the NTR operating personnel to handle abnormal events and to comply with the requirements and intent of 10CFR55.59. Refer to the "Requalification Program for the General Electric Nuclear Test Reactor" for details.

12.11 ENVIRONMENTAL REPORTS

Operation of the NTR has had minimal affect on the environment. Refer to "GETR Environmental Information Report," NEDO-12623, dated July 1976, and subsequent NTR annual reports to the NRC for information.

12.12 REFERENCES

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13.0 ACCIDENT ANALYSES

13.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

This chapter contains an evaluation of the facility response to certain events that can be reasonably postulated to occur at the NTR and which appear to have safety significance. The results of the analyses show that design features, equipment, and procedures are in place to ensure that the health and safety of the public and plant personnel are not jeopardized by the occurrence of any of the postulated events. The events analyzed include anticipated operational occurrences and potential accidents.

Reactor transients were analyzed by simulating reactor dynamics with a digital computer. The model used is discussed in Subsection 13.2.

Events categorized as anticipated operational occurrences are discussed in Subsection 13.3. Anticipated operational occurrences are the results of single equipment failures, or malfunction, or single operator errors that can reasonably be expected during any planned mode of facility operation. The anticipated operational occurrences analyzed in this chapter are:

- Loss of electric power
- Loss of secondary cooling
- Loss of facility air supply
- Inadvertent start of primary pump
- Fuel handling errors.

Unacceptable consequences for anticipated operational occurrences are:

- Release of radioactive material to the environs that exceeds the limits of 10CFR20;
- Radiation exposure of any person in excess of 10CFR20 limits; and
- Violation of an established safety limit.

Events categorized as accidents are discussed in Subsection 13.4. Accidents are defined as postulated events not expected to occur during the course of plant operation that appear to have the potential to affect one or more of the radioactive material barriers. The postulated accidents analyzed in this chapter are:

- Uncontrolled reactivity increases
- Loss of primary coolant flow (pump shaft seizure)
- Rod withdrawal
- Loss of primary coolant.

Unacceptable consequences for postulated accidents are:

- Radioactive material release to an extent that exceeds the guideline values of 10CFR100.
- Violation of a safety limit.

Subsection 13.5 is an evaluation of experiment safety and shows that procedures, limits, and safety equipment are in place to ensure the proposed experiment program can be carried out without undue risk to the health and safety of the public and plant personnel.

There is a close relationship between the safety analyses for anticipated operational occurrences and accidents and the safety limits and limiting safety system settings. Development of proposed safety limits and limiting safety system settings are discussed in Subsection 13.7.

The results of the analyses show that there are no credible events that could cause fuel melt or a significant release of fission products from the fuel. Even if catastrophic nonmechanistic failure of the NTR facilities is assumed, there are no potential consequences more severe than those associated with the accidents analyzed in this section. Compaction of the fuel, while essentially impossible mechanistically, would not cause the reactor to go critical since water loss, increased self shielding in the fuel, and the geometry change due to flattening of the cylindrical core are all negative reactivity effects. Loss of water shuts down the reactor and no fuel melting occurs, as discussed in Subsection 13.4.6. Also, deformation of the core, which causes the fuel to contact the core can structure, would improve heat-transfer and result in lower Loss-of-coolant Accident (LOCA) temperatures. The only accidents which could possibly cause fuel damage and release of fission products from the NTR fuel are those resulting from large reactivity insertions. Reactor configuration and the reactivity worth of experiments are controlled to ensure that destructive reactivity transients are not credible. Nevertheless, an assessment of the consequences of an assumed fission product release is presented in Subsection 13.6 to demonstrate the capability of the facility, even though such a release is not possible under the 0.76$ reactivity limit.

13.2 TRANSIENT MODEL

The reactor dynamics were simulated with a digital computer. The major features of the model are summarized below.

(1) The core was represented by three 5 node channels:

   a. A channel representing average power, flow, and temperature conditions.

   b. A channel with less than average power (90%) but much less than average coolant flow (53%).

c. A channel with highest power (130% of average) and highest flow (153% of average).

Each channel corresponded to a flow path from the inlet at the bottom of the core, upwards around the circumference of the core between adjacent fuel disks, to the flow exit at the top of the core.

(2) A circumferential power distribution was assumed which had the neutron flux skewed to one of the upper quadrants of the core. Conditions in channels b and c were taken as those in the hottest side of the core and differed only by the relative peaking along the axis of the core (chopped cosine with peak-to-average 1.15). Conditions in the average channel a represented the average of the power distributions on both sides of the core.

(3) The basic thermodynamic mass, volume, and energy balance equations were used to calculate the average fuel and water temperatures in each node. New heat-transfer coefficients were computed at each time step during the transient for the temperature and flow conditions existing at the time.

(4) Point kinetics were used since spatial neutron coupling is very strong. Six delayed neutron groups were included.

(5) Reactivity feedbacks included weighted values from the average channel water temperature and from steam voids in the hot and average channels. Doppler feedback was neglected for this high-enrichment fuel. Water temperature feedback was based on a coolant temperature coefficient of -5.7×10^{-3} (T-124) ¢/°F, where T is the water temperature in degrees Fahrenheit. Average channel void coefficient was –5.71 ¢/% voids, Steam formation in the hot channels was assumed to be worth 5% of average channel voids. Only bulk boiling feedback was included. Steam from subcooled surface boiling was neglected. No credit was taken for expulsion of primary coolant which resulted from thermal expansion of the fuel disks.

(6) Although temperature peaking was usually small within the uranium-aluminum alloy fuel, the center temperature was calculated at each node from average and surface temperatures. A parabolic distribution was assumed:

$$\text{Fuel Center Temperature} = 1.5\ (\text{Fuel Node Temperature}) - 0.5\ (\text{Surface Temperature}) \quad (2)$$

(7) For nonboiling nodes, the Seider-Tate relationship was used in the model to estimate the surface heat-transfer coefficient. The results were checked and found consistent with other laminar flow correlations. The average steady-state channel coefficients were calculated to be about 165 Btu/h-ft^2°F.

(8) Fuel surface nucleate boiling was simulated in the model, when fuel surface temperature reached the value given by the Jens-Lottes correlation.[7] During this condition, heat-transfer characteristics improve and the surface temperature can be calculated as follows:

$$T_{(Surf)} = T_{(Sat)} + 1.9\,(q/A)^{1/4}/e^{(P/900)} \tag{3}$$

where

T_{Sat} = coolant saturation temperature, °F,

q/A = surface heat flux, Btu/h-ft^2, and

P = system pressure, psia.

(9) Steam-blanketing was assumed to occur when a surface heat flux exceeds the critical heat flux. A critical value[7] of 450,000 Btu/h-ft^2 was used in the analysis. More recent work presented in Subsection 13.6 indicates a value of 600,000 Btu/h-ft is justified. During nucleate boiling, the effective heat-transfer coefficient is very large. If the critical heat flux was reached, the heat-transfer coefficient is very large. If the critical heat flux was reached, the heat-transfer coefficient[23] was conservatively dropped to 10 Btu/h-ft^2-°F.

(10) The safety rod insertion time was measured and is less than 300 milliseconds. In the analysis, a time of 200 milliseconds was assumed to include all electronic delays and the time required for the rods to move to the edge of the active core (the first 12 inches of their 30-inch stroke). The remainder of the insertion was fitted to an S-shaped curve of reactivity versus position.

An important aspect of the analysis is the heat-transfer characteristic by which steam is formed during excursions (the steam voids provide the strongest negative reactivity feedback in addition to scram). The general characteristics of the heat-transfer mechanisms have been described here and basic relations used in the heat-transfer analysis are given in Appendix B.

Figure 13-1 illustrates the qualitative hypothetical high-power excursion with out scram. For very fast transients which are not possible with the existing 0.76$ reactivity limit, some of the sequences shown may not be the same; however, most mechanisms which appear are illustrated in Figure 13-1. This hypothetical excursion develops according to the following sequence of events.

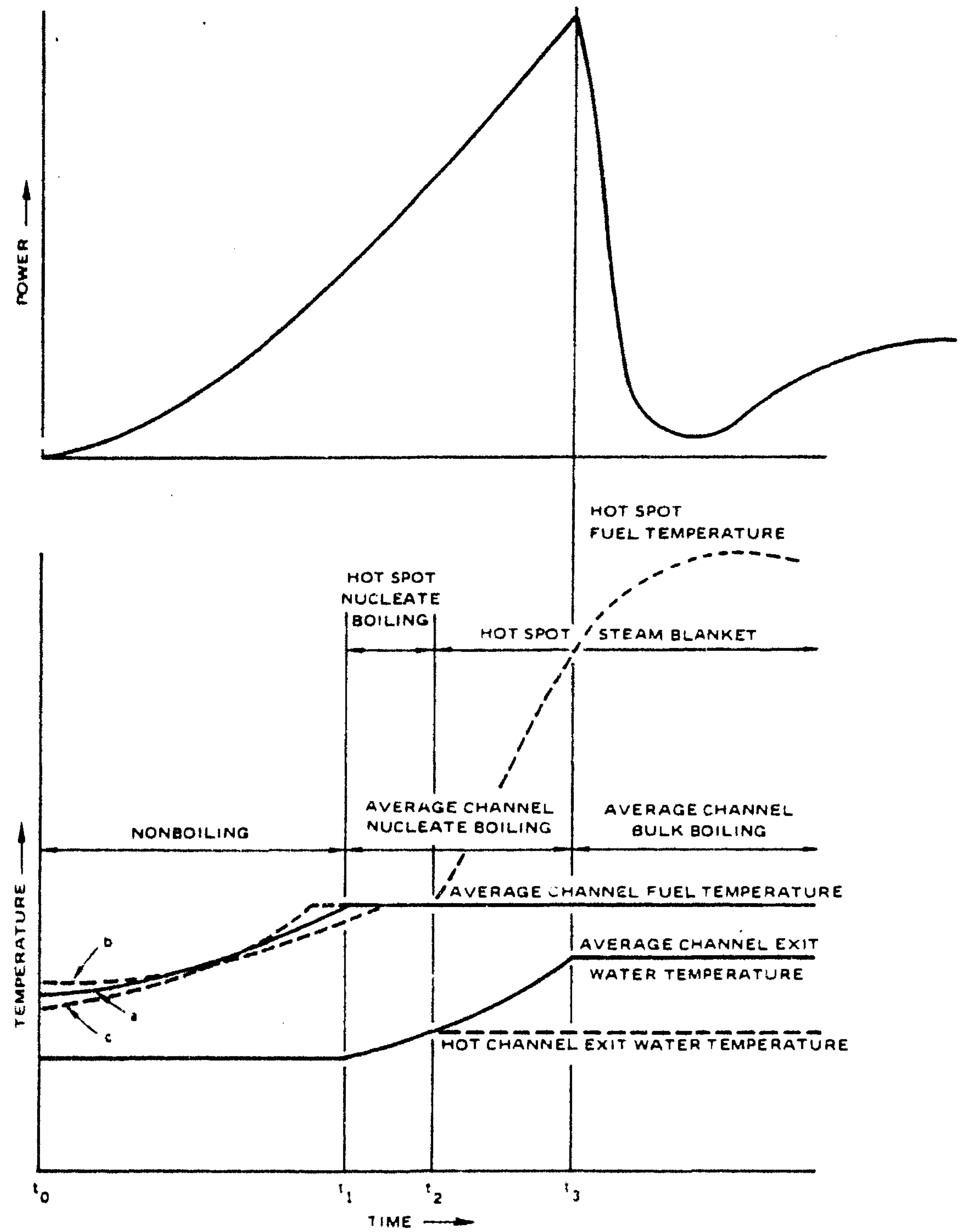


Figure 13-1. NTR Transient

(1) Initially, all channels are in laminar flow with heat-transfer coefficients near 165 Btu/h-ft^2-°F. Figure 13-1 shows three initial fuel temperatures for the three channels described in Subsection 13.2, Item 1. The highest power channel (c) usually produced the highest fuel temperature for transients in which the high-power peak was the dominant factor. As the transient progresses and power increases, fuel and water temperatures rise until the beginning of nucleate boiling.

(2) When the fuel surface temperature becomes high enough, nucleate boiling begins on the fuel surface. When this occurs at time, t_1, heat-transfer conditions improve greatly, holding the surface temperature essentially constant, and increasing the rate of rise of channel water temperature. The values of fuel surface temperatures during nucleate boiling conditions were estimated w n the Jens-Lottes correlation.[7]

(3) In the transient shown in Figure 13-1, average channel exit water temperature reached saturation at time The fuel temperature remained nearly constant throughout nucleate boiling. ormation of stream produced a large negative reactivity feedback, which turned the power excursion.

(4) In this example, the hot spot was steam-blanketed. This phenomenon is triggered if the power rises high enough to produce a surface heat flux greater than the critical value given in Subsection 13.2, Item 9. When this occurs, the surface heat-transfer coefficient drops to about 10 Btu/h-ft^2-°F. The fuel temperature will rise sharply since this condition almost insulates the fuel. As shown in Figure 13-1, this temperature will level off when the power is turned. The temperature will approach a new steady-state value which corresponds to the final power level. This power level is dependent upon the type of accident and the extent of steam formation. The reactivity events typified by this general type of behavior are presented in Subsections 13.4.1 through 13.4.4.

13.3 ANTICIPATED OPERATIONAL OCCURRENCES

13.3.1 Loss of Electrical Power

Other than the battery-operated emergency lights, there a no emergency power supplies, either ac or dc, for the NTR facility. Loss of ac power to the facility means a complete loss of electrical power and results in reactor scram through the following processes:

- The magnet power supply would be deenergized and cause loss of power to the safety rod electromagnets.
- The primary pump would stop and cause a low-flow scram signal (if power was greater than 100 watts at the time of loss of ac power).

- Numerous fail-safe circuits in the safety system would signal the power switches and scram relays to initiate a scram.

The only differences between this scram and a normal scram would be that the simultaneous loss of primary coolant flow (although secondary coolant flow is by gravity, it is stopped by a spring-closed, solenoid-activated valve) and the control rod and safety rod drives would not run in automatically. However, the safety rods would be inserted by their spring action (normal scram action) to shut down the reactor. The consequences from a loss of ac power would be no worse than a loss-of-flow scram. As discussed in Subsection 13.4.5, there are no unacceptable consequences, even from the worst case loss-of-flow accident.

13.3.2 Loss of Facility Air Supply

The only items affected by the loss of facilit ir are the air piston operator for the south cell door, and the radiation shield shutter for the horizontal facility in the south cell. The design specifications for the south cell door require that it be movable manually by one person. Loss of air to the beam shutter would have no effect on its position (i.e., it would remain in the position it was in at the time of air supply failure). Thus, there is no safety concern due to loss of facility air supply.

13.3.3 Loss of Secondary Coolant

Secondary coolant flows by gravity through the tube side of the primary heat exchanger, as described in Subsection 5.3. Loss of secondary coolant or loss-of-coolant flow when the power level is high enough to produce an appreciable heating rate will cause the reactor to scram from high primary coolant temperature. If the reactor power is not high enough to produce a heating rate that will soon cause a scram, the loss of secondary coolant will soon be evident to the operator by:

- The slightest changes in temperature, which cause an observable reactivity effect.
- The temperature monitor system readout at the console.
- The secondary flow control in the control room.

13.3.4 Primary Pump Inadvertent Start

If the primary pump were inadvertently started, the effect would be to change the reactor inlet temperature. A decrease in inlet temperature will cause reactor power to drop. An increase in inlet temperature will produce a rising power transient – a hot water transient. This transient is comparable to a cold-water accident for reactors that operate with a negative temperature coefficient of reactivity. Normally, there is no source of energy to produce an increase in the primary coolant temperature; however, the system is designed to accommodate an electric heater. The amount of positive reactivity which could be added is less than 0.10$ (from room temperature to turnover temperature); therefore, the resultant transient would, and could, be controlled by manipulation of the control rods. Although the 5-kW heater has been removed from the system, it could be replaced, if needed.

The worst possible case would be a coolant heatup to 124°F from shut-down power and temperature conditions. The temperature, net reactivity, and power characteristics are shown in Figure 13-2. The electric heater was assumed to be shut off (or regulated) after 1000 seconds, so that the inlet remained at 124°F. The power will continue to rise with the period between 80 and 100 seconds until power increases sufficiently to raise the core coolant temperature above 124°F and reduce the net reactivity.

For such a slow transient, a high-power scram would clearly stop the excursion without fuel damage. If the scram failed, bulk boiling would occur soon enough to prevent the power from reaching a level high enough to produce steam-blanketing. It has been shown that a step insertion of 0.76$ of reactivity would not cause fuel damage, even if the reactor failed to scram. Therefore, it can be concluded that a transient caused by the small amount of reactivity from the temperature would also be safely limited.

13.3.5 Fuel Handling Errors

Fuel handling equipment and procedures are discussed in Subsection 9.2. It should be reemphasized here that refueling for reactivity increase is not necessary, and fuel handling is very rare. The only fuel handling occurrence in the past 20 years was one core unloading and reloading associated with the core container replacement in 1976. The 16 fuel assemblies on hand completely fill the core reel assembly and fill the fuel container to the extent that the only remaining space of appreciable size is in the fuel loading chute. In other words, a fuel assembly, once inside the fuel container, must either be in the provided positions in the reel or in the fuel loading chute. The physical arrangement of the fuel container is such that an element located in the loading chute results in a worse core geometry 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 rods were withdrawn during loading so that the reactor was almost critical before adding fuel. Such an act is contrary to operating procedures and requires errors by the console operator and fuel loaders. The only other means of getting fuel close to the core is by inserting it into either the horizontal or vertical facilities. Use of these facilities is discussed in Section 10.

In addition to the inherent safety feature provided by having all existing NTR fuel elements in their most reactive configuration in the core, the following additional safety features ensure safety during all phases of fuel handling:

- Reactor design, fuel handling equipment, and administrative controls are such that not more than two elements can be handled at one time.
- All fuel movement must be performed in accordance with written procedures.
- The cell high-gamma-level alarm system will be in operation.

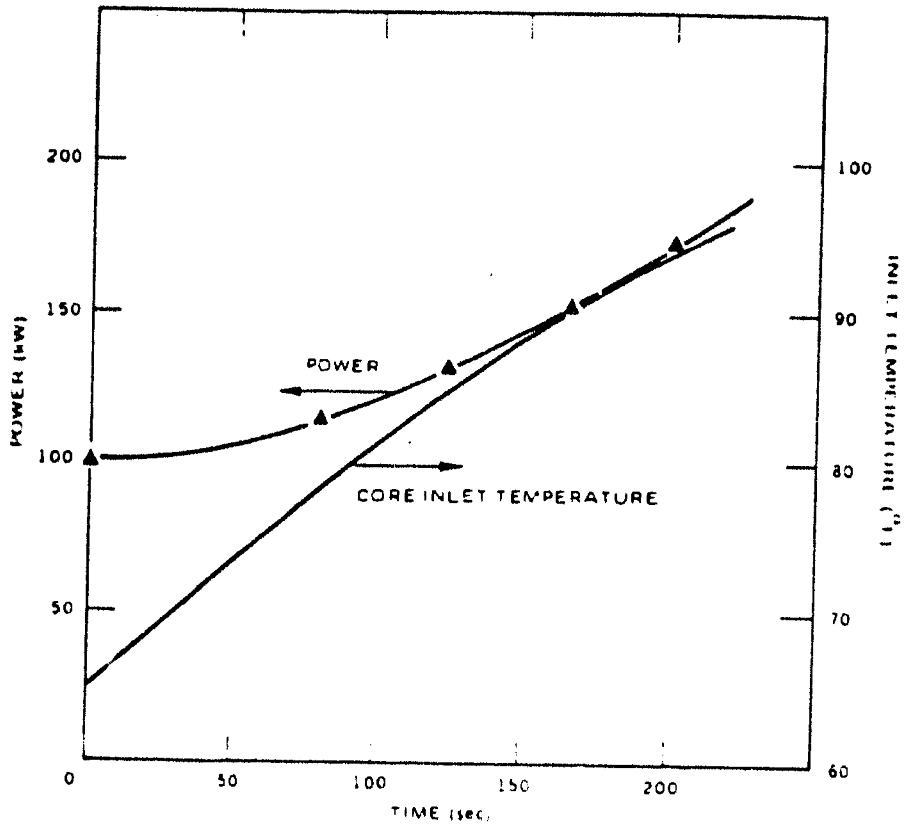


Figure 13-2. Primary Pump Inadvertent Start (from 65°F)

- By using all the manually positioned poison sheets, the core can be made 6.1$ subcritical (Table 4-1). Removal of the graphite plug from the fuel loading chute provides additional negative reactivity of approximately 1.25$.

- Any movement of source and special nuclear material within the NTR facility must have the approval of the licensed operator on duty.

- Any storage arrangement used will be analyzed to ensure a subcritical configuration.

13.4 POSTULATED ACCIDENTS

The transient model used to simulate the reactor dynamics is presented in Subsection 13.2.

13.4.1 Idealized Step Reactivity Insertions – with Scram

Transients resulting from step reactivity insertions up to 2.0$ were studied; a range of different initial reactor powers and flows were used. The results for steps with high power scram occurring at 150 kW are shown in Figure 13-3. Only a very slight fuel temperature increase was observed for steps up to 1.0$. In all cases, peak temperature rose sharply for reactivities above this value.

The transient due to a step reactivity insertion of 1.3$ while the reactor is at 100 kW and at rated flow is shown in Figure 13-4. The sequence of events for this transient follows.

Time (sec)	Event	Peak Fuel Temperature (°F)
0.0	1.3$ step insertion	195
0.0044	Scram circuit tripped	195
0.1460	Nucleate boiling began at hot spot	241
0.1538	Steam-blanketing occurred at hot spot	258
0.2046	Safety rods reached active core	504
0.218	(Power peak 1.04 x 10^5 kW)	652
1.0	Power dropping, temperature rising slowly	841

The transient is too fast for any channel bulk boiling to help the scram reduce power. The relatively high "tail" on the power curve is the result of delayed neutron groups which are controlling the rate of change of power. Even after an excursion has reached the steam-blanketed condition and the heat-transfer coefficient has dropped to 10 Btu/h-ft^2-°F, a power level of 100 kW can be maintained without melting at the hot spot. The peak temperature characteristic is very sharp. A peak temperature of only 400°F resulted from a 1.2$ step, compared to approximately 840°F for the 1.3$ step. The results for lower initial power and flow show that fuel temperatures are lower for these other cases.

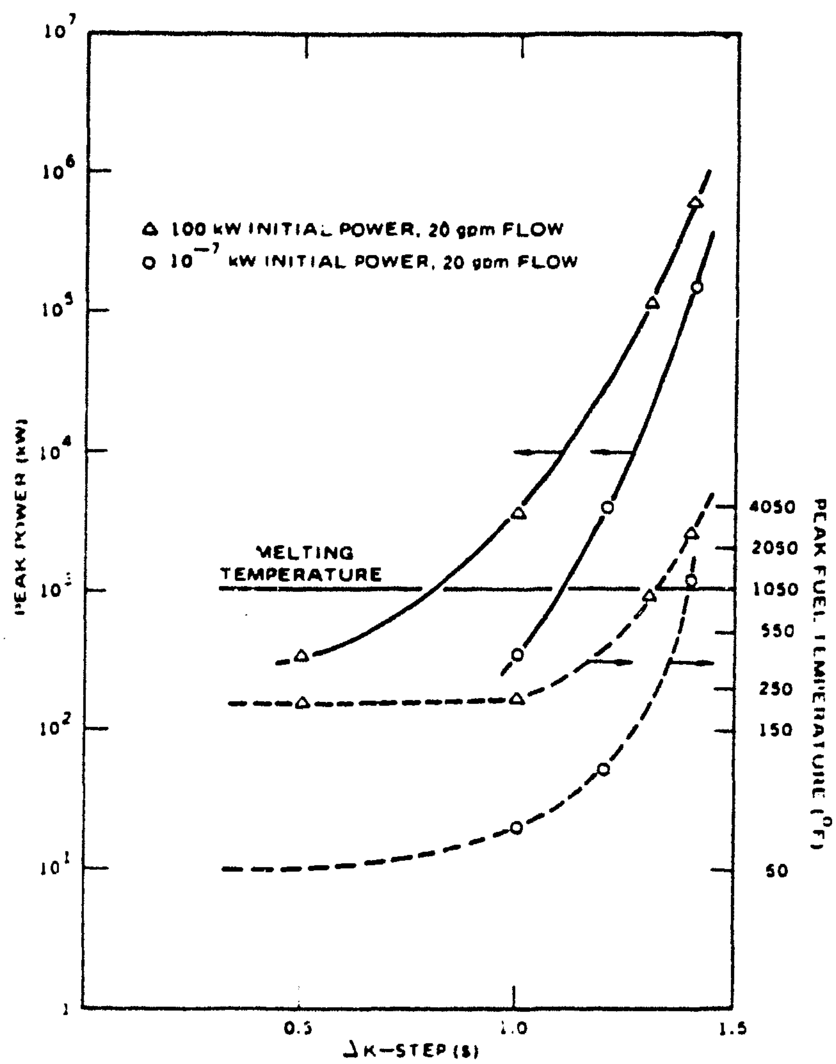
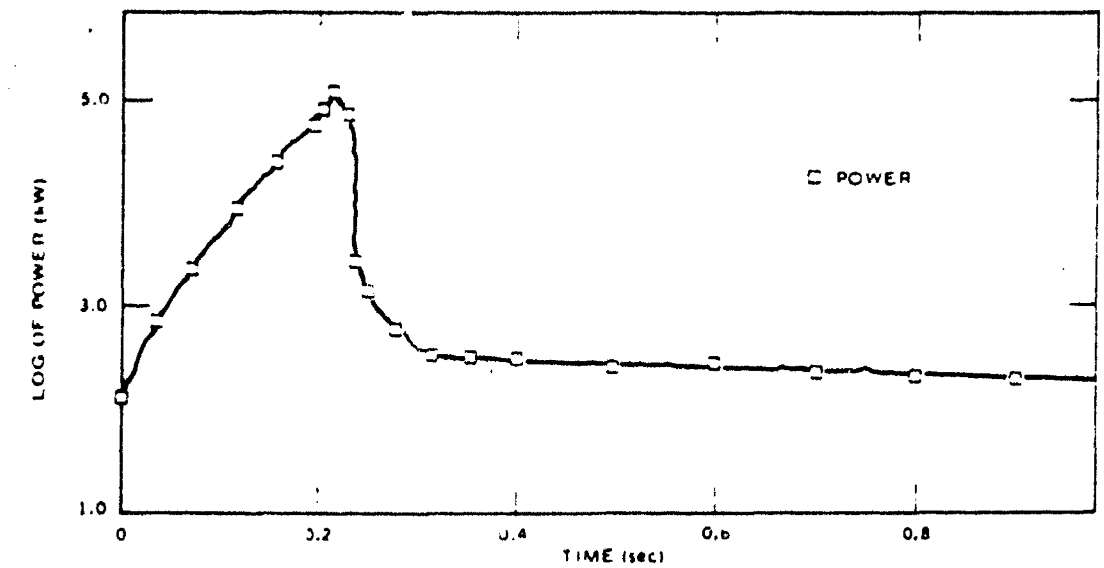
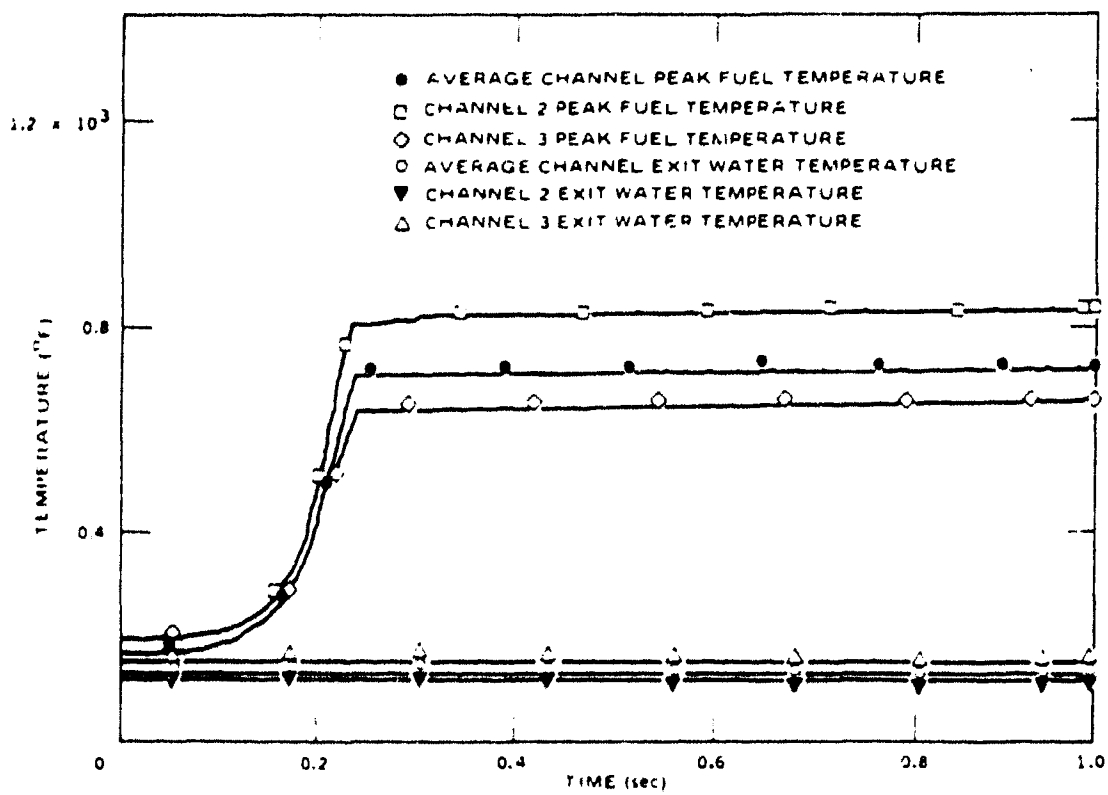


Figure 13-3. Δk-Steps with 15-kW Scram



a. Log of Power



b. Temperature

Figure 13-4. 1.3$ Step from 100 kW with Scram

13.4.2 Idealized Finite Ramp Reactivity Insertions – with Scram

Large reactivity insertions over short periods of time were studied for finite reactivity ramps. Reactivity insertions of 2$ and 4$, with durations from 0.2 to 0.6 second, were analyzed. The results for initial powers of 100 kW with the overpower scram occurring at 150 kW are given in Figure 13-5. For the 4$ insertion, fuel melting is not expected if the duration of the insertion is greater than 0.5 second. For the 2$ case, the minimum acceptable insertion time was 0.24 second. Figures 13-6 and 13-7 show near-limiting cases. In both cases, steam-blanketing and nucleate boiling occurred almost simultaneously so that fuel-surface, heat-transfer conditions were poor throughout the transients, and no bulk boiling was observed. In each case, power dropped below the level at which the hot spot is cooled even with steam-blanketed conditions before peak fuel temperature reached melting. For transients starting from lower power levels, the temperatures will be slightly less than those shown in Figure 13-5 because of the lower initial temperature. The sharp characteristic, however, places the limiting reactivity insertion time at nearly the same value. The consequence of inserting these large amounts of reactivity too fast, or if the scram failed, would be partial core destruction. The primary shut-down mechanisms would be associated with the expansion and dispersion of the fuel.

13.4.3 Reactivity Insertions – without Scram

It may be hypothesized that certain structures (used to support the control and safety rod mechanisms as well as experiments) might fail or move during a seismic event in such a manner as to withdraw the control rods and experiments from the core region and prevent operation of the safety rods. The cadmium poison sheets are manually positioned entirely within the graphite reflector, have no drive mechanisms, and are mechanically restrained so they will not move relative to the core during a seismic event. If the reactivity addition caused by control rod and experiment movement is sufficiently large, a power excursion not terminated by a scram could occur and result in fuel melting. The NTR will be operated in such a manner as to limit the potential excess reactivity to less than that required to cause fuel damage, assuming failure to scram.

From full power, the transient would be stopped by bulk boiling, even if all scrams fail, before fuel damage occurs for sizable step reactivity insertions. The results of a 0.76$ step reactivity insertion are shown in Figure 13-8. Power peaked at 4 x 10^3 kW, and bulk boiling began in the hot channel at about 2.3 seconds. The negative reactivity feedback from the voids was enough to drop power low enough that the temperature of the hot spot stayed below 255°F. The core did not steam-blanket and provided good boiling heat-transfer. The peak temperature characteristic versus magnitude of the reactivity step was very sharp (Figure 13-9). For small steps, the transient settled out with nucleate boiling characteristics.

To determine the effects of positive reactivity additions from less than full power, all transients were run with an initial power level of 1 x 10^{-7} kW. Inlet water temperatures ranged from 55 to 90°F and initial positive reactivity steps were varied from 0.60 to 0.80$. Results of the transient analyses led to two conclusions. First, the transients are relatively long, on the order of 40 or more seconds, which leads to the conclusion that the positive reactivity can be introduced in

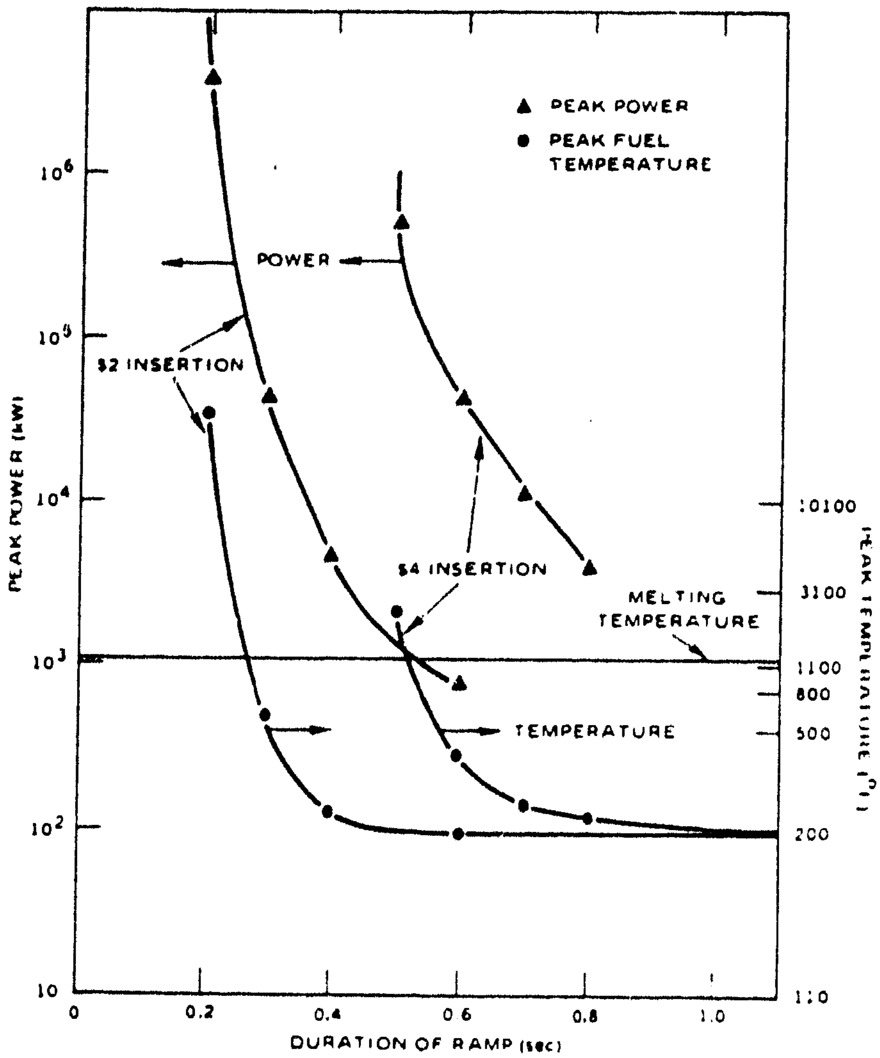
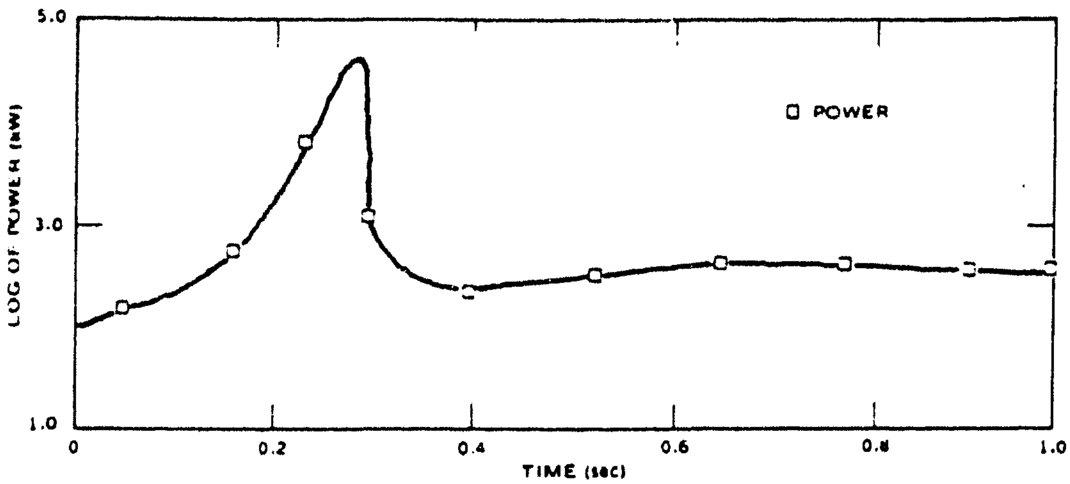
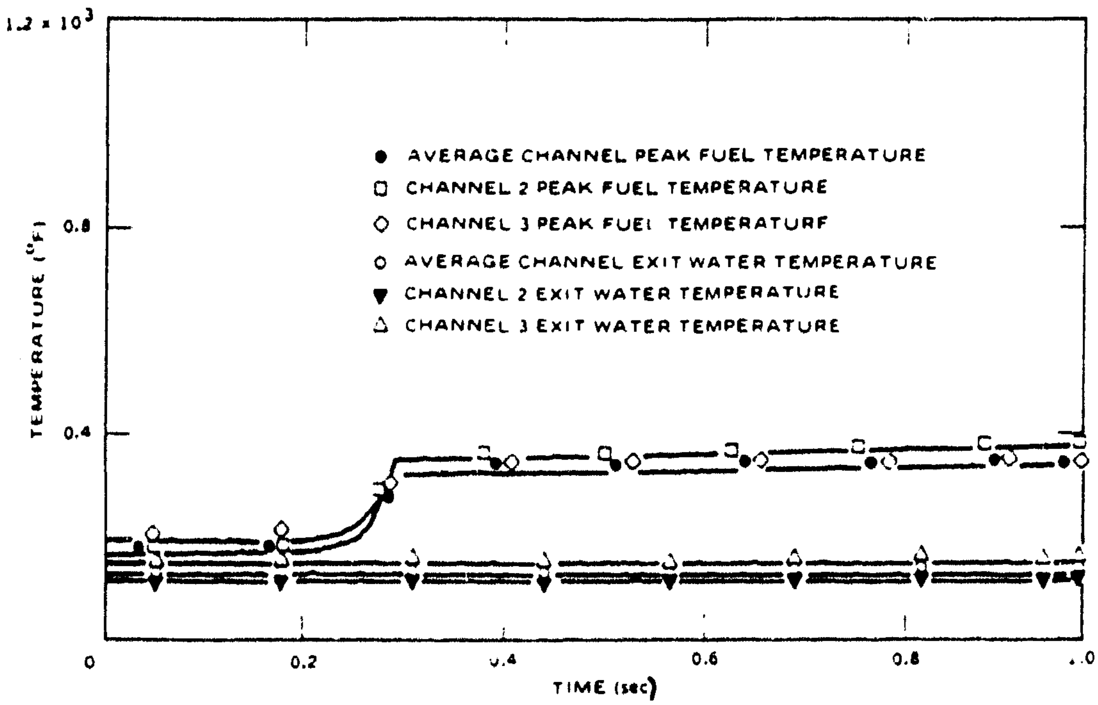


Figure 13-5. 100-kW Finite ramp Insertion with High-Flux Scram

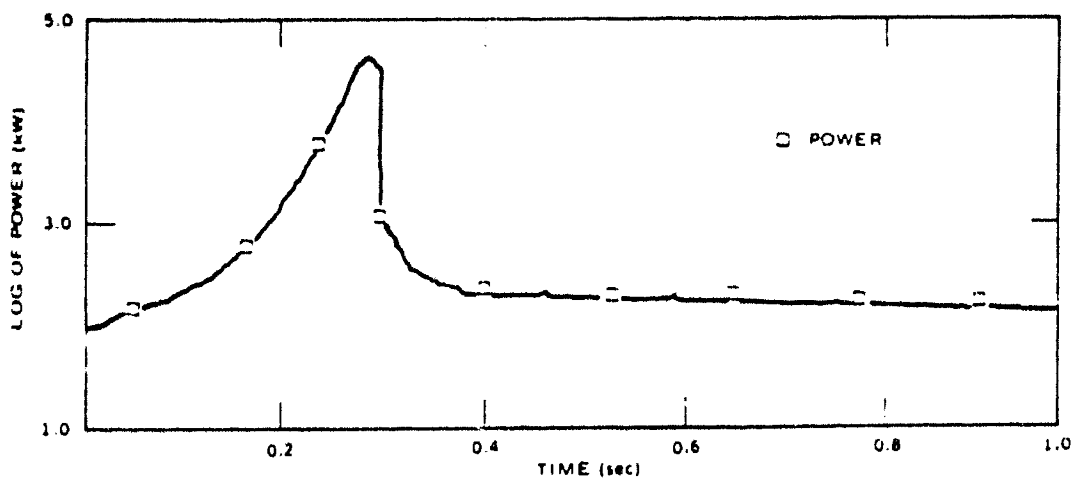


a. Log of Power

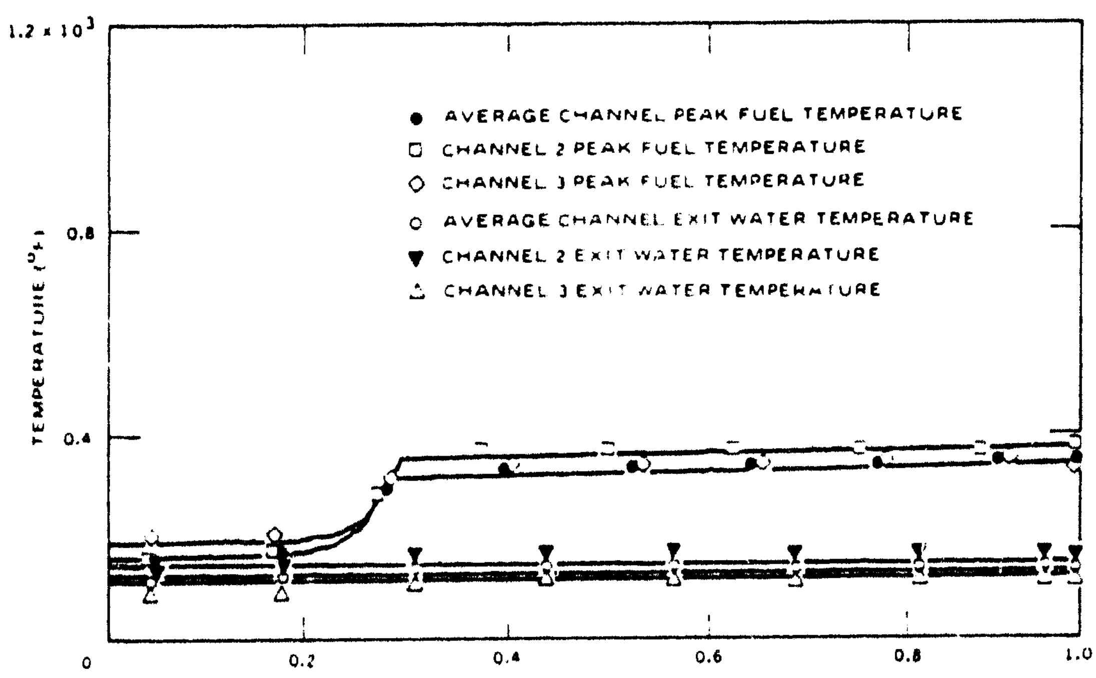


b. Temperature

Figure 13-6. 4$ Ramp in 0.6 Second from 100 kW with Scram



a. Log of Power



b. Temperature

Figure 13-7. 2$ Ramp in 0.3 Sec from 100 kW with Scram

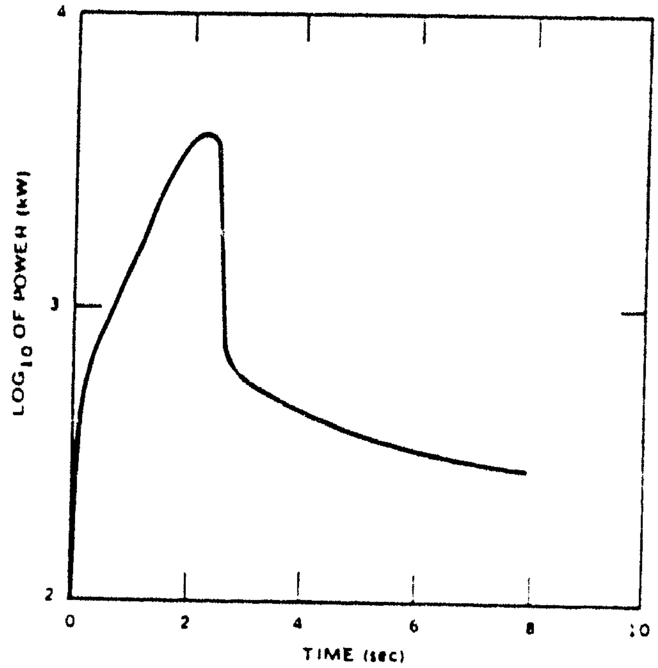


Figure 13-8. 0.76$ Step form 100 kW – No Scram

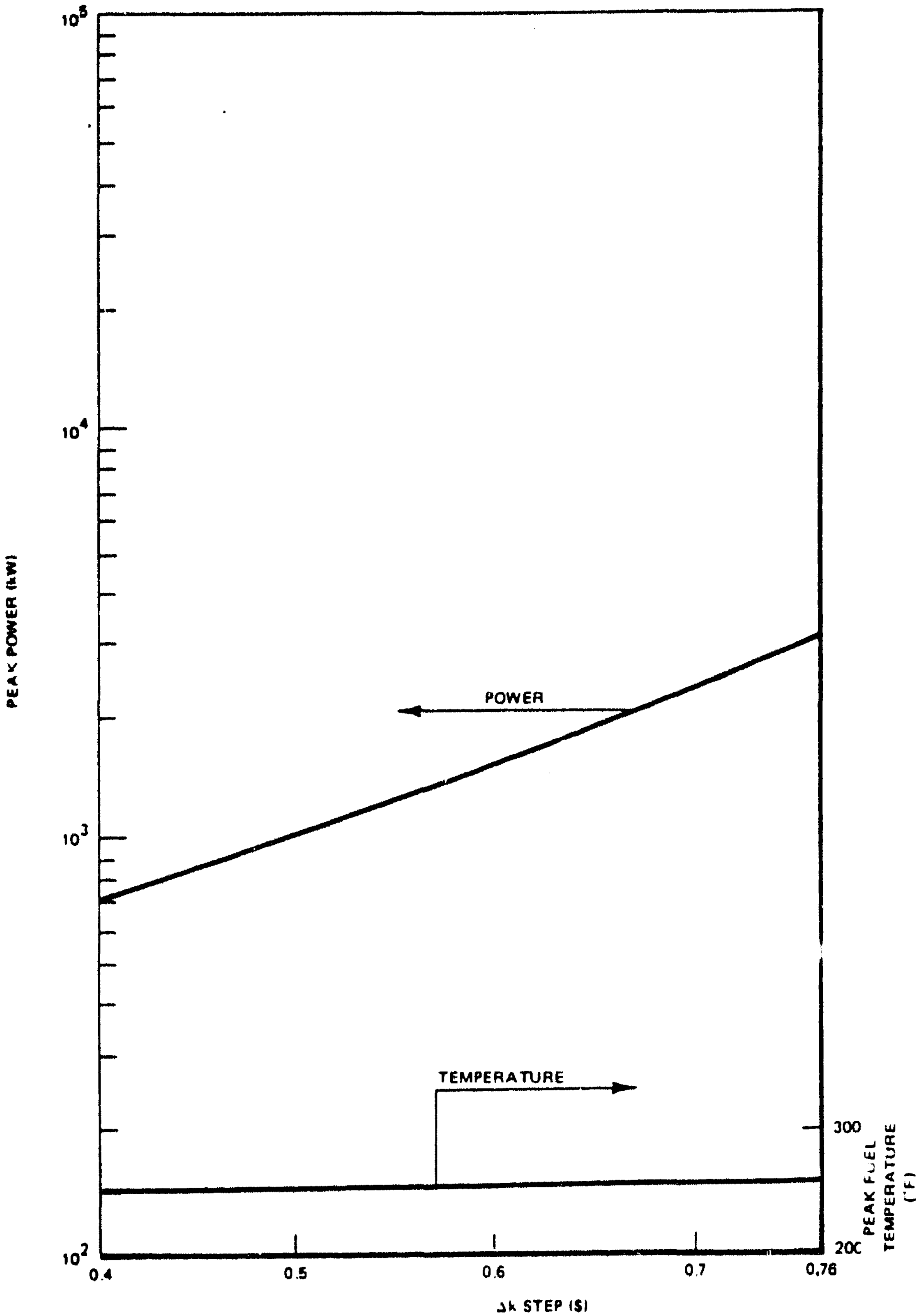


Figure 13-9. ΔK Steps from 100 kW – No Scram

either a step or relatively long ramp without affecting the outcome. Second, the positive reactivity feedback from the temperature coefficient, while not important for the full-power cases because the feedback is very small, is important for the zero-power cases.

Limiting values for the positive reactivity insertions were determined based on the acceptance criteria that the resultant transient was terminated by bulk boiling before any steam-blanketing occurred in the core. The limiting values based on this criterion are shown as the reactivity insertion limit values in Table 13-1 for various inlet water temperatures. Also in Table 13-1 are the maximum values of additional reactivity available from the temperature coefficient, which is positive at temperatures less than or equal to 124°F. As can be seen from the total reactivity values, limiting the total excess reactivity available from the temperature coefficient, control rods and experiments to 0.76$ or less ensures that there are no mechanisms available which will cause fuel damage.

Reactor power and peak fuel temperature versus time for a 0.66$ step insertion from 1 x 10^{-7} kW and 65°F inlet water temperature is given in Figure 13-10. While the time scale is different for other limiting reactivity insertions, the peak fuel temperature is virtually identical; it remains in the 240-250°F range during bulk boiling.

It should be stressed that these transient calculations are extremely conservative, since no credit is taken for the negative reactivity feedback from subcooled voids during nucleate boiling. With the large negative void coefficient of the NTR, it is felt that all the transients presented here would terminate before bulk boiling and realistic limits for reactivity insertions would be 0.90-1.00$.

Table 13-1

LIMITING REACTIVITY INSERTION VALUES

BASIS: Transient terminated by bulk boiling before any steam-blanketing in core; initial power 1 x 10^{-7} kW/no scram.

Reactivity Insertion Limit ($)	Inlet Water Temperature (°F)	Reactivity Addition from Temperature ($)*	Total Reactivity ($)
0.62	55	0.14	0.76
0.66	65	0.10	0.76
0.76	90	0.03	0.79

* Using the temperature coefficient of $d\rho/dt = -5.7 \times 10^{-3}\,(T-124)$ ¢/°F, where T is the water temperature in °F; the reactivity added by increasing the water temperature from T to 124°F is equal to $2.85 \times 10^{-3}\,(T-124)^2$ ¢

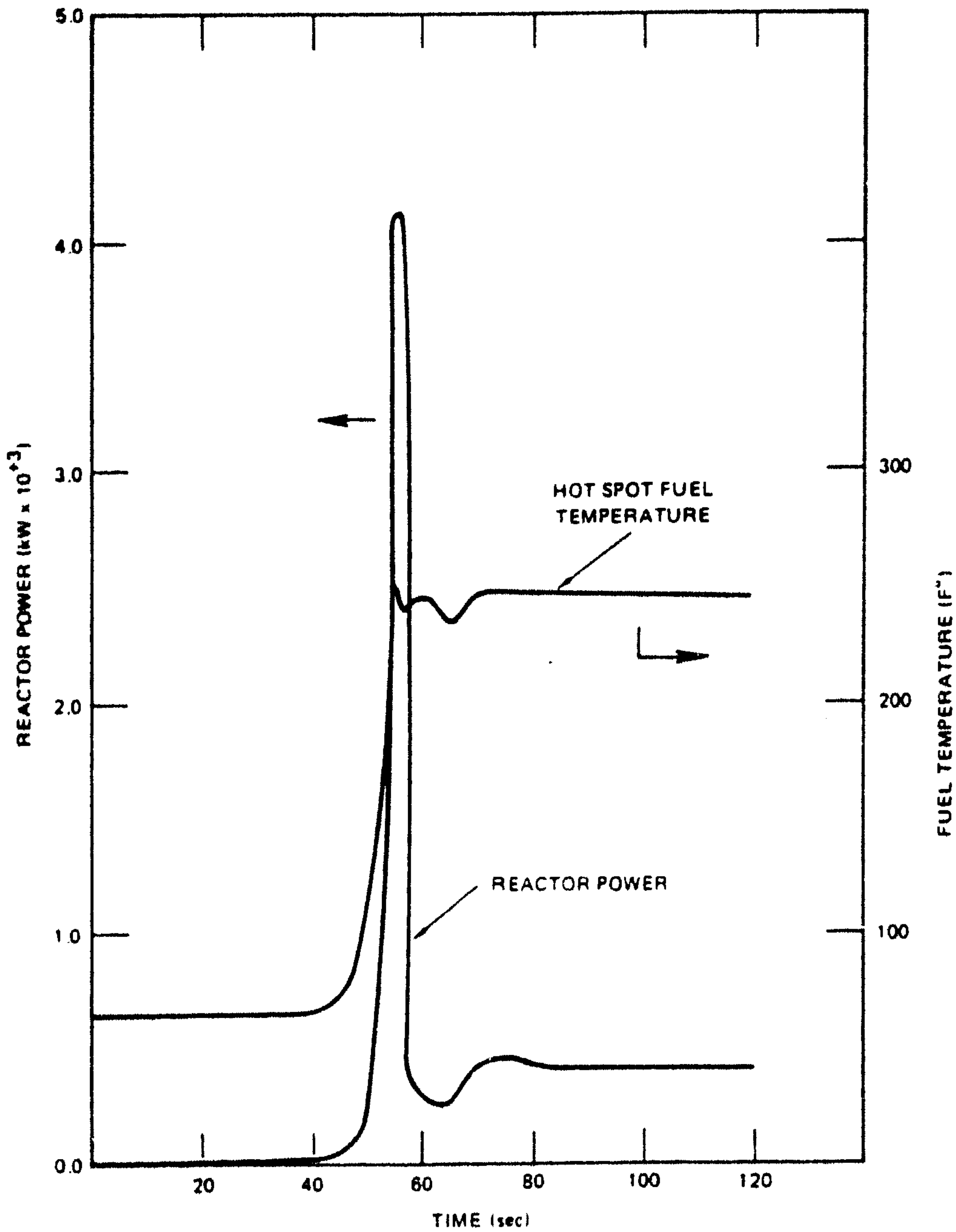


Figure 13-10. Reactor Power and Hot Spot Fuel Temperature Versus Time, 0.66$ Step from Source Level, 65°F Coolant Inlet Temperature – No Scram

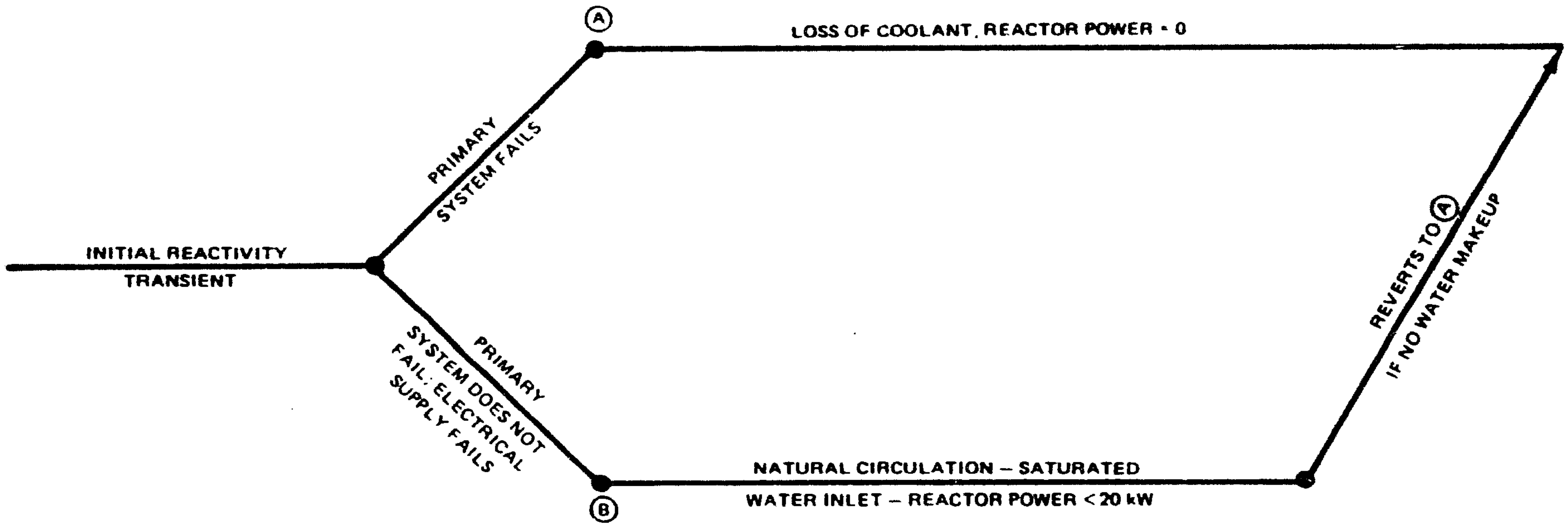


Figure 13-11. Possible Reactor States Following the Postulated Seismic Event

The discussion above described the sequence of events for the first 2 minutes of the hypothetical event. A discussion of several possible sequences of events from the 2-min point in time out to the final state for the reactor follows.

The diagram presented in Figure 13-11 shows the various possible states for the for the NTR following the initial reactivity transient. If no operator intervention is taken, the final state of the reactor will always be State A (reactor shutdown caused by loss-of-coolant). The extremely conservative loss-of-coolant analysis presented in Subsection 13.4.6 demonstrates the loss-of-coolant for the NTR has no significant consequences.

The performance of the reactor in the near term after the postulated seismic event depends on the extent of damage to the remainder of the reactor system. The most significant items are: a) the primary system piping, b) the primary pump, c) the secondary water supply system, and d) the electrical supply to the reactor system.

It is highly unlikely the primary system of the reactor would still be intact after a seismic event severe enough to result in the reactivity addition by the massive structural failure postulated here. If the primary system failed at the same time as the reactivity addition, the reactivity transient would not be significantly altered. The loss-of-coolant from the reactor results in shutdown by voiding the reactor core (Figure 13-11, State A).

We may also assume loss of electric power because: a) it is highly improbable that electric power to the site (including the NTR) would survive the event postulated here; and b) even in such an improbable circumstance, site emergency procedures call for the termination of all utility services to any buildings or facilities believed to have suffered damage.

As the loss of electrical supply automatically deactivates the primary system pump and automatically closes off flow to the secondary system, the structural fate of the secondary system becomes a moot question, and we need consider only the possibility of the primary system somehow surviving the event. If the primary system does not fail, or leaks at a very slow rate, the system will arrive at State B. For this state, the reactor will operate in a natural circulation mode at less than 3% of the pumped flow rate and at a power of less than 20 kW. Since there is no secondary cooling, the 20 kW of reactor power must be dissipated by the heat loss from the uninsulated reactor primary piping and by evaporation or boiloff of the primary coolant. If it is assumed that all of the heat is lost by boiloff, the rate of coolant loss is less than 4.5 lb/h. There are approximately 1000 of the 1800 gallons of water in the fuel storage tank which could drain into the reactor core can through the fuel loading chute to make up for the boiloff. If no water were made up to the system and no action were taken to shut down the reactor, it would operate for 70 days or more at a power level of 20 kW, or less. The loss-of-coolant by boiling will be a less severe event than the loss-of-coolant event described in Subsection 13.4.6 for two reasons. First, the reactor power is lower (20 kW, rather than 100 kW) and, if a primary system leak was not developed, the loss-of-coolant is not complete. In fact, the slow loss-of-coolant will result in a slow decrease in power and only a partial loss-of-coolant will occur. The core can be voided from the top by nearly 20% before any single fuel element is totally uncovered and the surface

heat flux would be so low that it could quite easily be cooled by convection to the steam and radiation to the inner surface of the core can.

As the maximum fuel temperature for a loss-of-coolant occurring at a power level of 100 kW is 620°F, no fuel melting, and hence, no fission product release, will occur from this accident.

13.4.4 Rod Withdrawal Accidents

The safety system and rod withdrawal procedures are designed to provide adequate control of the reactor at all times. Even if interlocks fail and the operator deviates from normal procedures so that the rate of power increase is not controlled by normal manual control rod movements, the reactor period and neutron flux level monitors would scram the reactor. If the reactor did not scram, the analysis in Subsection 13.4.3 is applicable. It is shown in the transient analysis the reactivity can be introduced in either a step or relatively long ramp without affecting the outcome. This analysis indicates that the transient which results from the total reactivity addition of the control rods, experiments, and temperature effect without scram (and the potential excess reactivity is ≤0.76$) does not melt fuel. Therefore, the transient which would be caused by the withdrawal of all the rods can be accommodated.

13.4.5 Reactor Loss of Flow Accident

To analyze the effects of a sudden loss of primary coolant recirculation pumping, it was assumed that the worst loss-of-flow accident (instantaneous seizure of the rotor in the single recirculation pump in the system) occurs. For such an accident, it is estimated that the pump flow will coast down to a natural circulation value within 0.1 second. The accident is assumed to occur while the reactor is operating at 100 kW. Although the transient would be terminated by the low-flow scram, in this analysis, it will be assumed that this scram does not function. After the flow has decreased to the natural circulation rate, the coolant temperature and the natural circulated flow rate will increase. This trend will continue until either a) bulk boiling at the hot spot produces enough voids to stop the power rise by reactivity feedbacks, or b) the average coolant temperature goes high enough to allow the negative temperature coefficient to halt the power rise. The initial core average coolant temperature is 110.6°F, and the initial excess reactivity is assumed to be zero. As the coolant temperature increases, the excess reactivity also increases to a turnaround temperature of 124°F, at which point the temperature coefficient becomes negative. Meanwhile, reactor power is on the rise, but will begin to slow down as the coefficient goes negative. The final steady-state operating point will correspond to a power and flow combination which gives the same reactivity contribution from temperature as for initial steady-state operation. Using the coolant temperature coefficient (Section 4, Equation 1), this final coolant temperature level is 138°F. Thus, there is no bulk boiling in the average channel. The heat flux is far below the heat flux necessary to initiate film blanketing. Moreover, the fuel plate surface temperature has been limited to a value well below the melting point, as a result of local surface boiling.

The times in the preceding discussion are all referenced to the time of initial flow reduction. Peak reactor power during the loss-of-flow condition is 101.2 kW and occurs 17 seconds after

instantaneous reduction of flow. Nucleate boiling begins in the outlet of the hot channel at 17.5 seconds. Maximum fuel temperature during the transient is 238°F. Bulk boiling begins in the outlet of the hot channel at 48 seconds. Equilibrium reactor power of 16 kW is reached at approximately 160 seconds. Maximum equilibrium fuel temperature is 226°F. After the very small power increase, the burnout ratio increases as power decreases during the transient and ends at an equilibrium value of approximately 26.

13.4.6 Reactor Loss-of-coolant Accident

The reactor loss-of-coolant accident involves the total loss-of-coolant inventory in the core as the result of a rupture in the primary system, combined with a failure to scram. The accident is postulated to occur as follows:

- Primary system ruptures at some point below the core entrance so that gross removal of core coolant supply occurs.
- As the water in the core is removed, the fuel is uncovered; the uncovering of the fuel acts to shut down power generation to a decay heat level.

The rupture is taken as being large enough to cause a very rapid coolant loss so that all water is lost and the core power is down to the decay heat level very shortly after the accident. It is assumed there is no post incident cooling system in the reactor and, as a result, the only cooling of the fuel plates occurs by any natural convection air currents that may be set up and by radiation heat-transfer from the core to the graphite. For simplicity and conservatism, convective heat removal by natural air currents is neglected. It is further assumed that no heat escapes from the graphite stack to the outside environment.

The initial power level of the reactor is taken to be 100 kW, and the subsequent decay heating rates are given in Figure 13-12 as a fraction of the initial power. A power-peaking factor of 1.30 was assumed, which includes both the normal axial peaking and severe azimuthal skewing. The calculation was performed using a version of the Transient Heat-Transfer (THT) computer program.[24] The nodal structure for the problem is shown in Figure 13-13. Axial heat-transfer was neglected.

The peak fuel temperature and the volume-averaged graphite temperature are shown in Figure 13-14 as a function of time after coolant loss. The fuel temperature reaches a maximum of about 570°F 100 minutes after coolant loss and then begins to decline. The rise of the graphite temperature is almost imperceptible – only 15°F in 3 hours.

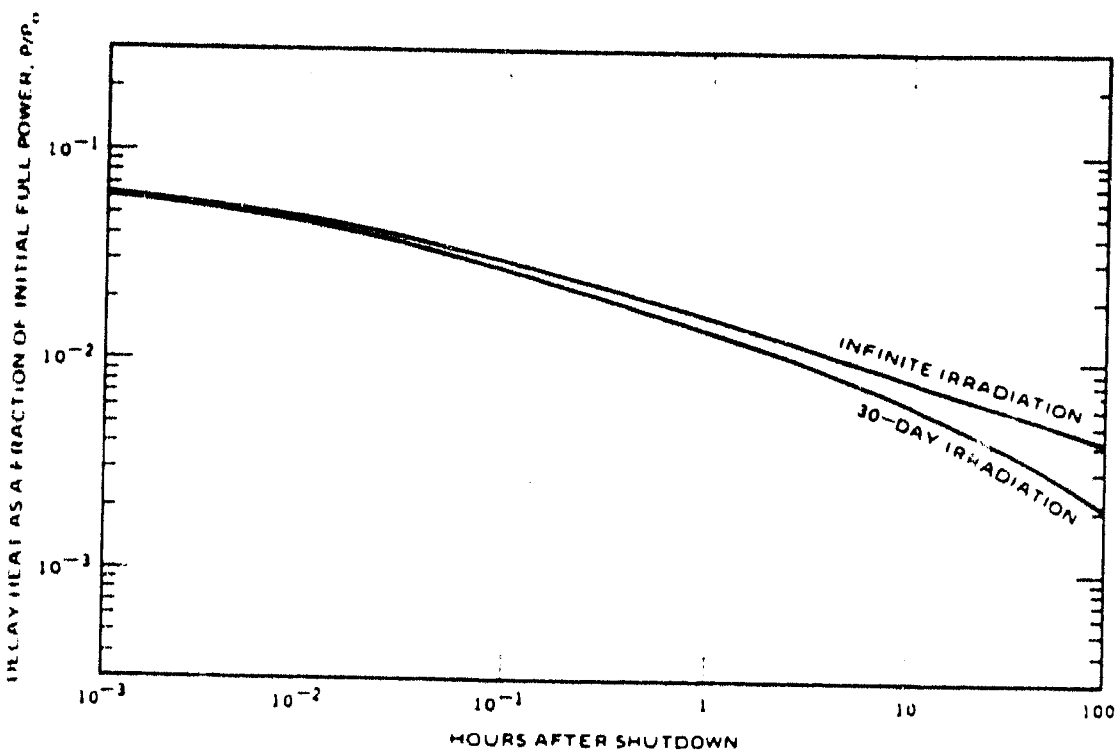


Figure 13-12. Decay Heat Rate

[redacted]

Figure 13-13. Node Structure for TNT Analysis

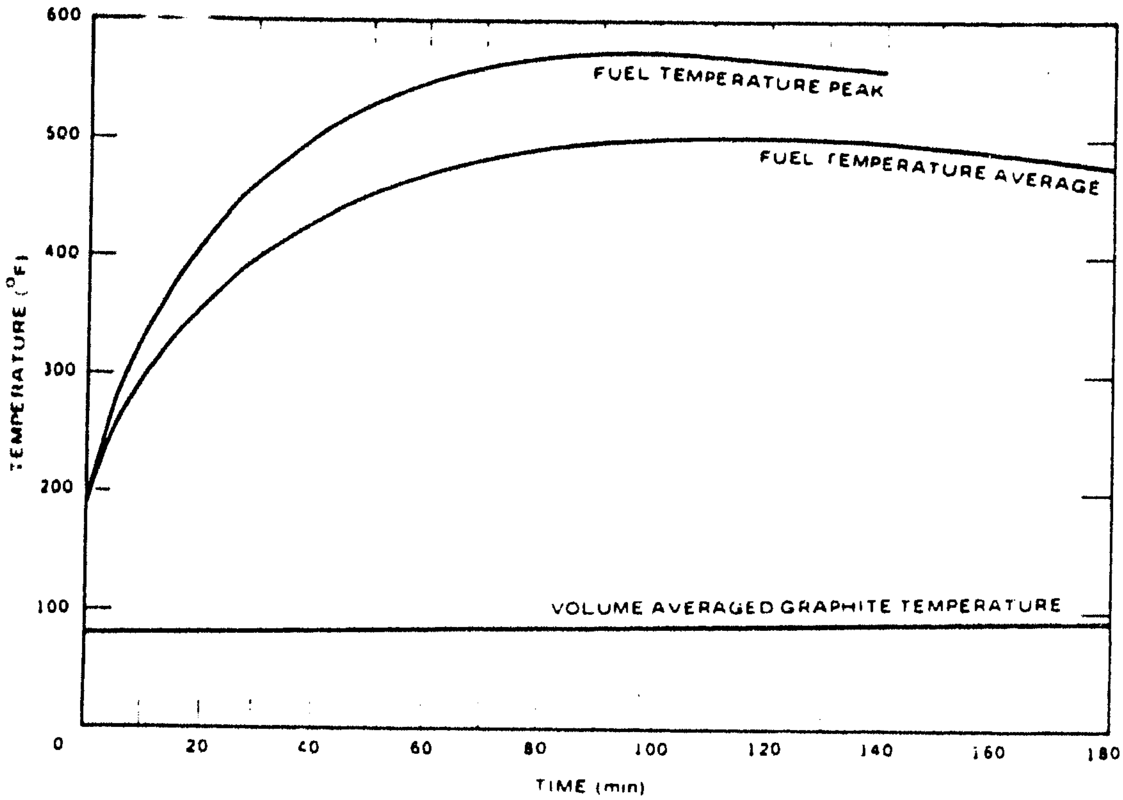


Figure 13-14. Fuel and Graphite Heatup Following Loss of Coolant (Power Calculated to 200 Seconds)

The analysis was repeated using a higher peaking factor. The maximum fuel temperature for a loss-of-coolant accident with a 1.58 peaking factor is 620°F at 1-1/2 hours. Reactor power at the time of the peak fuel temperature is 1.5 kW. It has been shown that this power could be tolerated indefinitely without increasing graphite temperatures to over 150°F, assuming a natural convection heat-transfer coefficient of 0.6 Btu/h-ft^2-°F on the exposed surface of the reactor. Therefore, a second fuel temperature peak greater than 150°F is not possible.

13.5 EXPERIMENT SAFETY ANALYSIS

13.5.1 Introduction

Descriptions of the NTR experiment safety programs and associated facilities, equipment, and procedures are presented in Section 10. As stated, before any experiment may be conducted, there must be a review and approval of a written description and safety analysis.

The purpose and requirements for experiment safety analyses are described in Subsection 13.5.2. Considerations that will be addressed are identified and discussed.

The potential mechanical and radiological consequences of postulated accidents involving explosive material at the NTR facility are described in Subsection 13.5.3. The details of the accident analysis are contained in Appendix A.

The limitations that will apply to all types of experiments are discussed. Adherence to these limitations is mandatory and will provide assurance of safe performance of experiment programs within imposed regulatory restrictions.

13.5.2 Safety Analysis

The purpose of the safety analysis for experiments is to ensure that consideration is given to any feature of the design or conduct of an experiment, including intended functions and possible malfunctions, which could create, directly or indirectly, a radiological exposure hazard. When applicable, the analysis will consider:

- Any interaction of an experiment with the reactor system that has potential for causing fission product release from the fuel.

- Any interaction that could adversely affect an engineered safety feature or control system feature designed to protect the public from fission product release.

- Inherent features of an experiment that could create beams, radiation fields, or unconfined radioactive materials.

- Potentially adverse interaction with concurrent experimental and operational activities.

The safety evaluation for each experiment utilized in experimental facilities will consider:

- The physical conditions of the design and conduct of the experiment.
- The content of the material.
- The administrative controls employed to evaluate, authorize, and carry out the experiment.

A description of specific items that will be addressed, when applicable, follows.

13.5.2.1 Reactivity Effects

The principal concern is with a net positive reactivity effect, whether it is caused by the insertion of an experiment with a positive effect, or by the removal of an experiment having a negative reactivity effect. Every experiment or type of experiment, as appropriate, will be evaluated for:

- Its potential reactivity worth.
- The rate of change of reactivity of unsecured experiments and movable experiments.

13.5.2.2 Thermal and Hydraulic Effects

An experiment will be evaluated to ensure its thermal limits are not exceeded and for its actual and potential thermal effects on reactor components and coolant. This evaluation will be made for the reactor at the extremes of its operating margin, as defined by limiting safety system settings.

The experiment design will be evaluated to ensure it will not adversely affect flux shape or reactor coolant flow considerations that were used to define or are implicit in the reactor safety limits.

13.5.2.3 Mechanical Stress Effects

Materials of construction and fabrication and assembly techniques utilized in experiments will be evaluated, as appropriate, to provide assurance that no stress failure will occur from manipulation and conduct of the experiment or as a result of unintended but credible changes of, or within, the experiment. Every experiment or type of experiment, as appropriate, will be evaluated with respect to storage and possible uncontrolled release of any mechanical energy.

13.5.2.4 Material Content of Experiments

Certain kinds of materials which may be used in experiments possess properties which could have significant safety implications. Limitations on the amounts of such materials can limit the consequences of experiment failures. The material content of an experiment will be analyzed and limited, as required, utilizing the following classifications as a guide:

- Radioactive material
- Trace elements and impurities
- High cross section materials
- Highly reactive chemicals (explosives)
- Corrosive chemicals
- Radiation sensitive materials
- Flammable material
- Toxic material
- Cryogenic liquids
- Unknown materials

13.5.2.5 Administrative Controls

Administrative controls are in place to ensure a written description and safety analysis are generated for each experiment type. Each experiment type must be reviewed by a technically competent independent review unit and approved by the Facility Manager or his designated alternate. An experiment type includes repetitive experiments that involve common safety considerations and a similar reactor setup. Acceptance criteria for an experiment include compliance with regulatory requirements (including 10CFR20 and Technical Specifications), GE procedures, and good safety practices. The independent review unit and its modulus operandi is discussed in Section 12.

Administrative controls applicable to all experiments are listed below:

1. Every experiment must have prior explicit written approval by an NTR Licensed Senior Reactor Operator, as determined by the experiment type.

2. Every person not part of NTR Operations who is to carry out an experiment must be certified by the Manager NTR or an NTR Licensed Senior Reactor Operator, as determined by the experiment type, as to the sufficiency of his knowledge and training in procedures required for the safe conduct of the experiment.

3. Detailed written procedures must be provided for the use, or operation of, each experiment facility and each experiment type.

4. The Licensed Reactor Operator at the console must be notified just prior to moving any experiment (or series of experiments as specified by procedures) within the NTR facility.

5. Every experiment removed from the reactor must be subject to a radiation and contamination monitoring procedure, as applicable, that anticipates levels greater than those predicted.

13.5.3 Consequences of Accidental Explosions

The facilities, equipment, and procedures used for experiment programs that involve explosive material are described in Section 10. To provide safe limits for the amounts of explosives permitted in the NTR handling and radiography areas, separate Design Basis Accidents (DBAs) were defined for the south cell, the north room, and the set up room. In general, these DBAs assumed a highly improbable accidental detonation of all explosive devices in the particular area and the consequences are evaluated in terms of both radiological and mechanical effects.

13.5.3.1 Radiological Consequences

The radiological consequences of an accidental detonation of an explosive device are essentially nonexistent. Induced activities in explosive materials, structural materials containing the explosive, or structures used in neutron radiography are extremely small considering thermal neutron fluxes of 2×10^6 nv and normal exposure times of 10^3 seconds. However, if sufficient other sources of radioactive materials are present in the immediate area and become dispersed or airborne during the accidental detonation, the radiological consequences could be serious. Operations at the NTR include neutron radiography of plutonia fuel pins and capsules containing significant amounts of fission products. Evaluation of the DBAs indicates that while it is virtually impossible to involve these materials in the accident, it is prudent to exclude these large sources of radioactive material from any area in which explosive devices are being handled.

Small amounts of radioactive materials (e.g., uranium contained in fission chambers or irradiated samples used in various experimental programs) may be safely stored in the south cell or the north room during the neutron radiography of explosives. By limiting these quantities to 10 curies of radioactive materials and to 50 grams of uranium, the health and safety of the general public will in no way be compromised. Storage locations are at least 5 feet from any explosive handling position and are normally either in concrete block caves or small lead casks. While accidental detonation of explosive devices might cause minor damage to the storage structures, the probability of releasing even a small percentage of the radioactive material from their contents is negligible. Assuming a 1% release and stable atmospheric conditions (inversion), maximum site boundary doses are less than 20 mRem to the thyroid and 1 mRem to the whole body under this most pessimistic combination of circumstances. No radioactive materials other than those produced by neutron radiography are permitted in the set up room if explosive devices are present.

13.5.3.2 Mechanical Consequences

The primary safety criterion is that complete simultaneous detonation of all explosive devices in a particular area will not increase the probability or consequences of accidents previously analyzed or create the possibility of a different type of accident not previously analyzed. While minor structural damage and possible injury to personnel will occur in the immediate area, damage to the reactor core, graphite pack, or control system is not expected, and injury to personnel is minimized. Damage to the reactor is prevented by limiting the amount of explosive material allowed in the particular areas (south cell, north room, and set up room) and by design

and construction of an additional shield structure (south cell). Potential injury to personnel is minimized by strict adherence to safe explosive handling procedures. The mechanical safety analyses are discussed in detail in Appendix A and show that neutron radiography of explosives can be accomplished safely in the reactor facility by limiting both the total quantity of explosive materials in pounds of equivalent TNT and the distance of the explosive material from sensitive components and structures.

Summarizing Appendix A gives the following limits:

South Cell $W = (D/2)^2$

> Where W is the weight of explosive in pounds of equivalent TNT; D is the distance from blast shield in feet; and $W \leq 9$ pounds, $D \geq 3$ feet.

North Room (without MSM) $W = D^2$

> Where W is the weight of explosive in pounds of equivalent TNT;
> D is the distance from north room wall in feet; and
> $W \leq 16$ pounds, $D \geq 1$ foot.

North Room (with MSM)	$W \leq 2$ pounds of equivalent TNT in the MSM and a maximum 16 pounds of equivalent TNT in the north room
Set up Room	$W \leq 25$ pounds of equivalent TNT

13.5.3.3 TNT Equivalence

The equivalence of an explosive material to TNT on a gram basis is determined by ratioing various parameters of the explosive to those of TNT. These parameters include brisance, ballistic mortar, trauzel test, and detonating velocity, and are described in "Properties of Explosives of Military Interest," AMCP 706-177. This report contains pertinent data on many types of explosives and is used as a primary reference document. The equivalent grams of TNT for an explosive being handled or radiographed is determined by the following:

$$\text{Gram equivalent TNT} = \text{grams of explosive} \times \frac{\text{Parameter of explosive}}{\text{Parameter of TNT}}$$

where the ratio of parameter is chosen to be the highest value of the brisance, ballistic mortar, trauzel test, or detonating velocity ratios.

If data are not available on the explosive, or the composition is proprietary, a factor of 2 is used for the parameter ratio, which is conservative and higher than any value found in AMCP 706-177.

13.5.3.4 Reactivity Effects

There are no reactivity effects directly associated with neutron radiography of explosive or other materials. Objects undergoing inspection are located at relatively large distances from the reactor and have no effect on core reactivity. Even the large shutter in the south cell may be moved during reactor operation without affecting core reactivity. Some minor reactivity effects are associated with the neutron radiography beam preparation devices. Under normal circumstances, shock waves from accidental detonation of explosives will be attenuated sufficiently to make movement of the beam preparation device highly improbable. It is also noted that the reactivity added during removal or expulsion of the beam preparation device from the core region is included in the total amount that would be available, as discussed in Subsection 13.4.3. Therefore, the consequences would be less severe than those analyzed, which assumed 0.76$ step insertion both with and without scram.

13.5.4 Experiment Limitations

Safety oriented limits and restrictions applicable to experiment facilities and experiment programs follow. The limits and restrictions presented are derived from the reactor and experiment safety analyses, nearly 40 years of experience in conducting experiments at the NTR, and sound engineering practice. The majority of these limits are contained in the Technical Specifications. Adherence to the limits and restrictions below is mandatory and provides assurance that:

(1) There is no anticipated mode of experiment operation that will endanger the health or safety of the general public or plant personnel.

(2) No experiment will be performed that involves a technical specification change or an unreviewed safety question (as defined in 10 CFR, Section 50.59).

(3) A proposed experiment type will be evaluated in detail and its execution controlled so as to reduce any radiation exposure to the public and plant personnel to the lowest practicable level.

13.5.4.1 General Experiment Requirements

(1) 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. Records of such evaluation and approval shall be maintained.

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

(3) Experimental capsules to be utilized in the experimental facilities shall be designed or tested to ensure that the pressure transients, if any, produced by any possible chemical

reaction of their contents and leakage of corrosive or flammable materials will not damage the reactor.

(4) No experimental objects shall be inside the core tank when the reactor is operating at a power greater than 0.1 kW.

(5) Experimental objects located in the fuel loading chute shall be secured to prevent their entry into the core region.

13.5.4.2 Reactivity Limits

(1) Requirements pertaining to the reactivity worth of experiments are as follows:

a. The sum of the potential reactivity worths of all experiments which coexist plus the reactivity available from control rods and coolant temperature shall not exceed 0.76$.

b. No experimental object shall be moved during reactor operation unless its potential reactivity worth is known to be less than 0.5$ and the operation is performed with the knowledge of the licensed operator at the console. 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.

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

(2) The potential reactivity worth of experiments shall be assessed before irradiation. If the assessment warrants, the reactivity worth of the experiment shall be measured and determined acceptable before reactor full-power operation.

13.5.4.3 Explosive and Flammable Material Lists

(1) The maximum amounts of explosives permitted in the NTR facilities are as follows:

a. South cell: $W = (D/2)^2$ with $W \leq 9$ pounds and $D \geq 3$ feet

b. North room (without MSM): $W = D^2$ with $W \leq 16$ pounds and $D \geq 1$ foot

c. North room (with MSM): W = 2 pounds in the MSM and a maximum 16 pounds in the north room

Set up room: W = 25 pounds

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

D = Distance in feet from the south cell blast shield or the north room wall.

(2) A maximum of 10 curies of radioactive material and up to 50 grams 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 5 feet from any explosive device. Radioactive materials other than those produced by neutron radiography of the explosive devices and imaging systems are not permitted in the set-up room if explosive devices are present.

(3) Unshielded high-frequency generating equipment shall not be operated within 50 feet of any explosive device.

(4) The c nulative radiation exposure for any explosive device shall not exceed 3×10^{12} n/cm^2 from thermal neutrons and 1×10^4 roentgens from gamma.

(5) 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 watt. Experimental facilities containing flammable materials shall be vented external to the reactor graphite pack.

13.6 EXPERIMENT DESIGN BASIS ACCIDENT

13.6.1 Introduction

The material quantity limits, clad requirements, operating limits, and required safety equipment for irradiation experiments at the NTR has been developed based on the radiological criteria given in Regulatory Guide 2.2.[25] This analysis specifically addresses the limits for singly and doubly clad plutonium-fueled capsules and shows the capability of the facility and site to accommodate a radioactive material release. The limits are dependent on the physical form of the plutonium and the filtration efficiency of the NTR stack HEPA filter system.

13.6.2 Accident Description

Regulatory Guide 2.2, Part c.2.a ("Material Content of Experiments") describes the release event as:

> ". . . .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. . . ."

and 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."

Therefore, the design basis accident for an experiment in the NTR is described as follows:

(1) Experiment material is Pu-239. Limits have been 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 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 local close-capture system to ensure release is through the HEPA filters. For purposes of this evaluation, the NTR HEPA filter system filtration efficiency for 0.3-μm-diameter particles is conservatively assumed to be only 99%.

3. The release fractions of Pu fuel and fission products to the environment are assigned as follows:

	Powder (%)	Pellet (%)
Release from capsule to reactor cell:		
Pu-239	100	0
Noble Gas	100	100
Iodine	100	25
All Remaining Fission Products	100	0
Release from reactor cell to stack:		
Pu-239	1	1
Noble Gas	100	100
Iodine	100	100
All Remaining Fission Products	1	1

4. 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

A more complete list of critical organ dose limits is given in Table 13-2.

5. The unrestricted area exposure will result from the diluted-dispersed cloud of isotopes released from the NTR stack, which reaches the nearest site boundary under type F meteorological conditions at 1 m/sec over a 2-hour period.

6. The restricted area exposure will result from the submersion in and inhalation of 3.06 x 10^{-3}%, of the isotopes released from the NTR stack at a flow rate of 1000 cfm* for a period of 5 minutes. The bases for this postulated exposure are as follows:

   a. It is assumed that the total release which will occur will be uniformly distributed over the two hours following the experiment failure.

*The stack flow rate may be as high as 3000 cfm, but 1000 cfm is used for conservatism in calculating the concentration at the stack and the resulting exposure.

Table 13-2

MAXIMUM ALLOWABLE ORGAN DOES 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	0.15	1.5
Lower Large Intestines	0.15	1.5
Skin of Body	0.3	3.0

b. The fission products from this release will cause high-activity alarms on the stack monitors.

c. The NTR operator will respond to the stack alarms and announce an area evacuation over the building public address system if the stack noble gas monitor indicates a concentration of $>6 \times 10^{-3}$ µCi/cc (6×10^{-10} amps). The basis for this action point is that a 5-min exposure to 6×10^{-3} µCi/cc of Kr-87 would be roughly equivalent to the maximum allowable quarterly average of 520 MPC hours, i.e.,

$$\frac{6\ 10^{-3}\ \mu\mathrm{Ci/cc}}{1 \times 10^{-6}\ \mu\mathrm{Ci/cc/MPC}} \times \frac{5\ \mathrm{min}}{60\ \mathrm{min/h}} = 500\ \mathrm{MPC-hours}$$

d. Evacuation to an upwind location will remove personnel from the stack concentration of released isotopes. On-site exposures can be controlled by use of the site alarm system and evacuation procedures.

13.6.3 Calculation Method

Computer Code DOSE77 is used 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 which results from irradiation of fissionable material. The input required for DOSE77 to calculate the fission product inventory and organ doses resulting from exposure to the released isotopes from a Pu-239 experiment in the NTR is:

(1) Capsule operating power = 60 watts (for 1 gram Pu-239 at a thermal neutron flux of 10^{12} n/cm^2-sec).

(2) Capsule operating time = 1 day (arbitrary selection).

(3) U-235 thermal fission cross section = 572 barns (Note: The code inputs the Pu-239 fission cross section as a function of the U-235 cross section).

(4) The initial ratio of Pu-239 fissions to U-235 fissions = 10^6.

(5) Thermal neutron flux = 1×10^{12} n/cm^2-sec.

(6) Decay time after shutdown = 2 minutes (arbitrary).

(7) Meteorology type is Pasquill Type F with a wind speed of 1 m/sec (for the boundary dose).

(8) Diameter of radioactive particles = 1 micron.

(9) Effective release height is at ground level.

(10) The breathing rate of the exposed subject is 347 cc/sec.

(11) The dose commitment time = 50 years.

(12) The distance from the nearest boundary point to the NTR stack is 510 meters.

(13) The fraction of the uniform 2-hour release which is inhaled in 5 minutes of exposure to the NTR stack concentration of nuclides is:

$$\text{Fraction of release inhaled} = \frac{(30\ \text{sec})\,(347\ \text{cc/sec})}{(120\ \text{min})(1000\ \text{ft}^3/\text{min})(2.83\text{x}10^4\ \text{cc/ft}^3} = 3.06\text{x}10^{-5}$$

(14) The various release or inhalation quantities and fractions used for the various DOSE77 cases are:

Boundary Evaluation – 2-hour release quantities

	Powder Form	Pellet Form
Noble Gas	100%	100%
Halogens (except Iodine)	1%	0%
Iodines	100%	25%
Volatile Solids	1%	0%
All Remaining Fission Products	1%	0%
Pu-239	6.17E-4 Ci	0%

Restricted Area (Stack Concentration) Evaluation – 5-min inhalation and exposure

	Powder Form	Pellet Form
Noble Gas	3.06E-3%	3.06E-3%
Halogens (except Iodine)	3.06E-5%	0%
Iodines	3.06E-3%	7.65-4%
Volatile Solids	3.06E-5%	0%
All Remaining Fission Products	3.06E-5%	0%
Pu-239	1.89E-8 Ci	0%

(15) The soluble forms of the isotopes were used to evaluate the doses to most organs, but insoluble forms were used in a few cases to check the doses to the lungs, stomach, small intestines, upper large intestines, and lower large intestines.

13.6.4 Results

DOSE77 was run by using a unit quantity of Pu-239 (1 gram) for the various cases of interest. The cases are:

(1) The 510-meter-boundary doses due to the mixed fission products released from both:

a. the power form, and

b. the pellet form of fuel.

(2) The 510-meter-boundary doses due to the Pu-239 released from the powder form of fuel.

(3) The doses resulting from exposure to stack concentrations of fixed fission products from both:

a. the powder form and,

b. the pellet form of fuel.

(4) The doses resulting from exposure to stack concentrations of Pu-239 from the powder form of fuel.

The organ doses from these runs are tabulated in Table 13-3. These doses were then compared with the maximum allowable doses given by the Regulatory Guide 2.2, Code of Federal Regulations 10CFR20, and the International Commission on Radiological Protection (ICRP 9).[26] These maximum allowable organ doses are shown in Table 13-2.

The maximum allowable quantities of Pu-239 (in grams) allowed in both single- and double-clad experiments in both solid and powder forms, assuming a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec, were calculated as the ratio of the maximum allowable dose to the highest appropriate calculated organ dose for one gram of Pu-239. These limits are given in Table 13-4. For example:

> The maximum allowable thyroid dose for an on-site employee from failure of a single-clad experiment is 3 Rem, and the calculated thyroid dose to an on-site employee resulting from the failure of a singly clad, pellet-form NTR experiment containing one gram of Pu-239 and irradiated in the center of the reactor (experiment fission power = 60 W) is 6.58 Rem. Therefore, the Pu-239 limit for a singly clad experiment operated at a 10^{12} n/cm^2-sec flux for one day is 3 Rem x 1 gram/6.58 Rem = 0.46 gram; since the double encapsulation dose limit is 10 times the single-clad limit, the Pu-239 limit for a doubly clad experiment operated at a 10^{12} n/cm- sec flux for one day is 4.6 grams.

In the case of a pellet-form experiment, the thyroid dose due to the iodine isotopes is the limiting factor for establishing capsule operating limits. Therefore, pellet- form experiment limits are flexible as long as the iodine inventory and resulting thyroid dose at the stack do not exceed those which result from a 24-hour irradiation of 0.46 gram (for a single-clad) of Pu-239 at a thermal neutron flux of 10^{12} n/cm^2-sec (at a power of approximately 28 watts); i.e., if the flux and/or irradiation time is reduced, the fuel quantity can be increased, as presented later. The same criteria apply to solid-form uranium- fueled experiments.

In the case of powder-form experiments, over 99% of the limiting liver dose results from the plutonium and less than 1% is due to iodine isotopes. Therefore, powder-form capsules are limited by the quantity of fuel which is released from the stack. Again, the same criteria apply to powder forms of uranium-fueled experiments, and uranium weight limits are higher than plutonium due to, among other things, the specific activity difference between the two fuel types.

Table 13-3

ORGAN DOSE SUMMARIES FOR BOUNDARY AND RESTRICTED AREA EXPOSURE TO NTR EXPERIMENT ISOTOPES

(1 gram Pu-239, Experiment Fission Power = 60 watts, Irradiation Time = 24 hours)

Organ	Soluble[a] Isotopes 50 Year Organ Dose (Rem)			
	Pellet-form Capsule		Powder-form Capsule	
	Boundary (MFP Only)	At Stack (MFP Only)	Boundary (MFP+Pu-239)	At Stack (MFP+Pu-239
Total Body				
Inhalation	2.92E-4	1.17E-2	3.59E-2	1.45E0
Submersion	4.54E-3	2.41E-1	9.89E-3	4.72E-1
Kidneys	1.44E-3	5.85E-2	1.53E-1	6.19E0
Liver	8.63E-4	3.50E-2	4.77E-1	1.93E-1
Bone	4.37E-4	1.76E-2	7.42E-1	3.00E-1
Lung[a]	3.54E-3	1.40E-1	2.55E-1	1.03E1
Thyroid	1.61E-1	6.58E0	6.47E-1	2.63E-1
Stomach	5.84E-4	2.47E-2	2.39E-3	1.01E-1
Small Intestine[a]	2.62E-4	1.10E-2	1.37E-3	5.72E-2
Upper Large Intestine[a]	1.02E-4	4.15E-3	9.27E-4	3.77E-2
Lower Large Intestine[a]	2.16E-4	8.77E-3	2.18E-3	8.81E-2
Skin, Submersion	7.59E-3	4.74E-1	1.54E-2	8.13E-1

[a]Most organ doses are calculated for soluble forms of the isotopes; the lung, small intestine, pper large intestine and lower large intestine dose are calculated for insoluble forms.

Table 13-4

SINGLE CLAD NTR EXPERIMENT Pu-239 LIMITS BASED ON CALCULATED ORGAN DOSES AND ALLOWABLE DOSES

	Calculated Limits, Grams Pu-239			
	Based on Pellet-form		Based on-powder Form	
Organ	Boundary	Stack	Boundary	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	686.50	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 to 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

The uranium limits in experiments are established using the same methods described for the plutonium evaluation. The uranium and plutonium experiment limits are shown as both quantity limits and one-day operating power limits in Table 13-5, assuming a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec. The operating power is calculated from the quantity limits with the following equation:

$$P = (8.3 \times 10^{10})\,(g)\,(\sigma_f)\,(\phi)$$

where

P = operating power of experiment (Watts)

g = mass of fissile material (grams)

σ_f = thermal fission cross section of fissile material (cm^2)

ϕ = thermal neutron flux (c/cm^2 sec)

For the NTR experiment, the values used are

$U^{235}\ \sigma_f = 572 \times 10^{-24}\ cm^2$

$Pu^{239}\ \sigma_f = 742 \times 10^{-24}\ cm^2$

$\phi = 1 \times 10^{12}\ n/cm^2\text{-sec}$

Figure 13-15 gives gram limits for a single clad, powder-form U-235 experiment in the NTR test facilities as a function of the thermal neutron flux and the continuous irradiation time of the experiment. These curves were generated by plotting the calculated limits resulting from 15 different power/time cases run with the RIBD/DOSE77 codes. These curves expand the single-condition limit to a large range of limits, depending on irradiation conditions. Double clad and/or solid-form U-235 limits can be derived from these curves as a direct ratio of the limits given for the specific cases in Table 13-5.

For example, the limit for a doubly clad, solid pellet of U-235 irradiated in a flux of 1×10^{11} n/cm^2-sec for 1 hour would be:

$$(28.6 \text{ grams}) \frac{6.0 \text{ g}}{0.15 \text{ g}} = 1{,}144 \text{ grams}$$

The 15 conditions and resulting thyroid doses per gram of U-235 for the restricted area exposure are:

Thyroid Dose per Gram U-235 (Rem)

Flux	10 seconds	1 hour	1 day	7 days	1 year
3×10^8	3.196E-7	3.145E-4	5.992E-3	1.352E-2	2.018E-2
1×10^{11}	1.066E-4	1.049E-1	1.998E0	4.508E0	6.732E0
2×10^{12}	2.132E-3	3.097E0	3.995E-1	9.016E-1	1.355E-2

The U-235 limits for these cases, based on a 3-Rem thyroid dose limit are:

U-235 Limits (Grams)

Irradiation Time

Flux	10 seconds	1 hour	1 day	7 days	1 year
3×10^8	9.385E-6	9.539E-3	5.007E-2	2.219E-2	1.487E-2
1×10^{11}	2.814E-4	2.860E-1	1.502E0	6.655E-1	4.458E-1
2×10^{12}	1.497E-3	1.431E0	7.509E-2	3.327E-2	2.214E-2

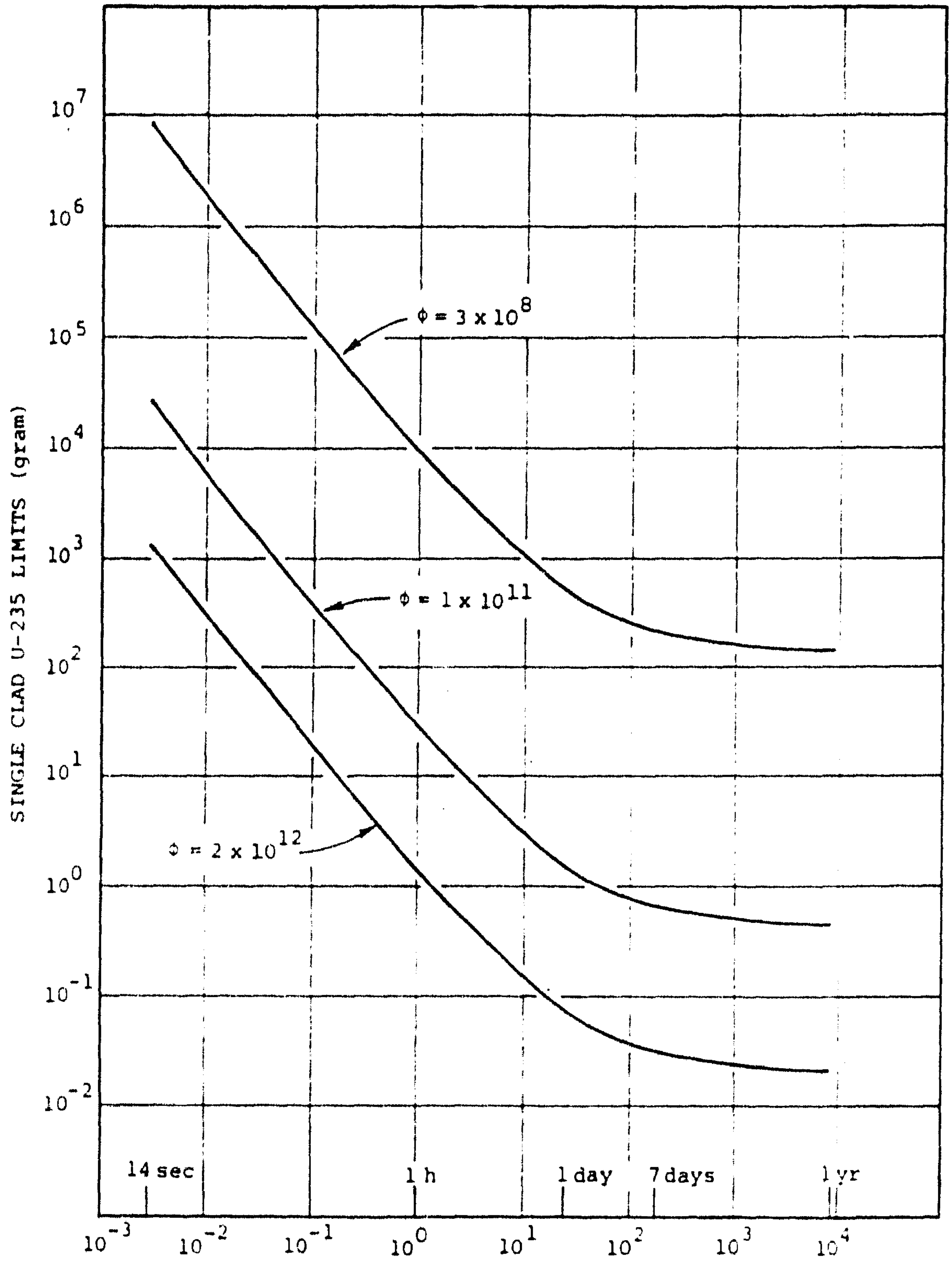


Figure 13-15. NTR Single-Clad, Powder-form U-235 Experiment Quantity Limits as a Function of Thermal Neutron Flux and Irradiation Time

Table 13-5

Pu-239 AND U-235 EXPERIMENT LIMITS

	Pu-239 Limits		U-235 Limits	
Capsule Form	**Quantity**	**24-hour Power**	**Quantity**	**24-hour Power**
Single-Clad				
Solid Pellet	0.46 g	28 watts	0.60 g	28 watts
Powder	0.078 g	5 watts	0.15 g	7 watts
Double-Clad				
Solid Pellet	4.6 g	280 watts	6.0 g	280 watts
Powder	0.78 g	50 watts	1.5 g	70 watts

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

13.6.5 Summary

The NTR experiment design basis accident analysis is based on the single-mode failure of a capsule which contains Pu-239 and/or U-235. Based on the conservatively assumed 5-min restricted area exposure of an on-site employee to the stack concentration of the released isotopes (averaged over 2 hours), the on-site organ dose determines the maximum quantity of Pu-239 and/or U-235 allowable in a capsule. The limits are dependent on the physical form of the fuel and the number of clads used for fuel containment.

The limit for a power-form plutonium experiment is determined by the plutonium release, i.e., greater than 99% of the controlling organ dose (liver) results from the released Pu -239. All other cases (solid Pu or any form of uranium) are limited by the thyroid dose caused by the released iodine isotopes. Therefore, the fueled experiment limits for all but powdered forms of plutonium are flexible with respect to the total quantity of material, but must be controlled with respect to operating power and time based on the iodine inventory.

13.7 REACTOR SAFETY LIMITS

13.7.1 Introduction

Safety limits for operation of the NTR are developed in this section. The safety limits presented also provide the basis for determining and specifying the Limiting Safety System Settings (LSSS) for important process variables.

Safety limits are developed for the reactor power, the only important measurable process variable with safety significance for reactor operation. Other process variables, namely core coolant inlet temperature and reactor primary flow rate, have no significant effect upon the safety criterion over the entire range of core flow conditions, including natural circulation.

In this section, the safety criterion of Departure from Nucleate Boiling is discussed. The critical heat flux relationship and the thermal-hydraulic computer model used in the NTR safety limit analysis are described. The resultant safety limit curves are presented. Instrument uncertainties are applied to the safety limit curves to provide the LSSS for steady-state reactor operation.

13.7.2 Criterion for Development of Safety Limits

Departure from Nucleate Boiling (DNB) has been selected as the most relevant criterion for development of safety limits for operation of the NTR. DNB is that stage of the boiling phenomenon when sufficient liquid is unable to reach the heating surface due to the rate at which vapor is leaving the surface. This restriction of the liquid flow causes an abrupt surface temperature rise above the saturation temperature in a heat-flux controlled situation.

The safety limits for the reactor power are chosen to restrict the actual heat flux in the hottest fuel element coolant passage below the DNB surface heat flux to preclude any subsequent fuel cladding damage due to a rise in surface temperature. The Departure from Nucleate Boiling Ratio, DNBR, is the ratio between the surface heat flux at DNB and at operating conditions; thus

$$\text{DNBR} = \frac{\text{DNB surface heat flux}}{\text{operating surface heat flux}}$$

It was necessary to use two different correlations to evaluate the DNB for the NTR. The steady-state DNB condition is found to occur with saturated bulk boiling in a substantial portion of the core and is accompanied by a significant void fraction. The postulated reactivity transients presented in Subsection 13.4 reach a DNB heat flux with the core coolant significantly subcooled.

13.7.3 Analysis for Development of Safety Limits

13.7.3.1 Steady-State Critical Heat Flux Relationship

The steady-state safety limit analysis required a DNB heat flux correlation which is applicable to low-velocity, low-pressure saturated boiling with a significant void fraction. As cited in Reference 27, Macbeth[28] developed an empirical correlation of experimental data which presents the critical heat flux[29] as a linear function of the mass quality at the hottest surface location. This correlation, which accounts for steam quality, is superior to other correlations which ignore the effect of void fractions and consider only other physical properties of the coolant. The Macbeth correlation states that the critical heat flux is proportional to the mass velocity in the low mass velocity region. The NTR core operates in the low mass velocity region for all operating conditions. The optimized correlation is

$$\text{DNB} = \frac{(H_{fg})\,(G \times 10^{-6})^{0.5}}{(135)}\,(1 - X_{max})$$

where

DNB = departure from nucleate boiling critical heat flux (Btu/h-ft^2)

H_{fg} = latent heat of vaporization (Btu/lb)

G = mass velocity (lb/h-ft^2)

X_{max} = maximum quality = $\dfrac{\text{mass of vapor}}{\text{mass of vapor and mass of liquid}}$

The critical heat flux is calculated for the hottest location of the hottest channel.

13.7.3.2 Transient Critical Heat Flux Relationship

The DNB correlation used to evaluate the reactor safety under transient conditions must be applicable to subcooled boiling. Macbeth[28] developed the following empirical correlation for the DNB critical heat flux under subcooled, low-pressure, low-flow conditions:

$$DNB = (0.247)(H_{fg})\left(\frac{\rho v}{\rho \ell}\right)^{0.024}\left(\frac{DG}{L}\right) +$$

$$+(0.00213)\,(\rho\ell)^{1/2}\,(H_{fg})^{1/6}\,(10^{6})^{1/3}\left(\frac{DG}{L}\right)^{2/3}\Delta Hi$$

where

H_{fg} = latent heat of vaporization (Btu/lb)

ρv = density of the vapor (lb/ft^3)

ρ_ℓ = density of the liquid (lb/ft^3)

D = hydraulic diameter (inches)

G = mass velocity (lb/h-ft^2)

L = overall length (inches)

ΔHi = enthalpy difference between saturation temperature and channel temperature (Btu/lb).

For a typical hypothesized transient, as presented in Subsection 13.4, the hottest surface location within the NTR core will attain DNB above a heat flux of 600,000 Btu/h-ft^2.

The postulated accidents presented in Subsection 13.4 were analyzed using a DNB heat flux of 450,000 Btu/h-ft^2).

13.7.3.3 Thermal Hydraulic Computer Model

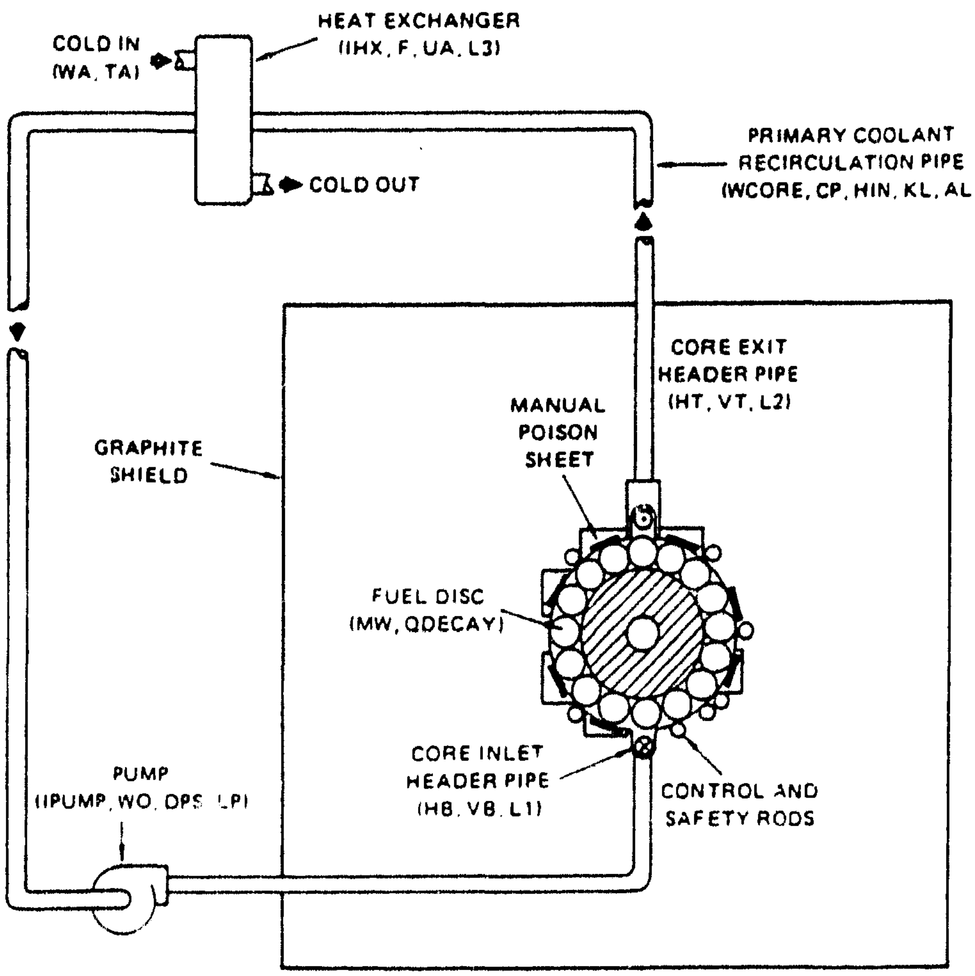
The computer model CORLOOP[30] was developed for the NTR natural circulation analysis. It is also used for forced convection analysis. CORLOOP includes a multi-channel core model and a circulation loop which includes the core, a heat exchanger, and a pump. The core model is illustrated in Figure 13-16; the circulation loop is illustrated in Figure 13-17. Situations with the secondary coolant flow to the heat exchanger on or off and the primary coolant pump on or off were analyzed using the program. When the primary pump is off, the core is cooled by natural circulation. The core model is adapted to the NTR from the GETR multi-channel core model, CORFLO[31].

CORLOOP represents the parallel flow channels between the vertical fuel discs in the NTR core as four channels with six nodes per channel. This provides an adequate grid for determining the values for the measurable process variables. The CORLOOP program has been checked against hand calculations for steady-state conditions, and has been verified against actual operating conditions for the NTR.

13.7.4 Safety Limits

The safety limit for the NTR was determined for two different kinds of events. The first analysis considers the steady-state high power operation of the reactor for various boundary conditions. The second type of analysis considers the behavior of the reactor during various postulated transient events. This second analysis involves the indirect application of the safety limit concept. For the transient analysis, a scram trip point is assumed for important process variables, mainly reactor power, and a value is chosen for the DNB heat flux. After the transient analysis is performed the integrity of the reactor fuel is evaluated, and the validity of the safety limit is determined.

The steady-state safety limits for the NTR were determined using the CORLOOP computer program. The analysis shows that the critical heat flux for the NTR is a strong function of the reactor power. Figure 13-18 shows, as one would expect, that the departure from nucleate boiling ratio approaches unity as the reactor power increases. Figure 13-19 shows the trend of increasing void fraction with increasing reactor power. The analysis shows, however, that the critical heat flux for the NTR is not significantly affected by the core flow rate, or the core inlet temperature (bottom plenum temperature), as shown in Figures 13-20 and 13-21. The reactor power, therefore, is the only important measurable process variable to be limited. The safety limit for the reactor power assures that the actual heat flux never approaches the DNB heat flux.



(For definitions of variables listed above, see Reference 30, <u>CORLOOP Multi-Channel Core and Loop Model</u> by A. I. Yang)

Figure 13-16. Multi-Channel Core Model of NTR (CORLOOP)

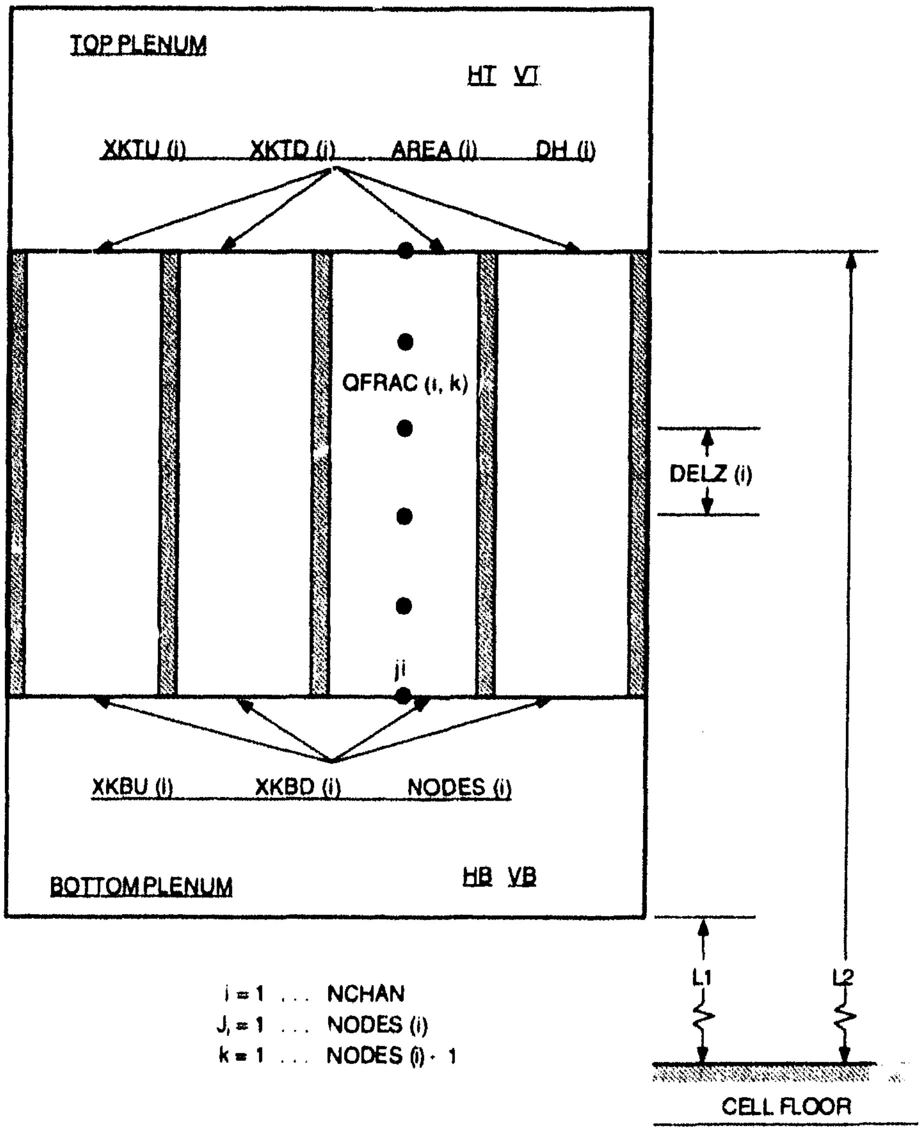


Figure 13-17. Schematic Diagram of the NTR Circulation Loop Model (CORLOOP)

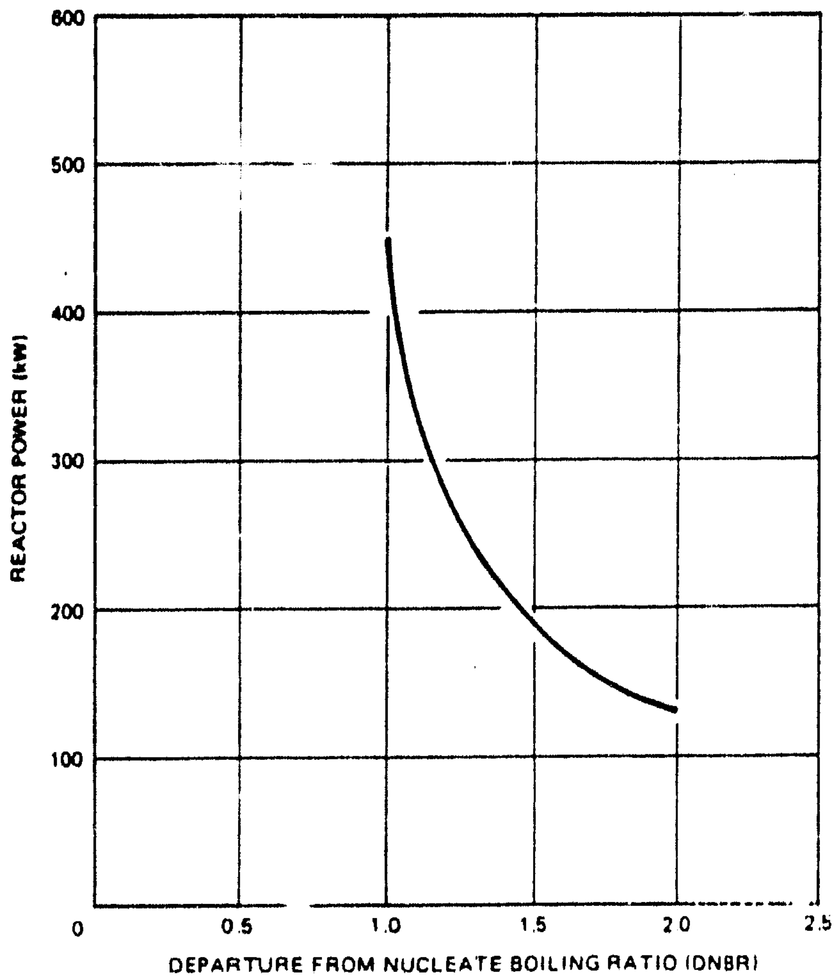


Figure 13-18. Reactor Power Versus DNBR = Depart from Nucleate Boiling Ratio

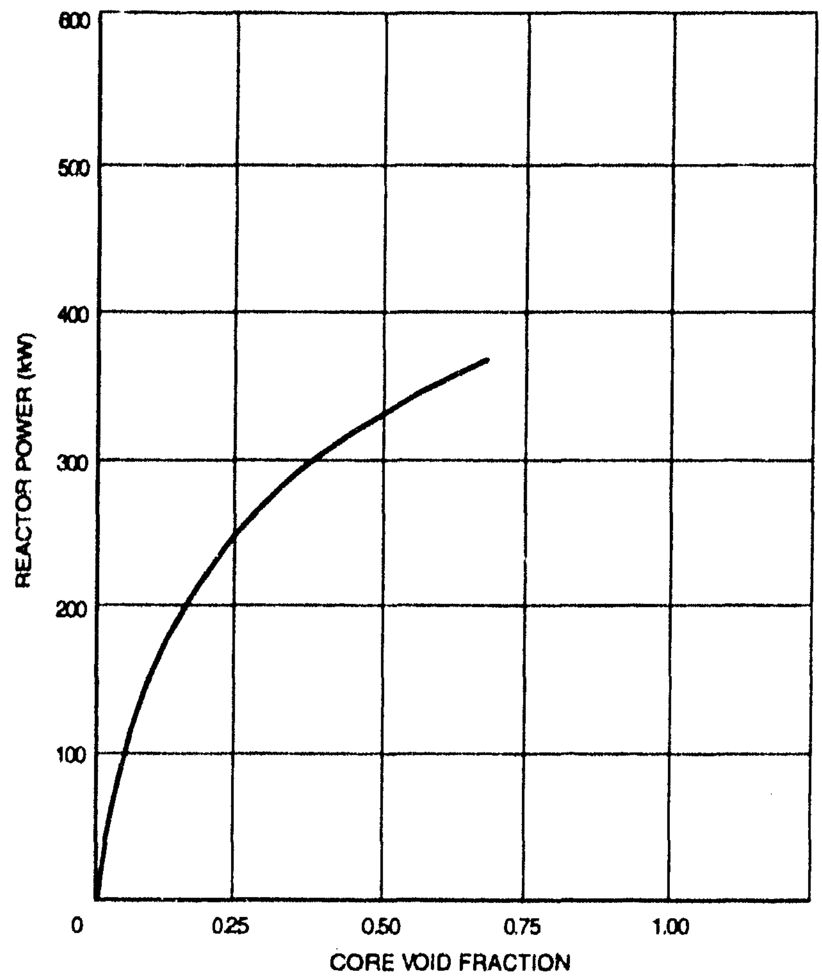


Figure 13-19. Reactor Power Versus Core Void Fraction

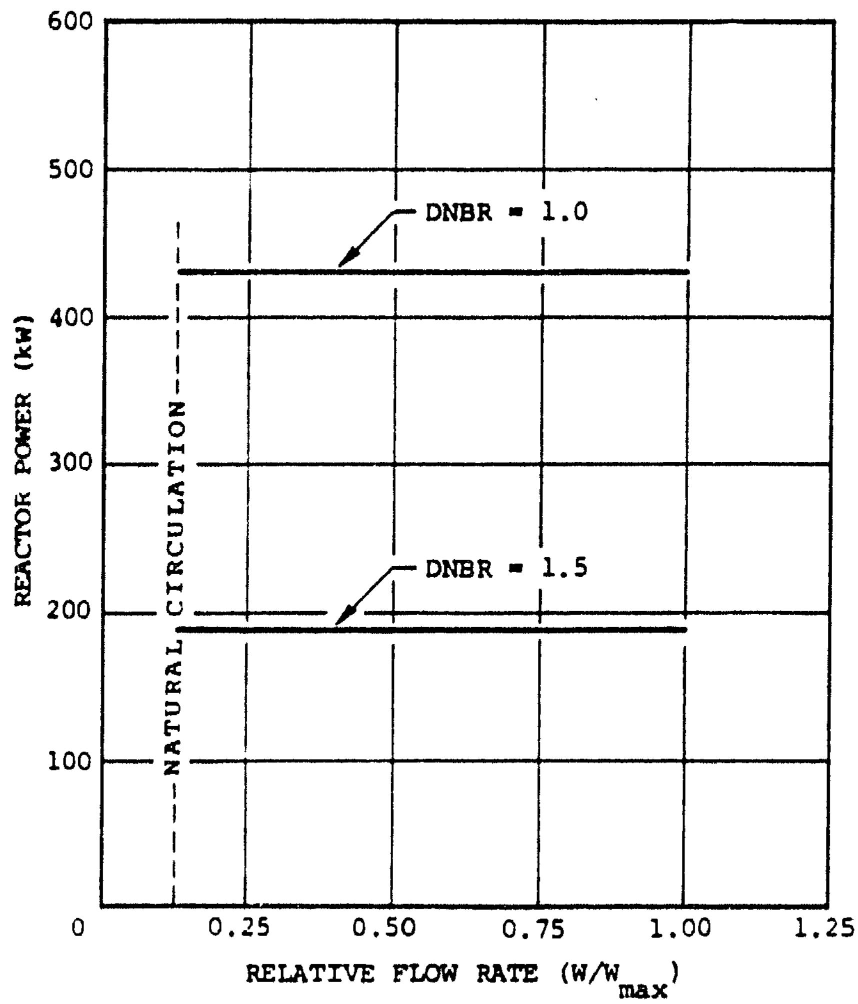


Figure 13-20. Reactor Power Versus Relative Flow Rate

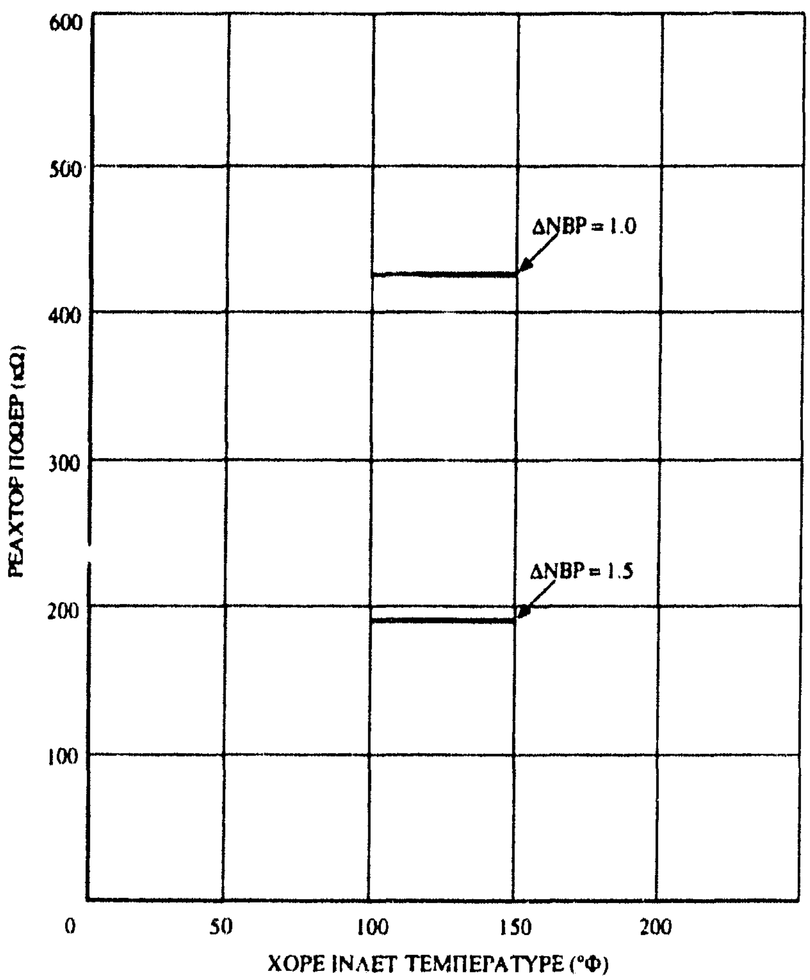


Figure 13-21. Reactor Power Versus Core Inlet Temperature

In the analysis, the actual heat flux has been determined from the CORLOOP computer model. The DNB heat flux has been determined from the Macbeth correlation for pool boiling conditions. At a reactor power of 430 kW, the DNBR reaches unity. At a reactor power of 190 kW, the DNBR = 1.5. As shown in Figure 13-20, a 30% decrease in DNBR corresponds to more than a 100% increase in reactor power. The analytical uncertainties present in the results represent the RMS error of the empirical correlation, the physical differences in the flow conditions between the NTR core and the experimental apparatus used in two phase flow research, and the assumptions incorporated in the four-channel computer model of the core. A safety limit which corresponds to a minimum allowable value of DNBR = 1.5 provides a conservative and satisfactory margin to more than compensate for any analytical uncertainties. The steady-state safety limit for reactor power is 190 kW, as shown in Figures 13-20 and 13-22.

The curves presented in Figures 13- 20 and 13-22 do not extend below a relative flow rate of ~0.12. This flow rate is the value which would exist if the reactor is operated at 190 kW with the pump turned off. Steady-state operation below this flow rate at a power level of 190 kW or greater is not possible. Likewise, the steady-state operation of the reactor with inlet temperatures of less than 100°F, or greater than 150°F, is not possible at these power levels with reasonable secondary coolant inlet temperatures. Values of reactor power, flow rate, and core inlet temperature which fall outside these bounds do not represent steady-state conditions and should be evaluated on the basis of the transient safety limits and analyses.

The transient analysis presented in Subsection 13.4, which required a reactor scram, were all performed assuming a scram occurred at 150 kW, a scram delay time of 0.200 seconds, and a DNB heat flux of 450,000 Btu/h-ft^2. None of the anticipated abnormal occurrences or postulated accidents resulted in fuel damage using these values. The transients were not reevaluated using the more realistic DNB heat flux of 600,000 Btu/h-ft^2 presented in Subsection 13.6.4.2.

13.7.5 Instrument Uncertainties

The instrument uncertainties are presented in Table 13-6 for each of the measured variables under consideration. These uncertainties, determined when the process variables were at their normal values and assumed unchanged over all acceptable LSSS, are both the systematic and random types. In general, systematic uncertainties include biases in calibrations, "standards," signal transmitters, and recorders. Random errors include drift of instrument settings, signal-to-noise ratio of instrument electrical output, instrument instability, and operator-to-operator variation in interpretation within least count.

The uncertainty values for the three measurable process variables used in the heat balance for reactor thermal power determination were determined by extracting the square root of the sum of the squares of the individual uncertainties in the contributing measurements.

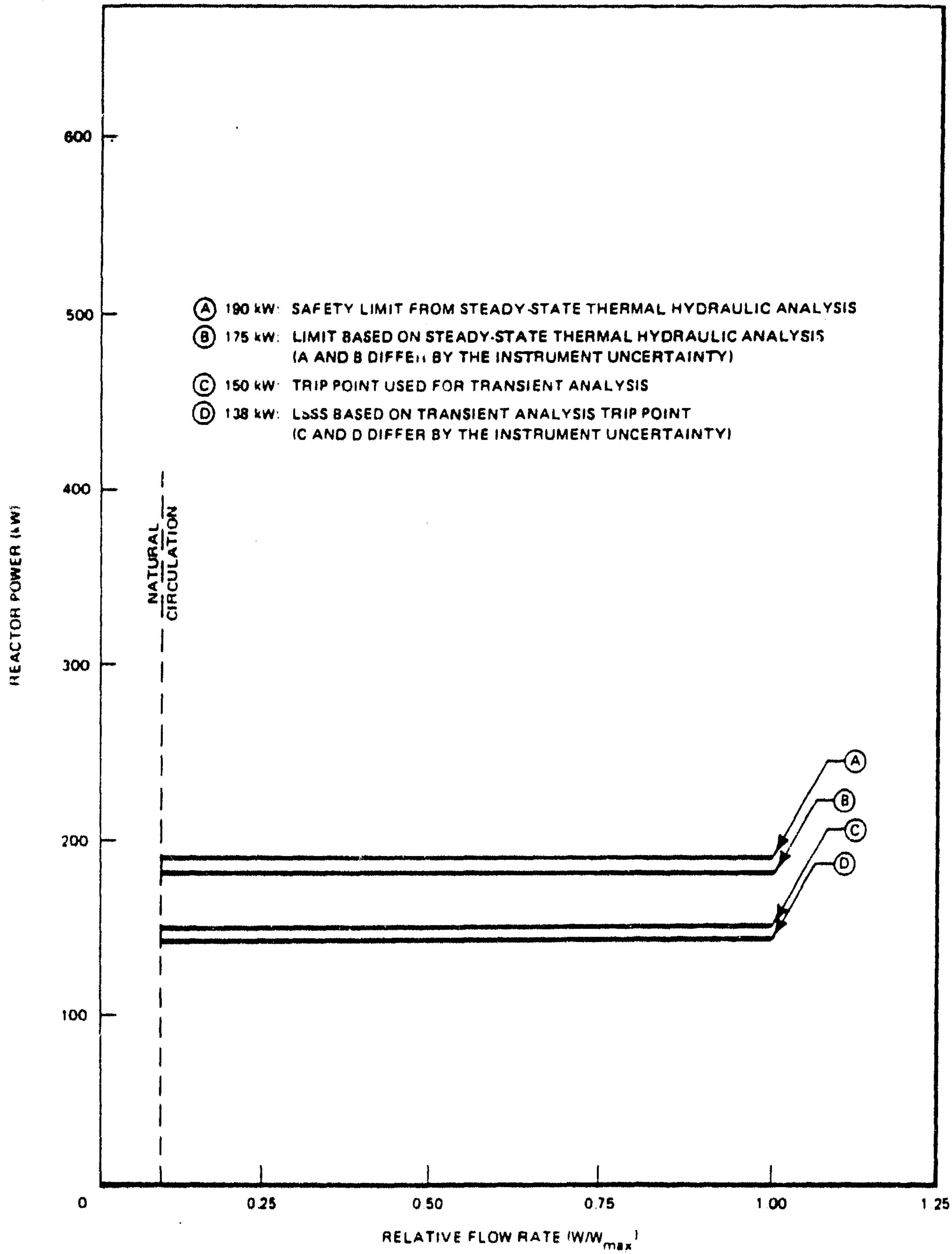


Figure 13-22. LSSS and Safety Limit for Reactor Power in Terms of Relative Core Flow Rate

Table 13-6

UNCERTAINTIES IN THE PRESENT METHODS FOR MEASURING IMPORTANT PROCESS VARIABLES

Reactor Power (Flux Monitoring)		
Compensated Ion Chamber	negligible	
High Voltage Power Supply	negligible	
Picoammeter Setting Accuracy	±0.25%	
Picoammeter Calibration Accuracy	±4%	
Picoammeter Long-Term Drift	±0.25%	
Net Reactor Power Uncertainty		±4.0%
Reactor Coolant Core Inlet and Outlet Differential Temperature		±3%
Reactor Primary Flow		±1.0%
Overall Instrument Uncertainty for Reactor Power =		±8.0%

13.7.6 Limiting Safety System Settings (LSSS)

The LSSS have been chosen to ensure that reactor scram is initiated in time to prevent exceeding the safety limit for reactor thermal power during normal operation and anticipated abnormal occurrences, or violation of safety criteria during postulated accidents. The safety margin (the difference between the safety limit and the LSSS) includes systematic and random types of instrument uncertainties, and, for transient events, also includes the effect of safety system delay times. The LSSS appears as Curve D in Figure 13-22. The limiting safety system setting for the reactor power is 125 kW over the entire range of core flow conditions, including natural circulation. The value of 125 kW is a conservative setpoint well below the trip point of 150 kW used in the transient analysis of the postulated accidents and is used rather than the 190 kW steady-state safety limit because it is more restrictive.

Any quasi-steady event comprising changes in any process variables may be analyzed using the safety limit curves regardless of the rationale used for postulating the event. The most severe anticipated off-normal, quasi-steady event is one in which the reactor power is at its least favorable value of 150 kW. For this highly unlikely operating condition, the DNBR = 1.8.

As a result of certain postulated accidents, the reactor power may exceed the specified safety limit without causing damage to the reactor fuel. The amount by which the safety limit may be exceeded is a time-dependent variable, with each case evaluated individually. Application of the limiting safety system settings for reactor power ensures that no damage to the fuel will occur for any transient resulting from the postulated accidents.

Curve A in Figure 13-22 shows the safety limit based upon the steady-state thermal hydraulics analysis. Curve B shows the safety limit curve adjusted to account for instrument uncertainties. Curve C in Figure 13-22 shows the scram trip point used in the transient analysis. Curve D, the LSSS curve, represents Curve C adjusted to account for instrument uncertainties.

14.0 TECHNICAL SPECIFICATIONS

Technical Specifications have been developed for the NTR which follow the format of the 1990 revision to American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1. Operation of the reactor within the limits of the Technical Specifications will not result in offsite radiation exposure in excess of 10CFR20 guidelines. Operation within the Technical Specifications also limits the likelihood and consequences of malfunctions and assures the health and safety of the on-site personnel and the public, and protection of the environment.

15.0 FINANCIAL QUALIFICATIONS

15.1 FINANCIAL ABILITY TO CONSTRUCT A NON-POWER REACTOR

Not Applicable

15.2 FINANCIAL ABILITY TO OPERATE A NON-POWER REACTOR

The actual costs and estimates of costs to operate the NTR for the first five years of the renewal period are considered by General Electric Nuclear Energy to be proprietary. However, based on 40 years of experience in operating the NTR at VNC, costs of NTR operation are well known and understood. General Electric is a multi-billion net worth, global corporation which is capable of assuming total operating costs for the NTR for the duration of the license renewal period.

Current and anticipated sources of funding for the NTR, would come from the sales of services to various customers in the areas of neutron radiography, irradiation of test and research materials, and reactivity testing. The sales of these services cover all of the NTR operating costs, excluding certain landlord type costs, which are required for operations of the site as a whole, and the expense of which is not directly attributed to the NTR. These landlord costs include: cleaning, landscape maintenance, utilities, facilities maintenance, and security.

15.3 FINANCIAL ABILITY TO DECOMMISSION THE FACILITY

General Electric Company submits a letter to the NRC, to update decommissioning costs for all GE facilities under NRC regulatory oversight, and to demonstrate financial responsibility under the *Self Guarantee Rule* (58 FR 68726; 12/29/93) which became effective on January 28, 1994.

The last letter submitted to the NRC dated March 17, 1997, guaranteed as a *self-guaranteeing licensee* and as *parent-guarantor*, the decommissioning closure care for the NTR, license R-33. The current cost estimate for decommissioning the NTR was stated in that letter as 1.23 million dollars. The letter was submitted by Dennis D. Dammerman, Senior Vice President – Finance, and included a copy of GE's 1996 Annual Report.

16.0 OPERATING EXPERIENCE

The NTR first went critical on November 15, 1957. The reactor was designed as an experimental physics tool to advance the company's nuclear energy programs. It has performed innumerable experiments, sample irradiations (including rocks from the moon) reactivity measurements, sensor calibrations and uranium enrichment analyses.

On August 30, 1966, the first neutron radiograph was performed. In July, 1969, the maximum operating power was increased from 30 kW to 100 kW.

The NTR is an easy to operate and easy to maintain facility. It is a low temperature, low pressure, low heat reactor so components are not unduly stressed. The primary system is constructed of aluminum and stainless steel components and the primary coolant system is maintained at a high purity so corrosion is not an accelerated concern. The reactor is also very accessible so that control rod and safety rod drives may be inspected and maintained regularly. These inspections and tests have demonstrated that the NTR can be operated safely and that components with degraded performance may be detected and replaced.

16.1 REACTOR FUEL

The reactor fuel is uranium-aluminum alloy with an aluminum cladding. This type of fuel has an excellent history and does not release large amounts of fission gas without melting. Corrosion of the aluminum in water is minimized when the pH of the water is 6.5. The reactor primary coolant pH is maintained between 4.8 and 8.7 by maintaining the water purity below 10 μmhos/cm. High specific conductivity can be tolerated for short time durations during unusual circumstances.

Fuel failure is monitored by the primary coolant water conductivity, a gamma monitor near the primary coolant pipe in the reactor cell and the stack effluent monitor. Smaller failures may be detected by annual sampling of the primary coolant for tramp uranium.

16.2 SAFETY RODS

The safety rods have an exceptional performance history. The rods are accessible for inspection and testing. The tests include scram times, low-current magnet separation, rod withdrawal time, limit switches and interlocks, residual magnetism of the electromagnets, sliding friction, and spring force. Any component of the Safety Rods may be replaced if required.

16.3 CONTROL RODS

The Control Rods have also had an exceptional performance history. The rods are accessible for inspection and testing. The tests include limit switches and interlocks, automatic insertion, rate of withdrawal, and position indication. Any component of the Control Rods may be replaced if required.

16.4 CONFINEMENT

The reactor cell or confinement building is largely a passive system. The door and seal are inspected periodically and the delta pressure gage and alarm are periodically calibrated and tested. In addition, the stack discharge air flow is measured periodically and its effluent monitoring system is periodically calibrated and the alarm setpoints checked. Any component of the confinement system may be replaced if required.

16.5 INSTRUMENT AND CONTROL

The instrument and control system is accessible for periodic inspection, calibration and testing. Any component may be replaced when required. The testing includes periodic tests of each instrument which initiates a scram or alarm and a neutron source check of the picoammeters.

16.6 RADIATION AREA MONITORS (RAM)

The RAMs are accessible for periodic inspection, calibration and testing. Any component may be replaced if required. The testing includes alarm setpoints and a radiation source check.

16.7 CONCLUSION

The NTR is a simple, compact, accessible reactor. This is evidenced by the replacement of the primary, core can which occurred in 1976. At that time, the control rods, safety rods, startup sources and reactor fuel were removed. Most of the graphite blocks were relocated and the aluminum core can and fuel reel assembly were replaced with new units and the reactor was reassembled. The extensive surveillance, testing and calibration program has resulted in a facility with an outstanding record of safe operation. Components with degraded performance can be detected and replaced easily. This assures that the NTR will continue to be a safe facility.

APPENDIX A

EVALUATION OF THE CONSEQUENCES OF ACCIDENTAL EXPLOSIONS

A.1 INTRODUCTION AND GENERAL CONCLUSIONS

A.1.1 Introduction

The use of explosive material within a reactor facility has been recognized as a significant safety concern by GE. Careful evaluation of the consequences of accidental detonation of such devices has been performed. In response to a letter of June 23, 1971 from the Nuclear Regulatory Commission (NRC), GE submitted an evaluation of the consequences of accidental explosions and proposed facility operation restrictions and safety controls for review and approval. Following is a summary of that submittal, plus some updated information.

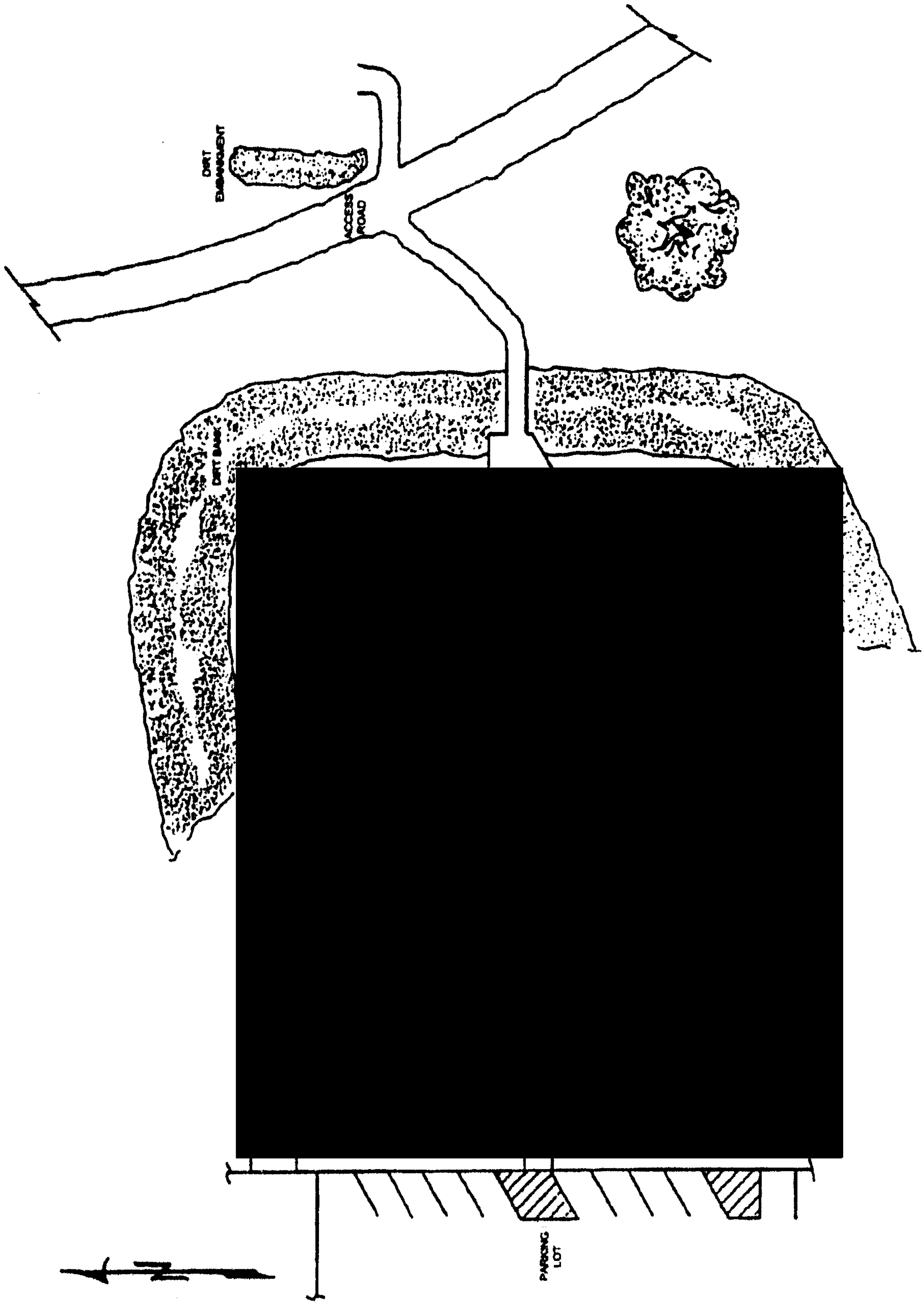
A.1.2 General Conclusions

Because of the many safety features provided and the strong administrative controls applied to the operation of the facility, the possibility of an accident involving explosive material is considered remote. On the basis of the descriptive and analytical information provided, and the proven performance of the facility over an extended operating period, it is concluded the operating and safety methods of the Nuclear Test Reactor (NTR) facility provide the reasonable assurance required by the regulations that the health and safety of the public will not be endangered as a consequence of explosive material handling and inspection at the facility.

A.2 FACILITY AND PROCESS DESCRIPTION

A.2.1 General

The Nuclear Test Reactor (NTR) facility is located at the east end of Building 105. The reactor and its control mechanisms are located within a concrete-shielded room designated as the reactor cell. Operation of the reactor is from a console located in the control room. Figure A.2-1 illustrates the floor plan of the NTR and adjoining related facilities. The south cell and north room areas are utilized for neutron radiography exposure facilities. The set-up room is utilized for preparation of material for neutron radiography. Darkroom and office areas are provided to service the facility.



[redacted]

Figure A.2-1. Building 105 Floor Plan

A.2.2 Neutron Radiography of Pyrotechnic and Explosive Devices

Neutron radiography has practical application in the nondestructive quality control inspection of pyrotechnic and explosive devices, particularly for military and aerospace programs. Inspection of pyrotechnic and explosive material is performed on finished or test sample devices. No work is performed which involves loose powder handling, explosive device repair, or modifications. Thermal neutrons are the main component of the reactor flux in the process area. The integrated thermal flux will not exceed 3×10^{12} n/cm^2 and the integrated gamma exposure will not exceed 1×10^4 roentgens. Investigations by Urizar,[1] et al., have shown that these levels should have no effect on explosive materials undergoing neutron radiography.

The exposure of material is controlled by the use of automatic shielding shutters, which close off the access to the reactor beam tube.

A.2.3 Neutron Radiography Facilities

The neutron beams emitted from the horizontal facility of the reactor are used for neutron radiographic inspection. The south beam enters the south cell where facilities are available to perform neutron radiographic inspection of explosive material. The north beam enters the North Room where it is also available for neutron radiographic inspection of explosive devices. These two facilities represent the only locations where neutron radiographic inspection of explosive material is performed.

The neutron beam from either the north or south beam ports is generated and shaped, using beam preparation devices which occupy part of the horizontal facility as well as the beam port penetration through the graphite pack and thermal column. The only portion of the beam preparation devices which could affect the core reactivity is that portion which is in the core section of the horizontal facility. The devices are normally constructed of graphite, lead, and polystyrene. The potential reactivity worth is normally very small, but in any case is included in the potential excess reactivity and is controlled accordingly.

A.2.4 Setup Room

Explosive material received at the facility which is to be processed is normally prepared for radiography in the Setup Room. Most material received at the NTR is processed as soon as possible after receipt and therefore would routinely be considered to be work-in progress.

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[1] J. Urizar, E. D. Loghran, and L. C. Smith, A Study of the Effects of Nuclear Radiation on Organic Explosives (TID-12491).

The set-up room is located outside the Building 105 east exit door and is primarily a preparation room for neutron radiography. Explosive devices which are scheduled to be inspected are brought to this location from customers or Site transportation services. The total explosive material inventory present in the room (or facility, as appropriate) is recorded for the information of all personnel.

The set-up room contains electrically grounded work benches and the floor is painted with electrically conductive paint. Nonsparking tools are utilized when working near and with explosives. Appropriate restriction and warning signs are posted at all room entrances.

A.2.5 Transportation Routes

While explosive material is in process, it will be transported down the hallway between the set-up room and the reactor control room. The material will pass through the south portion of the control room into the south cell for neutron radiography. The south portion of the control room may also be utilized as a staging area for preparing the next neutron radiographic exposure while one is already in process.

Explosive material to be inspected in the north room may be transported directly to the north room. All routes are premarked, as required, with appropriate warning signs for the material being moved.

A.2.6 Radiographic Process Areas

The north room and the south cell are designated as the two process areas for explosive material. The north room is located at the northeast corner of Building 105. It is a sheet-metal building approximately 25 feet by 41 feet by 35 feet high. The neutron beam from the reactor is isolated from the room by a shutter assembly and the Modular Stone Monument (MSM) which is discussed below. When a radiographic exposure is to be made, the shutter control is actuated, the shutter opens, and the neutron beam passes through the 24-inch-diameter reactor cell wall penetration (containing additional shielding for beam improvement) to the object in the MSM being processed. The MSM is a dual neutron radiography facility, allowing the capability of performing neutron radiography on unirradiated or irradiated objects (Figures 10-2 and 10-5*). The design involves six concrete blocks that make up the shield and structural unit. A 12-inch-i.d. stainless steel pipe capped off the bottom penetrates into the ground beneath the MSM for 20 feet. This penetration allows neutron radiography of long objects to be performed by lowering them into the pipe.

Irradiated objects normally arrive at the NTR in large casks which are placed on top of the MSM using the overhead crane. The objects can then be lowered down into the MSM in front of an imaging foil and the neutron radiography is then performed.

*Section 10

Unirradiated objects are moved into a facility on the north end of the MSM, usually by a trolley arrangement. The imaging system is placed behind the object and neutron radiography or irradiation is performed. Appropriate radiation shielding, radiation alarms, air monitors, warning signs, communication systems, and locks are provided in the area for safe operation of the facility.

The south cell is a room approximately 17 feet long by 9 feet wide by 8 feet high which serves as a neutron radiography process area. The cell walls are thick shielding structures to minimize radiation exposure to personnel in the adjoining control room. Objects may be moved into the cell, usually by a trolley arrangement, either from the control room or through a penetration in the south cell east wall. The cell is serviced by radiation alarms, a visual personnel monitoring system and an electric eye entry warning system. The cell is isolated from the reactor neutron beam by a shutter mechanism. When the object has been positioned and the imaging system installed, the shutter mechanism can be opened for the radiographic exposure.

A.3 EXPLOSIVE MATERIALS DESCRIPTION

A.3.1 Classification of Explosive Material

Explosive materials are Class 1.1 through 1.6, as described in Title 49, of the Code of Federal Regulations, regarding transportation of explosives and other dangerous materials. These classifications are assigned to packaged material and are not normally associated with the unpackaged explosive. Thus, explosive material may be noted as Class 1.4 on shipping containers, yet may be considered to be Class 1.1 when removed from that container. Explosive material classifications 1.1 through 1.6 relate to the consequences of initiation and not to initiation sensitivity. As a result, many Class 1.1 explosives are safer to handle than Class 1.3 or 1.4 because of their insensitivity to initiation. Facility handling procedures are based upon the assumption that all explosive material is initiation-sensitive.

A.3.2 Types of Explosive Material Handled

All explosive material handled at the facility is normally in the form of encapsulated powder or pellets. This material is usually installed in a finished manufactured device ready for quality control inspection before use. Sample portions of explosive material and some bulk forms are also inspected. No loose-powder-loaded device modification or detonation testing will be performed at this facility.

Normally, all explosive material will be in the form of squibs, initiators, bolts, blade charges, delays, and fuse material. The major portion of the devices are manufactured for NASA and military applications.

Explosive material received at the facility is reviewed to establish its characteristics, amount, and related safety precautions concerning the device it is loaded within. Following the safety review, the material is carefully uncrated and each piece inspected to assure all safety devices are installed, as required. Only grounded facilities and nonsparking tools will be used in

unpackaging these materials. Strict enforcement of facility explosive handling procedures is mandatory during all explosive material work.

The major portion of the explosive material handled at the facility is insensitive to initiation. Drop tests are performed by customers at their facilities and other safety-related information such as explosive type and amount are a part of the safety review. Some of the explosive devices may be held in the hand when detonated without injury to the holder. The main hazard to personnel in the immediate area is from fine metallic fragments which may become missiles upon detonation of the device. Since strict explosive material handling procedures are enforced at all times, the accidental detonation of a device is highly improbable.

A large portion of the explosive material handled is in the form of fuses or propellants. Should accidental detonation of these devices occur, there would only be smoke and sound emission. This type of material represents a minimal hazard to personnel and the facility.

Personnel handling ordnance devices are provided with special safety equipment, as required. The operation of unshielded, high-frequency generating equipment within 50 feet of any explosive device is prohibited, and radio transmissions are not allowed within 15 feet of explosive material. Appropriate warning signs are posted conspicuously to control transmission, smoking, and other unsafe acts.

A.3.3 Explosive Material Procedures and Controls

The Vallecitos Nuclear Center (VNC) maintains written safety standards governing industrial safety controls on explosive material. Detailed procedures are maintained in the area of routine operations, safety, and emergencies at the facility.

Audit functions are performed, as required, by Regulatory Compliance personnel and the Specialist, Industrial Safety and Hygiene. Additionally, facility inspections are made by insurance inspectors and customer Quality Assurance inspectors and the Defense Logistics Agency to ensure a high degree of safety at the facility.

The NTR facility has a 35-year record of safe explosive material handling. During this period there have been no accidents or injuries associated with explosive material. A continuing program of personnel training provides the NTR with highly trained and disciplined personnel.

A.4 SAFETY ANALYSIS

A.4.1 Design Basis Accidents

To provide safe limits for the amounts of explosives permitted in the NTR handling and radiography areas, separate Design Basis Accidents (DBA) were defined for the south cell, the north room and the set-up room. In general, the DBA assumed a highly improbable accidental detonation of all explosive devices in the particular area and evaluated the consequences in terms of both radiological and mechanical effects

A.4.2 Radiological Consequences

The radiological consequences of an accidental detonation of explosive devices are essentially nonexistent. Induced activities in explosive materials, structural materials containing the explosive, or structures used in neutron radiography are extremely small, considering thermal neutron fluxes of approximately 2×10^6 nv and normal irradiation times of 10^3 seconds. However, if sufficient other sources of radioactive materials are present in the immediate area and become dispersed or airborne during the accidental detonation, the radiological consequences could be serious. Operations at the NTR include neutron radiography of plutonia fuel pins and capsules containing significant amounts of fission products. Evaluations of the DBA indicate that, while it is virtually impossible to involve these materials in the accident, it is prudent to exclude these large sources of radioactive material from any area in which explosive devices are being handled.

Small amounts of radioactive materials (such as uranium contained in fission chambers and irradiated samples used in various experimental programs) may be safely stored in the south cell or the north room during the neutron radiography of explosives. By limiting these quantities to 10 curies of radioactive materials and 50 grams of uranium, the health and safety of the general public will in no way be compromised. Storage locations are at least 5 feet from any explosive handling position and are normally either concrete block caves or small lead casks. While accidental detonation of explosive devices might cause minor damage to the storage structures, the probability of releasing even a small percentage of the radioactive material from their contents is negligible. Assuming a 1% release and stable atmospheric conditions (inversion), maximum Site boundary doses are less than 20 mRem to the thyroid and 1 mRem to the whole body under this most pessimistic combination of circumstances. No radioactive materials other than those produced by neutron radiography are permitted in the set-up room if explosive devices are present.

A.4.3 Mechanical Consequences

The primary safety criterion is that complete simultaneous detonation of all explosive devices in a particular area will not increase the probability or consequences of accidents previously analyzed or create the possibility of a different type accident not previously analyzed. While minor structural damage and possible injury to personnel will occur in the immediate area, damage to the reactor core, the graphite pack, or control systems is not expected and injury to personnel is minimized. Damage to the reactor is prevented by limiting the amount of explosive material allowed in the particular areas (south cell, north room, and set-up room) and by design and construction of an additional shield structure (south cell). Potential injury to personnel is minimized by strict adherence to safe explosive handling procedures. The mechanical safety analyses are discussed in detail in Attachment A-1 and show that neutron radiography of explosives can be accomplished safely in the reactor facility by limiting both the total quantity of explosive materials in pounds of equivalent TNT and the distance of the explosive material from sensitive components and structures.

A summary of Attachment A-1 gives the following limits:

1. South cell $W = (D/2)^2$

   where W is the weight of explosive in pounds of equivalent TNT, D is the distance from blast shield in feet, and $W \leq 9$ pounds, $D \geq 3$ feet.

2. North room (without MSM) $W = D^2$

   where W is the weight of explosive in pounds of equivalent TNT, D is the distance from the north room wall in feet, and $W \leq 16$ pounds, $D \geq 1$ foot.

3. North room (with MSM) $W \leq 2$ pounds of equivalent TNT. Since the distance is stationary, there is no value for D (Figure 10-4*). The limit in the north room outside of the MSM is 16 pounds.

4. Set-up room $W \leq 25$ pounds of equivalent TNT

A.4.4 TNT Equivalence

The equivalence of an explosive material to TNT on a gram basis is determined by ratioing various parameters of the explosive to those of TNT. These parameters include brisance, ballistic mortar, Trauzl test, and detonating velocity, and are described in "Properties of Explosives of Military Interest," AMCP 706-177. This report contains pertinent data on many types of explosives and is used as a primary reference document. The equivalent grams of TNT for an explosive being handled or radiographed is determined by:

$$\text{Gram equivalent TNT} = \text{grams of explosive} \times \frac{\text{parameter of explosive}}{\text{parameter of TNT}}$$

where the ratio of parameters is chosen to be the highest value of the brisance, ballistic mortar, test, or detonating velocity ratios.

If data are not available on the explosive or the composition is proprietary, a factor of 2 is used for the parameter ratio, which is conservative and higher than any value found in AMCP 706-177.

*Section 10.

A.4.5 Reactivity Effects

No reactivity effects are directly associated with neutron radiography of explosive or other materials. Objects undergoing inspection are located at relatively large distances from the reactor and have no effect on core reactivity. Even the large shutter in the south cell may be moved during reactor operation without affecting core reactivity. As discussed in Subsection 2.3 of this appendix, some minor reactivity effects are associated with the neutron radiography beam preparation devices. Since these devices are considered to be experiments, their reactivity worth is controlled in accordance with Subsection 10.2.

Under normal circumstances, shock waves from accidental detonation of explosives will be attenuated sufficiently to make movement of the beam preparation device highly improbable. It is also noted that removal or expulsion of the beam preparation device has already been taken into account in the transient experiment value which is utilized and has been analyzed in the Safety Analysis Report.

ATTACHMENT A1

ENGINEERING ANALYSIS FOR NTR NEUTRON RADIOGRAPHY OF EXPLOSIVES

A1.1 INTRODUCTION AND SUMMARY

The following analyses of blast effects due to highly improbable accidental detonation of explosives during NTR neutron radiography operations provide the basis for establishing limits on the quantity of explosives which may be present in the facility. The safety criteria for the various NTR operating areas are as follows:

South cell	No damage to reactor, thermal column, or graphite pack.
North room	No damage to graphite pack, control rod mechanisms, or support structure.
Set-up room	No damage to south cell or reactor cell.

Based on the determination of the critical impulse capacities of the sensitive reactor structures, the maximum amounts of explosives permitted in the NTR facilities are as follows:

South cell	$W = (D/2)^2$; $W \leq 9$ pounds, $D \geq 3$ feet
North room (without MSM)	$W = D^2$; $W \leq 16$ pounds, $D \geq 1$ foot
North room (with MSM)	$W \leq 2$ pounds; 16 pounds outside the MSM
Set-up room	$W \leq 25$ pounds

where W is the total weight of explosive in pounds of equivalent TNT and D is the distance in feet from the south cell blast shield or the north room wall.

While the actual quantities of explosives handled and inspected by neutron radiography are normally much less than the safe quantities shown above, these limits will not increase the probability or consequences of accidents previously analyzed for the NTR.

A1.2 IMPULSE CAPACITIES

A1.2.1 General

To determine safe quantities of explosives permitted in the NTR neutron radiography and explosive handling areas, the impulse capacities of structures related to safety criteria in Section 1 must be analyzed. These structures are the blast shield in the south cell, the control rod support plate in the reactor cell, and the south cell shield wall, the latter of which is shown in Figure A1-1.

The impulsive loading delivered to surfaces exposed to blast fronts is generally expressed in terms of impulse per unit area, i.e.,

$$\Gamma = \frac{1}{A}\int_{0}^{t} Fdt$$

so that the momentum imparted to a structural component will be the product of the areal impulse density of the reflected blast front and the exposed area of the component, i.e.,

$$\Gamma A \int_{0}^{t} \quad Fdt = mv_0$$

Since

$$(\Gamma A)^2 = m^2 v_0^2 \rightarrow v_0^2 = 1/2\left(\frac{\Gamma A}{m}\right)^2$$

the component gains a kinetic energy of

$$K = 1/2\ mv_0^2 = 1/2\,m\left(\frac{\Gamma A}{m}\right)^2$$

The resistive forces restraining the affected component must be capable of dissipating this kinetic energy within an envelope which precludes destructive transfer of kinetic energy to adjacent sensitive components.

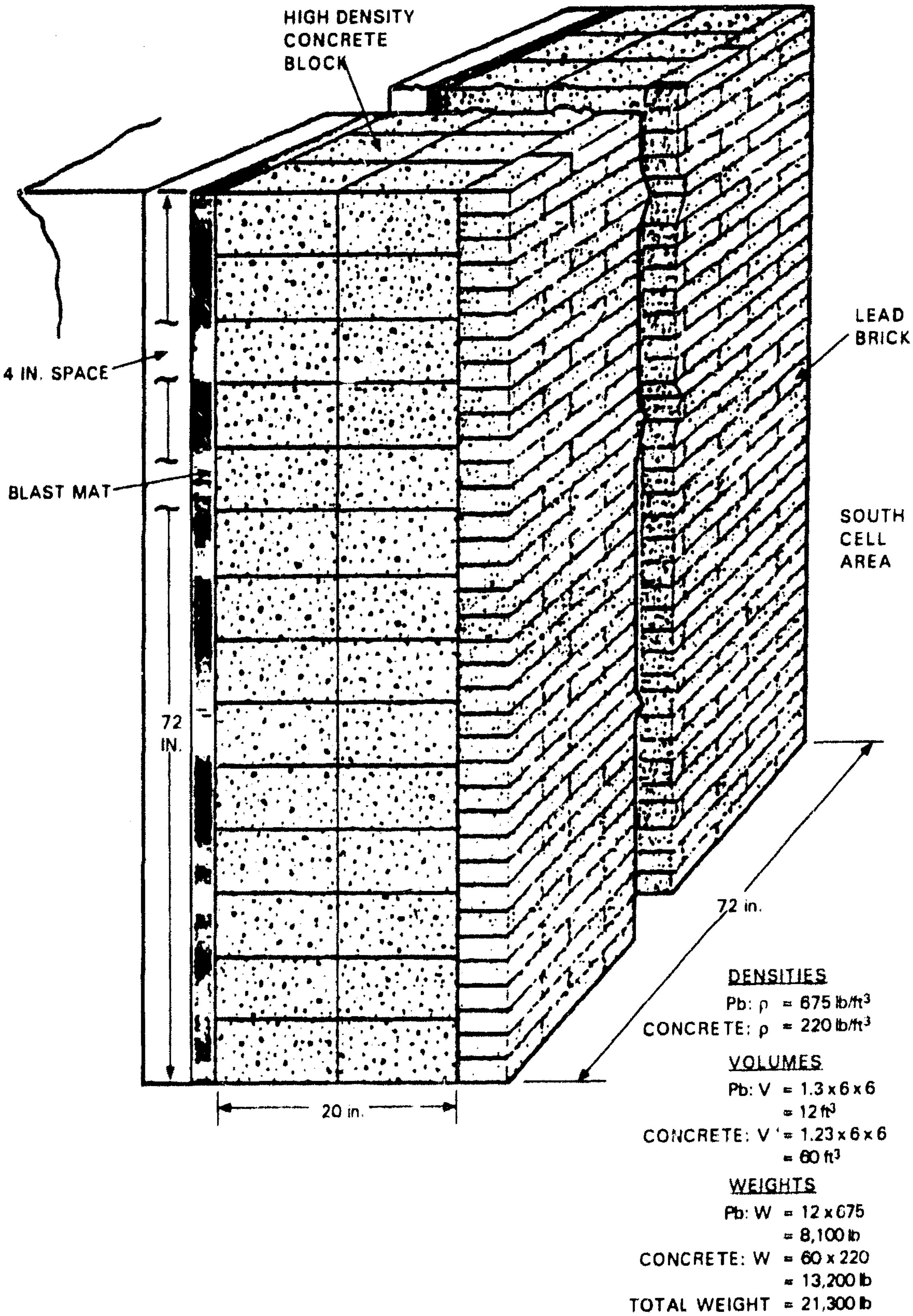


Figure A1-1. Shield Wall

The blast shield is a 67-in. by 69-in. vertical grid containing 14 1-in. by 1-in. by 3.1875-in. angles in a 3-in. by 2-in. by 0.25-in. angle frame. It is positioned 4 inches from the face of the NTR thermal column and secured in place by 14 0.75-in. anchor bolts in a friction hook arrangement between the frame and concrete walls.

The resultant translation of the wall will be accomplished against a resistive force at least as great as the friction force developed at the wall bottom. (The torque on the blast grid friction hooks ensures the actual resistive forces are much greater.) Since the inner face of the blast grid is separated from the reactor face by 4 inches, the work done by the restraint during the 4-in. traversal would be

$$E = E_f \; S = \mu_f \; WS$$

where

F_f = Resistive friction force

S = Displacement of shield wall

μ_f = Frictional coefficient

W = Weight of shield wall

Therefore,

$$K \leq \mu_f \; WS \text{ or } 1/2 \; \frac{g}{W} (I\,A)^2 \leq \mu_f \; WS \text{ and}$$

$$(I\,A)^2 \leq \frac{2\,\mu_f \; W^2 \; S}{g} \text{ or } I \leq \frac{W}{A} \left(\frac{2\,\mu_f \; s}{g} \right)^{1/2}$$

where A is the shield wall front area.

With

$$W = 21{,}300 \text{ lb}, \; A = 5{,}184 \text{ in.}^2, \; \mu_f \geq 0.2 \text{ and } S = 4 \text{ in.}$$

$$I = \frac{21{,}300}{5{,}184} \left(\frac{2 \times 0.2 \times 4}{32.2 \times 12} \right)^{1/2} \; lb_f \,/\, in.^2 = sec$$

$I_{allow} = 0.265$ psi – sec. ←

A1.2.3 North Room

The two major areas of concern regarding potential reactor damage as a result of blast effects arising from detonation of an explosive charge in the north room are:

1. Possible projection of missiles toward the support plate housing the reactor control rod cluster.

2. Possible blast wave impact loadings on the control rod support plate which might result in sufficient deformation to preclude proper functioning of the control rod mechanisms.

The shock wave arising from an explosion at the neutron radiography station in the north room (without utilizing the MSM) would be propagated through the 24-inch-diameter beam penetration in the 5-foot-thick concrete wall separating the reactor cell from the north room. An air control flange covers the beam penetration at the inner face of the reactor cell north wall. This flange is designed with a hinge mounting to allow it to swing open under a significant external overpressure condition, thereby precluding its becoming a missile source.

The problem of determining safe quantities of explosives in the north room then becomes one of restricting charge weight and distance parameters to values such that the propagated shock wave loading on the control rod support structure does not exceed its impulsive load capacity.

The 0.75-inch-thick control rod support plate is taken to be the critical reactor component since its entire frontal area is exposed to the blast wave propagated through the beam penetration and across the reactor cell. Deformation and failure of this support plate would jeopardize the mechanical integrity of the control rods.

Structural damage to the support plate would occur as a result of an impulse loading exceeding the resistance of the support plate. The top, bottom, and sides of the support plate are rigidly supported relative to the center and their inertia is sufficient to preclude their moving significantly relative to the center. For deformation of the support plate center to occur under an impulse loading, the loading must be sufficient to accelerate the central material particles to velocities such that stresses in the membrane exceed the dynamic yield strength for aluminum.

The critical areal impulse density associated with this critical velocity is given by Rinehart[1] as

$$I_c = \rho \, \delta \, V_c$$

[1] J. S. Rinehart and J. Pearson, *Behavior of Metals Under Impulsive Loads*, Dover Publications, New York, 1965.

where

ρ = density of the membrane material

δ = tnickness of membrane

V_c = critical velocity

A stress wave associated with an impulsive load propagates through the material such that

$$\sigma = \rho\, C\, V$$

where

σ = stress

C = velocity of sound in the material

V = relative particle velocity associated with the stress wave.

These two relations may be combined to express the critical impulse in terms of the dynamic yield strength for the material, i.e.,

$$I'_c = \delta\, \sigma_y / C$$

and since C is equivalent to $(E/\rho)^{1/2}$, where E is the modulus of elasticity, the resultant critical impulse loading becomes:

$$I'_c = (\rho / E)^{1/2}\, \delta\, \sigma_y$$

where

ρ = 5.22 slugs/ft^3

E = 1.58×10^9 lb$_f$/ft^2

σ_y = 2.02×10^7 lb$_f$/ft^2

for aluminum, and since $\delta = 0.75$ inch or 0.0574 feet, the critical load becomes:

$$I_c = \left(\frac{3.22 \text{ slug / ft}^3}{1.58 \times 10^9 \dfrac{\text{slug / ft}}{\text{sec}^2 - \text{ft}^2}} \right) \left(0.0574 \text{ ft} \times 2.02 \times 10^7 \text{ lb}_f \text{ / ft}^2\right)$$

i.e.,

$$I'_c = 72 \text{ psf-sec or } 0.504 \text{ psi-sec.}$$

Because of uncertainties arising from the fact that the supports other than at the support plate bottom are not truly rigid and the fact that the numerous penetrations of the plate provide opportunities for stress concentrations, an over-all safety factor of 10 will be applied in calculating safe TNT charge weights at the object position in the north room, i.e.,

$$I'_c = 0.0504 \text{ psi-sec.}$$

The following pertains to an accidental explosion where the explosive device is confined in the Modular Stone Monument (MSM). Since the explosive is confined, it is conservatively assumed that all the blast energy is directed through the port. The degree of conservatism in this assumption is difficult to ascertain without a more extensive analysis. It is our judgment that this might be a factor of 2 or 3, but is unlikely to be an order of magnitude greater.

Having assumed all the blast energy is directed through the port, the analysis assumes that the wave follows a spherical divergence. The scaling laws were used to predict the total impulse from normal reflection at the control rod support plate. This was checked by comparison to the curve in Figure A1-2 and gave identical results for small charge weights and slightly more conservative results for larger charge weights.

The allowable loading of the plate was evaluated by treating the plate as a simply supported rectangular plate subjected to a uniform impulse. Since the energy absorption capacity of the plate is small compared to the energy in the blast wave, the plate must reflect essentially all of the blast energy and the impulse is computed based upon this total reflection. The maximum stress (membrane plus bending) and the plate deflection were set at reasonable limits to assure integrity. The resulting allowable specific impulse of 0.041 lb.sec/in.^2 is within 20% of the allowable selected 0.05 lb.sec/in.^2. The approach taken to select this value was used instead of using one-tenth the critical impulse (based upon causing material scabbing).

A1.2.4 Set-up Room

Laing[2] has investigated the response of 24-in. thick reinforced concrete dividing walls separating explosive storage bays. In the case of walls 20 feet long and 12 feet high (approximate dimensions of the 24-in. reinforced concrete separating the south cell from the set-up room), Laing shows that a TNT-equivalent charge weight of 800 pounds detonated at a distance of 3 inches from the wall would bring the wall to incipient failure.

The set-up room TNT-equivalent charge weights will be restricted to a total of 25 pounds. In view of these circumstances, no damage to the south cell or the reactor would be expected as the result of an accidental explosion of 25 pounds of TNT in the set-up room.

A1.3 SAFE CHARGE WEIGHTS AND DISTANCES

A1.3.1 General

Considerable data are available on blast parameters for TNT explosions in free air at sea level. Furthermore, the 2W method for determining surface-reflected shock front parameters from free air data is widely accepted as valid. For this analysis, the allowable TNT weight will be taken as one-half the weight whose free air explosive output delivers critical areal impulse densities to the exposed surface of vulnerable components.

By using the data from Figure A1-2 obtained from Ammann and Whitney,[3] and by knowing the value of the critical impulse, the allowable TNT weights and corresponding separation distances are determined.

A1.3.2 South Cell Blast Shield

For the critical impulse value of 0.265 psi-see, the following data were determined from Figure A1-2.

Allowable Weight (lb)	Free Air Weight (lb)	$I/W^{1/3}$ (psi-ms/lb$^{1/3}$)	Z (ft/lb$^{1/3}$)	R (ft)
2.5	5.0	155	1.65	2.8
4.0	8.0	132	1.85	3.7
6.4	12.8	113	2.12	5.0
9.0	18.0	101	2.25	5.9

[2] E. B. Laing, *Design of Ammunition Maintenance Facility*, Annals of the New York Academy of Sciences, Vol. 152, 1968, p. 556.

[3] Ammann and Whitney, *Industrial Engineering Study to Establish Safety Design Criteria for Use in Engineering of Explosive Facilities and Operations*, a report submitted to Picatinny Arsenal, Dover, New Jersey, 1963

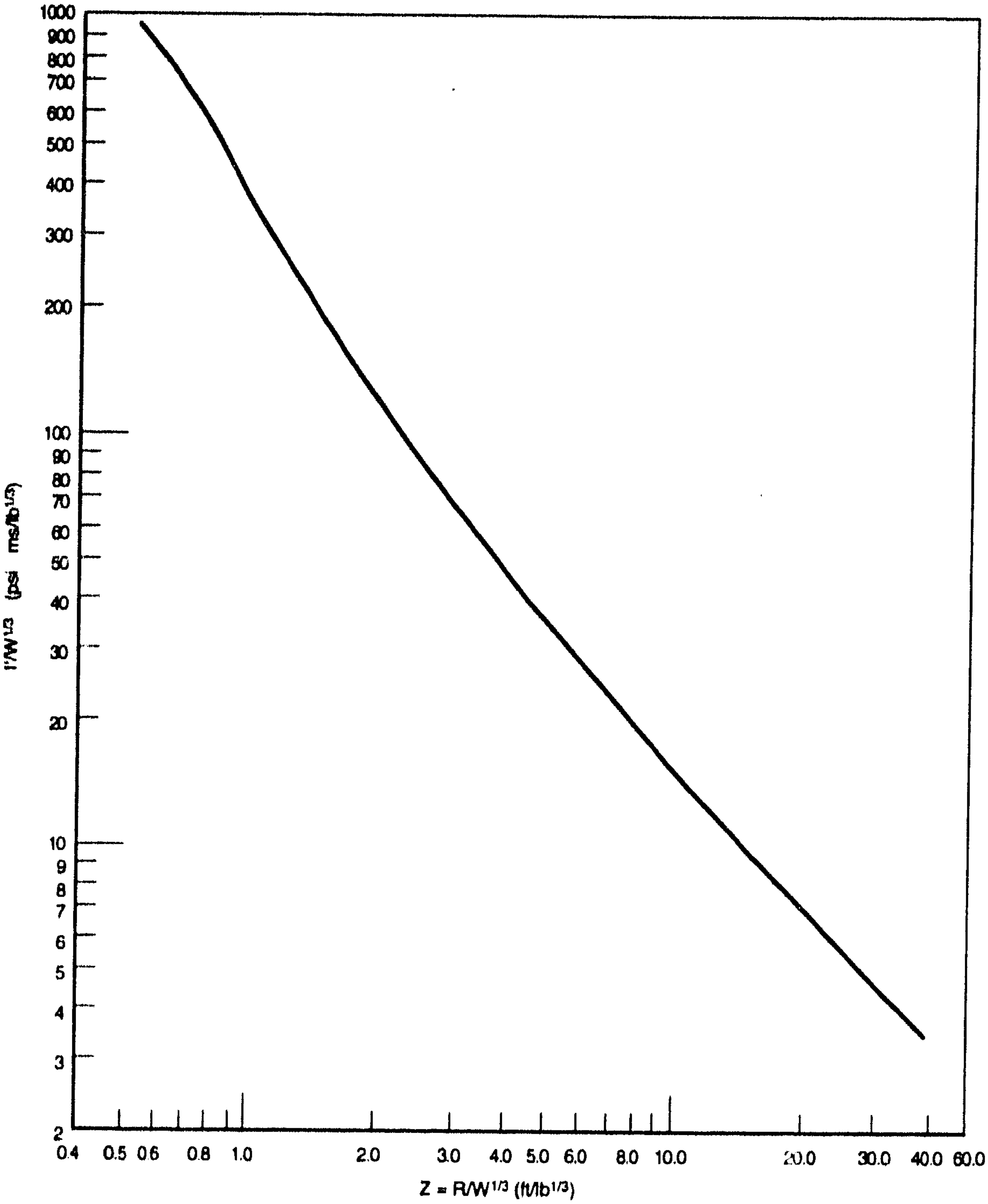


Figure A1-2. Explosive Criteria

A close approximation to these data takes the form of

$$W = (D/2)^2$$

and is conservative at all distances. Normal operating procedures for the south cell indicate that the additional limits of $D \geq 3$ feet and $W \leq 9$ pounds are appropriate.

A1.3.3 North Room (Without MSM)

Although extremely conservative, the areal impulse density of the blast wave emerging from the 24-inch-diameter beam port into the reactor cell will be assumed to be attenuated only by an expansion of its frontal area to a value equal to the area of the control rod support plate (~50 ft^2); i.e., credit will be taken for attenuation of the blast wave areal impulse density by a factor of 16 in undergoing the large, abrupt expansion at the cell north wall and propagation across the 12-foot space separating the wall from the control rod support structure. It should be noted this is a factor of 2 less than the geometric attenuation associated with an explosion occurring at the inner face of the wall.

No credit will be taken for attenuation of the blast wave in its traversal of the 5-foot-long beam port through the wall so that the criteria areal impulse density at the mouth of the north room penetration becomes:

$$I_i = 16\, I'_c = 16 \times 0.05 = 0.8 \text{ psi-sec.}$$

Again, using Figure A1-2 the following data were determined:

Allowable Weight (lb)	Free Air Weight (lb)	$1/W^{1/3}$ (psi-ms/$lb^{1/3}$)	Z (ft/$lb^{1/3}$)	R (ft)
1.35	2.7	575	0.75	1.0
4.75	9.5	378	0.96	2.0
9.75	19.5	296	1.1	3.0
17.0	34.0	247	1.23	4.0

A close approximation to these data takes the form of

$$W = D^2$$

and it is conservative for all distances. Normal operating procedures for the north room without the MSM indicate the additional limits of $D \geq 1$ foot and $W \leq 16$ pounds are appropriate.

A1.3.4 North Room (With MSM)

This analysis summarizes the safe charge weight for a confined explosion in the north room MSM.

A1.3.4.1 Determination of Pressure and Impulse at the Control Rod Support Plate

Using scaling laws where d = 12 ft

$$d_r = \frac{d}{W_S^{1/3}} (2 \times 10^6)^{1/3} = \frac{1510}{W^{1/3}}$$

Duration of positive pressures

$$t_s^+ = \frac{t_{rs}^+ \; W^{1/3}}{(2 \times 10^6)^{1/3}} = \frac{t_{rs}^+ \; W^{1/3}}{126} \qquad \text{Subscript s} \equiv \text{static}$$

$$t_d^+ = t_{rd}^+ \; \frac{W^{1/3}}{126} \qquad \text{Subscript d} \equiv \text{dynamic}$$

As an approximation, the incident impulses associated with the static over pressure and dynamic pressure is:

$$I_{so} = \frac{1}{2} P_{so} \; t_s^+$$

$$I_d = \frac{1}{2} P_d \; t_d^+$$

The total impulse during impact (assumes most of the energy reflected)

$$I_t = 2\,(I_{so} + I_d) = P_{so} \; t_s^+ + P_d \; t_d^+$$

A1.3.4.2 Allowable Impulse on Plate

The applied impulse will give velocity and kinetic energy to the plate. Let V = plate velocity

$$(I_t A) = MV = \rho_m h A V$$

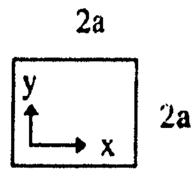
and

$$KE = U_K = \frac{1}{2} M V^2 = \frac{1}{2} \frac{(I_t A)^2}{M} = \frac{1 (I_t A)^2}{2 \rho_m h A}$$

$$U_K = \frac{I_t^2 \; A}{2 \rho_m h}$$

To determine allowable U_K, assume the deformation shape is

$$u = u_o \cos\frac{\pi x}{2a} \cos\frac{\pi y}{2a}$$



u = deflection

If P_e = equivalent static pressure which would cause deflection u

$$U = \iint p_e \, u dx dy = 4 \int_0^a\int_0^a P_e \, u_o \frac{\pi x}{2a} \cos\frac{\pi y}{2a}$$

$$= 4 \, P_e \, u_o \frac{a^2}{\pi^2} = 715 \, P_e \, u_o \text{ for } a = 42 \text{ in.}$$

Consider plate as simply supported

Consider bending only

Bending stress at center

$$f_b = 0.221 \, p \frac{a^2}{h^2} (1 + \nu)$$

For a = 42   h = 0.75

$$\underline{f_b = 3620 \, p^b}$$

$$U_{bo} = 0.0487 \frac{pu^4 \, (1 - \nu^2)}{E_{h^3}} \qquad E = 10 \times 10^6, \quad \nu = 0.3$$

$$U_{bo} = 0.52 \, p^b \qquad (f_b = 6950 \, U_{bo})$$

Consider membrane action

$$f_m = 0.396 \left(\frac{Ea^2}{h^2}\right) (p^m)^{2/3}$$

$$U_{mo} = 0.802\ a \left(\frac{p_m{}^3\ a}{E_h} \right)^{1/2}$$

$$f_m = 1250\ p_m{}^{2/3}$$

$$U_{mo}^3 = 0.214\ p_m \qquad \left(f_m = 3480\ U_{mo}^2 \right)$$

For combined bending plus membrane action

$$U_{mo} = U_{bo} = U_o$$

$$P_e = \left(\frac{u_o}{0.52} + \frac{u_o{}^3}{0.213} \right)$$

$$U = 715\ P_e u_o = 715 \left(\frac{u_o{}^2}{0.52} + \frac{u_o{}^4}{0.213} \right)$$

Hence

$$\frac{I_t{}^2 A}{2\ p_m h} = 715 \left(\frac{u_o{}^2}{0.52} + \frac{u_o{}^4}{0.213} \right)$$

$$\frac{A}{2\ p_m h} = \frac{84 \times 84\ 386}{2 \times 0.1 \times 0.75} = 1.82 \times 10^7$$

$$25400\ I_t{}^2 = 1.92\ u_o{}^2 + 4.7\ u_o{}^4$$

$$160.5\ I_t = \sqrt{1.93\ u_o{}^2 + 4.7\ u_o{}^4}$$

Limit $\quad f_m$ to $f_{yield}/2 = 10000$ psi

$$u_o = \left(\frac{10000}{3480} \right)^{1/2} = \underline{1.7 \text{ in.}} \quad \underline{I_t = 0.041}$$

To limit I_t of ~0.041 lb. sec/in.^{2}, W_c must be limited to 2 pounds from Table A1-2. Therefore, the analysis indicates that a safe charge weight of 2 pounds of TNT equivalent could be exposed in the north room MSM. The limit for the North Room explosion inventory outside of the MSM will remain at 16 pounds TNT equivalent.

Table A1-2

SAFE CHARGE WEIGHT

W_c	W_s (lb)	$W^{1/3}$	d_r (ft)	P_{so} (P_{do}) (psi)	P_{refc} (psi)	t^+_{rs} (t^+_{rd}) (sec)	t^+_s (t^+_d) (sec)	I_s (I_d) (lb.sec/in.^{2})	I_t (lb.sec/in.^{2})	I^1 Fig. A1-1 Ref. 1
0.5	1	1	1510	5 (0.4)	11	0.32 (0.40)	0.0025 (0.0032)	0.0063 (0.0006)	0.0138	0.013
1	2	1.26	1200	7.5 (1.0)	17	0.26 (0.36)	0.0026 (0.0036)	0.0097 (0.0018)	0.0230	0.024
2	4	1.59	950	10.5 (3.0)	27	0.23 (0.35)	0.0029 (0.0044)	0.0152 (0.0066)	0.0436	0.033
3	6	1.82	840	14.0 (4.0)	36	0.21 (0.34)	0.0030 (0.0049)	0.0210 (0.0098)	0.0616	0.044
5	10	2.15	700	20 (8.0)	60	0.18 (0.34)	0.0031 (0.0058)	0.0310 (0.0232)	0.1084	0.064

Appendix B

BASIC RELATIONS USED IN HEAT TRANSFER ANALYSIS

B.1 LAMINAR HEAT-TRANSFER COEFFICIENT

The Sieder-Tate correlation for laminar heat transfer inside tubes (and channels) was used to compute fuel-surface-to-coolant heat-transfer coefficients.[1] To demonstrate that the flow is laminar at all points in the core, the Reynolds number is calculated:

$$N_{RE} = \frac{\overline{W} D_H}{\mu\ A_F}$$

where:

$\overline{W}$ = maximum rate in a single channel, lb/in = 1.557 x average flow rates

μ = dynamic viscosity of the coolant, lb/ft-h

D_H = equivalent hydraulic diameter of channel, ft

A_F = channel cross-sectional area for flow, ft^2

For rated operation at 20-gpm total recirculation flow, the average Reynolds number is:

$$N_{RE} = 597$$

The flow in all regions is laminar, because the Reynolds number is much less than 2600, the number at which transition from laminar to turbulent flow occurs.

The Sieder-Tate correlation is:

$$\frac{h\ D_H}{K} = 1.86\ N_{RE}{}^{1/3}\ N_{PR}{}^{1/3} \left[\frac{D_H}{L}\right]^{1/3}$$

where:

h = film heat-transfer coefficient, Btu/h-ft^2-°F

D_H = channel hydraulic diameter, ft

N_{RE} = Reynolds number

N_{PR} = Prandtl number

L = flow length of channel, ft (average chord length of disk)

The heat-transfer coefficient computed from this relation is an average value over the disk but is used at all axial positions in the channel. All thermal properties are evaluated at the bulk coolant temperature.

B.2 FILM BOILING HEAT-TRANSFER COEFFICIENT

In those instances in which the surface heat flux locally rises above the burnout heat flux, it is necessary to recalculate the heat-transfer coefficient,[2] since burnout is the point at which local surface boiling changes to film boiling. The film boiling coefficient is computed from:

$$h = 0.724\left[\frac{\lambda\ g k_v^{3}\ \rho_v(\rho_\ell \rho_v)}{D_m \mu_v \Delta T}\right]^{1/4}$$

where:

λ = heat of vaporization, Btu/lb

g = gravitational constant, ft/h^2

K_V = thermal conductivity of vapor, Btu/h-ft-°F

ρ_v = density of saturated vapor, lb/ft^3

ρ_ℓ = density of saturated liquid, lb/ft^3

D_H = hydraulic diameter, ft

μ = dynamic viscosity of saturated vapor, lb/h-ft

T = temperature difference between surface and coolant, °F

B.3 JENS-LOTTES SURFACE BOILING CORRELATION

The fuel disk surface temperature that will exist in the presence of local surface film boiling was predicted from the Jens-Lottes[3] correlation:

$$T_{(Surf)} = T_{(Sat)} + \frac{1.9\ (q/A)^{1/4}}{e^{P/900}}$$

where:

T_{Sat} = coolant saturation temperature, °F

(q/A)= surface heat flux, Btu/h-ft^2

P = system pressure, psia

When the surface temperature locally rises to the value given by the above expression from fuel internal heat generation and laminar flow heat transfer, local surface boiling will start and thereafter, the surface temperature will be held at that value.

B.4 BURNOUT HEAT FLUX CORRELATIONS

The burnout correlation that was used to compute burnout heat fluxes[4] is:

$$(q/A)_{crit} = 7000\left(T_{(Sat)} - T_L\right) V^{0.5}$$

where:

T_{Sat} = coolant saturation temperature, °F

T_L = local coolant temperature, °F

V = coolant velocity, ft/sec

This correlation was proposed to predict low pressure burnout for subcooled water. Its range of application is pressure, 1 to 11 atm; velocity, 1 to 40 ft/sec; subcooling, 20 to 200 Btu/lb.

The preceding correlation is applied locally and the burnout heat flux varies from channel to channel and along the flow direction in each channel.

The channel velocities encountered in the NTR core are actually smaller than the recommended lower limit of application of the above correlation. However, the margin of conservatism can be examined by referring to existing data correlations on pool boiling. Pool boiling is boiling with no forced flow of the coolant, which would occur if the velocity were dropped to zero. The NTR maximum velocity is only 0.2 ft/sec; therefore, pool boiling data should serve as an indicator of a limit on the burnout heat fluxes.

A recent pool boiling correlation which shows good agreement with available data at all pressures is that of Zuber and Tribus.[5] The correlation is:

$$(q/A)_{crit} = \frac{\pi}{24} L\rho_v\gamma + \frac{2k}{\sqrt{\pi\,\alpha\,\tau}}\,(T_s - T_L)$$

with

$$\pi = \frac{\sqrt{2\pi^3}}{3\gamma}\left[\frac{\sigma}{g\,(\rho_\ell\ \rho_v)}\right]^{1/2}$$

and

$$\gamma = \left[\frac{\sigma g(\rho_\ell - \rho_v)}{\rho_v^{\,2}}\right]^{1/4}$$

where:

g = gravitational constant

k = thermal conductivity of liquid

L = latent heat of vaporization

α = thermal diffusivity of liquid

σ = surface tension

ρ = density

For a pressure of 14.7 psia (approximately the pressure level of the NTR), the following values of the burnout heat flux have been calculated from the expression:

T_{liquid} (°F)	$(q/A)_{crit}$ (Btu/f-ft^2)
100	1.14×10^6
150	0.78×10^6
200	0.42×10^6
212	0.32×10^6

As indicated by these values when compared to the values computed from the first burnout correlation presented above, the use of the first correlation is conservative.