50-602

DEPARTMENT OF MECHANICAL ENGINEERING
THE UNIVERSITY OF TEXAS AT AUSTIN

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July 10, 2000

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
ATTN: Warren J. Eresian, Mail Stop O12-D1

SUBJECT: Reference materials for license examination scheduled the week of 9/11/00

Dr. Eresian,

Enclosed you will find the following materials for the preparation of an SRO licensing examination at the University of Texas Nuclear Engineering Teaching Laboratory:

Volume I	UT-TRIGA Training Manual and Task Analysis
Volume II	UT-TRIGA Training Manual General Technical Information
Volume III	UT-TRIGA Safety Analysis Report and Technical Specifications
Volume IV	UT-TRIGA Standard Operating Procedures and Plans

Please contact me if you require additional information or materials at 512-232-5373 or SOKELLY@mail.utexas.edu.

Sincerely,

Sean O'Kelly
Associate Director

cc: Training File
Central File

ADD. Warren Eresian to RIDS

M003

Volume I

UT-TRIGA Training Manual and Task Analysis

UT-TRIGA

TRAINING PROGRAM FILES

Nuclear Engineering Teaching Laboratory

J.J. Pickle Research Campus
The Universty of Texas at Austin

Directory of Training Disk Files

Disk Name: NETL-Tasks

Subject	Filename	Version
Disk information	DSKinfo	1.10
Operator Task List		
List Description	TASKLD	2.10
Reactor Operator	TASKRO	2.10
Senior Operator	TASKSO	2.10
Performance Objectives		
Objectives Summary	PRFOS	1.00
Terminal Objectives	PRFTO	1.00
Enabling Objectives	PRFEO	1.00
Principles and Theory		
CH Chemistry	OBJCH	
MS Material Science	OBJMS	
NP Nuclear Physics	OBJNP	
ME Mechanical Science	OBJMS	
PR Engineering Symbology	OBJPR	
HP Health Physics	OBJHP	
Plant Features and Systems		
NB NETL Building	OBJNB	
RP Reactor Pool	OBJRP	
VC Ventilation Control	OBJVC	
RM Radiation Monitoring	OBJRM	
TR TRIGA Reactor	OBJTR	
XX Experiments	OBJXX	
Facility Written Procedures		
WP Procedures - 1	OBJWP1	
WP Procedures - 2	OBJWP2	
Performance Skills	SKILLS	
Question Files	QNETL???	

Summary of Training Components

Tasks and Performance Objectives

Operator Task Lists
- Reactor Operator
- Senior Operator

Performance Objectives
- Terminal Objectives
- Enabling Objectives

Subject Classifications

A - Theory and principles of reactor operation
B - General and specific plant operating characteristics
C - Reactor instrumentation and control systems
D - Reactor protection systems and engineered safety systems
E - Normal, abnormal and emergency operating procedures
F - Radiation control and safety
G - Technical specifications
H - Applicable portions of 10CFR

Principles and Theory	(A,~~E~~) F
Plant Features and Systems	(B,C,D)
Facility Written Procedures	E (~~F~~,G,H)
Performance Skills	(all A-H)

Training Materials

DOE Handbooks

Training Manuals

Volume 1	-	NETL - UT-TRIGA
Volume 2	-	References
Volume 3	-	In development, "STNP" version

UT-TRIGA Documents
- Safety Analysis Report, Safety Evaluation Report
- Instrumentation, Control and Safety System Volume 1
- Operating license, technical specifications

NETL Procedures
- ADMN, HP, MAIN, SURV, OPER, EXP, NETL

Facility Handouts

Summary of Performance Objectives
DOE-HNDBK Training Modules

Chemistry (5): HNDBK-1015/1-2

Section	*Title*	*# of TO*	*# of EO*	*Totals*
CH01	Fundamentals of Chemistry	3	7,9,4	20
CH02	Corrosion	1	28	28
CH03	Reactor Water Chemistry	1	6	6
CH04	Principles of Water Treatment	1	10	10
CH05	Hazards of Chemicals & Gas	1	16	16
		7		80

Key Words:

Terminal Objectives (5): balance, calculate, describe, discuss, explain

Enabling Objectives (11): balance, calculate, define, describe, discuss, explain, identify, list, state, summarize, write

Material Sciences (5): HNDBK-1017/1-2

Section	*Title*	*# of TO*	*# of EO*	*Totals*
MS01	Structure of Metals	1	15	15
MS02	Properties of Metals	1	22	22
MS03	Thermal Shock	1	9	9
MS04	Brittle Fracture	1	16	16
MS05	Plant Materials	1	28	28
		5		90

Key Words:

Terminal Objectives (2): describe, explain

Enabling Objectives (9): define, describe, distinguish, explain, identify, interpret, list, state

Atomic and Nuclear Physics (4): HNDBK-1019/1-2

Section	*Title*	*# of TO*	*# of EO*	*Totals*
NP01	Atomic and Nuclear Physics	5	8,10,3,10,2	33
NP02	Reactor Theory (Neutron Characteristics)	4	3,14,6,3	26
NP03	Reactor Theory (Reactor Parameters)	5	12,12,5,15,8	52
NP04	Reactor Theory (Reactor Operations)	3	4,10,11	25
		17		136

Key Words:

Terminal Objectives (2): describe, explain

Enabling Objectives (14): calculate, characterize, convert, define, describe, determine, estimate, explain, express, identify, list, plot, state, write

Engineering Symbology, Prints, and Drawing (6): HNDBK-1016/1-2

Section	Title	# of TO	# of EO	Totals
PR01	Introduction to Print Reading	1	5	5
PR02	Engineering Fluid Diagrams and Prints	1	12	12
PR03	Electrical Diagrams and Schematics	1	6	6
PR04	Electronic Diagrams and Schematics	1	2	2
PR05	Logic Diagrams	1	5	5
PR06	Engineering Fabrication, Construction, and Architectural Drawings	1	3	3
		6		65

Key Words:

Terminal Objectives (3): identify, interpret, read

Enabling Objectives (5): determine, develop, explain, identify, state

Mechanical Sciences (5): HNDBK-1018/1-2

Section	Title	# of TO	# of EO	Totals
ME01	Diesel Engine Fundamentals	1	8	8
ME02	Heat Exchangers	1	8	9
ME03	Pumps	2	9,6	15
ME04	Valves	1	6	6
ME05	Misc Mechanical Components	1	17	17
		6		65

Key Words:

Terminal Objectives (1): describe

Enabling Objectives (7): define, describe, differentiate, explain, identify, list, state

NETL Training Modules

Section Title		*# of TO*	*# of EO*	*Totals*
NB	NETL Building			
RP	Reactor Pool			
VC	Ventilation Control			
RM	Radiation Monitoring			
TR	TRIGA Reactor			
XX	Experiments			

Key Words:
- Terminal Objectives (1):
  - describe, explain
- Enabling Objectives (1):
  - calculate, compare, define, explain, identify, list, state

Section Title		*# of TO*	*# of EO*	*Totals*
HP	Health Physics			

Key Words:
- Terminal Objectives (2):
  - describe, explain
- Enabling Objectives (7):
  - calculate, compare, define, explain, identify, list, state

Section Title		*# of TO*	*# of EO*	*Totals*
WP	Procedures			

Key Words:
- Terminal Objectives (1):
  - describe
- Enabling Objectives (1):
  - define

NETL Question Data Bank

Question Bank Files

Principles and Theory		
CH	Chemistry	QNETLCH
MS	Material Science	QNETLMS
NP	Nuclear Physics	QNETLNP
ME	Mechanical Science	QNETLMS
PR	Engineering Symbology	QNETLPR
HP	Health Physics	QNETLHP
Plant Features and Systems		
NB	NETL Building	QNETLNB
RP	Reactor Pool	QNETLRP
VC	Ventilation Control	QNETLVC
RM	Radiation Monitoring	QNETLRM
TR	TRIGA Reactor	QNETLTR
XX	Experiments	QNETLXX
Facility Written Procedures		
WP	Procedures	QNETLWP
Performance Skills		QNETLPS

Question Bank Numbering Scheme

This version of the NETL Question data bank is based on performance objectives. The objectives are taken from two sources. One source is a DOE-HNDBK series. Numbering of questions corresponds to the training module, terminal objective and enabling objective.

Question numbering scheme:

Module.TO.EO.#

where

Module	Section number such as NP01
TO	Terminal objective number
EO	Enabling objective number
#	Question variation a - z

List of Terminal Objectives

Key Words:
Terminal Objectives ():

Enabling Objectives ():

CH Chemistry

CH01.1.0	Without reference, DESCRIBE the characteristics of an atom.
CH01.2.0	Given an incomplete chemical equation, BALANCE the equation by the method presented.
CH01.3.0	Given sufficient information about a solution, CALCULATE the pH and pOH of the solution.
CH02.1.0	Without references, DESCRIBE the causes and effects of corrosion on metals and the type of chemistry used in a plant to minimize corrosion.
CH03.1.0	Without references, DESCRIBE the effects of radiation on reactor water and methods of treatment for the products.
CH04.1.0	Without references, EXPLAIN the concept and application of ion exchange in water purification.
CH05.1.0	Without references, DISCUSS the hazards associated with chemicals (liquid and gas) found in a nuclear plant.

MS Material Sciences

MS01.1.0	Without references, DESCRIBE the bonding and patterns that effect the structure of a metal.
MS02.1.0	Without references, DESCRIBE how changes in stress, strain, and physical and chemical properties effect the materials used in a reactor plant.
MS03.1.0	Without references, DESCRIBE the importance of minimizing thermal shock (stress).
MS04.1.0	Without references, EXPLAIN the importance of controlling heatup and cooldown rates of the primary coolant system.
MS05.1.0	Without reference, DESCRIBE the considerations commonly used when selecting material for use in a reactor plant.

NP Nuclear Physics

NP01.1.0	Given sufficient information, DESCRIBE atoms, including components, structure, and nomenclature.
NP01.2.0	Given necessary references, DESCRIBE the various modes of radioactive decay.
NP01.3.0	Without references, DESCRIBE the different nuclear interactions initiated by neutrons.
NP01.4.0	Without references, DESCRIBE the fission process.
NP01.5.0	Without references, DESCRIBE how the various types of radiation interact with matter.
NP02.1.0	Without references, EXPLAIN how neutron sources produce neutrons.
NP02.2.0	Given the necessary information for calculations, EXPLAIN basic concepts in reactor physics and perform calculations.
NP02.3.0	Without references, EXPLAIN the production process and effects on fission of prompt and delayed neutrons.
NP02.4.0	Without references, DESCRIBE the neutron energy spectrum for the type of reactor presented in the module.
NP03.1.0	Using appropriate references, DESCRIBE the neutron life cycle discussed in this module.
NP03.2.0	From memory, EXPLAIN how reactivity varies with the thermodynamic properties of the moderator and the fuel.
NP03.3.0	Without references, DESCRIBE the use of neutron poison.
NP03.4.0	Without references, DESCRIBE the effects of fission product poisons on a reactor.
NP03.5.0	Without references, DESCRIBE how control rods affect the reactor core.
NP04.1.0	Given the necessary information and equations, EXPLAIN how subcritical multiplication occurs.
NP04.2.0	Given the necessary information and equations, DESCRIBE how power changes in a reactor that is near criticality.
NP04.3.0	Without references, EXPLAIN the concepts concerning reactor startup, operation, and shutdown.

ME Mechanical Science

PR Engineering Symbology

HP Health Physics
review by AJ, put stuff here

List of Objectives

Chemistry

CH	Terminal Objective: Chemistry	SRO	RO
	Fundamentals of Chemistry		
CH01.1.0	Without references, DESCRIBE the characteristics of an atom.	x	x
CH01.2.0	Given an incomplete chemical equation, BALANCE the equation by the method presented.	x	
CH01.3.0	Given sufficient information about a solution, CALCULATE the pH and pOH of the solution.	x	
	Corrosion		
CH02.1.0	Without references, DESCRIBE the causes and effects of corrosion on metals at the research reactor facility and the type of chemistry to minimize corrosion.	x	
	Reactor Water Chemistry		
CH03.1.0	Without references, DESCRIBE the effects of radiation on reactor system water and methods of treatment for the products.	x	
	Principles of Water Treatment		
CH04.1.0	Without references, EXPLAIN the concept and application of ion exchange in water purification.	x	x
	Hazards of Chemicals and Gases		
CH05.1.0	Without references, DISCUSS the hazards associated with chemicals (liquid and gas) found in a research reactor.	x	x

CH	Enabling Objective: Fundamentals of Chemistry	SRO	RO
CH01.1.1	DEFINE the following terms: a. Mole b. States of matter c. Atomic weight d. Molecular weight e. Gram atomic weight f. Gram molecular weight	x	x
CH01.1.2	LIST the components of an atom, their relative sizes, and charges.	x	x
CH01.1.3	STATE the criterion used to classify an atom chemically.	x	x
CH01.1.4	DEFINE the following subdivisions of the periodic table: a. Periods of the periodic table b. Groups of the periodic table c. Classes of the periodic table	x	
CH01.1.5	Given a periodic table, IDENTIFY the following subdivisions: a. Periods of the periodic table b. Groups of the periodic table c. Classes of the periodic table	x	
CH01.1.6	LIST the characteristics that elements in the same group on the periodic table share.	x	
CH01.1.7	DEFINE the term valence.	x	

CH	Enabling Objective: Corrosion	SRO	RO
CH01.2.1	DEFINE the following terms: a. Ionic bonds b. Van der Waals forces c. Covalent bonds d. Metallic bonds	x	
CH01.2.2	DESCRIBE the physical arrangement and bonding of a polar molecule.	x	
CH01.2.3	DESCRIBE the three basic laws of chemical reactions.	x	
CH01.2.4	STATE how elements combine to form chemical compounds.	x	
CH01.2.5	EXPLAIN the probability of any two elements combining to form a compound.	x	
CH01.2.6	DEFINE the following terms: a. Mixture b. Solvent c. Solubility d. Solute e. Solution f. Equilibrium	x	
CH01.2.7	STATE Le Chatelier's principle.	x	
CH01.2.8	DEFINE the following terms: a. ppm b. Molarity c. Density d. Normality	x	
CH01.2.9	BALANCE chemical equations that combine elements and/or compounds.	x	

CH	Chemistry Enabling Objective:	SRO	RO
CH01.3.1	Define the following terms: a. Acid b. Salt c. Alkalii d. Base c. pOH d. pH g. Dissociation constant of water	x	
CH01.3.2	STATE the formula of pH.	x	
CH01.3.3	STATE the formula of pOH.	x	
CH01.3.4	CALCULATE the pH of a specified solution.	x	

CH	Chemistry Enabling Objective:	SRO	RO
CH02.1.1	Define the following terms: a. Ionization b. Conductivity c. Corrosion d. Electrolysis e. General corrosion	x	
CH02.1.2	DESCRIBE an electrochemical cell with respect to the corrosion of metals.	x	
CH02.1.3	STATE what happens to a metal during the oxidation step of the oxidation-reduction process.	x	
CH02.1.4	STATE what happens to a metal during the reduction step of the oxidation-reduction process.	x	
CH02.1.5	DEFINE the following terms: a. Passivity b. Polarization		
CH02.1.6	DESCRIBE the affects of passivity and polarization on the corrosion process.		
CH02.1.7	LIST the two conditions that contribute to general corrosion.	x	
CH02.1.8	DESCRIBE how the rate of corrosion occurring is affected by the following: a. Temperature b. Water velocity c. Oxygen d. pH e. Condition and composition of the metal surface.	x	
CH02.1.9	LIST the three products that are formed from the general corrosion of iron.		
CH02.1.10	IDENTIFY the action taken for initial fill of a reactor system to limit general corrosion.		
CH02.1.11	STATE the four methods used to chemically control general corrosion.	x	
CH02.1.12	LIST the six water chemistry conditions that limit corrosion of aluminum.	x	

CH02.1.13	DEFINE the following terms: a. Crud b. Scale c. Galvanic corrosion	x
CH02.1.14	IDENTIFY the five problems associated with the presence or release of crud into reactor coolant.	
CH02.1.15	STATE the four causes of crud bursts.	
CH02.1.16	STATE the two conditions that can cause galvanic corrosion.	x
CH02.1.17	EXPLAIN the mechanism for galvanic corrosion.	x
CH02.1.18	IDENTIFY the two locations that are susceptible to galvanic corrosion.	
CH02.1.19	State the five control measures to minimize galvanic corrosion.	
CH02.1.20	DEFINE the following terms: a. Pitting corrosion b. Crevice corrosion c. Stress corrosion cracking	x
CH02.1.21	STATE the two conditions necessary for pitting corrosion to occur.	
CH02.1.22	STATE the particular hazard associated with pitting corrosion.	
CH02.1.23	STATE the four controls used to minimize pitting corrosion.	
CH02.1.24	IDENTIFY the three conditions necessary for stress corrosion cracking to occur.	
CH02.1.25	DEFINE the term chemisorption.	
CH02.1.26	STATE the hazard of stress corrosion cracking.	
CH02.1.27	STATE the three controls used to prevent stress corrosion cracking.	
CH02.1.28	DESCRIBE the two types of stress corrosion cracking that are of major concern to nuclear facilities including: a. Conditions for occurrence b. Method(s) used to minimize the probability of occurrence.	

CH	Enabling Objective: Reactor Water Chemistry	SRO	RO
CH03.1.1	DESCRIBE the process of radiolytic decomposition and recombination of water.	x	x
CH03.1.2	DESCRIBE the process of radiolytic decomposition and recombination of nitric acid and ammonia.	x	x
CH03.1.3	STATE the advantage of maintaining excess hydrogen in reactor water.	x	x
CH03.1.4	STATE the three sources of radioactivity in reactor water and each one's decay product.	x	
CH03.1.5	STATE the following for reactor water chemistry. a. Nine parameters controlled b. Reason for controlling each parameter c. Method of controlling each parameter	x	
CH03.1.6	STATE the possible effects of abnormal chemistry on core conditions.	x	

CH	Enabling Objective: Principles of Water Treatment	SRO	RO
CH04.1.1	LIST the three reasons for removing impurities from water prior to use in reactor systems.	x	x
CH04.1.2	DEFINE the following terns: a. Ion exchange b. Demineralize c. Cation d. Anion e. Polymer f. Mixed-bed demineralizer g. Affinity h. Decontamination factor	x	x
CH04.1.3	DESCRIBE the following: a. Resin bead b. Cation resin c. Anion resin	x	x
CH04.1.4	DISCUSS the following factors of ion exchange: a. Relative affinity b. Decontamination factor	x	
CH04.1.5	WRITE the reaction for removal of NaCl and CaSO4 by a mixed-bed ion exchanger such as one containing HOH resin.	x	
CH04.1.6	EXPLAIN the three basic methods used to remove dissolved gases from water.		
CH04.1.7	LIST five filtration media used to remove suspended solids from waters	x	x
CH04.1.8	EXPLAIN how mixed-bed ion exchangers may be used to control pH.		
CH04.1.9	DISCUSS resin malfunctions, including the following: a. Channeling b. Breakthrough c. Exhaustion	x	x
CH04.1.10	LIST the maximum conductivity and approximate concentration of electrolyte for each level of purity for makeup water.	x	x

CH	Enabling Objective: Hazards of Chemicals and Gases	SRO	RO
CH05.1.1	STATE the hazards associated with the use of corrosives.	x	x
CH05.1.2	STATE the general safety precautions necessary for the handling, storage, and disposal of corrosives.	x	x
CH05.1.3	LIST the general safety precautions regarding toxic compounds.	x	x
CH05.1.4	LIST the criteria used to determine if a compound is a health hazard.	x	x
CH05.1.5	STATE the methods by which toxic compounds may enter the body.	x	x
CH05.1.6	SUMMARIZE the purpose and general contents of the following: a. Material Safety Data Sheets (MSDS) b. Toxic Substance List	x	x
CH05.1.7	DEFINE the following terms: a. Compressed gas b. Non-liquefied gases c. Liquefied gases d. Dissolved gases	x	x
CH05.1.8	STATE the five major families of gases.	x	
CH05.1.9	STATE the general safety precautions regarding the use, handling, and storage of gases.	x	
CH05.1.10	STATE the safety precautions for working with cryogenic liquids.	x	
CH05.1.11	LIST the physical properties and special precautions for the following gases: a. Hydrogen b. Oxygen c. Nitrogen	x	
CH05.1.12	DEFINE the following terns: a. Flammable liquid b. Combustible liquid	x	
CH05.1.13	STATE general safety precautions regarding the use, handling, and storage of flammable and combustible liquids.	x	
CH05.1.14	STATE the reasons for and techniques used in bonding and grounding of flammable liquid containers.	x	
CH05.1.15	LIST four sources of ignition of flammable liquids.	x	

CH05.1.16	STATE the health hazards associated with flammable and/or combustible liquids.	x

List of Objectives

Material Science

MS	Terminal Objective: Material Science	SRO	RO
MS01.1.0	Structure of Metals Without references, DESCRIBE the bonding and patterns that effect the structure of a metal.	x	x
MS02.1.0	Properties of Metals Without references, DESCRIBE how changes in stress, strain, and physical and chemical properties effect the materials used in a reactor facility.	x	
MS03.1.0	Thermal Shock Without references, DESCRIBE the importance of minimizing thermal shock (stress).	x	
MS04.1.0	Brittle Fracture Without references, EXPLAIN the importance of controlling heatup and cooldown rates of the primary coolant system.	x	
MS05.1.0	Plant Materials Without reference, DESCRIBE the considerations commonly used when selecting material for use in a reactor facility	x	x

MS	Enabling Objective: Structure of Metals	SRO	RO
MS01.1.1	STATE the five types of bonding that occur in materials and their characteristics.	x	x
MS01.1.2	DEFINE the following terms: a. Crystal structure b. Body-centered cubic structure c. Face centered cubic structure d. Hexagonal close packed structure	x	
MS01.1.3	STATE the three lattice-type structures in metals.	x	
MS01.1.4	Given a description or drawing, DISTINGUISH between the three most common types of crystalline structures.	x	
MS01.1.5	IDENTIFY the crystalline structure possessed by a metal.		
MS01.1.6	DEFINE the following terms: a. Grain b. Grain structure c. Grain boundary d. Creep	x	
MS01.1.7	DEFINE the term polymorphism.		
MS01.1.8	IDENTIFY the ranges and names for the polymorphism phases associated with uranium metal.		
MS01.1.9	IDENTIFY the polymorphism phase that prevents pure uranium from being used as fuel.		
MS01.1.10	DEFINE the term alloy.	x	x
MS01.1.11	DESCRIBE an alloy as to the three possible microstructures and the two general characteristics as compared to pure metals.	x	
MS01.1.12	IDENTIFY the two desirable properties of type 304 stainless steel.	x	
MS01.1.13	IDENTIFY the three types of microscopic imperfections found in crystalline structures.	x	
MS01.1.14	STATE how slip occurs in crystals.		
MS01.1.15	IDENTIFY the four types of bulk defects.		

MS	Enabling Objective: Properties of Metals	SRO	RO
MS02.1.1	DEFINE the following terms: a. Stress b. Tensile stress c. Compressive stress d. Shear stress e. Compressibility	x	x
MS02.1.2	DISTINGUISH between the following types of stresses by the direction in which stress is applied. a. Tensile b. Compressive c. Shear	x	
MS02.1.3	DEFINE the following terms; a. Strain b. Plastic deformation c. Proportional limit	x	
MS02.1.4	IDENTIFY the two common forms of strain.	x	
MS02.1.5	DISTINGUISH between the two common forms of strain as to dimensional change.	x	
MS02.1.6	STATE how iron crystalline lattice, gamma or alpha, structure deforms under load.		
MS02.1.7	STATE Hooke's Law.		
MS02.1.8	DEFINE Young's Modulus (Elastic Modulus) as it relates to stress.		
MS02.1.9	Given the values of the associated material properties, CALCULATE the elongation of a material using Hooke's Law.		
MS02.1.10	DEFINE the following terms: a. Bulk Modulus b. Fracture point		
MS02.1.11	Given stress-strain curves for ductile and brittle material, IDENTIFY the following specific points on a stress-strain curve. a. Proportional limit b. Yield point c. Ultimate strength d. Fracture point	x	x

MS02.1.12	Given a stress-strain curve, IDENTIFY whether the type of material represented is ductile or brittle.	x	x
MS02.1.13	Given a stress-strain curve, INTERPRET a stress-strain curve for the following: a. Application of Hookes's Law b. Elastic region c. Plastic region	x	
MS02.1.14	DEFINE the following terms: a. Strength b. Ultimate tensile strength c. Yield strength d. Ductility e. Malleability f. Toughness g. Hardness	x	
MS02.1.15	IDENTIFY how slip effects the strength of a metal.		
MS02.1.16	DESCRIBE the effects on ductility caused by: a. Temperature changes b. Cold working c. Irradiation	x	
MS02.1.17	IDENTIFY the reactor facility application for which high ductility is desirable.	x	
MS02.1.18	STATE how heat treatment effects the properties of heat-treated steel and carbon steel.	x	
MS02.1.19	DESCRIBE the adverse effects of welding on metal including types of stress and method(s) for minimizing stress.	x	
MS02.1.20	STATE the reason that galvanic corrosion is a concern in design and material selection.	x	
MS02.1.21	DESCRIBE hydrogen embrittlement including the two required conditions and the formation process.		
MS02.1.22	IDENTIFY why zircaloy-4 is less susceptible to hydrogen embrittlement that zircaloy-2.		

MS	Enabling Objective: Thermal Shock	SRO	RO
MS03.1.1	IDENTIFY the two stresses that are the result of thermal shock (stress) to facility materials.	x	
MS03.1.2	STATE the two causes of thermal shock.	x	
MS03.1.3	Given the material's coefficient of Linear Thermal Expansion, CALCULATE the thermal shock (stress) on a material using Hooke's Law.		
MS03.1.4	DESCRIBE why thermal shock is a major concern in reactor systems when rapidly heating or cooling a thick-walled vessel.		
MS03.1.5	LIST the three operational limits that are specifically intended to reduce the severity of thermal shock.		
MS03.1.6	DEFINE the term pressurized thermal shock.		
MS03.1.7	STATE how the pressure in a closed system effects the severity of thermal shock.		
MS03.1.8	LIST the four system transients that have the greatest potential for causing thermal shock.		
MS03.1.9	STATE the three locations in a reactor system that are of primary concern for thermal shock.		

MS	Enabling Objective: Brittle Fracture	SRO	RO
MS04.1.1	DEFINE the following terms: a. Ductile fracture b. Brittle fracture c. Nil-ductility Transition Temperature (NDT)	x	x
MS04.1.2	DESCRIBE the two changes made to reactor pressure vessels to decrease NDT.		
MS04.1.3	STATE the effect grain size and irradiation have on a material's NDT.		
MS04.1.4	LIST the three conditions necessary for brittle fracture to occur.	x	
MS04.1.5	STATE the three conditions that tend to mitigate crack initialization.	x	
MS04.1.6	LIST the five factors that determine the fracture toughness of a material.		
MS04.1.7	Given a stress-temperature diagram, IDENTIFY the following points: a. NDT (with no flaw) b. NDT (with flaw) c. Fracture transition elastic point d. Fracture transition plastic point		
MS04.1.8	STATE the two bases used for developing a minimum pressurization-temperature curve.		
MS04.1.9	EXPLAIN a typical minimum pressure-temperature curve including: a. Location of safe operating region b. The way the curve will shift due to irradiation		
MS04.1.10	LIST the normal actions taken, in sequence, if the minimum pressurization-temperature curve is exceeded during critical operations.		
MS04.1.11	STATE the precaution for hydrostatic testing.		
MS04.1.12	IDENTIFY the basis used for determining heatup and cooldown rate limits.		
MS04.1.13	IDENTIFY the three components that will set limits on the heatup and cooldown rates.		
MS04.1.14	STATE the action typically taken upon discovering the heatup or cooldown rate has been exceeded.		
MS04.1.15	STATE the reason for using soak times.		
MS04.1.16	STATE when soak times become very significant.		

MS	Enabling Objective: Plant Materials	SRO	RO
MS05.1.1	DEFINE the following terms: s. Machinability b. Formability c. Stability d. Fabricability	x	x
MS05.1.2	IDENTIFY the importance of a material property and its application in a reactor facility.	x	
MS05.1.3	LIST the four radioactive materials that fission by thermal neutrons and are used as reactor fuels.	x	
MS05.1.4	STATE the four considerations in selecting fuel material and the desired effect on the nuclear properties of the selected fuel material.	x	x
MS05.1.5	STATE the four major characteristics necessary in a material used for fuel cladding.	x	x
MS05.1.6	IDENTIFY the four materials suitable for use a fuel cladding material and their applications.	x	x
MS05.1.7	STATE the purpose of a reflector.	x	x
MS05.1.8	LIST the five essential requirements for reflector material in a thermal reactor.	x	x
MS05.1.9	STATE the five common poisons used as control rod material.	x	x
MS05.1.10	IDENTIFY the advantage(s) and/or disadvantages of the five common poisons used as control rod material.	x	
MS05.1.11	DESCRIBE the requirements of a material used to shield against the following types of radiation: a. Beta b. Gamma c. High energy neutron d. Low energy neutron	x	x
MS05.1.12	STATE the nuclear reactor core problems and causes associated with the following: a. Pellet-cladding interaction b. Fuel densification c. Fuel cladding embrittlement d. Fuel burnup and fission product swelling	x	

MS05.1.13	STATE the measures taken to counteract or minimize the effects of the following: a. Pellet-cladding interaction b. Fuel densification c. Fuel cladding embrittlement d. Fission product swelling of a fuel element	x
MS05.1.14	DEFINE the following terms: a. Fatigue failure b. Work hardening c. Creep	x
MS05.1.15	STATE the measures taken to counteract or minimize the effects of the following: a. Fatigue failure b. Work hardening c. Creep	x
MS05.1.16	STATE how the following types of radiation interact with metals: a. Gamma b. Alpha c. Beta d. Fast neutron e. Slow neutron	x
MS05.1.17	DEFINE the following terms: a. Knock-on b. Vacancy c. Interstitial	x
MS05.1.18	DEFINE the following terms: a. Thermal spike b. Displacement spike	
MS05.1.19	STATE the effect a large number of displacement spikes has on the properties of a metal.	
MS05.1.20	DESCRIBE how the emission of radiation can cause dislocation of the atom emitting the radiation.	
MS05.1.21	STATE the two effects on a crystalline structure resulting from the capture of a neutron.	x
MS05.1.22	STATE how thermal neutrons can produce atomic displacements.	x
MS05.1.23	STATE how gamma and beta radiation effect organic materials.	x

MS05.1.24	IDENTIFY the change in organic compounds due to radiation. a. Nylon b. Rubber c. High-density polyethylene marlex 50	x	
MS05.1.25	IDENTIFY the chemical bond with the least resistance to radiation.	x	
MS05.1.26	DEFINE the term polymerization.	x	
MS05.1.27	STATE the applications and the property that makes aluminum desirable in a reactor operating at: a. Low kilowatt power b. Low temperature ranges c. Moderate temperature range	x	x
MS05.1.28	STATE why aluminum is undesirable in high temperature research or power reactors.	x	

List of Objectives

Nuclear Physics and Reactor Theory

NP	Terminal Objective: Nuclear Physics and Reactor Theory	SRO	RO
	Atomic and Nuclear Physics		
NP01.1.0	Given sufficient information, DESCRIBE atoms, including components, structure, and nomenclature.	x	x
NP01.2.0	Given necessary references, DESCRIBE the various modes of radioactive decay.	x	x
NP01.3.0	Without references, DESCRIBE the different nuclear interactions initiated by neutrons.	x	x
NP01.4.0	Without references, DESCRIBE the fission process.	x	x
NP01.5.0	Without references, DESCRIBE how the various types of radiation interact with matter.	x	x
	Reactor Theory (Nuclear Parameters)		
NP02.1.0	Without references, EXPLAIN how neutron sources produce neutrons.	x	x
NP02.2.0	Given the necessary information for calculations, EXPLAIN basic concepts in reactor physics and perform calculations.	x	x
NP02.3.0	Without references, EXPLAIN the production process and effects on fission of prompt and delayed neutrons.	x	x
NP02.4.0	Without references, DESCRIBE the neutron energy spectrum for the type of reactor presented in the module.	x	x
	Reactor Theory (Nuclear Parameters)		
NP03.1.0	Using appropriate references, DESCRIBE the neutron life cycle discussed in this module.	x	x
NP03.2.0	From memory, EXPLAIN how reactivity varies with the thermodynamic properties of the moderator and the fuel.	x	x
NP03.3.0	Without references, DESCRIBE the use of neutron poison.	x	x

NP03.4.0	Without references, DESCRIBE the effects of fission product poisons on a reactor.	x	x
NP03.5.0	Without references, DESCRIBE how control rods affect the reactor core.	x	x
	<u>Reactor Theory (Reactor Operations)</u>		
NP04.1.0	Given the necessary information and equations, EXPLAIN how subcritical multiplication occurs.	x	x
NP04.2.0	Given the necessary information and equations, DESCRIBE how power changes in a reactor that is near criticality.	x	x
NP04.3.0	Without references, EXPLAIN the concepts concerning reactor startup, operation, and shutdown.	x	x

NP	Enabling Objective: Atomic and Nuclear Physics	SRO	RO
NP01.1.1	STATE the characteristics of the following atomic particles, including mass, charge, and location within the atom: a. Proton b. Neutron c. Electron		
NP01.1.2	DESCRIBE the Bohr model of an atom.	x	x
NP01.1.3	DEFINE the following terms: a. Nuclide b. Isotope c. Atomic number d. Mass number	x	x
NP01.1.4	Given the standard superscript-subscript, a-z X notation for a particular nuclide, DETERMINE the following: a. Number of protons b. Number of neutrons c. Number of electrons	x	x
NP01.1.5	DESCRIBE the three forces that act on particles within the nucleus and affect the stability of the nucleus.	x	x
NP01.1.6	DEFINE the following terms: a. Enriched uranium b. Depleted uranium	x	x
NP01.1.7	DEFINE the following terms: a. Mass defect b. Binding energy	x	x
NP01.1.8	Given the atomic mass for a nuclide and the atomic masses of a neutron, proton, and an electron, CALCULATE the mass defect and binding energy of the nuclide.	x	x

NP	Enabling Objective: Atomic and Nuclear Physics	SRO	RO
NP01.2.1	DESCRIBE the following processes: a. Alpha decay b. Beta-plus decay c. Beta-minus decay d. Electron capture e. Internal conversion f. Isomeric transitions	x	x
NP01.2.2	Given a Chart of the Nuclides, WRITE the radioactive decay chain for a nuclide.	x	x
NP01.2.3	EXPLAIN why one or more gamma rays typically accompany particle emission.	x	x
NP01.2.4	Given the stability curve on the Chart of the Nuclides, DETERMINE the type of radioactive decay that the nuclides in each region of the chart will typically undergo.	x	x
NP01.2.5	DEFINE the following terms: a. Radioactivity b. Curie c. Becquerel d. Radioactive half-life e. Radioactive decay constant	x	x
NP01.2.6	Given the number of atoms and either the half-life or decay constant of a nuclide, CALCULATE the activity at any later time.	x	x
NP01.2.7	Given the initial activity and either the half-life or decay constant of a nuclide, CALCULATE the activity at any later time.	x	x
NP01.2.8	CONVERT between the half-life and decay constant for a nuclide.	x	x
NP01.2.9	Given the Chart of the Nuclides and the original activity, PLOT the radioactive decay curve for a nuclide on either linear or semi-log coordinates.	x	x
NP01.2.10	DEFINE the following radioactivity terms: a. Radioactive equilibrium b. Transient equilibrium	x	x

NP	Enabling Objective: Atomic and Nuclear Physics	SRO	RO
NP01.3.1	DESCRIBE the following scattering interactions between a neutron and a nucleus: a. Elastic scattering b. Inelastic scattering	x	x
NP01.3.2	STATE the conservation laws that apply to an elastic collision between a neutron and a nucleus.	x	x
NP01.3.3	DESCRIBE the following reactions where a neutron is absorbed in a nucleus: a. Radiative capture b. Particle ejection	x	x

NP	Enabling Objective: Atomic and Nuclear Physics	SRO	RO
NP01.4.1	Explain the fission process using the liquid drop model of a nucleus.	x	x
NP01.4.2	Define the following terms: a. Excitation energy b. Critical energy	x	x
NP01.4.3	Define the following terms: a. Fissile material b. Fissionable material c. Fertile material	x	x
NP01.4.4	Describe the processes of transmutation, conversion, and breeding.		x
NP01.4.5	Describe the curve of Binding Energy per Nucleon versus mass number and give a qualitative description of the reasons for its shape.	x	x
NP01.4.6	Explain why only the heaviest nuclei are easily fissioned.	x	x
NP01.4.7	Explain why uranium-235 fissions with thermal neutrons and uranium-238 fissions only with fast neutrons.	x	x
NP01.4.8	CHARACTERIZE the fission products in terms of mass groupings and radioactivity.	x	
NP01.4.9	Given the nuclides involved and their masses, Calculate the energy released from fission.	x	
NP01.4.10	Given the curve of Binding Energy per Nucleon versus mass number, Calculate the energy released from fission.	x	

NP	Enabling Objective: Atomic and Nuclear Physics	SRO	RO
NP01.5.1	Describe interactions of the following with matter: a. Alpha particle b. Beta particle c. Positron d. Neutron	x	x
NP01.5.2	Describe the following ways that gamma radiation interacts with matter: a. Compton scattering b. Photoelectric effect c. Pair production	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP02.1.1	Define the following terms: a. Intrinsic neutron source b. Installed neutron source	x	x
NP02.1.2	List three examples of reactions that produce neutrons in intrinsic neutron sources.	x	x
NP02.1.3	List three examples of reactions that produce neutrons in installed neutron sources.	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP02.2.1	Define the following terms: a. Atom density b. Neutron flux c. Microscopic cross section d. Macroscopic cross section e. Mean free path f. Barn	x	x
NP02.2.2	Express macroscopic cross section in terms of microscopic cross section.	x	x
NP02.2.3	Describe how the absorption cross section of typical nuclides varies with neutron energy at energies below the resonance absorption region.	x	x
NP02.2.4	Describe the cause of resonance absorption in terms of nuclear energy levels.	x	x
NP02.2.5	Describe the energy dependence of resonance absorption peaks for typical light and heavy nuclei.	x	x
NP02.2.6	Express the mean free path in terms of macroscopic cross section.	x	x
NP02.2.7	Given the number densities (or total density and component fractions) and microscopic cross sections of components, calculate the macroscopic cross section for a mixture.	x	x
NP02.2.8	Calculate a macroscopic cross section given a material density, atomic mass, and microscopic cross section.	x	x
NP02.2.9	Explain neutron shadowing or self-shielding.	x	x
NP02.2.10	Given the neutron flux and macroscopic cross section, calculate the reaction rate.	x	x
NP02.2.11	Describe the relationship between neutron flux and reactor power.	x	x
NP02.2.12	Define the following concepts: a. Thermalization b. Moderatory c. Moderating ratio d. Average logarithmic energy decrement e. Macroscopic slowing down power.	x	x
NP02.2.13	List three desirable characteristics of a moderator.	x	x

NP02.2.14	Given an average fractional energy loss per collision, calculate the energy loss after a specified number of collisions.	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP02.3.1	State the origin of prompt neutrons and delayed neutrons.	x	x
NP02.3.2	State the approximate fraction of neutrons that are born as delayed neutrons from the fission of the following nuclear fuels: a. Uranium-235 b. Plutonium-239	x	x
NP02.3.3	Explain the mechanism for production of delayed neutrons.	x	x
NP02.3.4	Explain prompt and delayed neutron generation times.	x	x
NP02.3.5	Given prompt and delayed neutron generation times and delayed neutron fraction, calculate the average generation time.	x	x
NP02.3.6	Explain the effect of delayed neutrons on reactor control.	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP02.4.1	State the average energy at which prompt neutrons are produced.	x	x
NP02.4.2	Describe the neutron energy spectrum in the following reactors: a. Fast reactor b. Thermal reactor	x	x
NP02.4.3	Explain the reason for the particular shape of the fast, intermediate, and slow energy regions of the neutron flux spectrum for a thermal reactor.	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP03.1.1	Define the following terms: a. Infinite multiplication factor, b. Effective mulitplication factor, keff c. Subcritical d. Critical e. Supercritical	x	x
NP03.1.2	Define each term in the six factor formula using the ratio of the number of neutrons present at different points in the neutron life cycle.	x	x
NP03.1.3	Given the macrscopic cross sections for various materials, calculate the thermal utilization factor.	x	
NP03.1.4	Given the microscopic cross sections for absorption and fission, atom density, and v, calculate the reproduction factor.	x	
NP03.1.5	Given the numbers of neutrons present at the start of a generation and values for each factor in the six factor formula, calculate the number of neutrons that will be present atr any point in the life cycle.	x	
NP03.1.6	List physical changes in the reactor, core that will have an effect on the thermal utilization factor, reporduction factor, or resonanace escape probability.	x	
NP03.1.7	Explain the effect that temperature changes will have on the following factors: a. Thermal utilization factor b. Resonance escape prebability c. Fast non-leakage probability d. Thermal non-leakage probability		
NP03.1.8	Given the number of neutrons in a reactor core and the effective multiplication factor, calculate the number of neutrons present after any number of generations.		
NP03.1.9	Define the term reactivity.		
NP03.1.10	Convert between reactivity and the associated value of keff.		

NP03.1.11	Convert measures of reactivity between the following units: a. D k/k b. % D k/k c. 10-4 D k/k d. Percent millirho (pcm)	x	x
NP03.1.12	Explain the relation ship between reactivity coefficients and reactivity defects.	x	x

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP03.2.1	EXPLAIN the conditions of over moderation and under moderation.	x	
NP03.2.2	Explain why many reactors are designed to be operated in an under moderated condition.	x	
NP03.2.3	State the effect that a change in moderator temperature will have on the moderator to fuel ration.	x	
NP03.2.4	Define the temperature coefficient of reactivity.	x	x
NP03.2.5	EXPLAIN why a negative temperature coefficient of reactivity is desirable.	x	x
NP03.2.6	Explain why the fuel temperature coefficient is more effective than the moderator temperature coefficient in terminating a rapid power rise.	x	x
NP03.2.7	Explain the concept of Doppler broadening of resonance absorption peaks.	x	
NP03.2.8	List two nuclides that are present in some types of reactor fuel assemblies that have significant resonance absorption peaks.	x	
NP03.2.9	Define the pressure coefficient of reactivity.	x	
NP03.2.10	Explain why the pressure coefficient of reactivity is usually negligible in a reactor cooled and moderated by a subcooled liquid.	x	
NP03.2.11	Define the void coefficient of reactivity.	x	
NP03.2.12	Identify the moderator conditions under which the void coefficient of reactivity becomes significant.	x	

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP03.3.1	DEFINE the following terms: a. Burnable poison b. Non-burnable poison c. Chemical shim	x	x
NP03.3.2	EXPLAIN the use of burnable neutron poison in a reactor core.	x	
NP03.3.3	LIST the advantages and disadvantages of chemical shim over fixed burnable poisons.	x	
NP03.3.4	STATE two reasons why fixed non-burnable neutron poisons are used in reactor cores.	x	
NP03.3.5	STATE an example of a material used as a fixed non-burnable neutron poison.	x	

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP03.4.1	LIST two methods of production and two methods of removal for xenon-135 during reactor operation.	x	x
NP03.4.2	STATE the equation for equilibrium xenon-135 concentration.	x	
NP03.4.3	DESCRIBE how equilibrium xenon-135 concentration varies with reactor power level.	x	x
NP03.4.4	DESCRIBE the causes and effects of a xenon oscillation.	x	
NP03.4.5	DESCRIBE how xenon-135 concentration changes following a reactor shutdown from steady-state conditions.	x	x
NP03.4.6	EXPLAIN the effect that pre-shutdown power levels have on the xenon-135 concentration after shutdown.	x	x
NP03.4.7	STATE the approximate time following a reactor shutdown at which the reactor can be considered "xenon free."	x	x
NP03.4.8	EXPLAIN what is meant by the following terms: a. Xenon precluded startup b. Xenon dead time	x	x
NP03.4.9	DESCRIBE how xenon-135 concentrations change following an increase or a decrease in the power level of a reactor.	x	x
NP03.4.10	DESCRIBE how samarium-149 is produced and removed from the reacor core during reactor operation.	x	
NP03.4.11	STATE the equation for equilibrium samarium-149 concentration.	x	
NP03.4.12	DESCRIBE how equilibrium samarium-149 concentration varies with reactor power level.	x	
NP03.4.13	DESCRIBE how samarium-149 concentration changes following a reactor shutdown from steady-state conditions.	x	
NP03.4.14	DESCRIBE how samarium-149 concentration changes following a reactor startup.	x	
NP03.4.15	STATE the conditions under which helium-3 will have a significant effect on the reactivity of a reactor.	x	

NP	Enabling Objective: Reactor Theory (Nuclear Parameters)	SRO	RO
NP03.5.1	DESCRIBE the difference between a "grey" neutron absorbing material and a "black" neutron absorbing material.	x	x
NP03.5.2	EXPLAIN why a "grey" neutron absorbing material may be preferable to a "black" neutron absorbing material for use in control rods.	x	x
NP03.5.3	EXPLAIN why resonance absorbers are sometimes preferred over thermal absorbers as a control rod material.	x	
NP03.5.4	DEFINE the following terms: a. Integral control rod worth b. Differential control rod worth.	x	x
NP03.5.5	DESCRIBE the shape of a typical differential control rod worth curve and explain the reason for the shape.	x	x
NP03.5.6	DESCRIBE the shape of a typical intergral control rod worth curve and explain the reason for the shape.	x	x
NP03.5.7	Given an integral or differential control rod worth curve, CALCULATE the reactivity change due to a control rod movement between two positions.	x	x
NP03.5.8	Given differential control rod worth data, PLOT differential and integral control rod worth curves.	x	x

NP	Enabling Objective: Reactor Theory (Reactor Operations)	SRO	RO
NP04.1.1	DEFINE the following terms: a. Subcritical multiplication b. Subcritical multiplication factor	x	x
NP04.1.2	Given a neutron source strength and a subcritical system of known keff, CALCULATE the steady-state neutron level.	x	x
NP04.1.3	Given an initial count rate and keff, CALCULATE the final count rate that will result from the addition of a known amount of reactivity.	x	x
NP04.1.4	Given count rates vs. the parameter being adjusted, ESTIMATE the value of the parameter at which the reactor will become critical through the use of a 1/M plot.	x	x

NP	Enabling Objective: Reactor Theory (Reactor Operations)	SRO	RO
NP04.2.1	DEFINE the following terms: a. Reactor period b. Doubling time c. Reactor startup rate	x	x
NP04.2.2	DESCRIBE the relationship between the delayed neutron fraction, average delayed neutron fraction, and effective delayed neutron fraction.	x	
NP04.2.3	WRITE the period equation and IDENTIFY each symbol.	x	x
NP04.2.4	Given the reactivity of the core and values for the effective average delayed neutron fraction and decay constant, CALCULATE the reactor period and the startup rate.	x	x
NP04.2.5	Given the initial power level and either the doubling or halving time, CALCULATE the power at any later time.	x	x
NP04.2.6	Given the initial power level and the reactor period, CALCULATE the power at any later time.	x	x
NP04.2.7	EXPLAIN what is meant by the terms prompt drop and prompt jump.	x	x
NP04.2.8	DEFINE the term prompt critical.	x	x
NP04.2.9	DESCRIBE reactor behavior during the prompt critical condition.	x	x
NP04.2.10	EXPLAIN the use of measuring reactivity in units of dollars.	x	x

NP	Enabling Objective: Reactor Theory (Reactor Operations)	SRO	RO
NP04.3.1	EXPLAIN why a startup neutron source may be required for a reactor.	x	x
NP04.3.2	LIST four variables typically involved in a reactivity balance.	x	x
NP04.3.3	EXPLAIN how a reactivity balance may be used to predict the conditions under which the reactor will become critical.	x	x
NP04.3.4	LIST three methods used to shape or flatten the core power distribution.	x	x
NP04.3.5	DESCRIBE the concept of power tilt.	x	x
NP04.3.6	DEFINEthe term shutdown margin.		
NP04.3.7	EXPLAIN the rationale behind the one stuck rod criterion.	x	x
NP04.3.8	IDENTIFY five changes that will occur during and after a reactor shutdown that will affect the reactivity of the core.	x	x
NP04.3.9	EXPLAIN why decay heat is present following reactor operation.	x	x
NP04.3.10	LIST three variables that will affect the amount of decay heat present following reactor shutdown.	x	x
NP04.3.11	ESTIMATE the approximate amount of decay heat that will exist one hour after a shudown from steady state conditions.	x	x

List of Objectives

Mechanical Science

ME	Terminal Objective: Mechanical Science	SRO	RO
	Diesel Engine Fundamentals		
ME01.1.0	Without references, DESCRIBE the components and theory of operation of a diesel engine.		
	Heat Exchangers		
ME02.1.0	Without references, DESCRIBE the purpose, construction and principles of operation for each major type of heat exchanger; parallel flow, counter flow , and cross flow.	x	x
	Pumps		
ME03.1.0	Without references, DESCRIBE the purpose, construction and principles of operation for a centrifugal pumps.	x	x
ME03.2.0	Without references, DESCRIBE the purpose, construction and principles of operation for a positive displacement pumps.	x	x
	Valves		
ME04.1.0	Without references, DESCRIBE the construction and operation of a given type of valve, valve component, or valve actuator, as presented in this module.	x	x
	Miscellaneous Mechanical Components		
ME05.1.0	Without references, DESCRIBE the purpose, construction and operation of miscellaneous mechanical components.	x	x

ME	Enabling Objective: Diesel Engine Fundamentals	SRO	RO
ME01.1.1	DEFINE the following diesel engine terms: a. Combustion chamber b. Compression ratio c. Bore d. Stroke		
ME01.1.2	Given drawing of a diesel engine, IDENTIFY the following: a. Piston/rod b. Cylinder c. Blower d. Crankshaft e. Intake ports or valves f. Exhaust ports or valves g. Fuel injector		
ME01.1.3	EXPLAIN now a diesel engine converts the chemical energy in the diesel fuel to mechanical energy.		
ME01.1.4	EXPLAIN how the ignition process occurs in a diesel engine.		
ME01.1.5	EXPLAIN the operation of a 4-cycle diesel engine to include the following events during a cycle: a. Intake b. Exhaust c. Fuel injection d. Compression e. Power		
ME01.1.6	EXPLAIN the operation of a 2-cycle diesel engine to include the following events during a cycle: a. Intake b. Exhaust c. Fuel injection d. Compression e. Power		
ME01.1.7	DESCRIBE how the mechanical-hydraulic governor on a diesel engine controls engine speed.		
ME01.1.8	LIST five protective alarms usually found on medium and large size diesel engines.		

ME	Enabling Objective: Heat Exchangers	SRO	RO
ME02.1.1	STATE the two types of heat exchanger construction.	x	x
ME02.1.2	Provided with a drawing of a heat exchanger, IDENTIFY the following parts: a. Tubes b. Tube sheet c. Shell d. Baffles	x	x
ME02.1.3	DESCRIBE hot and cold fluid flow in parallel flow, counter flow, and cross flow heat exchangers.	x	
ME02.1.4	DIFFERENTIATE between the following types of heat exchangers: a. Single-pass versus multi-pass heat exchangers. b. Regenerative versus non-regenerative heat exchangers.	x	
ME02.1.5	LIST at least three applications of heat exchangers.	x	
ME02.1.6	STATE the purpose of a condenser.		
ME02.1.7	DEFINE the following terms: a. Hotwell b. Condensate depression		
ME02.1.8	STATE why condensers in large steam cycles are operated at a vacuum.		

ME	Enabling Objective: Pumps	SRO	RO
ME03.1.1	STATE the purposes of the following centrifugal pump components: a. Impeller b. Volute c. Diffuser d. Packing e. Lantern ring f. Wearing ring	x	x
ME03.1.2	Given a drawing of a centrifugal pump, IDENTIFY the following major components: a. Pump casing b. Pump shaft c. Impeller d. Volute e. Stuffing box f. Stuffing box gland g. Packing h. Lantern ring i. Impeller wearing ring j. Pump casing wearing ring	x	
ME03.1.3	DEFINE the following terms: a. Net positive suction head available b. Cavitation c. Gas binding d. Shutoff head e. Pump runout	x	
ME03.1.4	STATE the relationship between net positive suction head available and net positive suction head that is necessary to avoid cavitation.	x	
ME03.1.5	LIST three indications that a centrifugal pump may be cavitating.		
ME03.1.6	LIST five changes that can be made in a pump or its surrounding system that can reduce cavitation.		
ME03.1.7	LIST three effects of cavitation.		
ME03.1.8	DESCRIBE the shape of the characteristic curve		

ME	Enabling Objective: Pumps	SRO	RO
ME03.2.1	STATE the difference between the flow characteristics of a centrifugal and positive displacement pumps.	x	x
ME03.2.2	Given a simplified diagram of a positive displacement pump, CLASSIFY the pump as one of the following: a. Reciprocating piston pump b. Gear-type pump c. Screw-type pump d. Lobe-type pump e. Moving vane pump f. Diaphragm pump	x	x
ME03.2.3	EXPLAIN the importance of viscosity as it relates to the operation of a reciprocating postive dispplacement pump.	x	
ME03.2.4	DESCRIBE the characteristic curve for a positive displacement pump.	x	
ME03.2.5	DEFINE the term slippage.	x	
ME03.2.6	STATE how postive displacement pumps are protected against overpressurization.	x	

ME	Enabling Objective: Valves	SRO	RO
ME04.1.1	DESCRIBE the four basic types of flow control elements employed in valve design.	x	x
ME04.1.2	DESCRIBE how valve stem leakage is controlled.	x	x
ME04.1.3	Given a drawing of a valve, IDENTIFY the following: a. Body b. Bonnet c. Stem d. Actuator e. Packing f. Seat g. Disk	x	x
ME04.1.4	Given a drawing of a valve, IDENTIFY each of the following types of valves: a. Globe b. Gate c. Plug d. Ball e. Needle f. Butterfly g. Diaphragm h. Pinch i. Check j. Stop check k. Safety/relief l. Reducing	x	
ME04.1.5	DESCRIBE the application of the following types of valves: a. Globe b. Gate c. Plug d. Ball e. Needle f. Butterfly g. Diaphragm h. Pinch i. Check j. Stop check k. Safety/relief l. Reducing	x	

ME04.1.6 DESCRIBE the construction and principle of operation for the following types of valve actuators: x

a. Manual
b. Electric motor
c. Pneumatic
d. Hydraulic
e. Solenoid

ME	Enabling Objective: Miscellaneous Components	SRO	RO
ME05.1.1	STATE the three common types of air compressors.	x	x
ME05.1.2	DESCRIBE the basic operation of the following types of air compressors: a. Reciprocating b. Centrifugal c. Rotary	x	x
ME05.1.3	STATE the reason for using cooling systems in air compressors.	x	
ME05.1.4	STATE three hazards associated with pressurized air systems.	x	
ME05.1.5	DESCRIBE the basic operation of a hydraulic system.	x	
ME05.1.6	Given the appropriate information, CALCULATE the pressure or force achieved in a hydraulic piston.	x	
ME05.1.7	DESCRIBE the basic operation of a boiler.	x	
ME05.1.8	IDENTIFY the following components of a typical boiler: a. Steam drum b. Distribution headers c. Combustion chamber d. Downcomer e. Risers	x	
ME05.1.9	STATE the purpose of cooling towers.	x	
ME05.1.10	DESCRIBE the operation of the following types of cooling towers: a. Forced draft b. Natural convection	x	
ME05.1.11	STATE the purpose of a demineralizer.	x	x
ME05.1.12	STATE the four purposes of a pressurizer.		
ME05.1.13	DEFINE the following terms attributable to a dynamic pressurizer: a. Spray nozzle b. Insurge c. Outsurge d. Surge volume		
ME05.1.14	STATE the purpose and general operation of a steam trap.		

ME05.1.15	IDENTIFY the following types of steam traps: a. Ball float steam trap b. Bellow steam trap c. Bucket steam trap d. Impulse steam trap		
ME05.1.16	DESCRIBE each of the following types of strainers and filters, including an example typical use: a. Cartridge filters b. Precoated filters c. Deep-bed filters d. Bucket strainer e. Duplex strainer	x	x
ME05.1.17	EXPLAIN the application and operation of a strainer or filter backwash.	x	x

List of Objectives

Engineering Symbology, Prints and Drawings

PR	Terminal Objective: Mechanical Science	SRO	RO
	Introduction to Print Reading		
PR01.1.0	Given an engineering print, READ and INTERPRET the information contained in the title block, the notes and legend, the revision block and the drawing grid.	x	
	Engineering Fluid Diagrams and Prints		
PR02.1.0	Given an engineering print, READ and INTERPRET facility engineering piping and instrument drawings.	x	
	Electrical Diagrams and Schematics		
PR03.1.0	Given an electrical print, READ and INTERPRET facility electrical diagrams and schematic drawings.	x	
	Electronic Diagrams and Schematics		
PR04.1.0	Given a block diagram, print or schematic, IDENTIFY the basic component symbols as presented in ths module.	x	
	Logic Diagrams		
PR05.1.0	Given a logic diagram, READ and INTERPRET the diagrams.	x	
	Engineering Fabrication, Construction, and Architectural Drawings		
PR06.1.0	Given an engineering fabrication, construction, or architectural drawing, READ and INTERPRET basic dimensional and tolerance symbology, and basic fabrication, construction or architectural symbology.	x	

PR	Enabling Objective: Introduction to Print Reading	SRO	RO
PR01.1.1	STATE the five types of information provided in the title block of an engineering drawing.	x	
PR01.1.2	STATE how the grid system on an engineering drawing is used to locate a piece of equipment.	x	
PR01.1.3	STATE the three types of information provided in the revision block of an engineering drawing.	x	
PR01.1.4	STATE the purpose of the notes and legend section of an engineering drawing.	x	
PR01.1.5	LIST the five drawing categories used on engineering drawings.	x	

PR	Enabling Objective: Engineering Fluid Diagrams and Prints	SRO	RO
PR02.1.1	IDENTIFY the symbols used on engineering P&IDs for the following types of valves: a. Globe valve b. Gate valve c. Ball valve d. Check valve e. Stop check valve f. Butterfly valve g. Relief vavle h. Rupture disk i. Three-way valve j. Four-way valve k. Throttle valve l. Pressure regulator	x	
PR02.1.2	IDENTIFY the symbols used on engineering P&IDs for the following types of valve operator: a. Diaphram valve operator b. Motor valve operator c. Solenoid valve operator d. Piston valve operator e. Hand valve operator f. Reach-rod valve operator	x	
PR02.1.3	IDENTIFY the symbols used on engineering P&IDs for educators and ejectors.		
PR02.1.4	IDENTIFY the symbols used on engineering P&IDs for the following lines: a. Process b. Pneumatic c. Hydraulic d. Inert gas e. Instrument signal f. Instrument capillary g. Electrical	x	
PR02.1.5	IDENTIFY the symbols used on engineering P&IDs for the following basic types of instrumentation: a. Differential pressure cell b. Temperature element c. Venturi d. Orifice e. Rotometer f. Conductivity g. Radiation detector	x	
PR02.1.6	IDENTIFY the symbols used on engineering P&IDs to denote the location, either local or board mounted, of instruments, indicators, and controllers.		

PR02.1.7 IDENTIFY the symbols used on engineering P&IDs for the following types of instrument controllers and modifiers:

a. Proportional
b. Proportional-integral
c. Proportional-integral-differential
d. Square-root extractor

PR02.1.8 IDENTIFY the symbols used on engineering P&IDs for the following types of system components:

a. Centrifugal pumps
b. Postive head displacement pumps
c. Heat exchangers
d. Compressors
e. Fans
f. Tanks
g. Filters/strainers

PR02.1.9 STATE how the valve conditions are depicted on an engineering P&ID:

a. Open valve
b. Closed vavle
c. Throttle valve
d. Combination valves (3 or 4 way)
e. Locked clesed valve
f. Locked open valve
g. Fail open valve
h. Fail closed valve
i. Fail as-is valve

PR02.1.10 Given an engineering P&ID, IDENTIFY components and DETERMINE the flow path(s) for a given valve lineup. x

PR02.1.11 IDENTIFY the symbols used on engineering fluid power drawings for the components: x

a. Pump
b. Compressor
c. Reservoir
d. Actuators
e. Piping ang piping junctions
f. Valves

PR02.1.12 Given a fluid power type drawing, DETERMINE the operation or resultant action of the stated component when hydraulic pressure is applied/removed.

PR	Enabling Objective: Electrical Diagrams and Schematics	SRO	RO
PR03.1.1	IDENTIFY the symbols used on engineering electrical drawings used for the following components: a. Single phase circuit breaker (open-closed) b. Three phase circuit breaker (open-closed) c. Thermal overload d. "a" contact e. "b" contact f. Time delay contacts g. Relay h. Potential transformer i. Current transformer j. Single-phase transformer k. Delta-wound transformer l. Wye-wound transformer m. Electric motor n. Meters o. Junctions p. In-line fuses q. Single switch r. Multiple position switch s. Pushbutton switch t. Limit switches u. Turbine driven generator v. Diesel-driven generator w. Motor generator set x. Generator (wye or delta) y. Battery	x	
PR03.1.2	Given an electrical drawing of a circuit containing a transformer, DETERMINE the direction of current flow, as shown by the transformers symbol.	x	
PR03.1.3	IDENTIFY the symbols and/or codes used on electrical drawings to depict the relationship between the following components: a. Relay and its contacts b. Switch and its contacts c. Interlocking device and its interlocked equipment		
PR03.1.4	STATE the condition in which all electrical devices are shown, unless otherwise noted on the drawing or schematic.		
PR03.1.5	Given a simple electrical schematic and initial conditions, DETERMINE the condition of the specified component, (i.e.,energized or de-energized, open or closed).		
PR03.1.6	Given a simple electrical schematic and initial conditions, IDENTIFY the power sources and/or loads and their status, (i.e.,energized or de-energized).		

PR	Enabling Objective: Electronic Diagrams and Schematics	SRO	RO
PR04.1.1	IDENTIFY the symbols used on engineering electronic block diagrams, prints, and schematics, for the following components. a. Fixed resistor b. Variable resistor c. Tapped resistor d. Fixed capacitor e. Variable capacitor f. Fixed inductor g. Variable inductor h. Diode i. Light emitting diode (LED) j. Ammeter k. Voltmeter l. Wattmeter m. Chassis ground n. Circuit ground o. Fuse p. Plug q. Headset r. Light bulb s. Silicon controlled rectifier (SCR) t. Half wave rectifier u. Full wave rectifier v. Oscillator w. Potentiometer x. Rheostat y. Antenna z. Amplifier aa. Transistors (PNP and NPN) bb. Junction	x	
PR04.1.2	STATE the purpose of a block diagram and an electronic schematic diagram.	x	

PR	Enabling Objective: Logic Diagrams	SRO	RO
PR05.1.1	IDENTIFY the symbols used on logic diagrams to represent the following components: a. AND gate b. NAND gate c. COINCIDENCE gate d. OR gate e. NOR gate f. EXCLUSIVE OR gate g. NOT gate or invertor h. Adder i. Time-delay j. Counter k. Shift register l. Flip-flop m. Logic memories	x	
PR05.1.2	EXPLAIN the operation of the three types of time delay devices.	x	
PR05.1.3	DEVELOP the truth tables for the following logic gates: a. AND gate b. OR gate c. NOT gate d. NAND gate e. NOR gate f. EXCLUSIVE OR gate	x	
PR05.1.4	DETERMINE the symbols used to denate a logical 1 (or high) and a logical 0 (or low) as used in logic diagrams	x	
PR05.1.5	Given a logic diagram and appropriate information, DETERMINE the output of each component and the logic circuit.	x	

PR	Enabling Objective: Engineering Fabrication, Construction, and Architectural Drawings	SRO	RO
PR06.1.1	STATE the purpose of engineering fabrication, construction, and architectural drawings.	x	
PR06.1.2	Given an engineering, fabrication, or architectural drawing, DETERMINE the specified dimensions of an object.	x	
PR06.1.3	Given an engineering, fabrication, or architectural drawing, DETERMINE the maximum and minimum dimensions or location of an object or feature from the stated drawing tolerance.	x	

List of Objectives

NETL Building

NB	Terminal Objective: NETL Building	SRO	RO
	NETL Building		
NB01.0	DESCRIBE the location of the NETL facility.	x	x
NB02.0	DESCRIBE the organizational structure.	x	x
NB03.0	DESCRIBE facilty safety features and programs.	x	x
NB04.0	REVIEW changes to facility equipment and written procedures.		
	RP - Reactor Pool Module		
RP01.0	Describe the reactor pool water system.	x	x
RP02.0	Describe the water system operating conditions.	x	x
	AC - Air Confinement Module		
AC01.0	Describe the reactor room ventilation system.	x	x
AC02.0	Describe the ventilation operating conditions.	x	x
	RM - Radiation Monitoring Module		
RM01.0	Describe the fixed radiation monitors for the detection of nuclear radiation.	x	x
RM02.0	Describe the portable survey instruments at the facility.	x	x
	HP - Health Physics Module		
HP01.0	Without references describe nuclear radiation effects on humans.	x	x
HP02.0	Describe the survey equipment that measures radiation exposure, radiation dose, and radiaoactivity contamination.	x	x
HP03.0	Explain the applicable exposure and dose limits for protection against the effects of radiation.	x	x

	<u>Written Procedures</u>		
WP01.0	DESCRIBE the Physical Security procedures.	x	x
WP02.0	DESCRIBE the Emergency Response procedures.	x	x
WP03.0	DESCRIBE the administrative procedures.	x	x
WP04.0	DESCRIBE the general requirements of the seven HP procedures.	x	x
WP05.0	LIST the Maintenance procedures and EXPLAIN the purpose of each procedure.	x	x
WP06.0	LIST the Surveillance procedures and EXPLAIN the purpose of each procedure.	x	x
WP07.0	DESCRIBE the procedures that control changes to configurations of the reactor core fuel and experiments.	x	x
WP08.0	LIST special facility activities that have written procedures.	x	x
WP09.0	DESCRIBE the requirements for Quality Assurance.	x	x
WP10.0	DESCRIBE the requirements for Operator Requalification.	x	x

NB	Enabling Objective: NETL Building	SRO	RO
NB01.1.1	Describe the facility location	x	x
NB01.1.2	Identify the access control requirements for the building	x	x
NB01.1.3	Identify the access control requirements for the reactor bay.	x	x

NB	Enabling Objective: NETL Building	SRO	RO
NB02.2.1	Identify the key organization components a. Department b. College c. University	x	x
NB02.2.2	Identify key facility personnel a. Director b. Emergency Director c. SRO (supervisor) d. SRO (shift), RO e. Health Physicist	x	x
NB02.2.3	Identify the notification requirements for a. access approval b. security event c. emergency event d. quality concern e. safety concern	x	x

NB	Enabling Objective: NETL Building	SRO	RO
NB03.3.1	Define a security event.	x	x
NB03.3.2	Define an emergency event.	x	x
NB03.3.3	Describe the steps to evacuate the building.	x	x
NB03.3.4	Explain the facility features for a. life safety b. operation safety c. radiation safety	x	x
NB03.3.5	Explain the requirements for written procedures	x	x

RP	Enabling Objective: RP - Reactor Pool Module	SRO	RO
RP01.1.1	Describe the components of the pool system: a. shield structure b. reactor tank c. dimensions of pool d. beam port penetrations e. construction materials	x	x
RP01.1.2	Describe the pool purification system: a. Pool Inlet valves b. Pool Outlet valves c. Demineralizer d. Pump e. Makeup f. Instrumentation	x	x
RP01.1.3	Describe the pool cooling system: a. Pool isolation valves b. N16 control valve c. heat exchanger d. pool (primary) pump e. process (secondary) pump f. chilled water isolation g. instrumentation	x	x
RP01.1.4	Draw a diagram of the two pool water systems.	x	x
RP02.2.1	State the pool water temperature limits	x	x
RP02.2.2	State the pool operating levels a. hi level (alarm) b. lo level (alarm) c. lo level (scram)	x	x
RP02.2.3	State the typical operating conditions of the pool purification loop: a. flow rate b. conductivity at resin tank inlet/outlet c. pressure drop across filter	x	x
RP02.2.4	State the typical operating conditions of the pool cooling system loop: a. flow b. temperature of heat sink (chilled water) c. differential pressure d. temperature control set point e. pressure at Hx	x	x

AC	Enabling Objective; AC - Air Confinement Module	SRO	RO
AC01.1.1	Describe the components of the ventilation system. a. isolation dampers b. control room panel c. argon exhaust d. room exhaust	x	x
AC01.1.2	Describe the reactor bay ventilation including a. negative pressure b. run mode switch c. emergency shutdown switch d. instrumentation	x	x
AC01.1.3	Describe the argon exhaust ventilation including a. o~~f~~n-off switch c. Beam port switch c. pool surface switch d. source point valves e. HEPA filters f. argon 41 continuous air monitor	x	x
AC02.2.1	State the operating differential pressure.	x	x
AC02.2.2	State the alarm conditions for differential pressure.	x	x
AC02.2.3	State the exhaust stack exit velocities.	x	x
AC02.2.4	State the operating conditions for the ????	x	x
AC02.2.5	State the set point limit for automatic operation of isolation dampers to high particulate radioactivity.	x	x
AC02.2.6	State the set point for action to correct the excess release of argon 41 gaseous activity.	x	x

RM	Enabling Objective: RM - Radiation Monitoring Module	SRO	RO
RM01.1.1	Identify the location and type of detectors that detect gamma rays in the reactor operating areas.	x	x
RM01.1.2	Describe the operation of the air particulate monitor	x	x
RM01.1.3	Describe the operation of the argon-41 gaseous monitor.	x	x
RM01.1.4	Explain the function of the hand and foot monitor.	x	x
RM01.1.5	Explain the purpose of the multi sample low background alpha-beta proportional counter (G5000).	x	x
RM02.2.1	Identify the instruments available for contamination surveys.	x	x
	a. RM-14 - GM probe		
	b. Bicron - GM probe		
	c. Bicron - scintillation probe		
RM02.2.2	Identify the instruments for neutron detection	x	x
RM02.2.3	Identify the instruments for area surveys	x	x
	a. R02		
	b. Victoreen ~~440~~ new		
	c. Thyac III - GM alpha		
	d. Technical Associates - GM neutron		
	e. Eberline		

HP	Enabling Objective: HP - Health Physics Module	SRO	RO
HP01.1.1	Define the terms a. exposure b. dose	x	x
HP01.1.2	Explain the following types of external exposure. a. whole body b. extremity c. lens of eye d. skin	x	x
HP01.1.3	Explain the following terms for internal exposures. a. organ dose b. biological half-life	x	x
HP01.1.4	Define the quality factor.	x	x
HP01.1.5	List the quality factor for the following radiations: a. gamma, x-ray b. fast neutron c. thermal neutron d. beta, electrons, positrons e. alpha, heavy charged particle	x	x
HP01.1.6	Calculate the radiation dose: a. gamma - whole body b. neutron - whole body c. beta skin dose d. alpha skin dose	x	x
HP01.1.7	Define the radiation protection terms: a. TEDE b. dose equivalent c. effective dose equivalent	x	x
HP01.1.8	Compare the risk of radiation dose to the risk of the following common activities a. transportation accidents b. industrial accidents c. teaching d. smoking	x	x
HP02.1.1	Describe the operation of gaseous detectors. a. ionization chambers b. proportional counters c. Geiger Mueller tubes	x	x
HP02.1.2	Describe the radiation detection with scintillators.	x	x

HP02.1.3	Identify the sensitivity of the personnel and area dosimeters. a. film badges b. pocket chambers c. thermal luminecsence detector	x	x
HP03.1.1	Explain the meaning of the radiation risk terms: a. stochastic b. deterministic	x	x
HP03.1.2	Define the regulatory limits for personnel: a. whole body b. extremities c. lens of the eye, skin	x	x
HP03.1.3	State the limits for aiborne exposures: a. ALI b. DAC	x	x
HP03.1.4	State the requirements for posting of radiation areas a. radiation area b. high radiation area c. very high radiation area	x	x
HP03.1.5	Describe the biological effects of a radiation dose of a. < 25 rem b. 25-400 rem c. >400 rem	x	x
HP03.1.6	Describe the training instructions to workers.	x	x

TASK LIST
for
UT TRIGA

Supervisory Operator
Reactor Operator

The University of Texas TRIGA

Revision 2.11

Nuclear Engineering Teaching Laboratory
J.J Pickle Research Campus
The University of Texas at Austin

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B. Supervisory Operator

1.0 Normal Reactor Operations
- 1.1 Assume Responsibility for
- 1.2 Assure Compliance with
- 1.3 Authorize Access to
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- 1.12 Schedule or Coordinate

2.0 Abnormal/emergency reactor operations
- 2.1 Analyze
- 2.2 Authorize
- 2.3 Determine
- 2.4 Direct
- 2.5 Procure
- 2.6 Remain
- 2.7 Report
- 2.8 Respond

3.0 Surveillance
- 3.1 Approve
- 3.2 Direct
- 3.3 Initiate/Revise/Rescind
- 3.4 Inspect
- 3.5 Perform
- 3.6 Review and Approve
- 3.7 Schedule
- 3.8 Supervise

4.0 Maintenance
- 4.1 Approve
- 4.2 Authorize
- 4.3 Initiate
- 4.4 Issue
- 4.5 Perform
- 4.6 Review and Approve
- 4.7 Schedule
- 4.8 Supervise

5.0 Administration
- 5.1 Accommodate
- 5.2 Attend
- 5.3 Authorize
- 5.4 Calculate
- 5.5 Control
- 5.6 Direct
- 5.7 Generate
- 5.8 Initiate
- 5.9 Prepare/Update
- 5.10 Review
- 5.11 Schedule
- 5.12 Submit

Tasklist for UT-TRIGA

A. Introduction

1.0 Purpose

1.1 Specify tasks associated with the operation of the UT-TRIGA facility.

1.1.1 The facility is part of the Nuclear Engineering Teaching Laboratory.

1.1.2 The facility is located at The University of Texas at Austin J.J. Pickle Research Campus.

1.2 Identify job tasks associated with each operator class.

1.2.1 NRC license permits are issued for two operator classifications.

1.2.1.1 SRO, senior reactor operator permit, (class A).
1.2.1.2 RO, reactor operator permit, (class B).

1.2.2 Facility operator tasks are identified by operator class as indicated by column S.

1.2.2.1 Supevisory operator tasks are to be performed or supervised by a specific class A operator except as noted in column S by an asterisk which identifies a general class A operator task.
1.2.2.2 Reactor operator tasks are to be performed or supervised by a class B operator except as noted in column S by an asterick which identifies a class A operator task.

1.3 Designate priority tasks related to safe facility operation.

1.3.1 Priority tasks are identified for safety related tasks by an asterisk in column P in each task list.

1.3.2 Criteria for the priority task designation are determined by the consequences of improper performance of that particular task.

1.3.2.1 Injury to personnel.
1.3.2.2 Radiation overexposure
1.3.2.3 Equipment damage.
1.3.2.4 Loss of facility utilization.
1.3.2.5 Event of noncompliance.
1.3.2.6 Release of radionuclides.

1.4 Establish link between task performance and training requirements.

1.4.1 A cross reference links the tasks to items of training material that should provide for proper task performance.

1.4.2 References are to five basic training areas.

1.4.2.1 Principles and theory for operation.
1.4.2.2 Reactor specific characteristics, parameters and systems.
1.4.2.3 Procedures for normal and abnormal operation.
1.4.2.4 Supervisor and administrative instructions.
1.4.2.5 Reactor operating skills.

2.0 Changes

2.1 Development of the task list was the result of several iterations on the analysis of past, present and future requirements for operation.

2.1.1 Documentation guidance was taken from publication ORAU-264, April 1986.

2.1.2 Although the general guidance of the documentation was applied some changes in form and specifics have been made to adapt to facility specific conditions.

2.1.3 A consistent use of task items, terminology, verb usage, and list order should be continued for changes to the list.

2.2 Supervisory Operator format, section A, applies several general headings (X.0) for major task categories and supervisor responsibilities.

2.2.1 The outline heading level (X.X) consist of action verbs for each major category of tasks.

2.2.2 Each task for a specific verb is listed at the next outline sublevel (X.X.X).

2.2.3 Exceptions to the format should be avoided.

2.3 Reactor Operator format, section B, applies general headings (X.X) to categorize the tasks into similar areas.

2.3.1 The next outline sublevel (X.X.X) consists of action verbs for each specific category of tasks.

2.3.2 Each task for a specific verb is listed at the next outline sublevel (X.X.X.X).

2.3.3 A few exceptions to the outline format maybe appropriate to accurately list a task.

2.4 The function of this task list and the changes of any task should be established so that changes to the list are infrequent.

TASK LIST

Section A
Supervisory Operator

The University of Texas TRIGA

Task List - Supervisory Reactor Operator	P	S	Comments
1.0 NORMAL REACTOR OPERATIONS			
1.1 Assume responsibility for:			
1.1.1 Overall facility safety	*	*	
1.1.2 User facility interface			
1.2 Assure compliance with:			
1.2.1 Licenses	*	*	
1.2.2 Technical Specifications	*	*	
1.2.3 Standard Operating Procedures	*	*	
1.2.4 Security Plan		*	
1.2.5 Emergency Plan		*	
1.2.6 Assurance Program: quality, training		*	
1.2.7 Safety Program: fire, radiation		*	
1.3 Authorize access to:			
1.3.1 Reactor facility			
1.3.2 Sensitive areas			
1. 3.3 Controlled materials			
1.3.4 Experiment systems	*		
1.3.5 Insert/remove samples for irradiation	*		
1.4 Direct actions to:			
1.4.1 Repair/replace auxiliary system components[1]	*	*	
1.4.2 Operate purification system	*	*	
1.4.3 Operate cooling system	*	*	
1.4.4 Install control rod drive	*	*	
1.4.5 Removal control rod drive	*	*	

[1]These systems include: Cooling, Purification, and Makeup; Fuel Storage; Material Handling; Waste Handling; Storage; Effluent; and Facility Utilities.

Task - List Supervisory Reactor Operator	P	S	Comments
1.0 NORMAL REACTOR OPERATIONS (cont.)			
1.4 Direct actions to (cont.):			
1.4.6 Test control rod	*	*	
1.4.7 Load core	*		
1.4.8 Position detectors	*		
1.4.9 Assemble experiment	*		
1.4.10 Install experiment (fixed)	*		
1.4.11 Install experiment (nonfixed)	*		
1.4.12 Perform experiment	*	*	
1.4.13 Prepare experiment (fixed)	*	*	
1.4.14 Prepare experiment (nonfixed)	*	*	
1.4.15 Move fuel	*		
1.4.16 Ship/receive fuel	*		
1.4.17 Change power level	*	*	
1.4.18 Repair/replace reactor component	*		
1.4.19 Repair/replace component in reactor control instrumentation or safety system	*		
1.4.20 Restart reactor after inadvertent scram	*	*	
1.4.21 Operate reactor	*	*	
1.4.22 Shutdown reactor	*	*	
1.4.23 startup reactor	*	*	
1.4.24 Change shift	*	*	
1.5 Maintain:			
1.5.1 As-built drawings/blueprints/ specifications			
1.5.2 Applicable regulations file			
1.5.3 Change/supervisor log			

Task List - Supervisory Reactor Operator	P	S	Comments
1.0 NORMAL REACTOR OPERATIONS (cont.)			
1.5 Maintain (cont.):			
1.5.4 Console/scram log	*	*	
1.5.5 Experiment/irradiation log	*	*	
1.5.6 Fuel measurements log	*	*	
1.5.7 Reactor calibration log	*	*	
1.5.8 Surveillance measurements log	*	*	
1.5.9 Water system log	*	*	
1.6 Assist:			
1.6.1 Experiment design		*	
1.6.2 Experiment safety review	*	*	
1.6.3 Facility modification		*	
1.6.4 Reactor modification	*	*	
1.7 Develop:			
1.7.1 Facility programs			
1.8 Operate:			
1.8.1 Facility computer		*	
1.9 Procure:			
1.9.1 Fuel elements	*		
1.9.2 Control elements	*		
1.9.3 Reactor components	*		
1.10 Provide:			
1.10.1 Experimenter training	*	*	
1.10.2 Fire, police training			
1.10.3 Emergency training	*		
1.10.4 Operator training	*		

Task List - Supervisory Reactor Operator	P	S	Comments
1.0 NORMAL REACTOR OPERATIONS (cont.)			
1.11 Review and Approve:			
1.11.1 Bridge changes	*		
1.11.2 Change of set points	*		
1.11.3 Core load configurations	*		
1.11.4 Experiments	*		
1.11.5 Facility tours/demonstrations			
1.11.6 Fuel movements	*		
1.11.7 Power calibration	*		
1.11.8 Radioactivity releases	*		
1.11.9 Reactor startup checklist		*	
1.11.10 Reactor shutdown checklist		*	
1.11.11 Console/scram log		*	
1.11.12 Use of bypass circuits	*		
1.12 Schedule or Coordinate:			
1.12.1 Emergency drills	*		
1.12.2 Emergency support training	*		
1.12.3 Experiment assembly	*	*	
1.12.4 Experiment maintenance	*	*	
1.12.5 Equipment operation		*	
1.12.6 Facility maintenance		*	
1.12.7 Pool maintenance	*	*	
1.12.8 Console maintenance	*	*	
1.12.9 Reactor maintenance	*	*	
1.12.10 Health physics support			
1.12.11 Reactor staff functions			

Task List - Supervisory Reactor Operator	P	S	Comments
1.12 Schedule or Coordinate (cont.):			
1.12.12 Tours/demonstrations			
2.0 ABNORMAL/EMERGENCY REACTOR OPERATIONS			
2.1 Analyze:			
2.1.1 Unusual circumstances	*	*	
2.2 Authorize:			
2.2.1 Personnel exposure above administrative limits	*		
2.3 Determine:			
2.3.1 Category of unusual events	*	*	
2.4 Direct:			
2.4.1 Corrective action of reportable circumstances	*		
2.4.2 Emergency response in accordance with emergency plan	*	*	
2.4.3 Security response in accordance with security plan	*	*	
2.5 Procure:			
2.5.1 Parts and components	*	*	
2.6 Remain:			
2.6.1 On 24-hour call	*	*	
2.7 Report:			
2.7.1 Reportable occurrences	*		
2.7.2 Change in fitness for duty	*		
2.7.3 Change in administrative personnel	*		
2.8 Respond:			
2.8.1 To unusual circumstances	*	*	

Task List - Supervisory Reactor Operator	P	S	Comments
3.0 SURVEILLANCE			
3.1 Approve:			
3.1.1 Auxiliary systems surveillance		*	
3.1.2 Confinement system surveillance		*	
3.1.3 Control rod calibration	*		
3.1.4 Control rod function	*		
3.1.5 Core excess reactivity	*		
3.1.6 Core shutdown reactivity	*		
3.1.7 Emergency equipment	*		
3.1.8 Security equipment	*		
3.1.9 Fuel/element measurements	*		
3.1.10 Power/energy calibration - steady-state pulse	*		
3.1.11 Radiation surveys/instrument	*	*	
3.1.12 Reactivity parameters	*	*	
3.1.13 Reactor controls	*	*	
3.1.14 Reactor instruments	*	*	
3.2 Direct:			
3.2.1 Measurement of reactor parameters		*	
3.3 Initiate/Revise/Rescind:			
3.3.1 Surveillance procedures	*		
3.4 Inspect:			
3.4.1 Facility		*	
3.4.2 Reactor and associated equipment		*	

Task List - Supervisory Reactor Operator	P	S	Comments
3.0 SURVEILLANCE (cont.)			
3.5 Perform:			
3.5.1 Auxiliary systems surveillance		*	
3.5.2 Confinement system surveillance		*	
3.5.3 Control rod calibration	*	*	
3.5.4 Control rod function	*	*	
3.5.5 Core excess reactivity	*	*	
3.5.6 Core shutdown reactivity	*	*	
3.5.7 Emergency equipment		*	
3.5.8 Security equipment		*	
3.5.9 Fuel/element measurements	*	*	
3.5.10 Power/energy calibration - steady-state pulse	*	*	
3.5.11 Radiation surveys/instrument		*	
3.5.12 Reactivity parameters	*	*	
3.5.13 Reactor controls	*	*	
3.5.14 Reactor instruments	*	*	
3.6 Review and Approve:			
3.6.1 Emergency equipment checklists		*	
3.6.2 Security detection checklists		*	
3.6.3 Surveillance procedures	*		
3.6 Review and Approve (cont.):			
3.6.4 Weekly, monthly, quarterly, semiannual annual checklists		*	
3.6.5 Calibraton methods, procedures	*		

Task List - Supervisory Reactor Operator	P	S	Comments
3.0 SURVEILLANCE (cont.)			
3.7 Schedule:			
3.7.1 Auxiliary systems surveillance		*	
3.7.2 Confinement system surveillance		*	
3.7.3 Control rod calibration	*		
3.7.4 Control rod function	*		
3.7.5 Core excess reactivity	*		
3.7.6 Core shutdown reactivity	*		
3.7.7 Emergency equipment		*	
3.7.8 Security equipment		*	
3.7.9 Fuel/element measurements	*		
3.7.10 Power/energy calibration - steady-state pulse	*		
3.7.11 Radiation surveys		*	
3.7.12 Reactivity parameters		*	
3.7.13 Reactor controls		*	
3.7.14 Reactor instruments		*	
3.8 Supervise:			
3.8.1 Auxiliary systems surveillance		*	
3.8.2 Confinement system surveillance		*	
3.8.3 Control rod calibration	*		
3.8.4 Control rod function	*		
3.8.5 Core excess reactivity	*		
3.8.6 Core shutdown reactivity	*		
3.8.7 Emergency equipment		*	
3.8.8 Security equipment		*	

Task List - Supervisory Reactor Operator	P	S	Comments
3.0 SURVEILLANCE (cont.)			
3.8 Supervise (cont.)			
3.8.9 Fuel/element measurements	*		
3.8.10 Power/energy calibration - steady-state pulse	*		
3.8.11 Radiation surveys		*	
3.8.12 Reactivity parameters		*	
3.8.13 Reactor controls		*	
3.8.14 Reactor instruments		*	
4.0 MAINTENANCE			
4.1 Approve:			
4.1.1 Auxilary system maintenance			
4.1.2 Bridge maintenance	*		
4.1.3 Control rod maintenance	*		
4.1.4 Experiment system maintenance	*		
4.1.5 General maintenance		*	
4.1.6 Pool and water system maintenance	*		
4.1.7 Reactor instrumentation maintenance	*		
4.2 Authorize:			
4.2.1 Return to service	*		
4.3 Initiate/Revise/Rescind:			
4.3.1 Repair actions	*		
4.4 Issue:			
4.4.1 Radiation exposure control instructions	*		
4.4.2 Status tags (includes all colors)	*	*	

Task List - Supervisory Reactor Operator	P	S	Comments
4.0 MAINTENANCE (cont.)			
4.5 Perform:			
4.5.1 Auxilary system maintenance		*	
4.5.2 Bridge maintenance	*	*	
4.5.3 Control rod maintenance	*	*	
4.5.4 Experiment system maintenance	*	*	
4.5.5 General maintenance		*	
4.5.6 Pool and water system maintenance	*	*	
4.5.7 Reactor instrumentation maintenance	*	*	
4.6 Review and Approve:			
4.6.1 Reactor operation outages	*		
4.6.2 Use of jumpers/replacements	*		
4.6.3 Maintenance procedures	*		
4.6.4 Auxilary systems maintenance log			
4.6.5 Control system log			
4.6.6 Calibration methods, results	*		
4.7 Schedule:			
4.7.1 Auxilary system maintenance		*	
4.7.2 Bridge maintenance	*		
4.7.3 Control rod maintenance	*		
4.7.4 Experiment system maintenance	*		
4.7.5 General maintenance		*	
4.7.6 Pool and water system maintenance	*		
4.7.7 Reactor instrumentation maintenance	*		

Task List - Supervisory Reactor Operator	P	S	Comments
4.0 MAINTENANCE (cont.)			
4.8 Supervise:			
4.8.1 Auxilary system maintenance		*	
4.8.2 Bridge maintenance	*	*	
4.8.3 Control rod maintenance	*	*	
4.8.4 Experiment system maintenance	*	*	
4.8.5 General maintenance		*	
4.8.6 Pool and water system maintenance	*	*	
4.8.7 Reactor instrumentation maintenance	*	*	
5.0 ADMINISTRATION			
5.1 Accommodate:			
5.1.1 Auditors		*	
5.1.2 Inspectors		*	
5.2 Attend:			
5.2.1 Committee meetings			
5.2.2 Staff meetings			
5.3 Authorize:			
5.3.1 Personnel access to facility			
5.4 Calculate:			
5.4.1 Fuel material burnup		*	
5.5 Control:			
5.5.1 Personnel access to confinement	*	*	
5.5.2 Personnel access to control area	*	*	
5.5.3 Personnel access to facility		*	

Task List - Supervisory Reactor Operator	P	S	Comments
5.0 ADMINISTRATION (cont.)			
5.6 Direct:			
5.6.1 Experimenters	*	*	
5.6.2 Operations personnel	*	*	
5.7 Generate:			
5.7.1 Administrative memos			
5.7.2 Standard Operating Procedures	*		
5.8 Initiate:			
5.8.1 Maintenance procedures		*	
5.8.2 Surveillance procedures		*	
5.8.3 Changes to standard operating procedures	*	*	
5.9 Prepare/Update:			
5.9.1 Safety reviews			
5.9.2 Annual report		*	
5.9.3 As-built drawings		*	
5.9.4 Program reports			
5.9.5 Licensee event reports			
5.9.6 Material balance reports			
5.9.7 Material transaction reports			
5.9.8 Monthly reports		*	
5.9.9 Standard operating procedures		*	
5.9.10 Quarterly reports and other material for reactor committee		*	
5.10 Review:			
5.10.1 Audit reports			
5.10.2 Emergency plans	*	*	

Task List - Supervisory Reactor Operator	P	S	Comments
5.0 ADMINISTRATION (cont.)			
5.10 Review (cont.):			
5.10.3 Security plans	*	*	
5.10.4 Standard operating procedures	*	*	
5.11 Schedule:			
5.11.1 Facility inspections			
5.11.2 Committee meetings			
5.11.3 Shift assignments			
5.12 Submit:			
5.12.1 License amendments			

TASK LIST

Section B
Reactor Operator

The University of Texas TRIGA

Task List - Reactor Operator	P	S	Comments
1.0 GENERAL			
1.1 Operator			
1.1.1 Maintain:			
1.1.1.1 Clean laboratory conditions			
1.1.1.2 Equipment design function	*		
1.1.1.3 Operation specifications	*		
1.1.2 Assist:			
1.1.2.1 Personnel involved in facility programs			
1.1.2.2 Performance of scheduled experiments			
1.1.2.3 Tours of facility			
1.1.3 Attend:			
1.1.3.1 Facility meetings			
1.1.3.2 Training sessions	*		
1.2 Trainee			
1.2.1 Learn:			
1.2.1.1 Basic theory and principles			
1.2.1.2 Facility characteristics and operation			
1.2.1.3 Normal and abnormal operation procedures			
1.2.2 Perform:			
1.2.2.1 Training tasks	*		
1.2.3 Develop:			
1.2.3.1 Operator skills	*		
1.2.4 Demonstrate:			
1.2.4.1 Knowledge of principles and operation	*		
1.2.4.2 Proficient operation skills	*		

Task List - Reactor Operator	P	S	Comments
1.0 GENERAL (cont.)			
1.2 Trainee (cont.)			
1.2.4 Demonstrate (cont.)			
1.2.4.3 Responsible attitude	*		
2.0 REACTOR CORE AND POOL STRUCTURE			
2.1 Reactor			
2.1.1 Inspect:			
2.1.1.1 Core	*		
2.1.1.2 Reflector	*		
2.1.1.3 Source assembly	*		
2.1.1.4 Detector canisters	*		
2.1.2 Perform:			
2.1.2.1 Fuel movements	*	*	
2.1.2.2 Source relocation	*	*	
2.2 Pool			
2.2.1 Check:			
2.2.1.1 Bridge	*		
2.2.1.2 Rod drives	*		
2.2.1.3 Pool covers			
2.2.1.4 Fuel storage	*		
2.2.1.5 Gamma irradiator	*		
2.2.2 Identify:			
2.2.2.1 Non standard conditions			
2.2.2.2 Unnecessary loose objects			

Task List - Reactor Operator	P	S	Comments
2.0 REACTOR CORE AND POOL STRUCTURE (cont.)			
2.3 Penetrations			
2.3.1 Examine:			
2.3.1.1 Beam tubes	*		
2.3.1.2 Irradiation tubes	*		
2.3.1.3 Experiment facilities	*		
2.3.2 Secure:			
2.3.2.1 Access points			
3.0 WATER PURIFICATION AND COOLING SYSTEMS			
3.1 Purification			
3.1.1 Operate:			
3.1.1.1 Startup and shutdown			
3.1.1.2 Purification pump			
3.1.1.3 Isolation valves			
3.1.2 Adjust:			
3.1.2.1 Flow rate			
3.1.3 Align:			
3.1.3.1 Valves			
3.1.4 Monitor:			
3.1.4.1 Flow rate			
3.1.4.2 Temperature			
3.1.4.3 Pressure drops			
3.1.4.4 Conductivity, pH	*		
3.1.4.5 Demineralizer radioactivity	*		
3.1.5 Log:			
3.1.5.1 Flow rate			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.1 Purification (cont.)			
3.1.5 Log (cont.)			
3.1.5.2 Temperature			
3.1.5.4 Demineralizer radioactivity			
3.1.6 Check:			
3.1.6.1 Inlet			
3.1.6.2 Outlet			
3.1.6.3 Skimmer			
3.1.6.4 Fittings			
3.1.6.5 Valve alignments			
3.1.6.6 Pump shaft and seals	*		
3.1.6.7 Motor bearings and coupling	*		
3.1.7 Replace:			
3.1.7.1 Resins			
3.1.7.2 Filters			
3.1.7.3 Gaskets			
3.1.8 Order:			
3.1.8.1 Resins			
3.1.8.2 Filters			
3.1.8.3 Gaskets			
3.2 Primary Coolant			
3.2.1 Operate:			
3.2.1.1 Startup and shutdown			
3.2.1.2 Primary pump			
3.2.1.3 Block valves			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.2 Primary coolant (cont.)			
3.2.2 Adjust:			
3.2.2.1 Flow rate		*	
3.2.3 Align:			
3.2.3.1 Valves			
3.2.4 Monitor:			
3.2.4.1 Flow rate			
3.2.4.2 Pressure at inlet and outlet			
3.2.4.3 Temperature at inlet and outlet			
3.2.5 Log:			
3.2.5.1 Flow rate			
3.2.5.2 Pressure at inlet and outlet			
3.2.5.3 Temperature at inlet and outlet			
3.2.6 Check:			
3.2.6.1 Inlet			
3.2.6.2 Outlet			
3.2.6.3 Diffuser			
3.2.6.4 Fittings			
3.2.6.5 Valve alignments			
3.2.6.6 Pump shaft and seals	*		
3.2.6.7 Motor bearings and coupling	*		
3.3 Secondary Coolant			
3.3.1 Operate:			
3.3.1.1 Startup and shutdown			
3.3.1.2 Secondary pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.2 Primary coolant (cont.)			
3.2.2 Adjust:			
3.2.2.1 Flow rate		*	
3.2.3 Align:			
3.2.3.1 Valves			
3.2.4 Monitor:			
3.2.4.1 Flow rate			
3.2.4.2 Pressure at inlet and outlet			
3.2.4.3 Temperature at inlet and outlet			
3.2.5 Log:			
3.2.5.1 Flow rate			
3.2.5.2 Pressure at inlet and outlet			
3.2.5.3 Temperature at inlet and outlet			
3.2.6 Check:			
3.2.6.1 Inlet			
3.2.6.2 Outlet			
3.2.6.3 Diffuser			
3.2.6.4 Fittings			
3.2.6.5 Valve alignments			
3.2.6.6 Pump shaft and seals	*		
3.2.6.7 Motor bearings and coupling	*		
3.3 Secondary Coolant			
3.3.1 Operate:			
3.3.1.1 Startup and shutdown			
3.3.1.2 Secondary pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.3 Secondary Coolant (cont.)			
3.3.1 Operate (cont.):			
3.3.1.3 Block valves			
3.3.2 Adjust:			
3.3.2.1 Flow rate		*	
3.3.3 Align:			
3.3.3.1 Valves			
3.3.4 Monitor:			
3.3.4.1 Flow rate			
3.3.4.2 Pressure at inlet and outlet			
3.3.4.3 Temperature at inlet and outlet			
3.3.5 Log:			
3.3.5.1 Flow rate			
3.3.5.2 Pressure at inlet and outlet			
3.3.5.3 Temperature at inlet and outlet			
3.3.6 Check:			
3.3.6.1 Chilled water supply			
3.3.6.2 Chilled water return			
3.3.6.3 Fittings			
3.3.6.4 Valve alignments			
3.3.6.5 Pump shaft and seals	*		
3.3.6.6 Motor bearings and coupling	*		
3.4 Pool Water			
3.4.1 Inspect:			
3.4.1.1 Appearance			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.4 Pool Water (cont.)			
3.4.1 Inspect (cont.):			
3.4.1.2 Penetrations	*		
3.4.2 Monitor:			
3.4.2.1 Pool level	*		
3.4.2.2 Temperature	*		
3.4.2.3 Conductivity	*		
3.4.2.4 Radioactivity	*		
3.4.3 Operate:			
3.4.3.1 Makeup			
3.4.3.2 Cleanup			
3.4.3.3 Skimmer			
3.4.3.4 Diffuser			
3.4.4 Collect:			
3.4.4.1 Water samples			
3.4.5 Measure:			
3.4.5.1 Power heat rate	*		
3.4.6 Log:			
3.4.6.1 Pool level changes for fill			
3.4.6.2 Pool level changes for drain			
3.4.6.3 Water activity			
3.4.6.4 Power heat rate			
3.5 Other			
3.5.1 Monitor:			
3.5.1.1 Foundation pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.5 Other (cont.)			
3.5.2 Operate:			
3.5.1 Sump pump		*	
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS			
4.1 General Operation			
4.1.1 Actuate:			
4.1.1.1 Control console power			
4.1.1.2 Power to other systems			
4.1.2 Confirm:			
4.1.2.1 Proper operation requirements		*	
4.1.2.2 Status of operation equipment	*	*	
4.1.2.3 Status of experiment components	*	*	
4.1.2.4 Ventilation exhaust conditions		*	
4.1.3 Check:			
4.1.3.1 Safety equipment			
4.1.3.2 Security equipment			
4.1.3.3 Emergency equipment			
4.1.3.4 Communication capability			
4.1.3.5 Non control console systems			
4.2 Control Console Operation Mode			
4.2.1 Switch:			
4.2.1.1 Console mode			
4.2.1.2 Magnet power			
4.2.1.3 Demand power			

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.2 Control Console Operation Mode			
4.2.2 Acknowledge:			
4.2.2.1 Status condition changes, modes	*		
4.2.2.2 Response to console display menus	*		
4.2.2.3 Valid system response to calibration	*		
4.2.3 Verify:			
4.2.3.1 Annunciation of console status panel			
4.2.3.2 Function of console display			
4.2.3.3 Function of entry keyboard			
4.2.4 Monitor:			
4.2.4.1 Graphic display information			
4.2.4.2 Direct parameter indicators			
4.2.4.3 Control rod operation panel			
4.2.5 Implement:			
4.2.5.1 DOS or scram operational			
4.2.5.2 Log process for valid operator			
4.2.5.3 Programmed calibration and test	*		
4.2.5.4 Operational mode; non pulse, automatic			
4.2.5.5 Operational mode; pulse, square wave			
4.2.5.6 Special function key processes			
4.2.5.7 System diagnostic routines			
4.2.6 Observe:			
4.2.6.1 Output trends			
4.2.7 Measure:			
4.2.7.1 Values at circuit test points	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.2. Control Console Operation Mode (cont.)			
4.2.8 Adjust:			
4.2.8.1 Values at circuit test points	*	*	
4.2.9 Determine:			
4.2.9.1 Circuit test error conditions	*	*	
4.2.10 Replace:			
4.2.10 Specified component failures	*	*	
4.2.11 Operate:			
4.2.11.1 Console printer for log records			
4.2.12 Log:			
4.2.12.1 System generated log inforamtion			
4.3 Rod Operation Interlock Logic			
4.3.1 Control:	*		
4.3.1.1 Rod up motion	*		
4.3.1.2 Rod down motion	*		
4.3.1.3 Rod drop insertion	*		
4.3.1.4 Magnet current power	*		
4.3.1.5 Transient fire switch	*		
4.3.2 Check:			
4.3.2.1 Control switch actuations	*		
4.3.2.2 Count rate interlock	*		
4.3.2.3 Pulse interlock	*		
4.3.2.4 Scram function	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.3 Rod Operation Interlock Logis (cont.)			
4.3.3 Verify:			
4.3.3.1 Supplemental interlocks	*		
4.3.3.2 Non pulse interlocks	*		
4.3.3.3 Pulse interlocks	*		
4.3.3.4 Rod rundown			
4.4 Scram Safety System Logic			
4.4.1 Specify:			
4.4.1.1 Scram set point changes	*	*	
4.4.1.2 Log entry for scram		*	
4.4.2 Check:			
4.4.2.1 Manual scram switch	*		
4.4.2.2 CSC processor failure	*		
4.4.2.3 DAC processor failure	*		
4.4.2.4 CSC/DAC communication loss	*		
4.4.2.5 Fuel temperature (2)	*		
4.4.2.6 Percent power (2)	*		
4.4.2.7 High Voltage (2)	*		
4.4.2.8 Amplifier/logic power (2)	*		
4.4.2.9 Power rate of change (2)	*		
4.4.2.10 Low source count (2)	*		
4.4.2.11 Pool level low	*		
4.4.2.12 External scram	*		
4.4.3 Determine:			
4.4.3.1 Scram condition	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.4 Scram Safety System Logic (cont.)			
4.4.3 Determine (cont.):			
4.4.3.2 Reason for condition	*	*	
4.4.3.3 Restart requirements	*	*	
4.4.3.4 Log entry for scram			
4.5 Fuel Temperature Channels			
4.5.1 Setup:			
4.5.1.1 Instrument element position	*	*	
4.5.1.2 Thermocouple source	*	*	
4.5.2 Check:			
4.5.2.1 Operability	*		
4.5.2.2 Calibration	*		
4.5.2.3 Amplifier/logic	*		
4.5.2.4 Set points (scram)	*		
4.5.3 Log:			
4.5.3.1 Non pulse temperature values			
4.5.3.2 Pulse temperature values			
4.6 Percent Power Channels			
4.6.1 Setup:			
4.6.1.1 Detector chamber position	*	*	
4.6.1.2 Non pulse operation values	*	*	
4.6.2 Check:			
4.6.2.1 Operability	*		
4.6.2.2 Calibration	*		
4.6.2.3 Detector HV	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.6 Percent Power Channels (cont.)			
4.6.2 Check (cont.):			
4.6.2.4 Amplifier/logic	*		
4.6.2.5 Set points (scram)	*		
4.6.2.6 Demand power value	*		
4.6.3 Log:			
4.6.3.1 At power operating conditions			
4.6.3.2 Safety channel status conditions			
4.7 Pulse Power Channel			
4.7.1 Setup:			
4.7.1.1 Detector chamber position	*	*	
4.7.1.2 Pulse transient conditions	*	*	
4.7.2 Check:			
4.7.2.1 Operability	*		
4.7.2.2 Calibration	*		
4.7.2.3 Detector HV	*		
4.7.2.4 Amplifier/logic	*		
4.7.2.5 Set points (scram)	*		
4.7.2.6 Transient rod position	*		
4.7.3 Monitor:			
4.7.3.1 Pulse peak and energy data			
4.7.4 Log:			
4.7.4.1 Pulse event data			

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.8 Pool Water System Channels			
4.8.1 Check:			
4.8.1.1 Set points			
4.8.1.2 Level sensors			
4.8.1.3 Temperature sensors	*		
4.8.1.4 Conductivity sensors	*		
4.8.1.5 Flow rate measurement	*		
4.8.2 Monitor:			
4.8.2.1 Pool level status	*		
4.8.2.2 Pool temperature			
4.8.2.3 Flow rate			
4.9 Radiation Monitor Channels			
4.9.1 Check:			
4.9.1.1 High set points	*		
4.9.1.2 Low set points			
4.9.1.3 Area monitor	*		
4.9.1.4 Gaseous monitor	*		
4.9.1.5 Particulate monitor	*		
4.9.2 Monitor:			
4.9.2.1 Counts per minute			
4.9.2.2 Millirem per hour			

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS			
5.1 General			
5.1.1 Verify:			
5.1.1.1 Correct experiment approval	*		
5.1.1.2 Classification of experiment	*		
5.1.2 Review:			
5.1.2.1 Previous related experiment	*	*	
5.1.2.2 Projected experiment results	*	*	
5.1.3 Estimate:			
5.1.3.1 Reactivity effect to the core of experiment structure	*	*	
5.1.3.2 Reactivity effect to the core of removeable materials	*	*	
5.1.3.3 Radiation levels of experiment structure	*	*	
5.1.3.4 Radiation levels of removeable materials	*	*	
5.1.4 Monitor:			
5.1.4.1 Experiment or irradiation conditions	*		
5.1.5 Encapsulate			
5.1.5.1 Materials for irradiation	*		
5.1.6 Insert:			
5.1.6.1 Samples for irradiation	*		
5.1.7 Remove:			
5.1.7.1 Samples after irradiation	*		
5.1.8 Log:			
5.1.8.1 Experiment equipment			
5.1.8.2 Irradiation samples			

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS (cont.)			
5.2 Beam tubes			
5.2.1 Manipulate:			
5.2.1.1 Beam tube components	*		
5.2.1.2 Beam tube experiments	*		
5.2.1.3 Radiation shield materials	*		
5.2.2 Monitor:			
5.2.2.1 Beam tube status			
5.2.2.2 Area radiation levels			
5.2.2.3 Experiment progress			
5.2.2.4 Area personnel access			
5.3 Rotary Specimen Rack			
5.3.1 Operate:			
5.3.1.1 Rack drive	*		
5.3.1.2 Sample removal device	*		
5.3.1.3 Sample insertion device	*		
5.3.2 Monitor:			
5.3.2.1 Contamination of access tube	*		
5.4 Pneumatic Transfer Tube			
5.4.1 Operate:			
5.4.1.1 Transfer terminal	*		
5.4.1.2 Permit switch	*		
5.4.1.3 Blower supply	*		
5.4.1.4 Propellant gas supply	*		

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS (cont.)			
5.4 Pneumatic Transfer Tube (cont.)			
5.4.2 Monitor:			
5.4.2.1 Access port radiation levels			
5.4.3 Instruct:			
5.4.3.1 Experimenter			
5.5 Tri-cut Core Position To be determined	*	*	
5.6 Hex-cut Core Position To be determined	*	*	
5.7 In-Core Irradiation To be determined	*	*	
5.8 Ex-core Irradiation To be determined	*	*	
5.9 Material Processing and Transfer			
5.9.1 Prepare:			
5.9.1.1 Material encapsulation	*		
5.9.1.2 Transfer documents			
5.9.1.3 Shipping papers			
5.9.2 Perform:			
5.9.2.1 Package labeling			
5.9.2.2 Package contamination smears			
5.9.2.3 Package radiation surveys			
5.9.3 Monitor:			
5.9.3.1 Transfer to authorized agent			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES			
6.1 HVAC Equipment			
6.1.1 Test:			
6.1.1.1 Purge air system fan	*		
6.1.1.2 Isolation damper function	*		
6.1.1.3 Supply, return, exhaust fans			
6.1.1.4 Filter differential pressure	*		
6.1.1.5 Air sample ports			
6.1.2 Start/stop:			
6.1.2.1 Reactor room HVAC fans			
6.1.2.2 Reactor room purge fan	*		
6.1.3 Monitor:			
6.1.3.1 Air radioactivity of purge system			
6.1.3.2 Room air activity confinement signal			
6.1.3.3 Differential pressure to reactor room			
6.1.4 Maintain:			
6.1.4.1 Reactor room pressure negative			
6.1.4.2 Reactor room confinement control			
6.1.5 Service:			
6.1.5.1 HEPA filter and prefilter	*		
6.2 Deionized Water			
6.2.1 Check:			
6.2.1.1 Water supply status			
6.2.1.2 Filter status			
6.2.1.3 Resin status			
6.2.1.4 Conductivity			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES (cont.)			
6.2 Deionized Water (cont.)			
6.2.2 Service:			
6.2.2.1 Filter, resin	*		
6.3 Compressed Air			
6.3.1 Check:			
6.3.1.1 Compressor status			
6.3.1.2 Filter status			
6.3.1.3 Dryer status			
6.3.2 Service:			
6.3.2.1 Filter, dryer	*		
6.4 Electrical			
6.4.1 Operate:			
6.4.1.1 Reactor area power distributions			
6.4.2 Locate:			
6.4.2.1 Electrical power circuit source			
6.5 Miscellaneous			
6.5.1 Operate:			
6.5.1.1 Compressed gas cylinders	*		
6.5.1.2 Vacuum pumps and valves	*		
6.5.1.3 Fire control equipment	*		
6.5.1.4 Overhead bridge crane	*		
6.5.1.5 Communication system	*		
6.5.2 Replace:			
6.5.2.1 Gas cylinders			
6.5.2.2 Liquid nitrogen			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES (cont.)			
6.5 Miscellaneous (cont.)			
6.5.2 Replace (cont.):			
6.5.2.3 First aid supplies			
6.5.3 Perform:			
6.5.3.1 Minor repairs			
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL			
7.1 TRIGA Fuel			
7.1.1 Operate:			
7.1.1.1 Fuel handling tools	*		
7.1.1.2 Fuel inspection device	*		
7.1.1.3 Fuel transfer cask	*		
7.1.2 Move:			
7.1.2.1 Fuel into reactor core	*	*	
7.1.2.2 Fuel out of reactor core	*	*	
7.1.2.3 Fuel followed control rods	*	*	
7.1.2.4 To or from storage racks	*		
7.1.2.5 To or from storage wells	*		
7.1.2.6 To or from transfer casks	*		
7.1.3 Clean:			
7.1.3.1 Elements			
7.1.4 Locate:			
7.1.4.1 Failed elements	*	*	
7.1.5 Connect/disconnect:			
7.1.5.1 Instrumented elements	*	*	

Task List - Reactor Operator	P	S	Comments
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL (cont.)			
7.1 TRIGA Fuel (cont.)			
7.1.6 Install/remove:			
7.1.6.1 Fuel follow control elements	*	*	
7.1.7 Measure:			
7.1.7.1 Element length	*		
7.1.7.2 Element bow	*		
7.1.7.3 Go gauge	*		
7.1.8 Document:			
7.1.8.1 Fuel movements		*	
7.1.8.2 Operation history		*	
7.1.8.3 Inspection results		*	
7.2 Subcritical Assembly			
7.2.1 Monitor:			
7.2.1.1 Fuel pellets used	*		
7.2.1.2 Use of core cylinder	*		
7.2.1.3 Use of reflector components			
7.2.1.4 Insertion of neutron sources	*	*	
7.2.2 Maintain:			
7.2.2.1 SNM material access control			
7.2.2.2 SNM material storage			
7.3 Isotopic Sealed Sources			
7.3.1 Monitor:			
7.3.1.1 Use of sealed sources	*	*	
7.3.1.2 Applications with sealed sources			

Task List - Reactor Operator	P	S	Comments
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL (cont.)			
7.3 Isotopic Sealed Sources (cont.)			
7.3.2 Maintain:			
7.3.2.1 Leak tests of sealed sources		*	
7.3.2.2 Inventory of sealed sources		*	
7.4 Standard or Reference Materials			
7.4.1 Monitor:			
7.4.1.1 Use of SNM materials	*		
7.4.1.2 Applications with SNM materials			
7.4.2 Maintain:			
7.4.2.1 Inventory of detectors with SNM		*	
7.4.2.2 Inventory of reference materials		*	
8.0 RADIOACTIVE EFFLUENT AND WASTE			
8.1 Gaseous			
8.1.1 Test:			
8.1.1.1 Confinement system (dampers, seals)	*		
8.1.1.2 Stack air monitor (gas, particulate)	*		
8.1.2 Check:			
8.1.2.1 Exhaust stack filters			
8.1.2.2 Purge air filters	*		
8.1.2.3 Hood fan filters	*		
8.1.2.4 Sample filters	*		
8.1.3 Operate:			
8.1.3.1 Reactor room ventilation system	*		
8.1.3.2 Purge air exhaust system	*		
8.1.3.3 Hood exhaust air systems	*		

Task List - Reactor Operator	P	S	Comments
8.0 RADIOACTIVE EFFLUENT AND WASTE (cont.)			
8.1 Gaseous (cont.)			
8.1.3 Operate (cont.):			
8.1.3.4 Air sample monitors	*		
8.1.4 Monitor:			
8.1.4.1 Reactor room negative pressure			
8.1.4.2 Continuous air monitors			
8.1.5 Log:			
8.1.5.1 Argon-41 release			
8.2 Liquid			
8.2.1 Check:			
8.2.1.1 Liquid tank levels	*		
8.2.2 Measure:			
8.2.2.1 Beta-gamma activity	*		
8.2.3 Release:			
8.2.3.1 Controlled quantity	*	*	
8.2.4 Log:			
8.2.4.1 Isotope			
8.2.4.2 Concentration			
8.2.4.3 Volume released			
8.3 Solid			
8.3.1 Specify:			
8.3.1.1 Isotopes present			
8.3.1.2 Physical or chemical form			
8.3.2 Dispose:			
8.3.2.1 Specific materials	*	*	

Task List - Reactor Operator	P	S	Comments
8.0 RADIOACTIVE EFFLUENT AND WASTE (cont.)			
8.3 Solid (cont.)			
8.3.2 Dispose (cont.):			
8.3.2.2 Irradiated samples	*	*	
8.3.2.3 General waste	*	*	
8.3.3 Log:			
8.3.3.1 Isotope and total activity			
8.3.3.2 Material forms			
8.3.3.3 Waste volume			
9.0 SURVEILLANCE AND MAINTENANCE			
9.1 Periodic Surveillance Schedule			
9.1.1 Verify:			
9.1.1.1 Limiting safety system settings	*	*	
9.1.1.2 Limiting conditions for operation	*	*	
9.1.2 Inspect:			
9.1.2.1 Fuel elements	*	*	
9.1.2.2 Control rods	*	*	
9.1.3 Calibrate:			
9.1.3.1 Control rods	*		
9.1.3.2 Reactor power	*		
9.1.3.3 Control console	*		
9.1.3.4 Radiation monitors	*		
9.1.4 Inventory:			
9.1.4.1 Special nuclear material			
9.1.4.2 Radioactive material			
9.1.4.3 Emergency supplies			

Task List - Reactor Operator	P	S	Comments
9.0 SURVEILLANCE AND MAINTENANCE (cont.)			
9.1 Periodic Surveillance Schedule (cont.)			
9.1.4 Inventory (cont.):			
9.1.4.4 General supplies			
9.1.5 Survey:			
9.1.5.1 Leak tests of radioactive source	*		
9.1.5.2 Surface contamination levels	*		
9.1.5.3 Area radiation levels	*		
9.1.5.4 Environment impact	*		
9.2 Non Periodic Maintenance Schedule			
9.2.1 Examine:			
9.2.1.1 Core components	*		
9.2.1.2 Flux detectors	*		
9.2.1.3 Rod drives	*		
9.2.1.4 Experiment systems	*		
9.2.2 Replace:			
9.2.2.1 Reactor core components	*	*	
9.2.2.2 Control console components	*	*	
9.2.2.3 Pool water system components	*		
9.2.2.4 Experiment system components	*		
9.2.3 Maintain:			
9.2.3.1 Experiment counting system			
9.2.3.2 Experiment support equipment			

Task List - Reactor Operator	P	S	Comments
10.0 PHYSICAL SECURITY AND EMERGENCY RESPONSE			
10.1 Controlled Access Area			
10.1.1 Check:			
10.1.1.1 Controlled access area integrity	*		
10.1.1.2 Authorized access to materials			
10.1.1.3 Authorized entry to area			
10.1.1.4 Changes in secure areas			
10.1.2 Control:			
10.1.2.1 Access to reactor room	*		
10.1.2.2 Access to control room	*		
10.1.2.3 SNM storage areas			
10.1.2.4 SNM in use areas			
10.1.2.5 Visitor escort			
10.2 Intrusion Detection			
10.2.1 Check:			
10.2.1.1 Intrusion detection system sensors	*		
10.2.1.2 Alarm status indication	*		
10.2.2 Maintain:			
10.2.2.1 Operation of alarm system			
10.2.2.2 Link for security response			
10.3 Emergency Equipment			
10.3.1 Check:			
10.3.1.1 Emergency supply inventory			
10.3.1.2 Emergency equipment status			
10.3.1.3 Call lists and telephone numbers			
10.3.1.4 Availability of emergency instructions			

Task List - Reactor Operator	P	S	Comments
10.0 PHYSICAL SECURITY AND EMERGENCY RESPONSE (cont.)			
10.3 Emergency Equipment (cont.)			
10.3.2 Recognize:			
10.3.2.1 Emergency conditions	*		
10.3.2.2 Conditions that are unsafe	*		
10.4 Emergency Drills			
10.4.1 Review:			
10.4.1.1 Emergency procedures			
10.4.2 Demonstrate:			
10.4.2.1 Proper response to practice exercise			
11.0 RADIATION SAFETY AND FIRE SAFETY			
11.1 Protective Barriers			
11.1.1 Provide:			
11.1.1.1 Gloves	*		
11.1.1.2 Coveralls	*		
11.1.1.3 Face masks	*		
11.1.1.4 Shoe covers	*		
11.1.1.5 Tape and bags	*		
11.1.1.6 Absorbent paper	*		
11.1.1.7 Cleaning materials	*		
11.1.1.8 Shielding blocks	*		
11.1.1.9 Signs or labels	*		
11.1.2 Control:			
11.1.2.1 Radiation areas	*		
11.1.2.2 Contamination areas	*		

Task List - Reactor Operator	P	S	Comments
11.0 RADIATION SAFETY AND FIRE SAFETY (cont.)			
11.1 Protective Barriers (cont.)			
11.1.3 Replace:			
11.1.3.1 Consumable supplies			
11.2 Portable Survey Instruments			
11.2.1 Calibrate:			
11.2.1.1 Response to reference source	*		
11.2.2 Check:			
11.2.2.1 Operability	*		
11.2.3 Monitor:			
11.2.3.1 Alpha, beta activity			
11.2.3.2 Gamma dose levels			
11.2.3.3 Neutron count rates			
11.3 Installed Monitor Systems			
11.3.1 Calibrate:			
11.3.1.1 Response to reference source	*		
11.3.2 Check:			
11.3.2.1 Operability	*		
11.3.3 Adjust:			
11.3.3.1 Alarm set points	*		
11.3.4 Monitor:			
11.3.4.1 Area radiation levels			
11.3.4.2 Air system count rates			
11.4 Personnel Exposure Monitoring			
11.4.1 Use:			
11.4.1.1 Certified badge dosimeter	*		

Task List - Reactor Operator	P	S	Comments
11.0 RADIATION SAFETY AND FIRE SAFETY (cont.)			
11.4 Personnel Exposure Monitoring (cont.)			
11.4.1 Use:			
11.4.1.2 Supplemental exposure device			
11.4.1.3 Pocket dosimeter			
11.4.1.4 TLD dosimeter			
11.4.2 Determine:			
11.4.2.1 Device sensitivity			
11.4.2.2 Pocket dosimeter exposure	*		
11.4.2.3 TLD dosimeter exposure	*		
11.4.3 Maintain:			
11.4.3.1 Alara principles	*	*	
11.4.3.2 Exposure records			
11.5 Laboratory Measurement Equipment			
11.5.1 Calibrate:			
11.5.1.1 Reference source value	*		
11.5.1.2 Background	*		
11.5.2 Check:			
11.5.2.1 Operability	*		
11.5.3 Measure:			
11.5.3.1 Samples	*		
11.6 Surveys			
11.6.1 Measure:			
11.6.1.1 Area gamma radiation levels	*		
11.6.1.2 Area neutron count rates	*		
11.6.1.3 Removeable count rates	*		

Task List - Reactor Operator	F	S	Comments
11.0 RADIATION SAFETY AND FIRE SAFETY (cont.)			
11.6 Surveys (cont.)			
11.6.2 Collect:			
11.6.2.1 Surface swipes	*		
11.6.2.2 Water sample	*		
11.6.2.3 Filters	*		
11.6.3 Count:			
11.6.3.1 Samples	*		
11.6.4 Log:			
11.6.4.1 Survey results			
11.7 Fire Prevention			
11.7.1 Recognize:			
11.7.1.1 Barriers	*		
11.7.1.2 Fire hazards	*		
11.7.1.3 Ignition sources	*		
11.7.1.4 Flammable materials	*		
11.7.2 Monitor:			
11.7.2.1 Risk areas			
11.8 Fire Protection			
11.8.1 Operate:			
11.8.1.1 Fire alarm system	*		
11.8.1.2 Fire extinguishers	*		
11.8.2 Observe:			
11.8.2.1 Smoke, heat sensors	*		
11.8.3 Evacuate:			
11.8.3.1 Area personnel	*		

Task List - Reactor Operator	P	S	Comments
11.0 RADIATION SAFETY AND FIRE SAFETY (cont.)			
11.8 Fire Protection (cont.)			
11.8.4 Identify:			
11.8.4.1 Entry hazards			
12.0 QUALITY CONTROL AND OPERATOR TRAINING			
12.1 Level One Quality			
12.1.1 Know:			
12.1.1.1 Q-list items	A	A	
12.1.2 Follow:			
12.1.2.1 Procedures	A	A	
12.1.3 Identify:			
12.1.3.1 Changes	A	A	
12.1.3.2 Standards	A	A	
12.2 Level Two Quality			
12.2.1 Check:			
12.2.1.1 Specifications	A	A	
12.2.2 Maintain:			
12.2.2.1 Equivalent standards	A	A	
12.3 Training Lectures			
12.3.1 Attend:			
12.3.1.1 Topical lectures			
12.3.1.2 Classroom presentations			
12.3.2 Work:			
12.3.2.1 Study assignments			
12.3.2.2 Example problems			

Task List - Reactor Operator	P	S	Comments
12.0 QUALITY CONTROL AND OPERATOR TRAINING (cont.)			
12.4 Operation Experience			
12.4.1 Perform.			
12.4.1.1 Facility checklists			
12.4.1.2 Reactor control function	*		
12.4.2 Demonstrate.			
12.4.2.1 Appropriate manipulation skills	*		
12.4.2.2 Knowledge of system performance	*		
12.5 Examination			
12.5.1 Answer:			
12.5.1.1 Questions			
12.5.2 Review:			
12.5.2.1 Mistakes			
13.0 RECORDS MANAGEMENT			
13.1 Logs			
13.1.1 Enter:			
13.1.1.1 Reactor console operation operator, time, mode, nv, nvt			
Calibration periodic, measurements, tests			
Maintenance failures, repairs, replacements			
13.1.1.2 Pool water systems Calibrations, checks Measurements Repairs			
13.1.1.3 Radiation monitor systems Calibrations, checks Measurements Repairs			

Task List - Reactor Operator	F	S	Comments
13.0 RECORDS MANAGEMENT (cont.)			
13.1 Logs (cont.)			
13.1.1 Enter (cont.).			
13.1.1.4 Facility Systems Physical security Emergency response Other systems			
13.1.2 List.			
13.1.2.1 Area access	A		
Keys, codes Visitors, dosimeters			
13.1.2.2 Experiments	A		
Facilities functional Irradiation samples			
13.1.2.3 Operation	A		
Log nv, nvt, daily total Log pulse, each event Scrams, safety system			
13.1.2.4 Fuel (elements, control rods)	A	A	
Movements locations, core, racks, experiment Inspections dimensions, length, bow, visual Burnup initial, loss, capture, fission			
13.1.3 Review.			
13.1.3.1 Log Requirements			
13.1.3.2 Recent entries			
13.2 Forms			
13.2.1 Complete.			
13.2.1.1 Irradiations	A		
13.2.1.2 Proposed experiment	A	A	
13.2.1.3 Approved experiment	A	A	

Task List - Reactor Operator	F	S	Comments
13.0 RECORDS MANAGEMENT (cont.)			
13.2 Forms (cont.)			
13.2.1 Complete (cont.).			
13.2.1.4 Requalification	*		
13.2.1.5 Startup checks	*		
13.2.1.6 Shutdown checks	*		
13.2.1.7 Quarterly reports			
13.2.1.8 Annual reports			
13.3 Data			
13.3.1 Tabulate.			
13.3.1.1 Reactor	*		
Reactivity parameters Core configurations Control rod worths Thermal power			
13.3.1.2 Radioactive	*		
Waste disposal Effluent releases Material transfers			
13.3.2 Calculate.			
13.3.2.1 Reactor parameters	*		
13.3.2.2 Radioactivity concentrations	*		
13.3.2.3 Radiation exposure rates	*		
13.4 Files			
13.4.1 Maintain.			
13.4.1.1 Routine facility records			
13.4.1.2 Schematics and drawings			
13.4.1.3 Material inventories			
13.4.1.4 Instruction manuals			

Task List - Reactor Operator	1	2	Comments
13.0 Records Management (cont.)			
13.4 Files (cont.)			
13.4.1 Maintain (cont.).			
13.4.1.5 Reference documents			
13.4.1.6 Product data sheets			
13.4.1.7 Correspondence			
13.4.2 Control.			
13.4.2.1 Access to files			
13.4.2.2 Return of files			

TASK LIST

Section B
Reactor Operator

The University of Texas TRIGA

Task List - Reactor Operator	P	S	Comments
1.0 GENERAL			
1.1 Operator			
1.1.1 Maintain:			
1.1.1.1 Clean laboratory conditions			
1.1.1.2 Equipment design function	*		
1.1.1.3 Operation specifications	*		
1.1.2 Assist:			
1.1.2.1 Personnel involved in facility programs			
1.1.2.2 Performance of scheduled experiments			
1.1.2.3 Tours of facility			
1.1.3 Attend:			
1.1.3.1 Facility meetings			
1.1.3.2 Training sessions	*		
1.2 Trainee			
1.2.1 Learn:			
1.2.1.1 Basic theory and principles			
1.2.1.2 Facility characteristics and operation			
1.2.1.3 Normal and abnormal operation procedures			
1.2.2 Perform:			
1.2.2.1 Training tasks	*		
1.2.3 Develop:			
1.2.3.1 Operator skills	*		
1.2.4 Demonstrate:			
1.2.4.1 Knowledge of principles and operation	*		
1.2.4.2 Proficient operation skills	*		

Task List - Reactor Operator	P	S	Comments
1.0 GENERAL (cont.)			
1.2 Trainee (cont.)			
1.2.4 Demonstrate (cont.)			
1.2.4.3 Responsible attitude	*		
2.0 REACTOR CORE AND POOL STRUCTURE			
2.1 Reactor			
2.1.1 Inspect:			
2.1.1.1 Core	*		
2.1.1.2 Reflector	*		
2.1.1.3 Source assembly	*		
2.1.1.4 Detector canisters	*		
2.1.2 Perform:			
2.1.2.1 Fuel movements	*	*	
2.1.2.2 Source relocation	*	*	
2.2 Pool			
2.2.1 Check:			
2.2.1.1 Bridge	*		
2.2.1.2 Rod drives	*		
2.2.1.3 Pool covers			
2.2.1.4 Fuel storage	*		
2.2.1.5 Gamma irradiator	*		
2.2.2 Identify:			
2.2.2.1 Non standard conditions			
2.2.2.2 Unnecessary loose objects			

Task List - Reactor Operator	P	S	Comments
2.0 REACTOR CORE AND POOL STRUCTURE (cont.)			
2.3 Penetrations			
2.3.1 Examine:			
2.3.1.1 Beam tubes	*		
2.3.1.2 Irradiation tubes	*		
2.3.1.3 Experiment facilities	*		
2.3.2 Secure:			
2.3.2.1 Access points			
3.0 WATER PURIFICATION AND COOLING SYSTEMS			
3.1 Purification			
3.1.1 Operate:			
3.1.1.1 Startup and shutdown			
3.1.1.2 Purification pump			
3.1.1.3 Isolation valves			
3.1.2 Adjust:			
3.1.2.1 Flow rate			
3.1.3 Align:			
3.1.3.1 Valves			
3.1.4 Monitor:			
3.1.4.1 Flow rate			
3.1.4.2 Temperature			
3.1.4.3 Pressure drops			
3.1.4.4 Conductivity, pH	*		
3.1.4.5 Demineralizer radioactivity	*		
3.1.5 Log:			
3.1.5.1 Flow rate			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.1 Purification (cont.)			
3.1.5 Log (cont.)			
3.1.5.2 Temperature			
3.1.5.4 Demineralizer radioactivity			
3.1.6 Check:			
3.1.6.1 Inlet			
3.1.6.2 Outlet			
3.1.6.3 Skimmer			
3.1.6.4 Fittings			
3.1.6.5 Valve alignments			
3.1.6.6 Pump shaft and seals	*		
3.1.6.7 Motor bearings and coupling	*		
3.1.7 Replace:			
3.1.7.1 Resins			
3.1.7.2 Filters			
3.1.7.3 Gaskets			
3.1.8 Order:			
3.1.8.1 Resins			
3.1.8.2 Filters			
3.1.8.3 Gaskets			
3.2 Primary Coolant			
3.2.1 Operate:			
3.2.1.1 Startup and shutdown			
3.2.1.2 Primary pump			
3.2.1.3 Block valves			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.2 Primary coolant (cont.)			
3.2.2 Adjust:			
3.2.2.1 Flow rate		*	
3.2.3 Align:			
3.2.3.1 Valves			
3.2.4 Monitor:			
3.2.4.1 Flow rate			
3.2.4.2 Pressure at inlet and outlet			
3.2.4.3 Temperature at inlet and outlet			
3.2.5 Log:			
3.2.5.1 Flow rate			
3.2.5.2 Pressure at inlet and outlet			
3.2.5.3 Temperature at inlet and outlet			
3.2.6 Check:			
3.2.6.1 Inlet			
3.2.6.2 Outlet			
3.2.6.3 Diffuser			
3.2.6.4 Fittings			
3.2.6.5 Valve alignments			
3.2.6.6 Pump shaft and seals	*		
3.2.6.7 Motor bearings and coupling	*		
3.3 Secondary Coolant			
3.3.1 Operate:			
3.3.1.1 Startup and shutdown			
3.3.1.2 Secondary pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.2 Primary coolant (cont.)			
3.2.2 Adjust:			
3.2.2.1 Flow rate		*	
3.2.3 Align:			
3.2.3.1 Valves			
3.2.4 Monitor:			
3.2.4.1 Flow rate			
3.2.4.2 Pressure at inlet and outlet			
3.2.4.3 Temperature at inlet and outlet			
3.2.5 Log:			
3.2.5.1 Flow rate			
3.2.5.2 Pressure at inlet and outlet			
3.2.5.3 Temperature at inlet and outlet			
3.2.6 Check:			
3.2.6.1 Inlet			
3.2.6.2 Outlet			
3.2.6.3 Diffuser			
3.2.6.4 Fittings			
3.2.6.5 Valve alignments			
3.2.6.6 Pump shaft and seals	*		
3.2.6.7 Motor bearings and coupling	*		
3.3 Secondary Coolant			
3.3.1 Operate:			
3.3.1.1 Startup and shutdown			
3.3.1.2 Secondary pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.3 Secondary Coolant (cont.)			
3.3.1 Operate (cont.):			
3.3.1.3 Block valves			
3.3.2 Adjust:			
3.3.2.1 Flow rate		*	
3.3.3 Align:			
3.3.3.1 Valves			
3.3.4 Monitor:			
3.3.4.1 Flow rate			
3.3.4.2 Pressure at inlet and outlet			
3.3.4.3 Temperature at inlet and outlet			
3.3.5 Log:			
3.3.5.1 Flow rate			
3.3.5.2 Pressure at inlet and outlet			
3.3.5.3 Temperature at inlet and outlet			
3.3.6 Check:			
3.3.6.1 Chilled water supply			
3.3.6.2 Chilled water return			
3.3.6.3 Fittings			
3.3.6.4 Valve alignments			
3.3.6.5 Pump shaft and seals	*		
3.3.6.6 Motor bearings and coupling	*		
3.4 Pool Water			
3.4.1 Inspect:			
3.4.1.1 Appearance			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.4 Pool Water (cont.)			
3.4.1 Inspect (cont.):			
3.4.1.2 Penetrations	*		
3.4.2 Monitor:			
3.4.2.1 Pool level	*		
3.4.2.2 Temperature	*		
3.4.2.3 Conductivity	*		
3.4.2.4 Radioactivity	*		
3.4.3 Operate:			
3.4.3.1 Makeup			
3.4.3.2 Cleanup			
3.4.3.3 Skimmer			
3.4.3.4 Diffuser			
3.4.4 Collect:			
3.4.4.1 Water samples			
3.4.5 Measure:			
3.4.5.1 Power heat rate	*		
3.4.6 Log:			
3.4.6.1 Pool level changes for fill			
3.4.6.2 Pool level changes for drain			
3.4.6.3 Water activity			
3.4.6.4 Power heat rate			
3.5 Other			
3.5.1 Monitor:			
3.5.1.1 Foundation pump			

Task List - Reactor Operator	P	S	Comments
3.0 WATER PURIFICATION AND COOLING SYSTEMS (cont.)			
3.5 Other (cont.)			
3.5.2 Operate:			
3.5.1 Sump pump		*	
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS			
4.1 General Operation			
4.1.1 Actuate:			
4.1.1.1 Control console power			
4.1.1.2 Power to other systems			
4.1.2 Confirm:			
4.1.2.1 Proper operation requirements		*	
4.1.2.2 Status of operation equipment	*	*	
4.1.2.3 Status of experiment components	*	*	
4.1.2.4 Ventilation exhaust conditions		*	
4.1.3 Check:			
4.1.3.1 Safety equipment			
4.1.3.2 Security equipment			
4.1.3.3 Emergency equipment			
4.1.3.4 Communication capability			
4.1.3.5 Non control console systems			
4.2 Control Console Operation Mode			
4.2.1 Switch:			
4.2.1.1 Console mode			
4.2.1.2 Magnet power			
4.2.1.3 Demand power			

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.2 Control Console Operation Mode			
4.2.2 Acknowledge:			
4.2.2.1 Status condition changes, modes	*		
4.2.2.2 Response to console display menus	*		
4.2.2.3 Valid system response to calibration	*		
4.2.3 Verify:			
4.2.3.1 Annunciation of console status panel			
4.2.3.2 Function of console display			
4.2.3.3 Function of entry keyboard			
4.2.4 Monitor:			
4.2.4.1 Graphic display information			
4.2.4.2 Direct parameter indicators			
4.2.4.3 Control rod operation panel			
4.2.5 Implement:			
4.2.5.1 DOS or scram operational			
4.2.5.2 Log process for valid operator			
4.2.5.3 Programmed calibration and test	*		
4.2.5.4 Operational mode; non pulse, automatic			
4.2.5.5 Operational mode; pulse, square wave			
4.2.5.6 Special function key processes			
4.2.5.7 System diagnostic routines			
4.2.6 Observe:			
4.2.6.1 Output trends			
4.2.7 Measure:			
4.2.7.1 Values at circuit test points	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.2. Control Console Operation Mode (cont.)			
4.2.8 Adjust:			
4.2.8.1 Values at circuit test points	*	*	
4.2.9 Determine:			
4.2.9.1 Circuit test error conditions	*	*	
4.2.10 Replace:			
4.2.10 Specified component failures	*	*	
4.2.11 Operate:			
4.2.11.1 Console printer for log records			
4.2.12 Log:			
4.2.12.1 System generated log inforamtion			
4.3 Rod Operation Interlock Logic			
4.3.1 Control:	*		
4.3.1.1 Rod up motion	*		
4.3.1.2 Rod down motion	*		
4.3.1.3 Rod drop insertion	*		
4.3.1.4 Magnet current power	*		
4.3.1.5 Transient fire switch	*		
4.3.2 Check:			
4.3.2.1 Control switch actuations	*		
4.3.2.2 Count rate interlock	*		
4.3.2.3 Pulse interlock	*		
4.3.2.4 Scram function	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.3 Rod Operation Interlock Logis (cont.)			
4.3.3 Verify:			
4.3.3.1 Supplemental interlocks	*		
4.3.3.2 Non pulse interlocks	*		
4.3.3.3 Pulse interlocks	*		
4.3.3.4 Rod rundown			
4.4 Scram Safety System Logic			
4.4.1 Specify:			
4.4.1.1 Scram set point changes	*	*	
4.4.1.2 Log entry for scram		*	
4.4.2 Check:			
4.4.2.1 Manual scram switch	*		
4.4.2.2 CSC processor failure	*		
4.4.2.3 DAC processor failure	*		
4.4.2.4 CSC/DAC communication loss	*		
4.4.2.5 Fuel temperature (2)	*		
4.4.2.6 Percent power (2)	*		
4.4.2.7 High Voltage (2)	*		
4.4.2.8 Amplifier/logic power (2)	*		
4.4.2.9 Power rate of change (2)	*		
4.4.2.10 Low source count (2)	*		
4.4.2.11 Pool level low	*		
4.4.2.12 External scram	*		
4.4.3 Determine:			
4.4.3.1 Scram condition	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.4 Scram Safety System Logic (cont.)			
4.4.3 Determine (cont.):			
4.4.3.2 Reason for condition	*	*	
4.4.3.3 Restart requirements	*	*	
4.4.3.4 Log entry for scram			
4.5 Fuel Temperature Channels			
4.5.1 Setup:			
4.5.1.1 Instrument element position	*	*	
4.5.1.2 Thermocouple source	*	*	
4.5.2 Check:			
4.5.2.1 Operability	*		
4.5.2.2 Calibration	*		
4.5.2.3 Amplifier/logic	*		
4.5.2.4 Set points (scram)	*		
4.5.3 Log:			
4.5.3.1 Non pulse temperature values			
4.5.3.2 Pulse temperature values			
4.6 Percent Power Channels			
4.6.1 Setup:			
4.6.1.1 Detector chamber position	*	*	
4.6.1.2 Non pulse operation values	*	*	
4.6.2 Check:			
4.6.2.1 Operability	*		
4.6.2.2 Calibration	*		
4.6.2.3 Detector HV	*		

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.6 Percent Power Channels (cont.)			
4.6.2 Check (cont.):			
4.6.2.4 Amplifier/logic	*		
4.6.2.5 Set points (scram)	*		
4.6.2.6 Demand power value	*		
4.6.3 Log:			
4.6.3.1 At power operating conditions			
4.6.3.2 Safety channel status conditions			
4.7 Pulse Power Channel			
4.7.1 Setup:			
4.7.1.1 Detector chamber position	*	*	
4.7.1.2 Pulse transient conditions	*	*	
4.7.2 Check:			
4.7.2.1 Operability	*		
4.7.2.2 Calibration	*		
4.7.2.3 Detector HV	*		
4.7.2.4 Amplifier/logic	*		
4.7.2.5 Set points (scram)	*		
4.7.2.6 Transient rod position	*		
4.7.3 Monitor:			
4.7.3.1 Pulse peak and energy data			
4.7.4 Log:			
4.7.4.1 Pulse event data			

Task List - Reactor Operator	P	S	Comments
4.0 REACTOR CONTROL AND INSTRUMENTATION SYSTEMS (cont.)			
4.8 Pool Water System Channels			
4.8.1 Check:			
4.8.1.1 Set points			
4.8.1.2 Level sensors			
4.8.1.3 Temperature sensors	*		
4.8.1.4 Conductivity sensors	*		
4.8.1.5 Flow rate measurement	*		
4.8.2 Monitor:			
4.8.2.1 Pool level status	*		
4.8.2.2 Pool temperature			
4.8.2.3 Flow rate			
4.9 Radiation Monitor Channels			
4.9.1 Check:			
4.9.1.1 High set points	*		
4.9.1.2 Low set points			
4.9.1.3 Area monitor	*		
4.9.1.4 Gaseous monitor	*		
4.9.1.5 Particulate monitor	*		
4.9.2 Monitor:			
4.9.2.1 Counts per minute			
4.9.2.2 Millirem per hour			

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS			
5.1 General			
5.1.1 Verify:			
5.1.1.1 Correct experiment approval	*		
5.1.1.2 Classification of experiment	*		
5.1.2 Review:			
5.1.2.1 Previous related experiment	*	*	
5.1.2.2 Projected experiment results	*	*	
5.1.3 Estimate:			
5.1.3.1 Reactivity effect to the core of experiment structure	*	*	
5.1.3.2 Reactivity effect to the core of removeable materials	*	*	
5.1.3.3 Radiation levels of experiment structure	*	*	
5.1.3.4 Radiation levels of removeable materials	*	*	
5.1.4 Monitor:			
5.1.4.1 Experiment or irradiation conditions	*		
5.1.5 Encapsulate			
5.1.5.1 Materials for irradiation	*		
5.1.6 Insert:			
5.1.6.1 Samples for irradiation	*		
5.1.7 Remove:			
5.1.7.1 Samples after irradiation	*		
5.1.8 Log:			
5.1.8.1 Experiment equipment			
5.1.8.2 Irradiation samples			

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS (cont.)			
5.2 Beam tubes			
5.2.1 Manipulate:			
5.2.1.1 Beam tube components	*		
5.2.1.2 Beam tube experiments	*		
5.2.1.3 Radiation shield materials	*		
5.2.2 Monitor:			
5.2.2.1 Beam tube status			
5.2.2.2 Area radiation levels			
5.2.2.3 Experiment progress			
5.2.2.4 Area personnel access			
5.3 Rotary Specimen Rack			
5.3.1 Operate:			
5.3.1.1 Rack drive	*		
5.3.1.2 Sample removal device	*		
5.3.1.3 Sample insertion device	*		
5.3.2 Monitor:			
5.3.2.1 Contamination of access tube	*		
5.4 Pneumatic Transfer Tube			
5.4.1 Operate:			
5.4.1.1 Transfer terminal	*		
5.4.1.2 Permit switch	*		
5.4.1.3 Blower supply	*		
5.4.1.4 Propellant gas supply	*		

Task List - Reactor Operator	P	S	Comments
5.0 PROPOSED OR APPROVED EXPERIMENTS (cont.)			
5.4 Pneumatic Transfer Tube (cont.)			
5.4.2 Monitor:			
5.4.2.1 Access port radiation levels			
5.4.3 Instruct:			
5.4.3.1 Experimenter			
5.5 Tri-cut Core Position To be determined	*	*	
5.6 Hex-cut Core Position To be determined	*	*	
5.7 In-Core Irradiation To be determined	*	*	
5.8 Ex-core Irradiation To be determined	*	*	
5.9 Material Processing and Transfer			
5.9.1 Prepare:			
5.9.1.1 Material encapsulation	*		
5.9.1.2 Transfer documents			
5.9.1.3 Shipping papers			
5.9.2 Perform:			
5.9.2.1 Package labeling			
5.9.2.2 Package contamination smears			
5.9.2.3 Package radiation surveys			
5.9.3 Monitor:			
5.9.3.1 Transfer to authorized agent			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES			
6.1 HVAC Equipment			
6.1.1 Test:			
6.1.1.1 Purge air system fan	*		
6.1.1.2 Isolation damper function	*		
6.1.1.3 Supply, return, exhaust fans			
6.1.1.4 Filter differential pressure	*		
6.1.1.5 Air sample ports			
6.1.2 Start/stop:			
6.1.2.1 Reactor room HVAC fans			
6.1.2.2 Reactor room purge fan	*		
6.1.3 Monitor:			
6.1.3.1 Air radioactivity of purge system			
6.1.3.2 Room air activity confinement signal			
6.1.3.3 Differential pressure to reactor room			
6.1.4 Maintain:			
6.1.4.1 Reactor room pressure negative			
6.1.4.2 Reactor room confinement control			
6.1.5 Service:			
6.1.5.1 HEPA filter and prefilter	*		
6.2 Deionized Water			
6.2.1 Check:			
6.2.1.1 Water supply status			
6.2.1.2 Filter status			
6.2.1.3 Resin status			
6.2.1.4 Conductivity			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES (cont.)			
6.2 Deionized Water (cont.)			
6.2.2 Service:			
6.2.2.1 Filter, resin	*		
6.3 Compressed Air			
6.3.1 Check:			
6.3.1.1 Compressor status			
6.3.1.2 Filter status			
6.3.1.3 Dryer status			
6.3.2 Service:			
6.3.2.1 Filter, dryer	*		
6.4 Electrical			
6.4.1 Operate:			
6.4.1.1 Reactor area power distributions			
6.4.2 Locate:			
6.4.2.1 Electrical power circuit source			
6.5 Miscellaneous			
6.5.1 Operate:			
6.5.1.1 Compressed gas cylinders	*		
6.5.1.2 Vacuum pumps and valves	*		
6.5.1.3 Fire control equipment	*		
6.5.1.4 Overhead bridge crane	*		
6.5.1.5 Communication system	*		
6.5.2 Replace:			
6.5.2.1 Gas cylinders			
6.5.2.2 Liquid nitrogen			

Task List - Reactor Operator	P	S	Comments
6.0 FACILITY UTILITIES (cont.)			
6.5 Miscellaneous (cont.)			
6.5.2 Replace (cont.):			
6.5.2.3 First aid supplies			
6.5.3 Perform:			
6.5.3.1 Minor repairs			
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL			
7.1 TRIGA Fuel			
7.1.1 Operate:			
7.1.1.1 Fuel handling tools	*		
7.1.1.2 Fuel inspection device	*		
7.1.1.3 Fuel transfer cask	*		
7.1.2 Move:			
7.1.2.1 Fuel into reactor core	*	*	
7.1.2.2 Fuel out of reactor core	*	*	
7.1.2.3 Fuel followed control rods	*	*	
7.1.2.4 To or from storage racks	*		
7.1.2.5 To or from storage wells	*		
7.1.2.6 To or from transfer casks	*		
7.1.3 Clean:			
7.1.3.1 Elements			
7.1.4 Locate:			
7.1.4.1 Failed elements	*	*	
7.1.5 Connect/disconnect:			
7.1.5.1 Instrumented elements	*	*	

Task List - Reactor Operator	P	S	Comments
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL (cont.)			
7.1 TRIGA Fuel (cont.)			
7.1.6 Install/remove:			
7.1.6.1 Fuel follow control elements	*	*	
7.1.7 Measure:			
7.1.7.1 Element length	*		
7.1.7.2 Element bow	*		
7.1.7.3 Go gauge	*		
7.1.8 Document:			
7.1.8.1 Fuel movements		*	
7.1.8.2 Operation history		*	
7.1.8.3 Inspection results		*	
7.2 Subcritical Assembly			
7.2.1 Monitor:			
7.2.1.1 Fuel pellets used	*		
7.2.1.2 Use of core cylinder	*		
7.2.1.3 Use of reflector components			
7.2.1.4 Insertion of neutron sources	*	*	
7.2.2 Maintain:			
7.2.2.1 SNM material access control			
7.2.2.2 SNM material storage			
7.3 Isotopic Sealed Sources			
7.3.1 Monitor:			
7.3.1.1 Use of sealed sources	*	*	
7.3.1.2 Applications with sealed sources			

Task List - Reactor Operator	P	S	Comments
7.0 SPECIAL NUCLEAR AND BY-PRODUCT MATERIAL (cont.)			
7.3 Isotopic Sealed Sources (cont.)			
7.3.2 Maintain:			
7.3.2.1 Leak tests of sealed sources		*	
7.3.2.2 Inventory of sealed sources		*	
7.4 Standard or Reference Materials			
7.4.1 Monitor:			
7.4.1.1 Use of SNM materials	*		
7.4.1.2 Applications with SNM materials			
7.4.2 Maintain:			
7.4.2.1 Inventory of detectors with SNM		*	
7.4.2.2 Inventory of reference materials		*	
8.0 RADIOACTIVE EFFLUENT AND WASTE			
8.1 Gaseous			
8.1.1 Test:			
8.1.1.1 Confinement system (dampers, seals)	*		
8.1.1.2 Stack air monitor (gas, particulate)	*		
8.1.2 Check:			
8.1.2.1 Exhaust stack filters			
8.1.2.2 Purge air filters	*		
8.1.2.3 Hood fan filters	*		
8.1.2.4 Sample filters	*		
8.1.3 Operate:			
8.1.3.1 Reactor room ventilation system	*		
8.1.3.2 Purge air exhaust system	*		
8.1.3.3 Hood exhaust air systems	*		

Task List - Reactor Operator	P	S	Comments
8.0 RADIOACTIVE EFFLUENT AND WASTE (cont.)			
8.1 Gaseous (cont.)			
8.1.3 Operate (cont.):			
8.1.3.4 Air sample monitors	*		
8.1.4 Monitor:			
8.1.4.1 Reactor room negative pressure			
8.1.4.2 Continuous air monitors			
8.1.5 Log:			
8.1.5.1 Argon-41 release			
8.2 Liquid			
8.2.1 Check:			
8.2.1.1 Liquid tank levels	*		
8.2.2 Measure:			
8.2.2.1 Beta-gamma activity	*		
8.2.3 Release:			
8.2.3.1 Controlled quantity	*	*	
8.2.4 Log:			
8.2.4.1 Isotope			
8.2.4.2 Concentration			
8.2.4.3 Volume released			
8.3 Solid			
8.3.1 Specify:			
8.3.1.1 Isotopes present			
8.3.1.2 Physical or chemical form			
8.3.2 Dispose:			
8.3.2.1 Specific materials	*	*	

Task List - Reactor Operator	P	S	Comments
8.0 RADIOACTIVE EFFLUENT AND WASTE (cont.)			
8.3 Solid (cont.)			
8.3.2 Dispose (cont.):			
8.3.2.2 Irradiated samples	*	*	
8.3.2.3 General waste	*	*	
8.3.3 Log:			
8.3.3.1 Isotope and total activity			
8.3.3.2 Material forms			
8.3.3.3 Waste volume			
9.0 SURVEILLANCE AND MAINTENANCE			
9.1 Periodic Surveillance Schedule			
9.1.1 Verify:			
9.1.1.1 Limiting safety system settings	*	*	
9.1.1.2 Limiting conditions for operation	*	*	
9.1.2 Inspect:			
9.1.2.1 Fuel elements	*	*	
9.1.2.2 Control rods	*	*	
9.1.3 Calibrate:			
9.1.3.1 Control rods	*		
9.1.3.2 Reactor power	*		
9.1.3.3 Control console	*		
9.1.3.4 Radiation monitors	*		
9.1.4 Inventory:			
9.1.4.1 Special nuclear material			
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Volume II

UT-TRIGA Training Manual
Facility Descriptions
Instrumentation Control and Safety
Radiological Safety, Reactor Physics
UT-TRIGA Technical Data

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DESCRIPTION OF TRIGA MARK II REACTOR

1. GENERAL DESCRIPTION

The TRIGA Mark II reactor was designed to effectively implement the various fields of basic nuclear research and education. It incorporates facilities for advanced neutron and gamma radiation studies as well as for isotope production, sample activation, and student training. The TRIGA Mark II reactor is installed entirely above ground (see Fig. 1-1). Its built-in safety permits installation at a minimum of expense in either an existing or a new building.

The reactor and experimental facilities are surrounded by a concrete shield structure (see Fig. 1-2). The reactor core and reflector assembly is located at the bottom of a 6.5x9.75ft.(2x3m) dia. tank 27.2-ft(8.3m) deep. Approximately 22.6ft(6.9m) of water above the core provides vertical shielding. The core is shielded radially by a minimum of 8 ft, (2.4m) concrete, 1.5 ft (45.7 cm) of water, and 10.2 in. (25.9 cm) of graphite reflector.

TRIGA reactors utilize solid fuel elements, developed by General Atomic, in which the zirconium-hydride moderator is homogeneously combined with the enriched uranium. The unique feature of these fuel-moderator elements is the prompt negative temperature coefficient of reactivity, which gives the TRIGA reactor its built-in safety by automatically limiting the reactor power to a safe level in the event of a power excursion. The reactor core consists of a lattice of cylindrical fuel-moderator elements and graphite (dummy) elements. The fuel elements have 3.5-in. (8.9-cm)-long graphite end sections that form the top and bottom reflector. A 10-in. (25.4-cm)-thick graphite radial reflector surrounds the core and is supported on an aluminum stand at the bottom of the tank. Water occupies about 1/3 of the core volume.

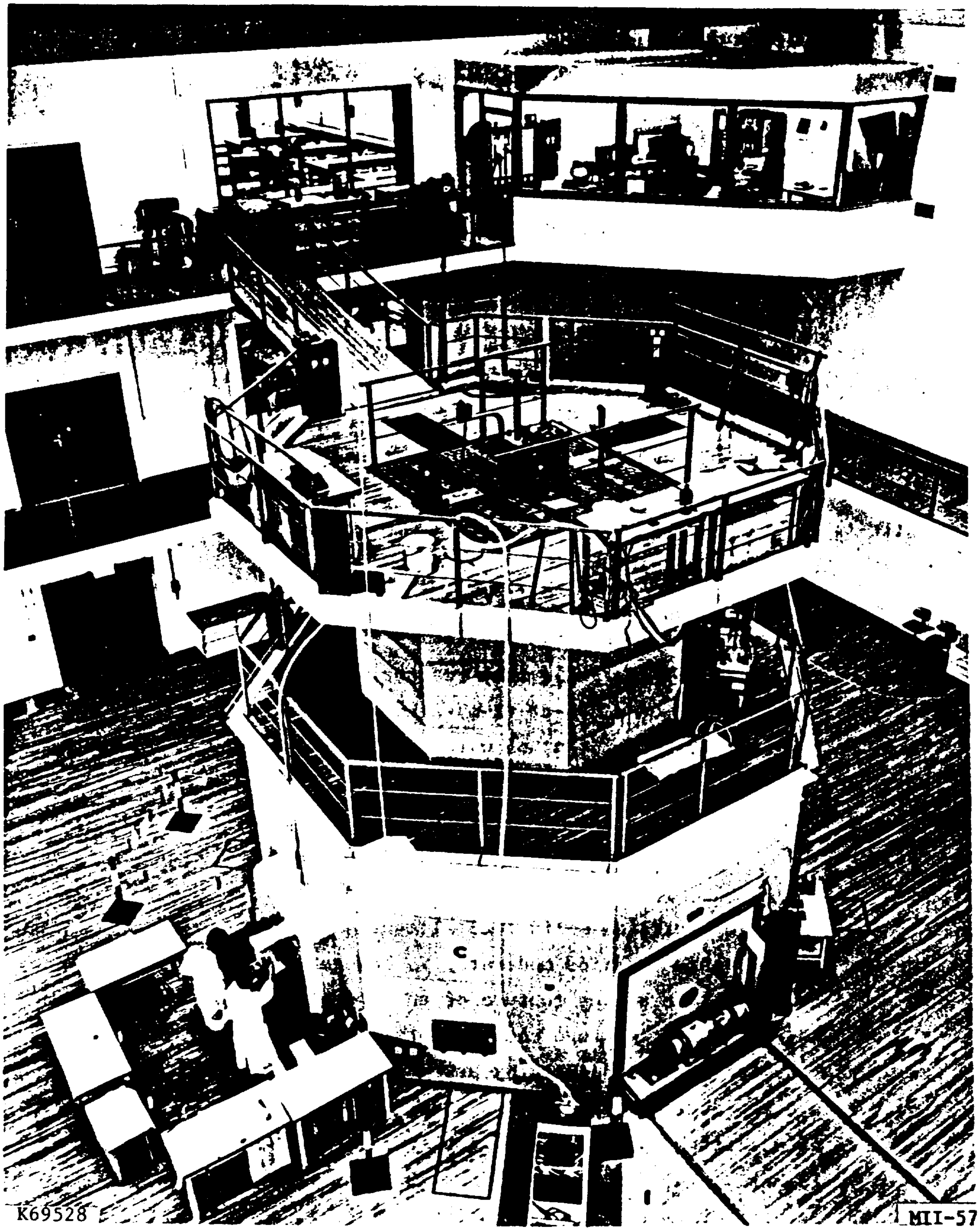


Fig. 2-1. Typical TRIGA Mark II reactor facility

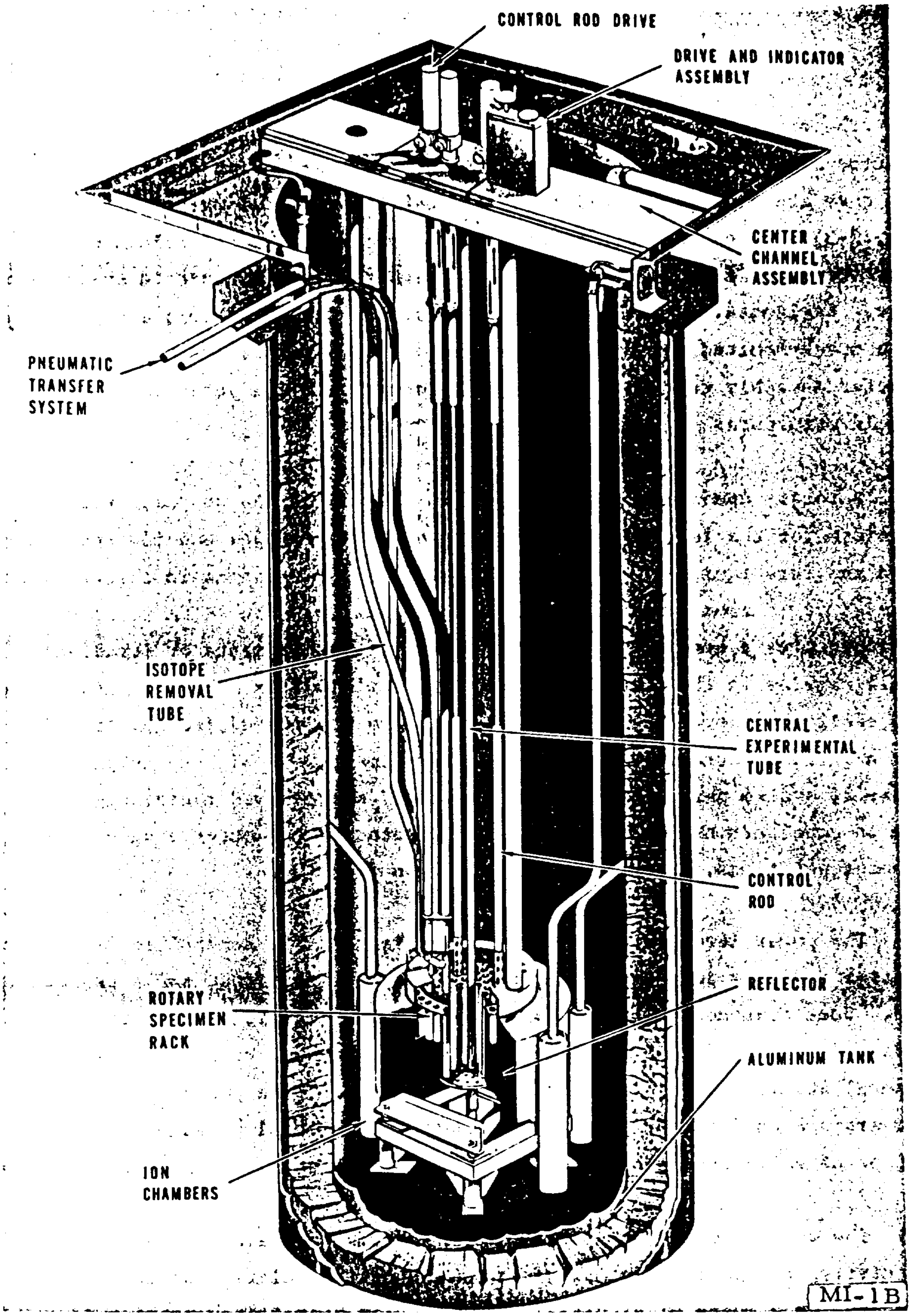


Fig. 2-2. Cutaway view of TRIGA reactor (beam ports and thermal column not shown)

The experimental and irradiation facilities of the TRIGA Mark II reactor are extensive and versatile. Physical access and observation of the core are possible at all times through the vertical water shield. A typical side view of a TRIGA MARK II reactor showing a beam tube is shown in Fig. 1-4. Thermal columns are not part of the shield structure.

Five beam ports extend from the reactor assembly through the water and concrete to the outer face of the shield structure.

A rotary specimen rack in a well in the top of the graphite reflector provides for the large-scale production of radioisotopes and for the activation and irradiation of small specimens. All 40 positions in this rack are exposed to neutron fluxes of comparable intensity. The TRIGA reactor is equipped with a central thimble for conducting experiments or irradiating small samples in the core at the point of maximum flux. Experimental tubes can easily be installed in the core region to provide additional facilities for high-level irradiation or in-core experiments. A high-speed pneumatic transfer system permits the use of extremely short-lived radioisotopes. The in-core terminus of this system is located in the outer ring of fuel element positions, a region of high neutron flux.

The power level of the TRIGA reactor is normally controlled with four control rods: a regulating rod, two shim rods and a transient rod.

Insofar as control for the sake of safety is concerned, transient tests at General Atomic have proved conclusively that the large prompt negative temperature coefficient of the fuel-moderator material provides a high degree of self-regulation without the assistance of external control devices.

The water cooling and purification systems maintain low water conductivity, remove impurities, maintain the optical clarity of the water, and provide a means of dissipating the reactor heat. They consist of a

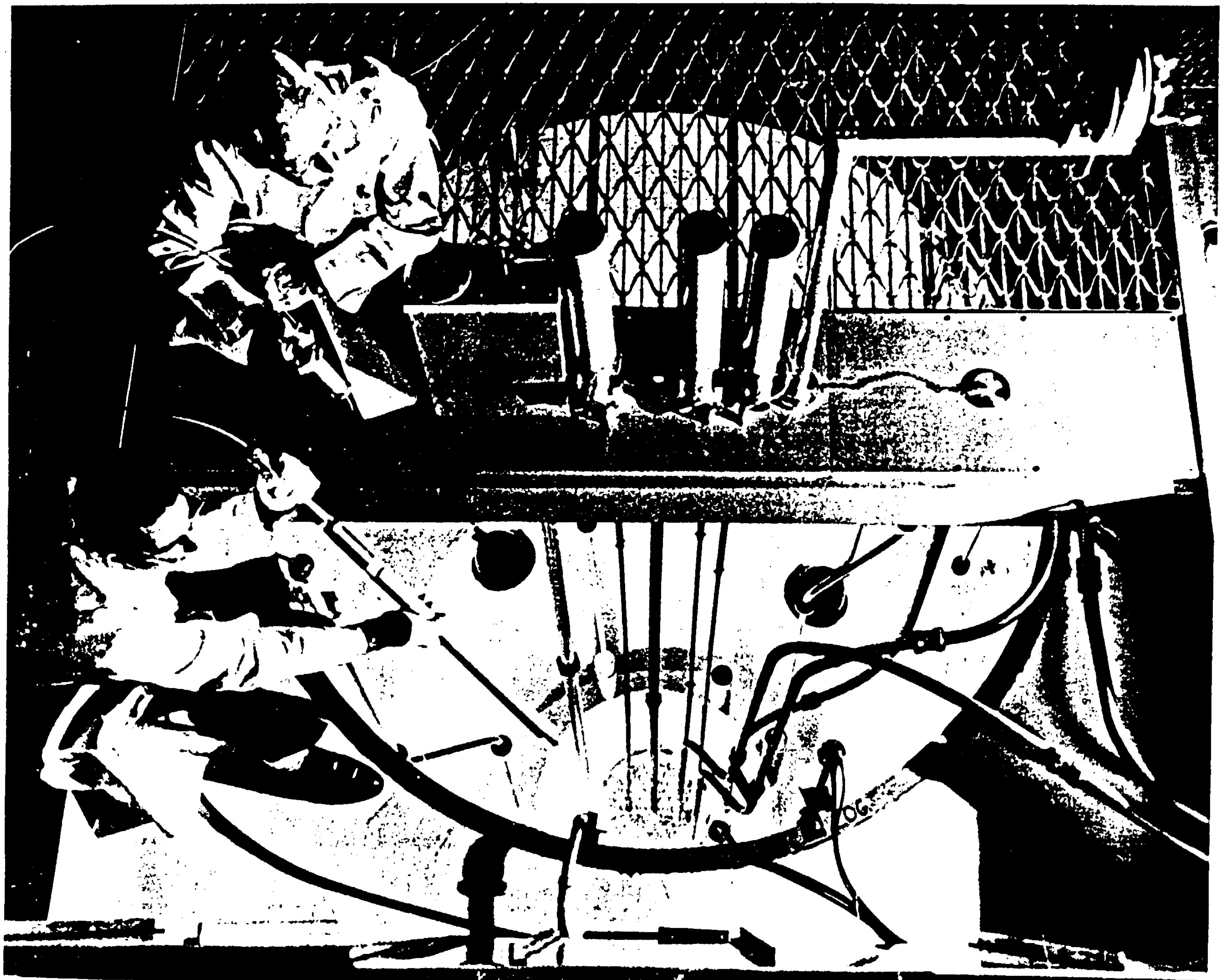


Fig. 1-3 TRIGA core visible th gh 20 ft of cooling and shielding water MI-70

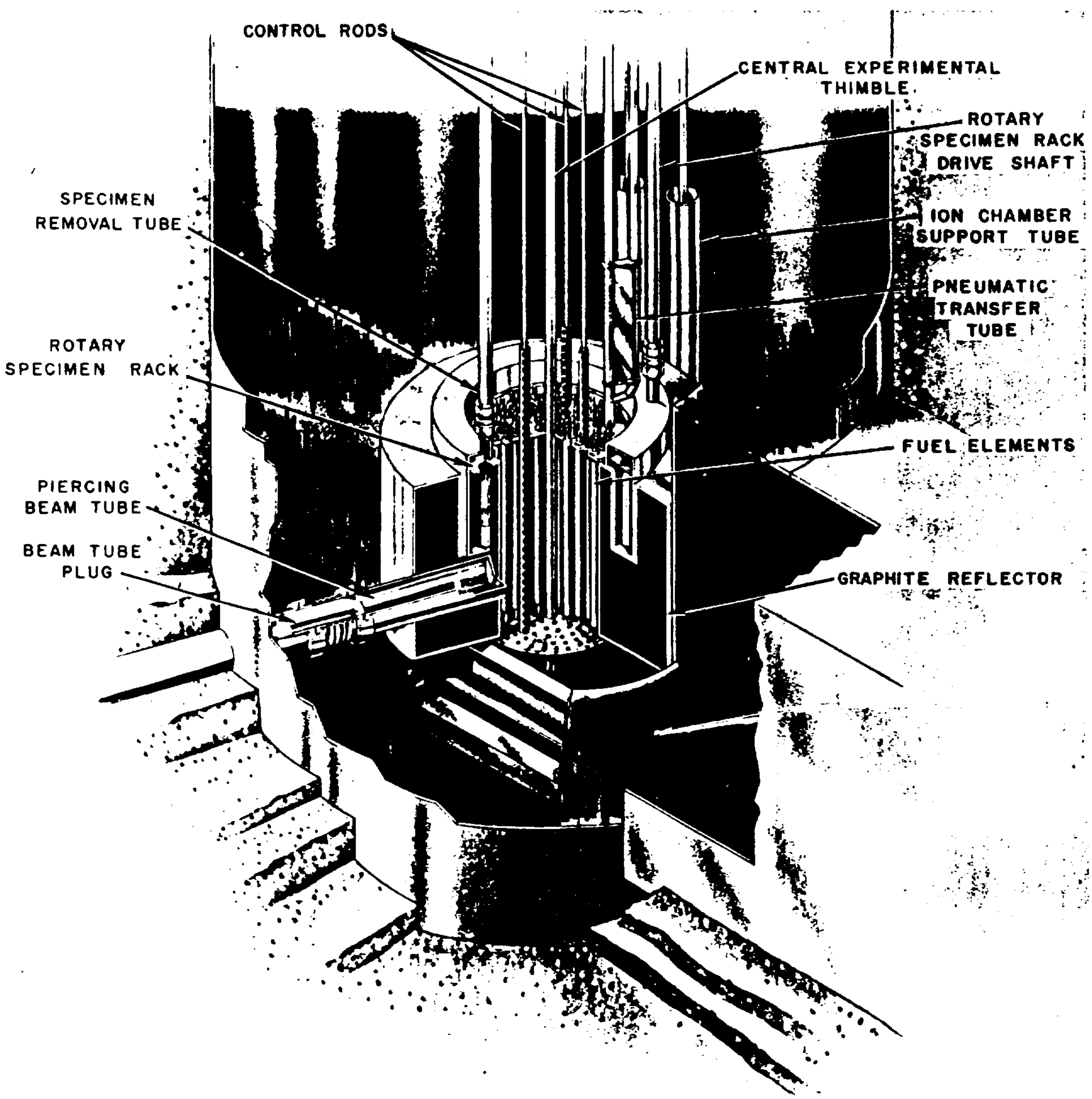


Fig. 1-4 . **Cutaway view showing TRIGA Mark II core arrangement**
(MII-27B)

water surface skimmer, pump, filter, demineralizer, heat exchange unit, associated piping and valving, and miscellaneous instrumentation.

2. REACTOR STRUCTURES

2.1. Reactor Shield Structure

The reactor shield is a reinforced concrete structure standing 28 ft. (8.5m) above the reactor room floor. In elevation, the structure contains a step about mid-height as shown in Fig. 1-1. The lower octagonal portion is 22 ft, 6 in. (7 m) across flats.

Four beam ports are installed in the shield structure. The beam ports are tubular penetrations through the concrete shield and the reactor tank water that terminate at either the reactor reflector assembly or at the edge of the core.

The lower 19.5 ft. (5.9m) of the shield is composed of high density 180 lb/ft^3 (2.9 gm/cc) concrete. The upper portion of the shield and the platform are composed of standard weight concrete 145 lb/ft^3 (2.3 gm/cc).

Small-diameter steel pipes embedded in the concrete are connected to the pool area and each of the beam ports. They terminate in a manifold at the base of the shield which may be connected to a venting system for radioactive argon.

A cantilevered platform surrounded by metal railings is provided for personnel and equipment at the top of the reactor shield. A metal stairway and railing extends from the floor to the cantilevered platform at the top of the shield structure.

2.2. Reactor Tank

The reactor tank consists of an aluminum vessel installed in the reactor shield structure. It has an inside diameter of approximately 6.5 ft (2 m) and a depth of 27.2 ft. (8.3 m) . The tank is waterproofed by continuous welded joints; the integrity of the joints is verified by X-ray testing, pressure testing, dye penetrant checking, and soap-bubble leak testing. An aluminum angle, used for mounting the ion chambers and underwater lights, is attached to the top of the tank. The reactor tank is pierced by five beam ports

2.3. Center Channel Assembly

The center channel assembly provides support for the isotope production facility drive and indicator assembly, the control rod drives, and the tank covers. It is located at the top of the reactor tank directly over the reactor core.

The assembly consists of two 8-in. (20.3 cm) structural steel channels covered with steel plates, 16 in. (40.7 cm) wide and 5/8 in. (1.6 cm) thick. This assembly has the shape of an inverted U and is 8 ft, 2 in. (2.5 m) long. The channel assembly is designed to support a shielded isotope cask weighing 3.5 tons (3175 kg) placed over the specimen removal tube.

2.4. Reactor Tank Covers

The top of the reactor tank is closed by 9 aluminum grating covers that are hinged and installed flush with the floor. Lucite plastic, 1/4-in.-thick, is attached to the bottom of each grating section to prevent foreign matter from entering the tank while still permitting visual observation. A gap between one edge of the plastic and the grating provides adequate ventilation for the small amount of radiolytic gases that may be released during reactor operation. Each cover has two flush lifting handles to facilitate its movement. The center channel assembly provides support for the unhinged end of the covers when they are closed.

3. BASIC REACTOR COMPONENTS

The reactor core and reflector assembly, shown in Fig. 1-5, is a cylinder approximately 43 in. (1.1 m) in diameter and 23 in. (0.6 m) high. Surrounded by the graphite reflector, the core consists principally of a lattice of fuel-moderator elements, graphite dummy elements (optional), and control rods. Submerged in water in a tank approximately 6.5 ft (2 m) in diameter and approximately 27.2 ft. (8.3 m) deep, the reflector assembly rests on a platform, which raises the lower edge of the reflector assembly about 2 ft (0.6 m) above the tank floor. Shielding above the core is provided by approximately 22.6ft.(6.9m)of water. Cooling of the core is by natural circulation of water. Coolant water occupies about 1/3 of the core volume.

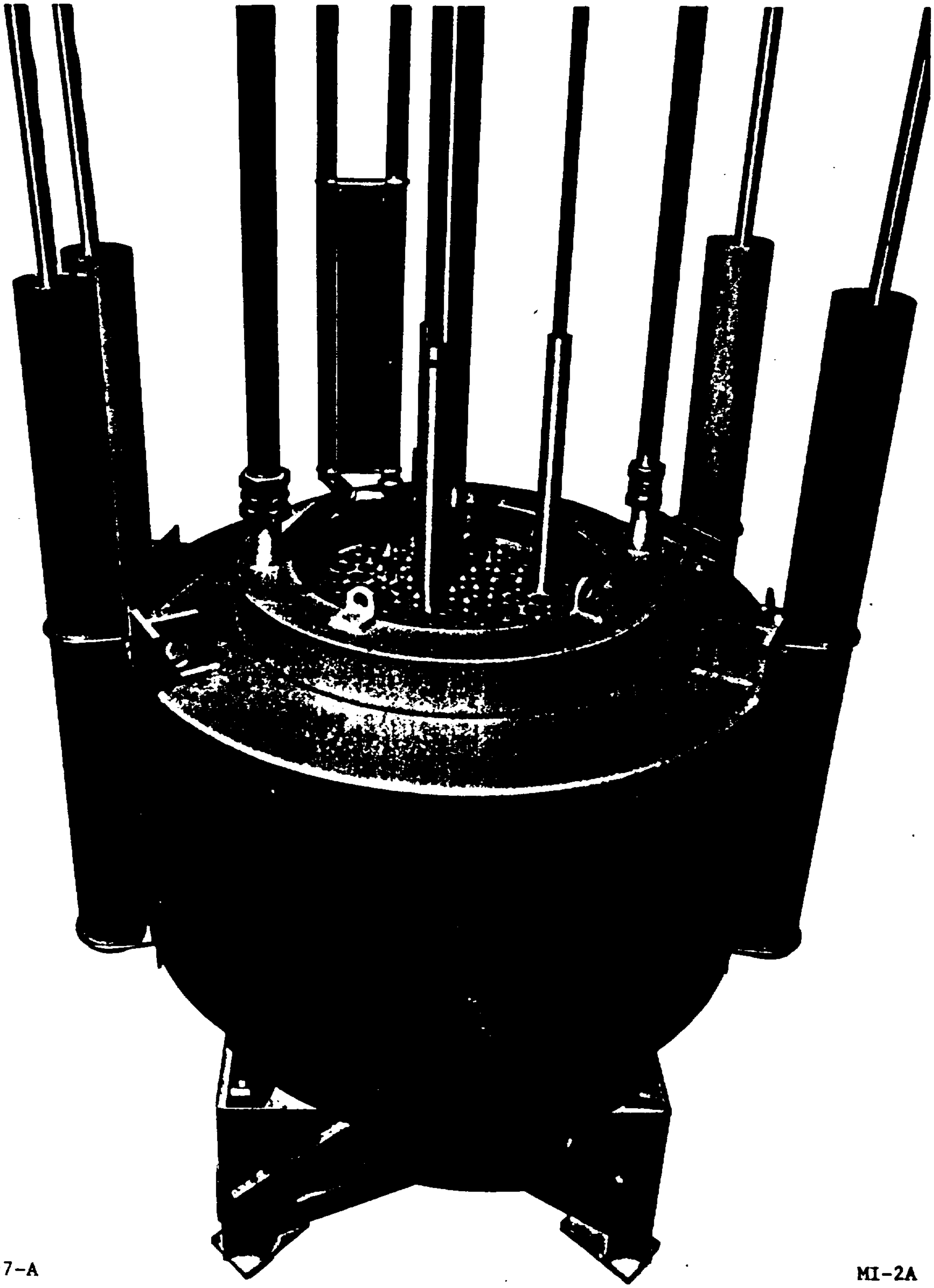
3.1. Reflector Platform

The reflector platform, shown at the bottom of Fig. 1-5, is a square all-welded aluminum-frame structure. It rests on the floor of the tank on four legs and is secured by aluminum anchor bolts welded to the tank bottom. The reflector assembly bolts to the reflector platform.

3.2. Reflector

The reflector is a ring-shaped block of graphite that surrounds the core radially. The graphite is 10.2 in. (25.9 cm) thick radially, with an inside diameter (below the lazy susan) of 21-7/8 in. (55.6 cm) and a height of 21-7/8 in. (55.6 cm) . The graphite is protected from water penetration by a leak-tight welded aluminum can.

A well on the inside diameter in the top of the graphite reflector is provided for the rotary specimen rack. This well is also aluminum-lined, the lining being an integral part of the aluminum reflector can. The rotary specimen rack is a self-contained unit and does not penetrate the sealed reflector at any point.



T-88,907-A MI-2A

Fig. 1-5. Typical reactor core and reflector assembly

The graphite and the outer surface of the aluminum can are pierced by an aluminum tube which forms the inner section of the piercing radial beam port. Two additional holes penetrate the graphite through a radial beam tube.

just outside the reflector can.

Vertical tubes attached to the reflector assembly permit accurate and reproducible positioning of the ion chambers used for monitoring reactor operation.

The reflector assembly rests on an aluminum platform at the bottom of the tank, and provides the support for the two grid plates and the safety plate. Lugs are provided for lifting the assembly.

3.3. Grid Plates and Safety Plate

The top grid plate is an anodized aluminum plate 5/8 in. (1.6 cm) thick [3/8 in. (1 cm) thick in the central region] that provides accurate lateral positioning for the core components. The plate is supported by a ring welded to the top inside surface of the reflector container and is anodized to resist wear and corrosion.

One hundred twenty six (126) holes, 1.505 in. in diameter (3.82 cm), are drilled into the top grid plate in six hexagonal bands around a central hole to locate the fuel-moderator and graphite dummy elements, the control rods and guide tubes, and the pneumatic transfer tube (see Fig. 1-6). A 1.505-in.-diameter (3.82 cm) center hole accommodates the central thimble. Small holes at various positions in the top grid plate permit insertion of foils into the core to obtain flux data.

A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 in. (11.2 cm) in diameter into the region of highest flux; this requires prior relocation of the six

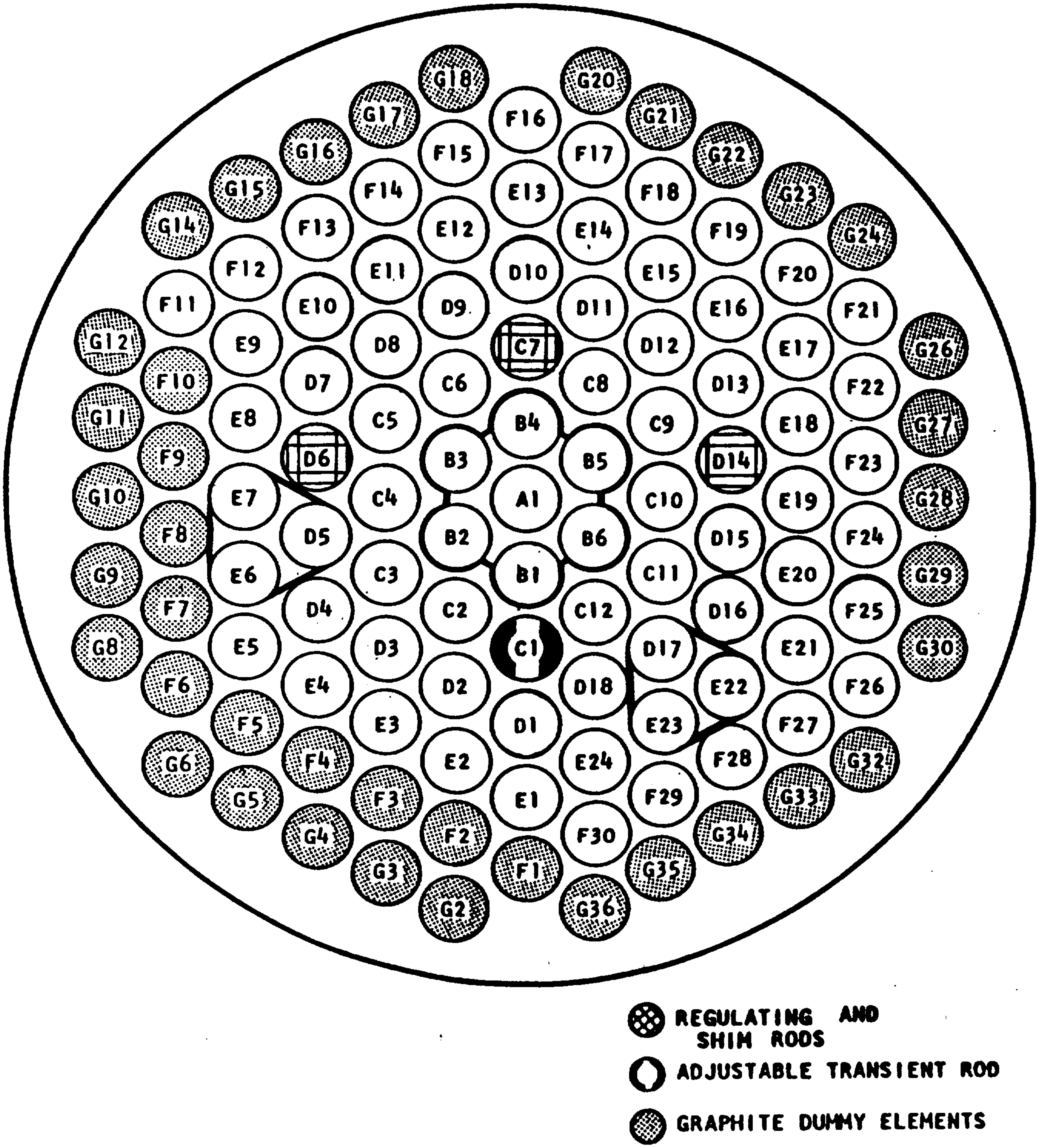


Fig. 1-6. Typical core arrangement for four control rods

fuel elements from the B-ring to the outer portion of the core and removal of the central thimble.

Two generally triangular-shaped sections are cut out of the upper grid plate. Each encompasses one D- and two E-ring holes. When fuel elements are placed in these locations, their lateral support is provided by a special fixture. When the fuel elements and support are removed, there is room for inserting specimens up to 2.4 in. (6.1 cm) in diameter.

Two 5/8in. (1.6cm) diameter holes between the F- and G-rings of the grid plate locate and provide support for the source holder at alternate positions.

The differential area between the triangular-shaped spacer blocks at the top of the fuel element and the round holes in the top grid plate permits passage of cooling water through the plate.

The bottom grid plate is an anodized aluminum plate 3/4 in. (1.9 cm) thick which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. Six pads welded to a ring which is, in turn, welded to the reflector container, support the bottom grid plate.

Holes in the bottom grid plate are aligned with fuel element holes in the top grid plate. They are countersunk to receive the adaptor end of the fuel-moderator elements and the adaptor end of the pneumatic transfer tube.

A central hole 1.505 in. (3.82 cm) in diameter in the lower grid serves as a clearance hole for the central thimble. Eight additional 1.505-in. (3.82-cm)-diameter holes are aligned with upper grid plate holes to provide passage of fuel follower control rods. Those holes in the bottom grid plate not occupied by control rod followers are plugged with removable fuel element adaptors that rest on the safety plate. These fuel element adaptors are aluminum cylinders 1.5 in. (3.8 cm) in diameter by about

17 in. (43.2 cm) long. At the lower end is a fitting that is accommodated by a hole in the safety plate. The upper end of the cylinder is flush with the upper surface of the bottom grid plate when the adaptor is in place. This end of the adaptor has a hole similar to that in the bottom grid plate for accepting the fuel element lower end fitting. With the adaptor in place, a position formerly occupied by a control rod with a fuel follower will now accept a standard fuel element. The adaptor can be removed with a special handling tool.

The safety plate is provided to preclude the possibility of control rods falling out of the core. It is a 1/2-in. (1.3-cm)-thick plate of aluminum attached to the extension of the inner reflector liner and placed about 16 in. (40.6 cm) below the bottom grid plate.

3.4. Fuel-Moderator Elements

The active part of each fuel-moderator element, shown in Fig. 1-7, is approximately 1.43 in. (3.63 cm) in diameter and 15 in. (38.1 cm) long. The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing about 8-1/2% by weight of uranium enriched to 20% U-235. The hydrogen-to-zirconium atom ratio is approximately 1.6. A small hole is drilled through the center of the active fuel section to facilitate hydriding; a zirconium rod is inserted in this hole after hydriding is complete.

Each element is clad with a 0.020-in. (0.051-cm)-thick stainless steel can, and all closures are made by heliarc welding. Two sections of graphite are inserted in the can, one above and one below the fuel, to serve as top and bottom reflectors for the core. Stainless steel end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element 28.8 in. (73.15 cm).

The lower end fixture supports the fuel-moderator element on the bottom grid plate. The upper end fixture consists of a knob for attachment of the

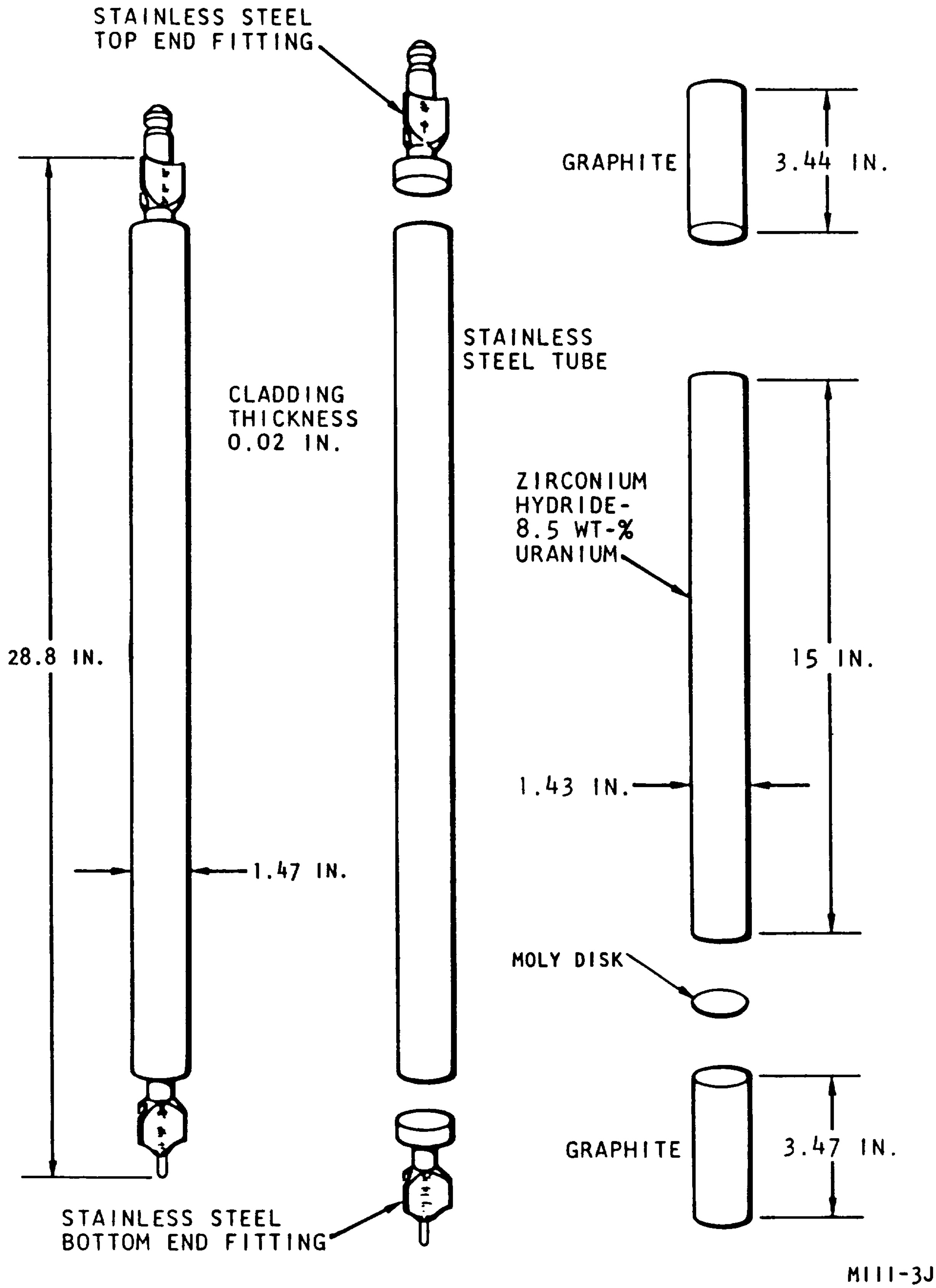


Fig. 1-7. TRIGA stainless steel-clad fuel element with triflute end fittings

fuel-handling tool and a triangular spacer, which permits cooling water to flow through the upper grid plate. The total weight of a fully loaded fuel element is about 7 lb (3.2 kg).

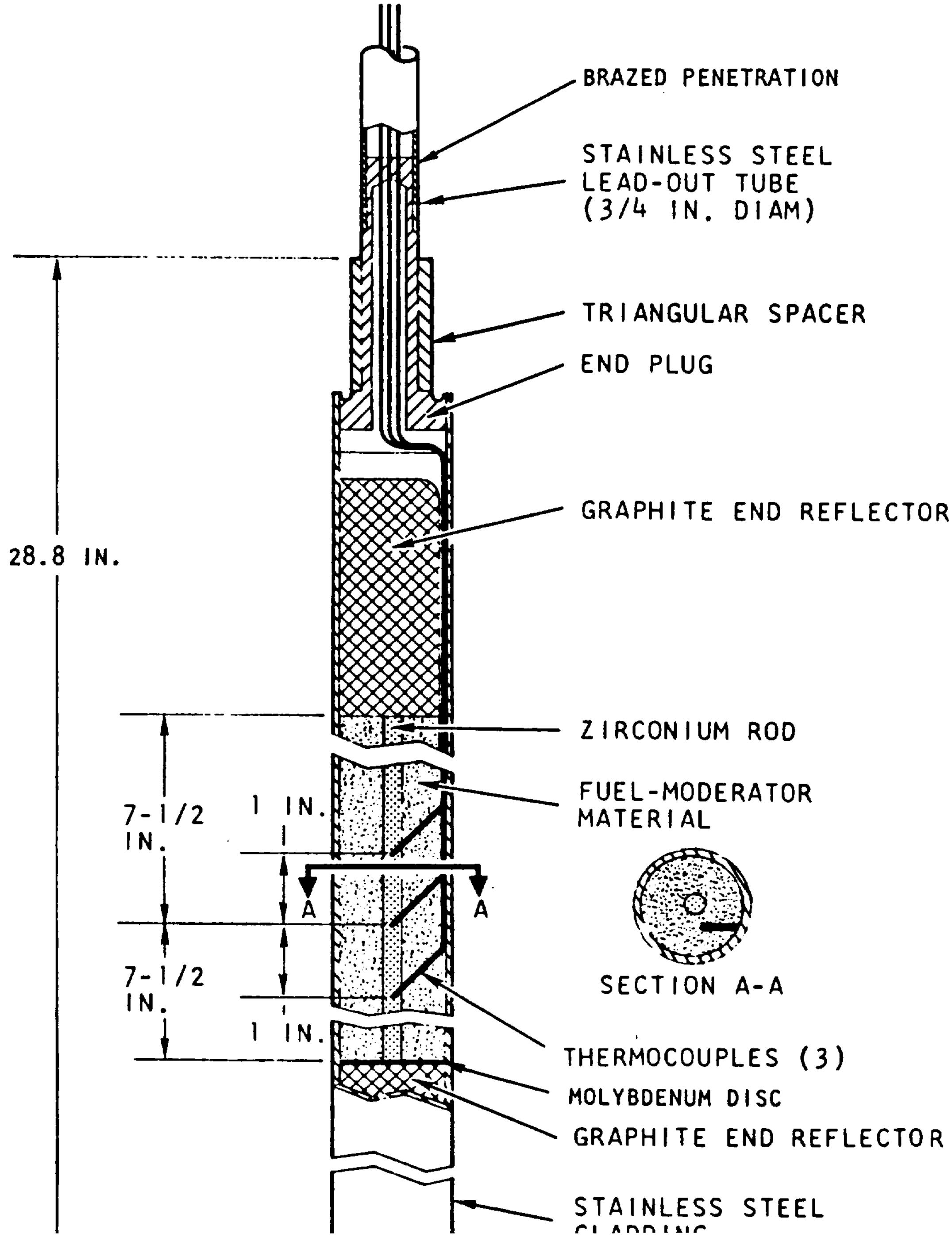
An instrumented fuel-moderator element will have three thermocouples embedded in the fuel. As shown in Fig. 1-8, the sensing tips of the fuel element thermocouples are located about two-thirds of the distance between the outer radius and the vertical centerline at the center of the fuel section and 1 in. (2.5 cm) above and below the horizontal midplane. The thermocouple leadout wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. This tube projects about 18 in. (46 cm) above the upper end of the element and is extended by two lengths of tubing connected by unions to provide a watertight conduit carrying the lead-out wires above the water surface in the reactor pool. In other respects the instrumented fuel-moderator element is identical to the standard element.

3.5. Graphite Dummy Elements

Graphite dummy elements may in some instances be used to fill grid positions not filled by the fuel-moderator elements or other core components. They are of the same general dimensions and construction as the fuel-moderator elements, but are filled entirely with graphite and clad with Al.

3.6. Neutron Source and Holder

The neutron source consists of a mixture of americium and beryllium, double encapsulated to ensure leak-tightness. Its initial strength at manufacture is 2 Ci. The upper and lower portions of the holder are screwed together to enclose a cavity that contains the source. A shoulder at the upper end of the neutron source holder supports the assembly in any of the grid locations in the upper grid plate. The upper end fixture


BRAZED PENETRATION
STAINLESS STEEL
LEAD-OUT TUBE
(3/4 IN. DIAM)
TRIANGULAR SPACER
END PLUG
GRAPHITE END REFLECTOR
28.8 IN.
ZIRCONIUM ROD
FUEL-MODERATOR
MATERIAL
7-1/2
IN.
1 IN.
A
A
SECTION A-A
7-1/2
IN.
1 IN.
THERMOCOUPLES (3)
MOLYBDENUM DISC
GRAPHITE END REFLECTOR
STAINLESS STEEL

of the source holder contains a knob similar to that of a fuel element so the source holder can be installed or removed with the fuel handling tool. In addition, the upper end fixture has a small hole through which one end of a stainless steel wire may be inserted to facilitate handling from the top of the tank.

3.7. Control System Design

The reactor uses four control rods: a regulating rod, two shim rods and a safety-transient rod. The regulating and shim rods are sealed Type 304 stainless steel tubes approximately 43 in. (109 cm) long by 1.35 in. (3.43 cm) in diameter in which the uppermost 6.5-in. (16.5-cm) section is an air void and the next 15 in. (38.1 cm) is the neutron absorber (boron carbide in solid form). Immediately below the neutron absorber is a fuel follower section consisting of 15 in. (38.1 cm) of U-$ZrH_{1.6}$ fuel. The bottom section of the rod is 6.5-in. (16.5-cm) air void.

The regulating, and shim rods pass through and are guided by 1.5-in. (3.8-cm)-diameter holes in the top and bottom grid plates. A typical control rod with fuel follower is shown in the withdrawn and inserted positions in Fig. 1-9.

The safety-transient rod is a sealed, 36.75-in. (93.35-cm)-long by 1-1/4-in. (3.18-cm)-diam aluminum tube containing solid boron carbide as a neutron absorber. Below the absorber is an air-filled follower section. The absorber section is 15 in. (38.1 cm) long and the follower is 20.88 in. (53.02 cm) long. The transient rod passes through the core in a perforated aluminum guide tube. The tube receives its support from the safety plate and its lateral positioning from both grid plates; it extends approximately 10 in. (25.4 cm) above the top grid plate. Water passage through the tube is provided by a large number of holes distributed evenly over its entire length.

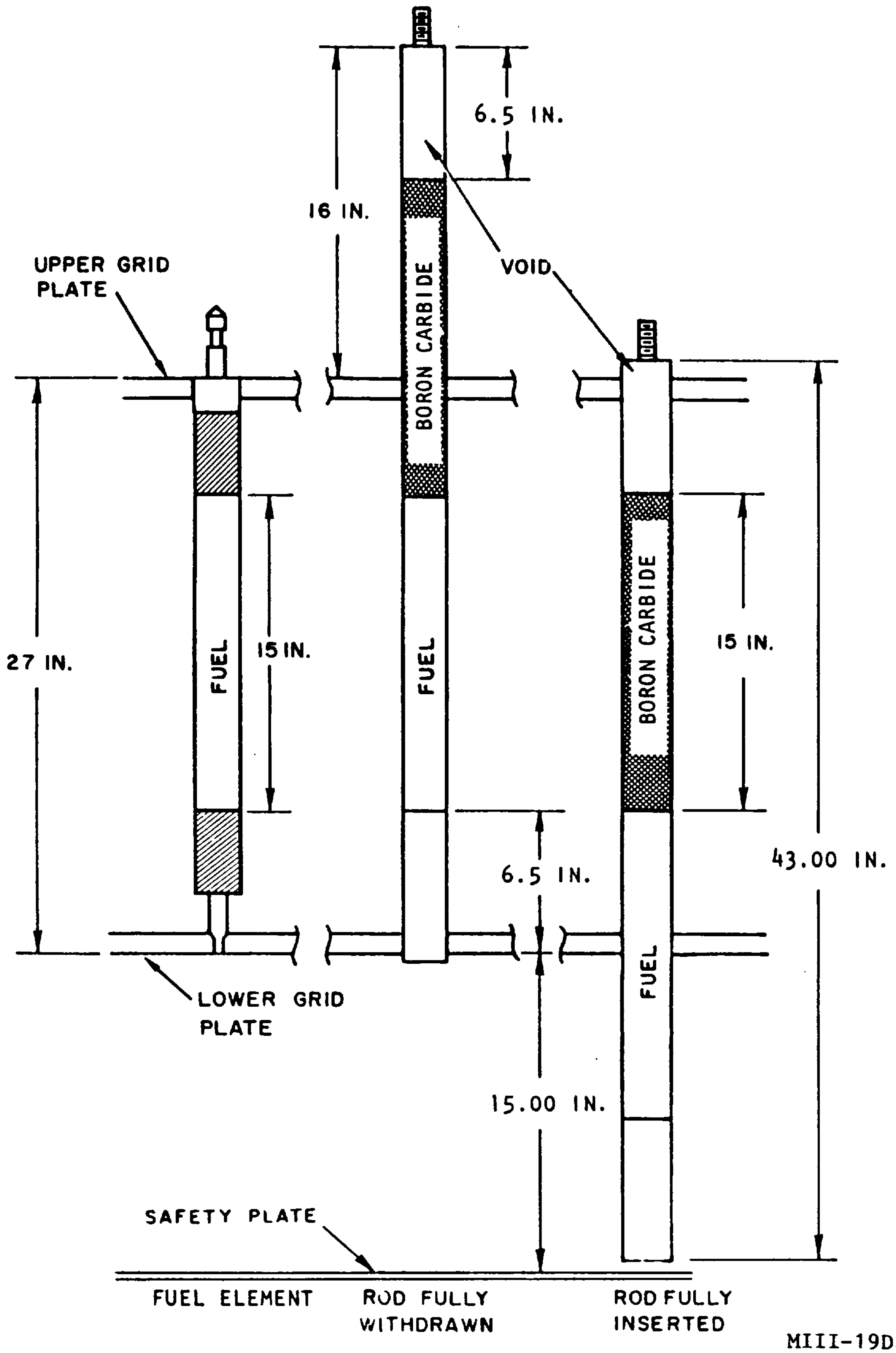


Fig. 1-9. Fuel follower control rod shown withdrawn and inserted

The control rods are connected to their individual drive units by screwing the upper end of the rod into a control rod drive assembly connecting rod. Vertical travel of each rod is approximately 15 in. (38.1 cm).

3.7.1. <u>Control Rod Drive Assemblies</u>. The control rod drive assemblies for the shim, and regulating rods are mounted on a bridge assembly over the pool and consist of a motor and reduction gear driving a rack-and-pinion, as indicated in Fig. 1-10. A helipot connected to the pinion generates the position indication. Each control rod has an extension tube that extends to a dashpot below the surface of the water. The dashpot and control rod assembly are connected to the rack through an electromagnet and armature. In the event of a power failure or scram signal, the control rod magnets are de-energized and the rods fall into the core. The time required for a rod to drop into the core from the full-out position is about 1 sec. The rod drive motor is nonsynchronous, single-phase, and instantly reversible, and will insert or withdraw the control rod at a rate of approximately ~~19~~ 18 in. (48 cm) per min for the shim rod or ~~24~~ 27 in. (61 cm) per min for the regulating rod. The regulating rod drive has a variable speed stepping motor. A key-locked switch on the control console power supply prevents unauthorized operation of all control rod drives. Electrical dynamic and static braking on the motor are used for fast stops. Control rod travel is 15 in. (38.1 cm).

Limit switches mounted on the drive assembly indicate the following:

1. The "up" and "down" position of the magnet
2. The "down" position of the rod
3. The magnet in contact with the rod

3.7.2. <u>Transient Rod Drive Assembly</u>. The safety-transient control rod on pulsing TRIGA Mark II reactors is operated with a pneumatic drive (see Fig. 1-11). The operation of the transient rod drive is controlled from the reactor console.

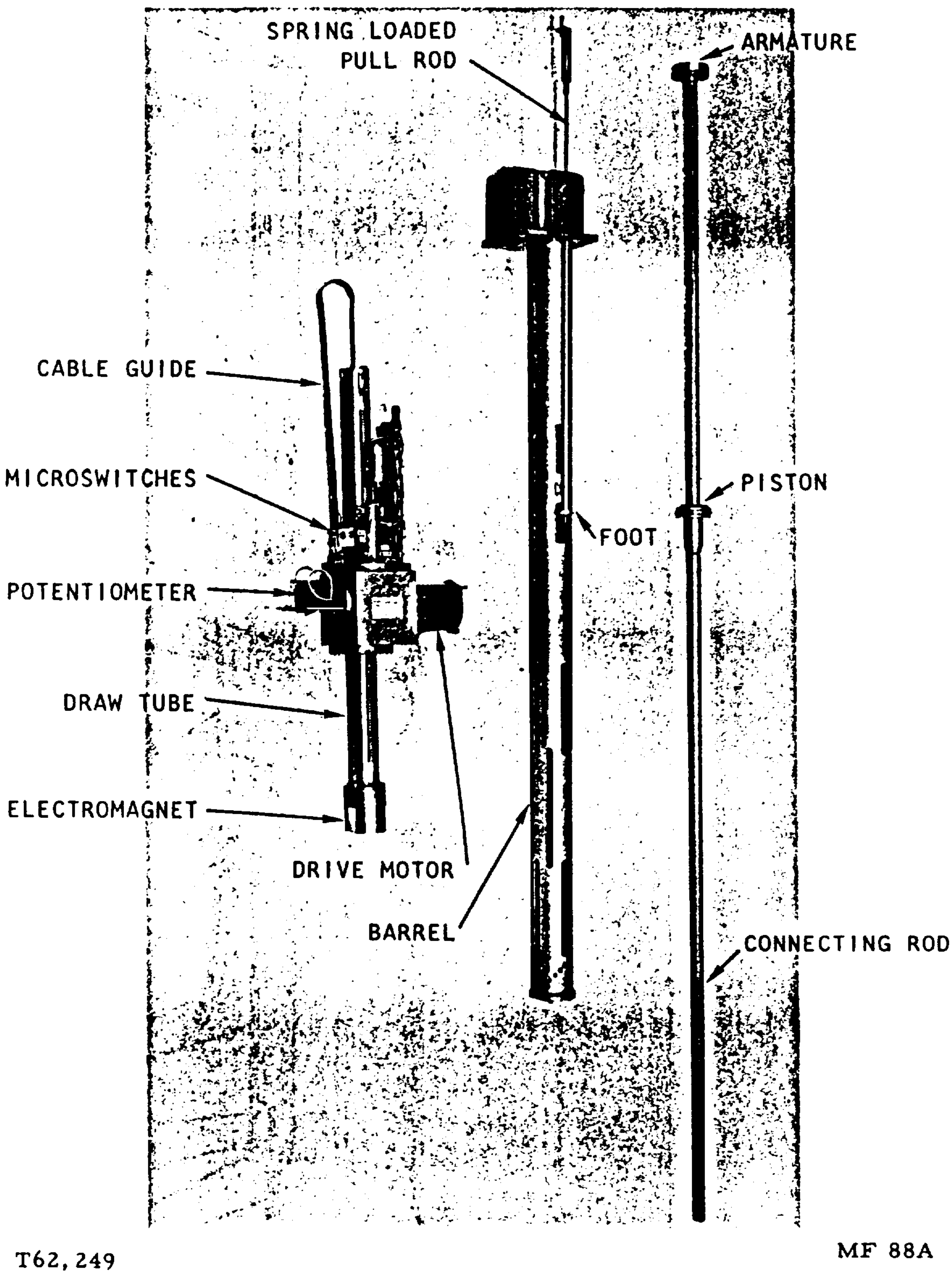
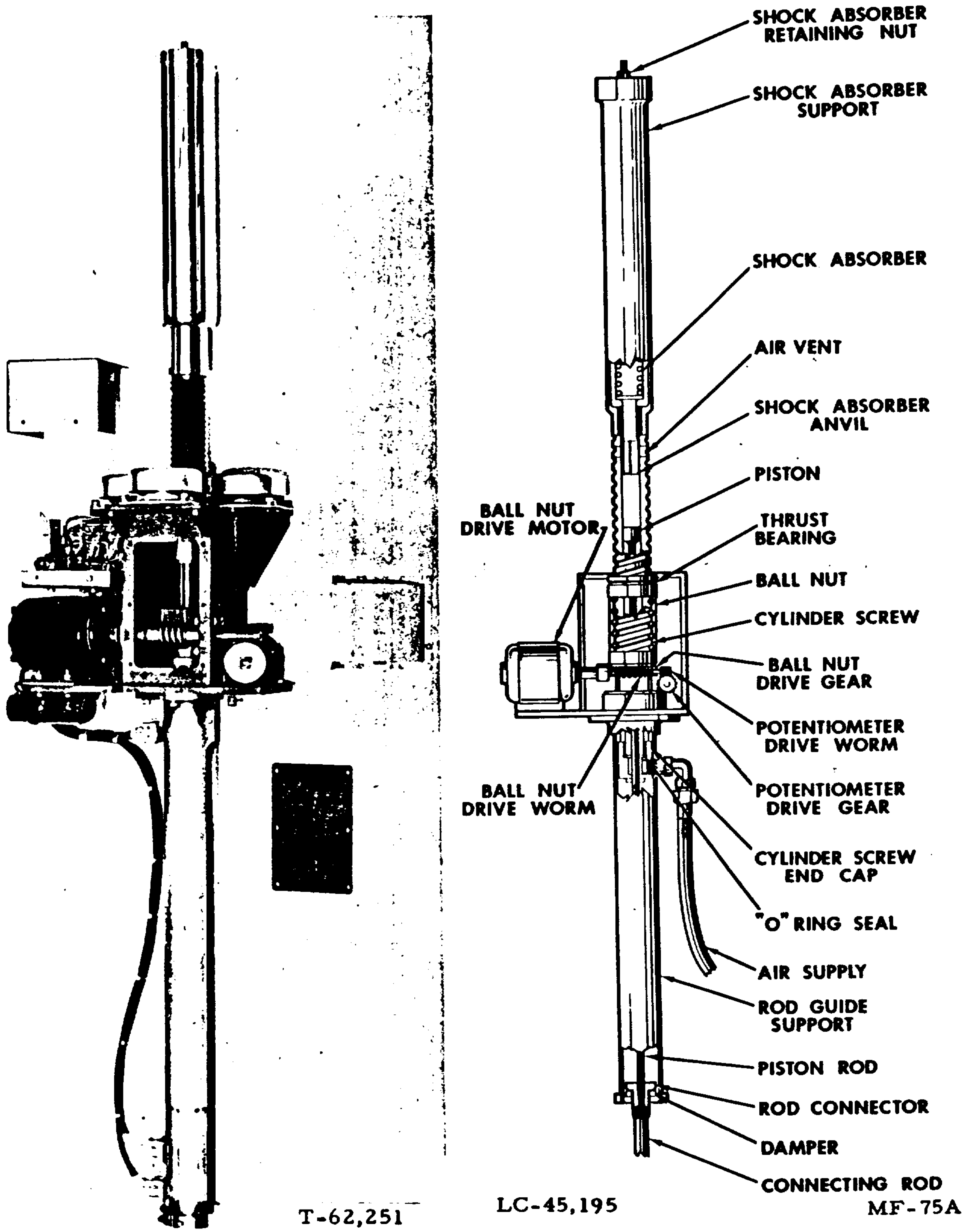


Fig. 1-10. Rack-and-pinion control rod drive (typical)



MF-75A

Fig. 1-11. Typical pneumatic/electromechanical transient drive

The transient rod drive is mounted on a steel frame that bolts to the bridge. Any value from zero to a maximum of 15 in. (38.1 cm) of rod may be withdrawn from the core.

The transient rod drive is a single-acting pneumatic cylinder with its piston attached to the transient rod through a connecting rod assembly. The piston rod passes through an air seal to the lower end of the cylinder (see Fig. 2-12). Compressed air is supplied to the lower end of the cylinder from an accumulator tank when a three-way solenoid valve located in the piping between the accumulator and cylinder is energized. The compressed air drives the piston upward in the cylinder and causes the rapid withdrawal of the transient rod from the core. As the piston rises, the air trapped above it is pushed out through vents at the upper end of the cylinder. At the end of its travel, the piston strikes the anvil of an oil-filled hydraulic shock absorber which has a spring return and which decelerates the piston at a controlled rate over its last 2 in. (5 cm) of travel. When the solenoid is de-energized, the valve cuts off the compressed air supply and exhausts the pressure in the cylinder, thus allowing the piston to drop by gravity to its original position and restore the transient rod to its fully inserted position in the reactor core.

The extent of transient rod withdrawal from the core during a pulse is determined by raising or lowering the cylinder, thereby controlling the distance the piston travels.

The cylinder has external threads running most of its length which engage a series of ball bearings contained in a ball nut mounted in the drive housing. As the ball nut is rotated by a worm gear, the cylinder moves up or down depending on the direction of worm gear rotation. A ten-turn potentiometer driven by the worm shaft provides a signal indicating the position of the cylinder and the distance the transient rod will be ejected from the reactor core. Motor operation for pneumatic cylinder positioning is controlled by a switch on the reactor control console. The magnet power key switch on the control console power supply prevents unauthorized firing of the transient rod drive.

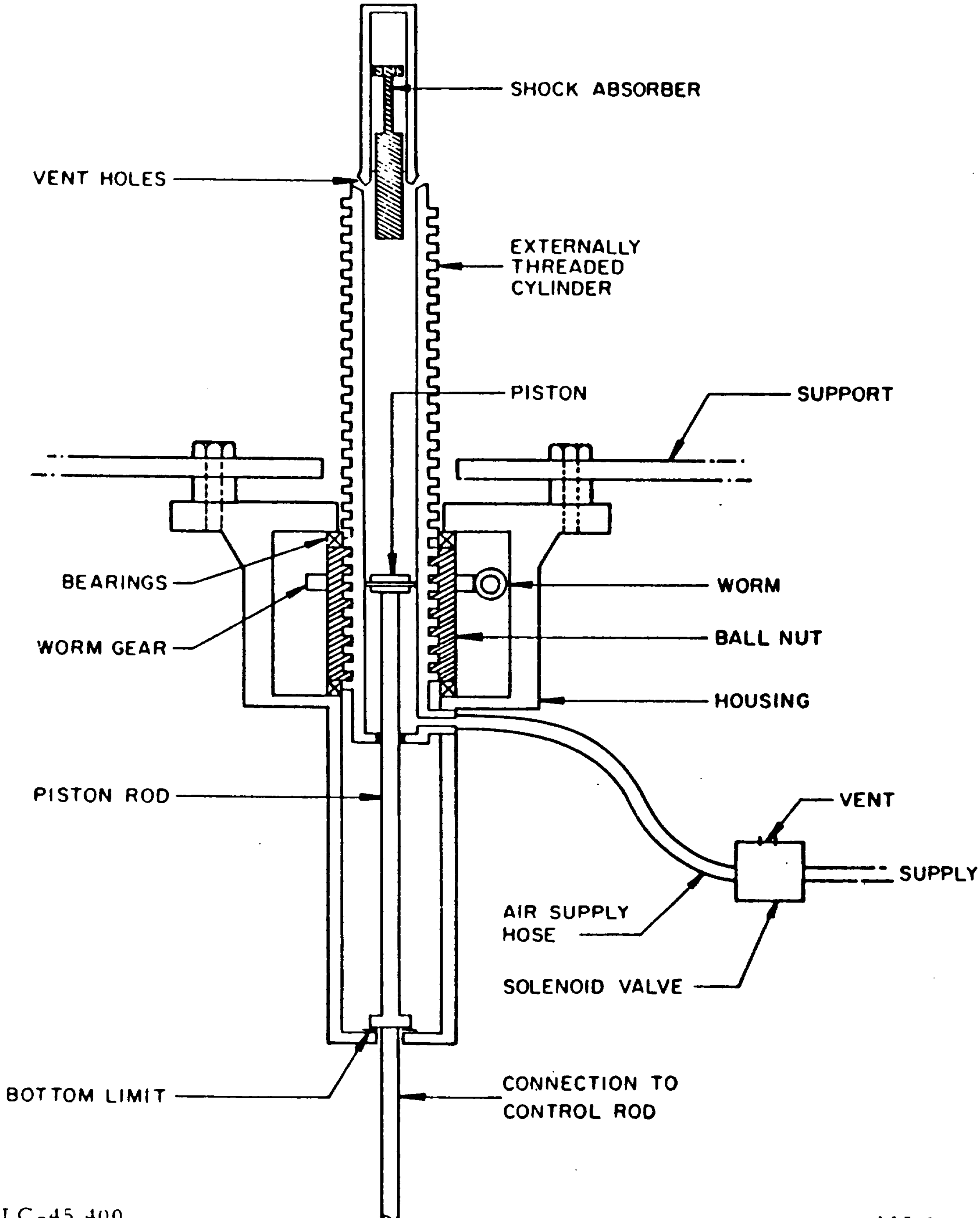


Fig. 1-12. Schematic diagram of transient-rod drive

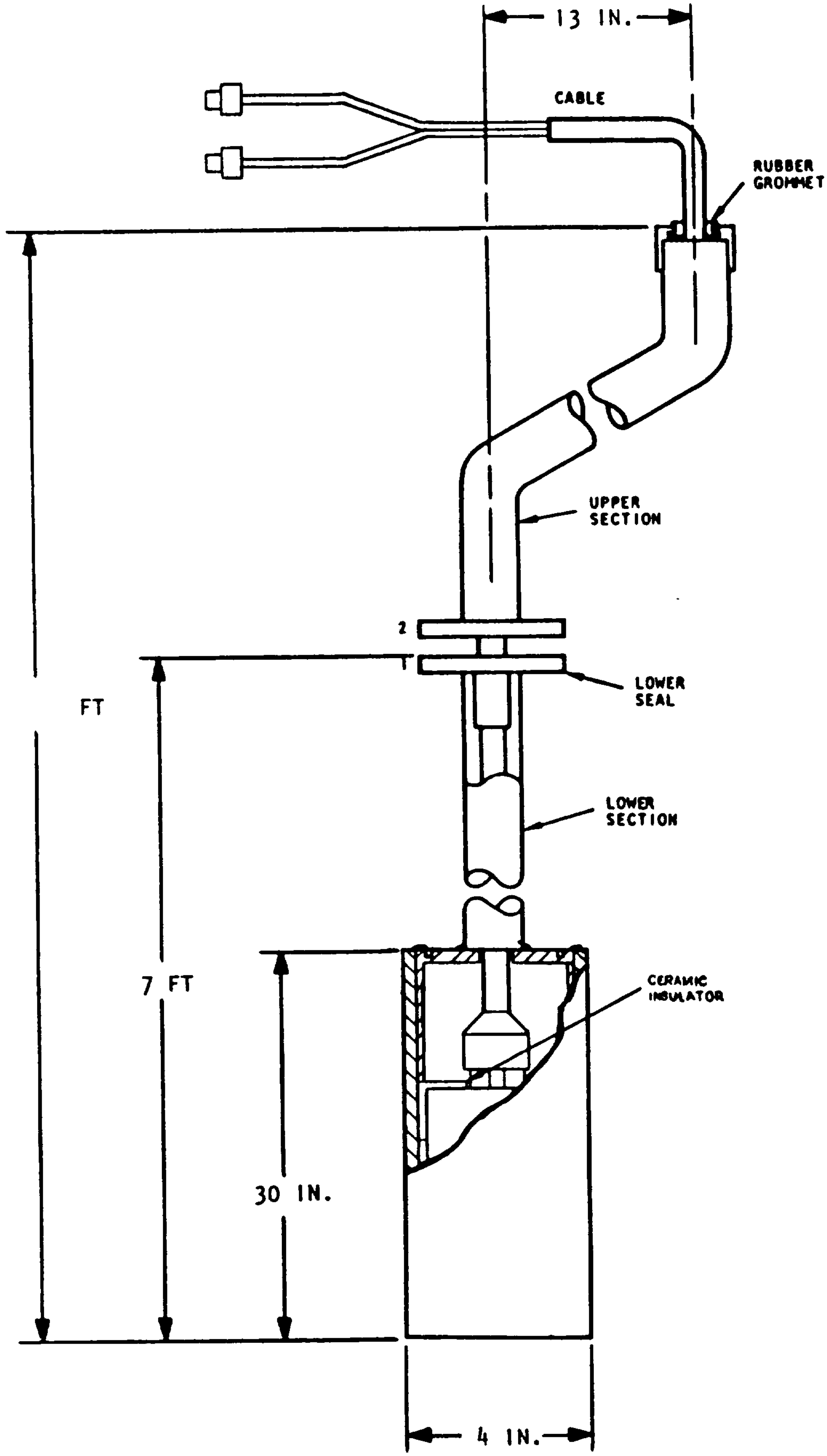
Attached to, and extending downward from, the transient rod drive housing is the rod guide support which serves several purposes. The air inlet connection near the bottom of the cylinder projects through a slot in the rod guide and prevents the cylinder from rotating. Attached to the lower end of the piston rod is a flanged connector that is attached to the connecting rod assembly that moves the transient rod. The flanged connector stops the downward movement of the transient rod when the connector strikes the damper pad at the bottom of the rod guide support. A microswitch is mounted on the outside of the guide tube with its actuating lever extending inward through a slot. When the transient rod is fully inserted in the reactor core, the flanged connector engages the actuating lever of the microswitch, and indicates on the instrument console that the rod is in the core.

In the case of the safety-transient rod, a scram signal de-energizes the solenoid valve which supplies the air required to hold the rod in a withdrawn position and the rod drops into the core from the full-out position in about 1 sec.

3.8. Detectors

The detectors (see Fig. 1-13) used with TRIGA reactors are standard commercial units which have been enclosed in seal-welded aluminum cans to provide shielding from spurious noise, complete moisture proofing, and physical support of detector and cable. This construction improves reliability and life of the detector assembly. Helium leak testing is performed on all assemblies after welding. No plastic or gasket type joints are used in a location where they are subject to radiation.

The electrical connections for each chamber are contained in a 3/4-in. offset aluminum pipe which terminates above the water level at the side of the tank. A flanged, gasketed joint is provided below the offset.



EL-2631

Fig. 2-13. Typical detector assembly

The chambers are located adjacent to the core reflector. To provide lateral positioning and to facilitate removal and reinstallation, each chamber is positioned inside an aluminum guide tube. These tubes are attached to the outer edge of the reflector assembly.

REACTOR INSTRUMENTATION AND CONTROL SYSTEMS

In Chapter I, the major components and systems of the TRIGA reactor were described. In this chapter, the control and instrumentation systems employed in operating the TRIGA are discussed. This discussion is divided into four sections:

1. Control system console data acquisition unit and power monitors.

2. Safety systems, control rod drives, and scram circuit.

3. Modes of operation, servo controller, and auxilary channels.

4. Software general descriptions.

Functional block diagrams, which indicate the methods of measurement, presentation of data to the operator, as well as operating principles of the reactor systems, are presented in this chapter.

2.1 TRIGA Instrumentation and Control System

The Instrumentation and Control System (ICS) used in the General Atomics TRIGA research reactor is a computer based reactor control system. The basic elements of this system are:

(1) A control system console (CSC);

(2) A data acquisition and control unit (DAC);

(3) Three independent power monitors and associated safety circuits;

(4) Two (or more) independent fuel temperature monitors and associated safety circuits.

Figure 2.1 is a simplified block diagram of the ICS.

2.1.1. Control System Console (CSC)

The CSC is a desk-type control console located in the Reactor Control Room (Figure 2.2). Reactor operations from this console are conducted a set of control switches, mode switches and a keyboard, and information is fed back to the operator through a microcomputer, color CRT, high resolution color monitor, and various indicators and annunciators.

The rod positions are adjusted by issuing commands to the CSC which in turn transmits these commands to the DAC vis high speed data communication

GENERAL ATOMICS

GA TRIGA CONTROL SYSTEM BLOCK DIAGRAM

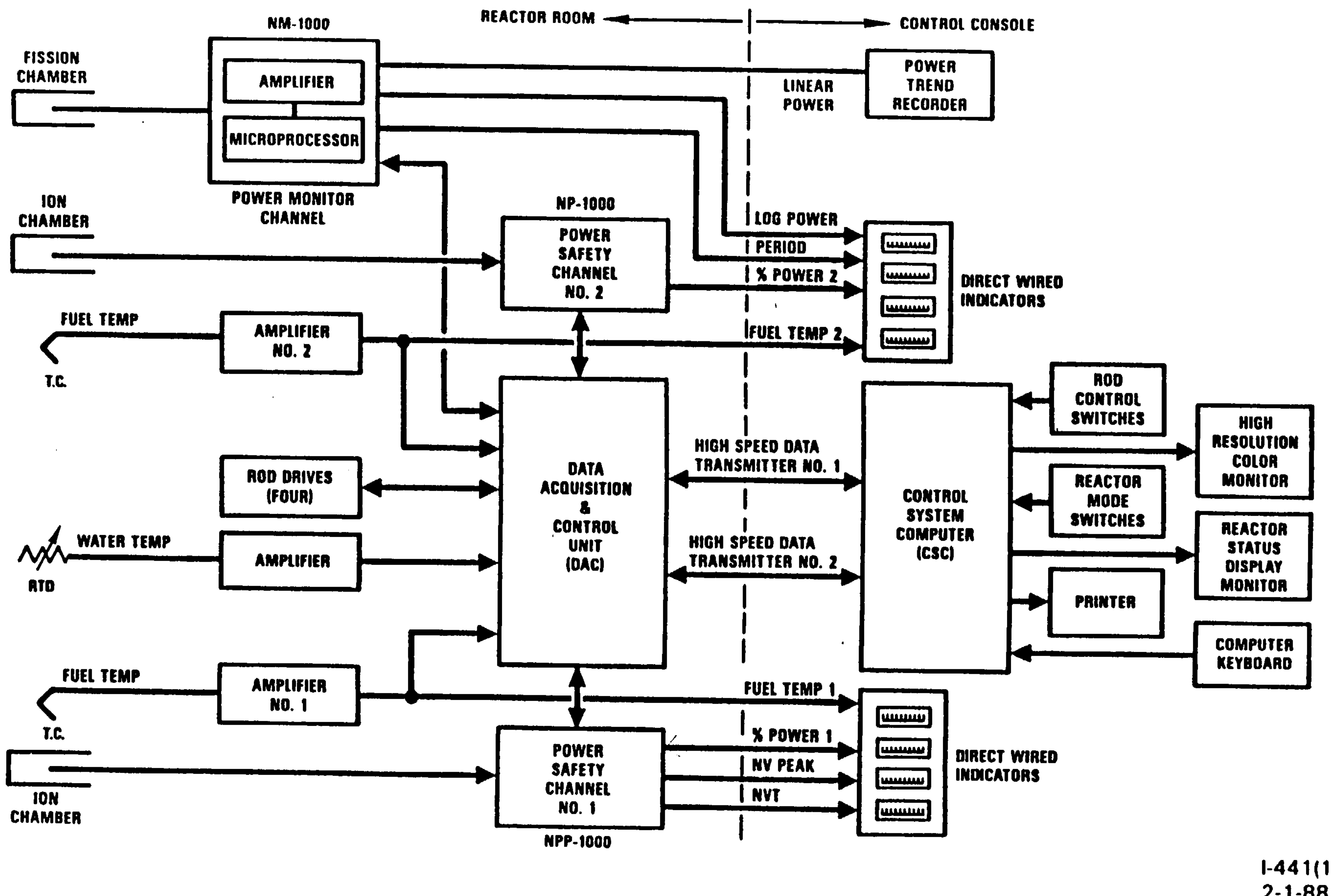


FIGURE 2.1.

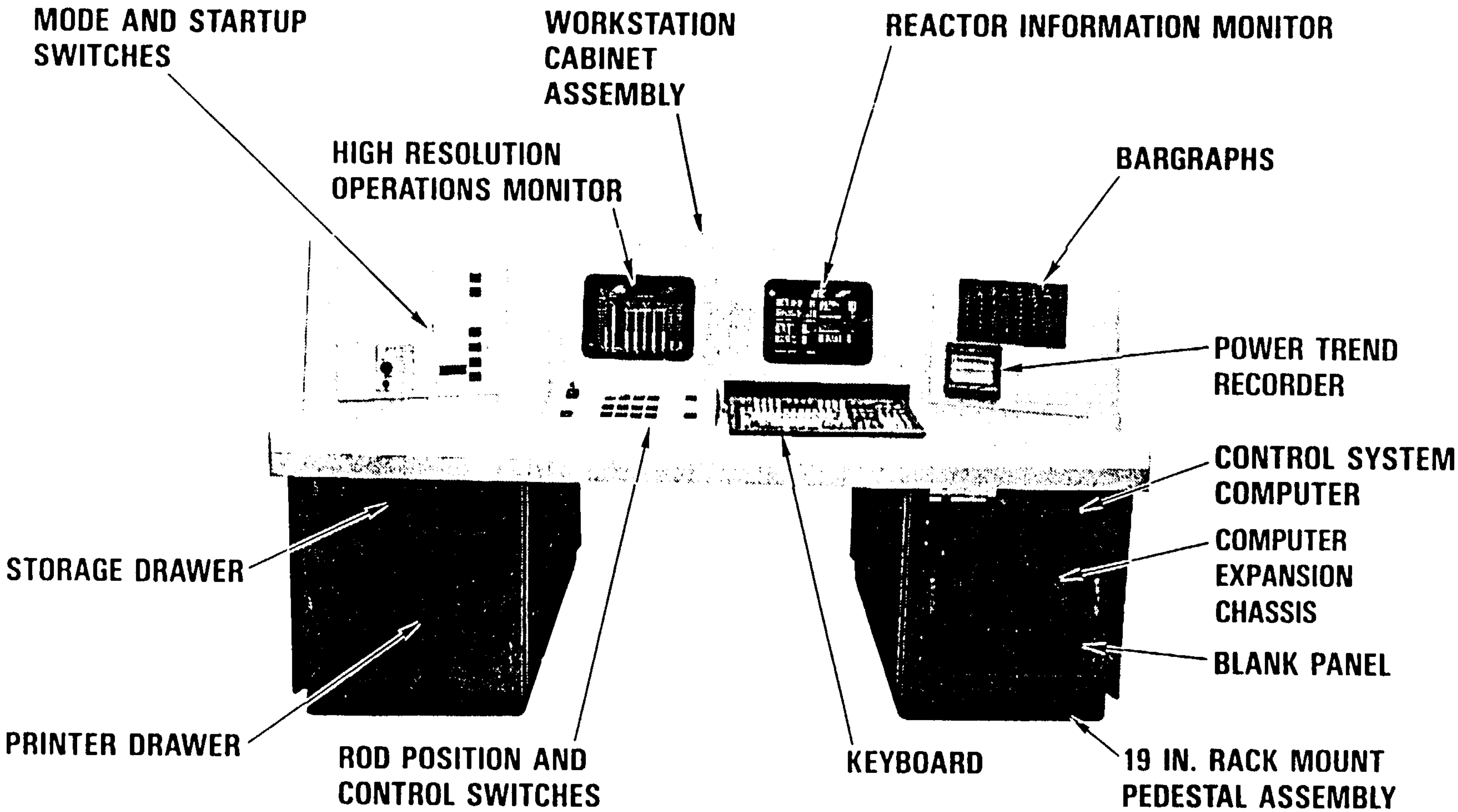


Figure 2-2 TRIGA Control System Console

cables. The DAC reissues these commands to the rod drive mechanisms. During reactor operations, the CSC receives raw data from the DAC, again via the communications network, processes this data, and presents the data in meaningful engineering units and graphic displays on a number of peripheral systems. The CSC also provides data storage and logging capabilities.

2.1.1.1. Control Panels and Indicators

The rod control panel, located directly below the high-resolution color monitor, contains the following:

1. A SCRAM switch;
2. A rod magnet ON/OFF/RESET key switch;
3. Manual control rod adjustment switches (UP, DOWN, CONTACT/ON)
4. An ALARM ACKNOWLEDGE switch.

To the right of the reactor status CRT are eight vertical LED bar graph displays indicating (i) percent power (safety channels 1 and 2), (ii) log power, (iii) period, (iv) fuel temperature (safety channels 1 and 2), (v) nv (peak power in a pulse), and (vi) nvt (energy released in a pulse). All bargraph displays are hardwired to their respective sensors and signal processors, and thus do not rely on computer generated output. This is part of the hardwired reactor safety system.

To the left of the reactor control CRT is the mode control panel containing (i) the console power-on switch, (ii) the prestart run switch, (iii) reactor mode switches, and (iv) thumb-wheel switches to set percent power demand in the automatic mode.

2.1.1.2. CRT Displays

The CSC incorporates two color CRT monitors. A high resolution color graphics (reactor control) CRT provides the operator with a real-time graphic display of the reactor status [Fig. 2.3(a)]. This CRT displays a number of strategic operating parameters using bar graphs and other digital displays, and alerts the operator to any abnormal or dangerous conditions. A second reactor status CRT displays pertinent diagnostic messages and reactor status and facility status information [Figure 2.3(b)].

2.1.1.3. Printer

The CSC also interfaces with a near-letter-quality (dot matrix) printer. The printer can be used to print out log information, historical data (such as pulse information), or any files stored in the CSC hard disk or floppy disk files. It also can be used to print the high resolution reactor status CRT, including pulses displays.

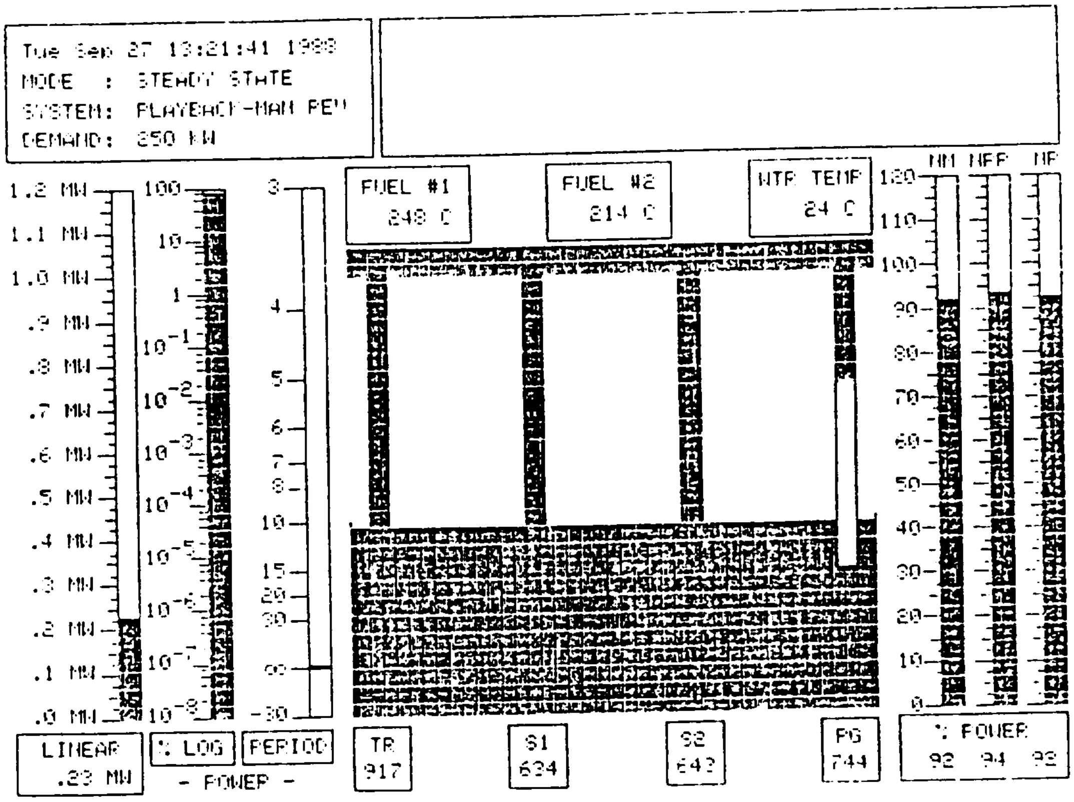


Figure 2-3(a). CSC High Resolution CRT Display of Reactor Operating Parameters.

STATUS WINDOW

Reactor Room Door	OPEN	Trans Rod Dn Limit Sw	YES
		Trans Drive Dn Lmt Sw	YES
Coolant Wtr Pool Temp	27.55 C	Trans Drive Up Lmt Sw	NO
Pri Coolant Condtvty	2.0e+0 Umh	Trans Rod Drive Positi	0
Pool Area monitor	.00 mR	Shim1 Rod Dn Limit Sw	YES
Beam Port 1 Area Monit	.00 mR	Shim1 Drive Dn Lmt Sw	YES
Beam Port 2 Area Monit	.00 mR	Shim1 Drive Up Lmt Sw	NO
Beam Port 3 Area Monit	.00 mR	Shim1 Rod Drive Positi	0
Beam Port 4 Area Monit	.00 mR		
NM1000 Log Power	-5.76	Shim2 Rod Dn Limit Sw	YES
NM1000 Linear Power	0 %	Shim2 Drive Dn Lmt Sw	YES
NPP1000 % Linear Power	0 %	Shim2 Drive Up Lmt Sw	NO
NP1000 % Linear Power	0 %	Shim2 Rod Drive Positi	2
Fuel Temp #1	23 C	Reg Rod Dn Limit Sw	YES
Fuel Temp #2	1 C	Reg Drive Dn Lmt Sw	YES
		Reg Drive Up Lmt Sw	NO
		Reg Rod Drive Position	0
Current Pulse Number	11146		
Pulse Mode Prohibit	OFF	Rod Withdrawal Prohibt	OFF

Figure 2-3(b). CSC Reactor and Facility Status CRT Display.

2.1.2. Data Acquisition and Control Unit

The DAC is located in the reactor room adjacent to the reactor and provides high speed data acquisition and control capability. It monitors the reactor power from the wide range power monitor (NM-1000), the safety channels (NP-1000 and NPP-1000), the fuel temperature safety channels, water temperature, and control rod position. The DAC receives commands from the CSC - and reissues these commands - to raise and lower the control rods or scram the reactor. It communicates with the CSC via a primary and a secondary serial data trunk; the secondary trunk serves as a backup should the primary trunk fail. These serial data trunks allow a drastic reduction in the wiring requirements between the reactor room and the control console.

The DAC is a multi-shelf cabinet which contains, in addition to a variety of signal processors, an industrially hardened microprocessor-based computer. Figure 2.4 shows the basic configuration of the cabinet. The DAC cabinet houses the following equipment, arranged on shelves from top to bottom:

1. Power supplies and power conditioning equipment;
2. Relay boards, optical isolator boards, and input scanner boards;
3. Action Paks, which monitor and condition signals for temperatures, conductivity as well as other parameters;
4. NP-1000 and NPP-1000 power safety channels;
5. Various analog termination and serial communications boards;
6. An IBM 7532 microcomputer.

2.1.2.1. Shelf One Components

Shelf one consists of power supplies and power conditioning equipment. The magnet power supply supplies magnet power to the control rod electromagnets in the SCRAM circuit. A potentiometer power supply supplies power to the potentiometers that monitor rod positions. The solenoid power supply provides power to the solenoid controlling the air for the transient rod mechanism. The auxiliary power supply furnishes power for control relays (RLY08s), the opto-isolators, and the DIS064 scanner board. The power conditioner contains a trip breaker and two line filters. It provides output to two 120 V ac strips. One is a direct output from the line filters; the other is switched by a remotely-controlled relay.

2.1.2.2. Shelf Two Components

The components of shelf two are shown in Fig. 2.5. These are:

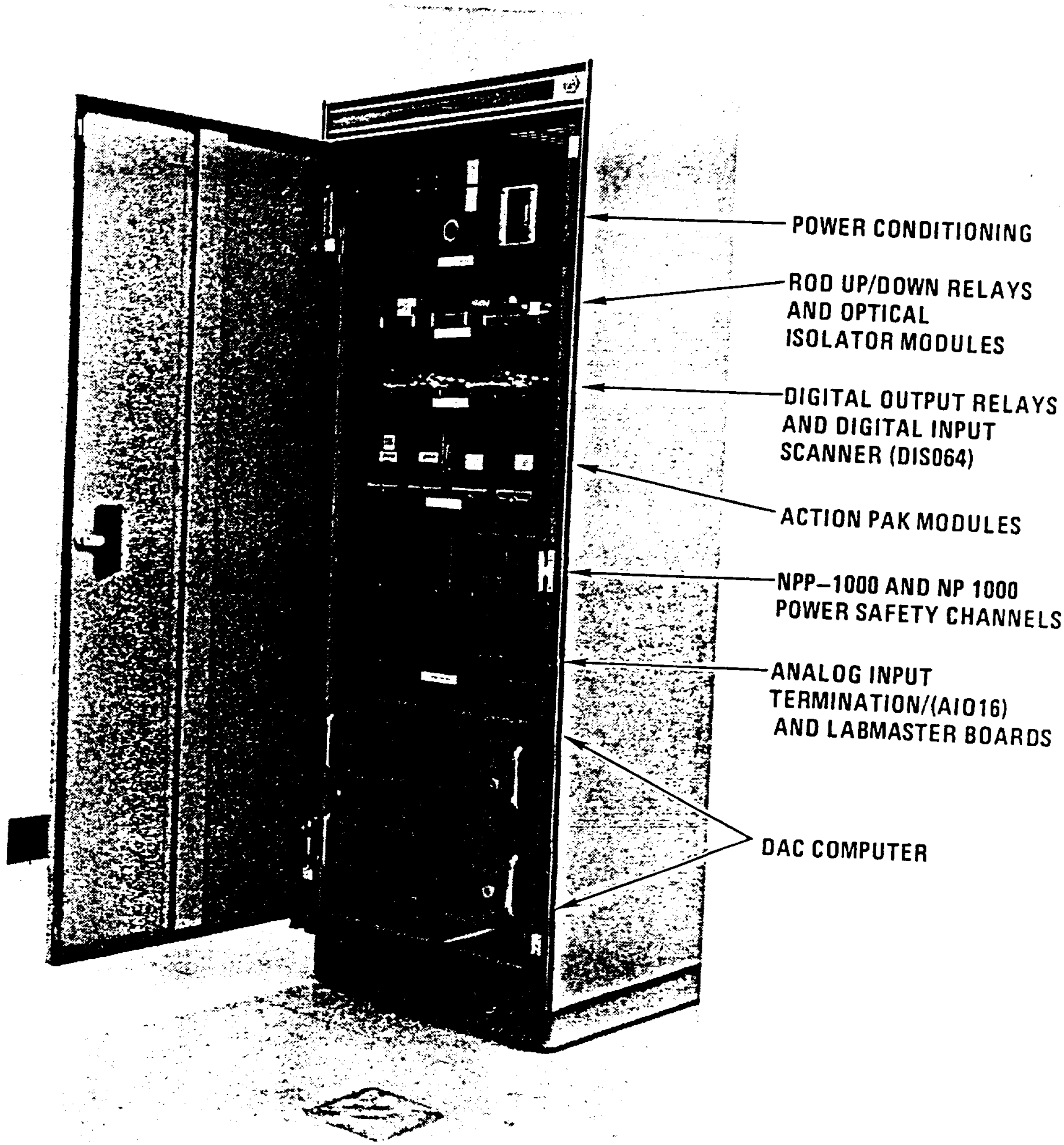


Figure 2-4. Configuration of DAC Cabinet.

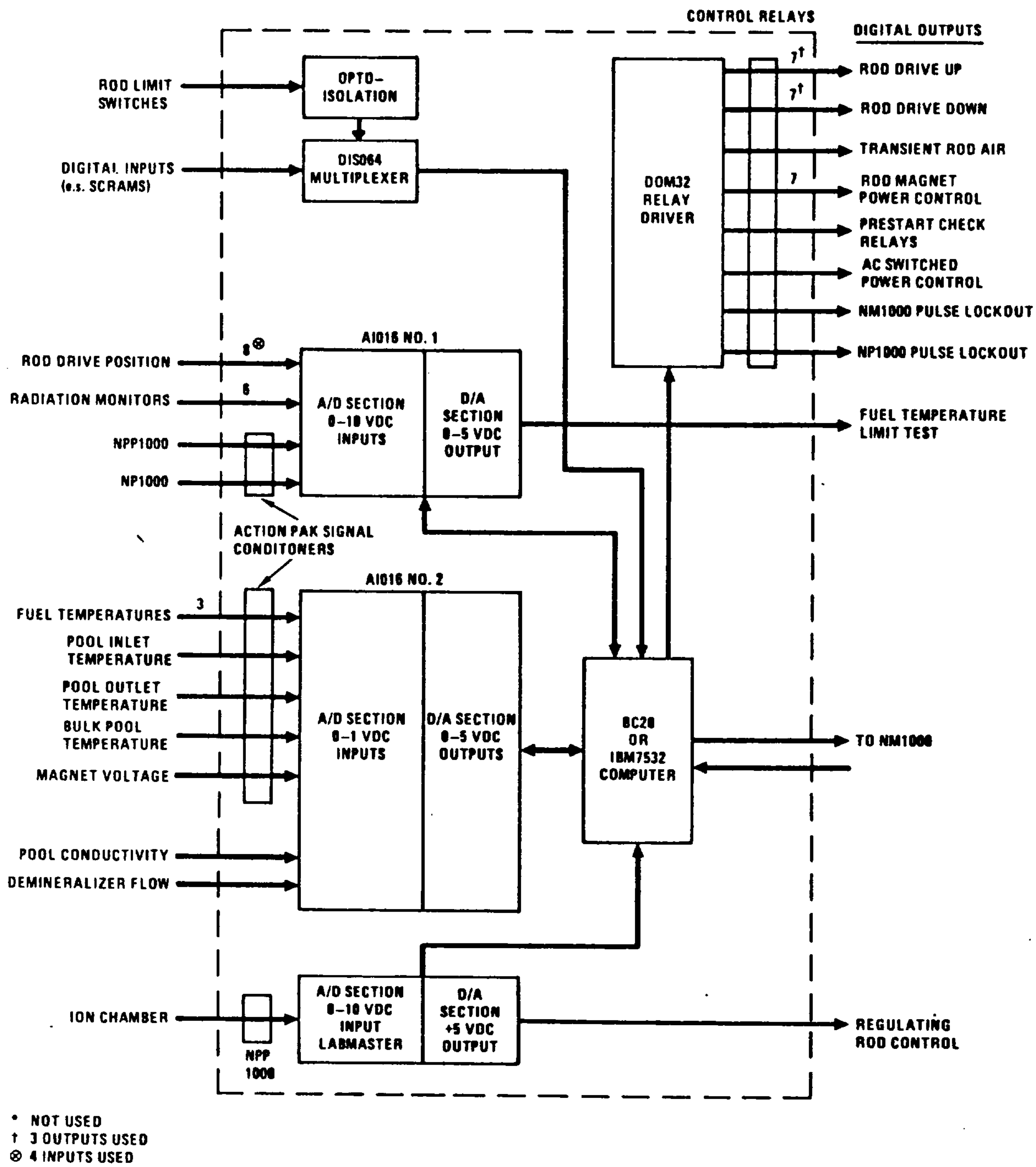


Figure 2-4 Summary of DAC Inputs and Outputs

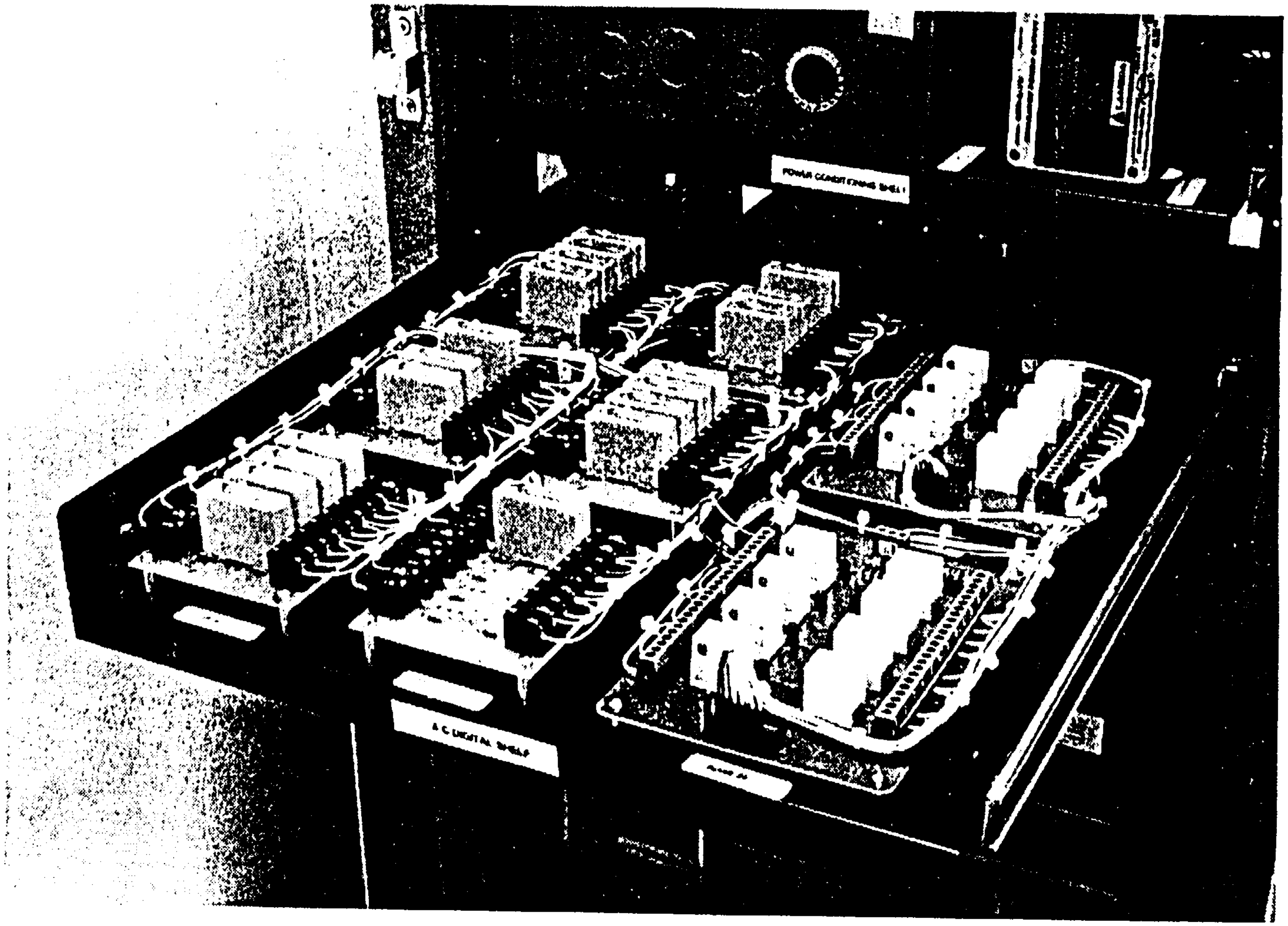


Figure 2-5. Components of Shelf 2 of DAC Cabinet.

1. Two RLY08 relay boards which handle output signals for rod movement. Board 1 controls the UP movement of the transient and standard rods; board 2 controls the DOWN movement of the transient and standard rods.

2. Six optical isolator module boards, each of which contain up to four opto-isolator signal conditioning modules. These boards interface the DAC components with external ac voltages.

2.1.2.3. Shelf Three Components

The components of shelf three are shown in Fig. 2.6. These are:

1. Three RLY08 relay boards which provide a variety of digital output signals. These include (i) control of the standard control rod electromagnetic current; (ii) transfer of test signal inputs into the Action Pak limit alarms during prestart test mode; (iii) DAC watchdog relay contacts in the SCRAM circuit; (iv) control of switched 120 V ac power; (v) NM-1000, NP-1000 scram lockouts during pulsing operation; (vi) NPP-1000 amplifier gain change control for pulse mode; and (vii) control of the transient rod air solenoid.

2. A digital input scanner board (DIS064) which has a capacity of 64 inputs. The module consists of a remote intelligent module (RIM808), which controls the DIS064, and a diode matrix/terminator board. The DIS064 scans up to 64 isolated contacts every 10 ms.

2.1.2.4. Shelf Four Components

Shelf four of the DAC (Figure 2.7) contains Action Pak modules and trip test relays which are used to monitor or condition reactor sensor signals (Action Paks perform the function of the bistable trips in the old analog TRIGA consoles). A summary of the functions of the various Action Paks is provided in Table 2.1.

2.1.2.5. Shelf Five Components

Shelf five (Figure 2.8) contains the NPP-1000 and the NP-1000 nuclear instruments that serve as power safety channels 1 and 2.

2.1.2.6. Shelf Six Components

Shelf six of the DAC contains the following components:

1. An analog input termination board (AI016) which receives all inputs to the AI016 board No. 2 in the microcomputer(shelf 7).

2. The Lab Master daughter board, which is an analog-to-digital (A/D) converter for the Lab Master mother board (located in

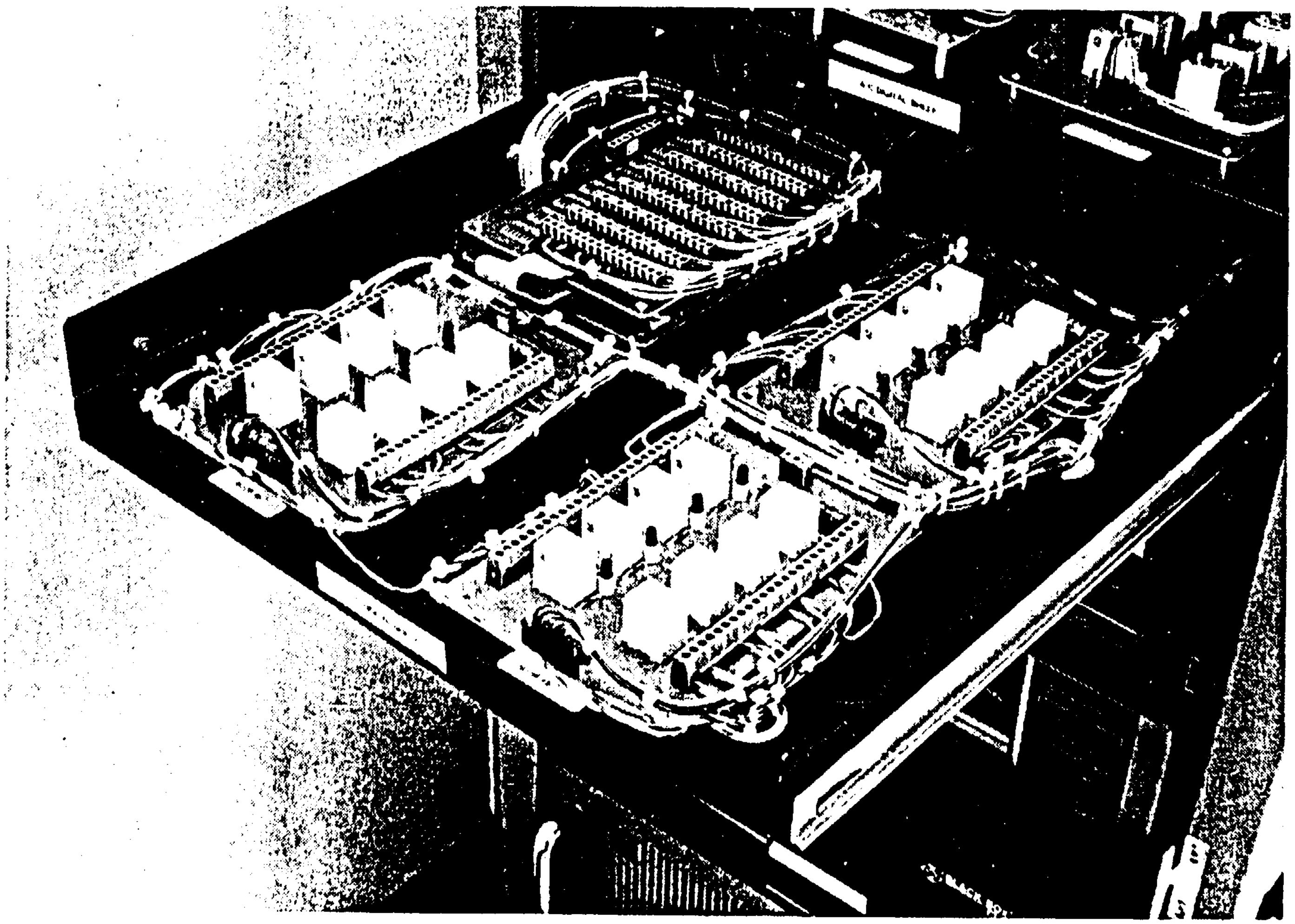


Figure 2-6. Components of Shelf Three in DAC.

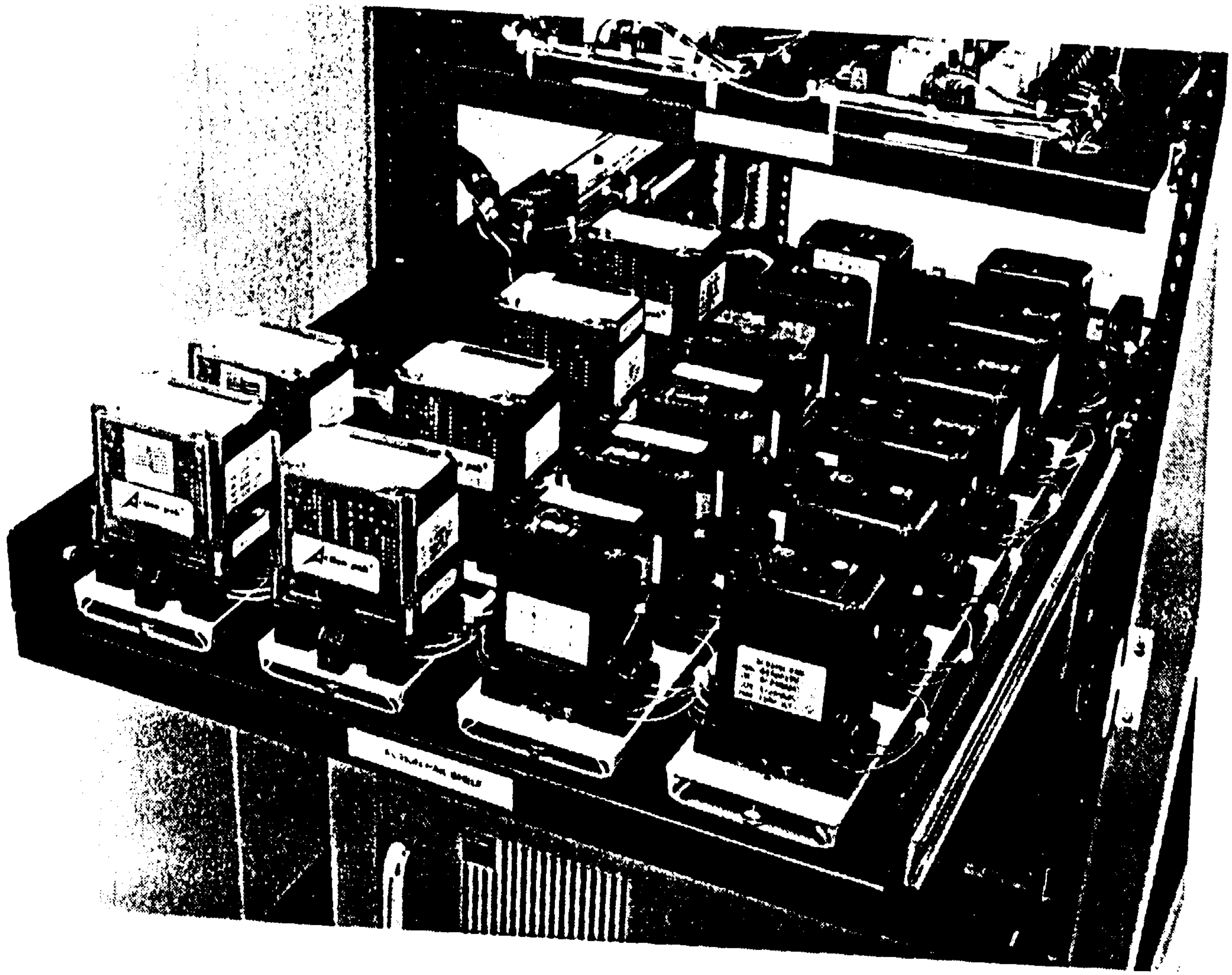
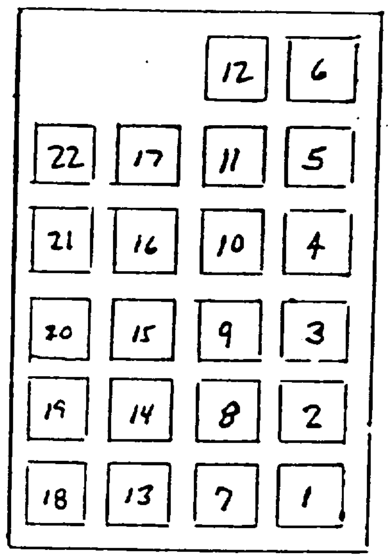


Figure 2-7. Components of Shelf Four in DAC.

Table 2.1. Summary of Action Pak Functions in DAC



Location	Action Pak Type	Function	Comments
1	4440 (-108)	Primary Flow	
2	4440 (-108)	Secondary Flow	
3	4157 (-177)	Water Temp In	
4	4157 (-177)	Water Temp Out	
5	4157 (-177)	Water Temp Pool	
6	4300 (-1206N)	Magnet Voltage Monitor	
7	4350 (-A0013)	Fuel Temp 1	
8	4350 (-A003)	Fuel Temp 2	
9	4350 (-A003)	Fuel Temp 3	Not Used
10	4010 (-144)	Fuel Temp Test	
11	----	Prestart Time Delay Relay	K2
12	4001.5	Conductivity	
13	1000 (-6016 L,N)	Fuel Temp 1 Limit	
14	1000 (-6016 L,N)	Fuel Temp 2 Limit	
15	1000 (-6016 L,N)	Fuel Temp 3 Limit	Not Used
16	1020 (-6074 L)	Ground Fault Detector	
17	4800 (-325)	Fuel Temp Ramp Timer	
18	4300 (-122)	NP-1000 High Power Limit	
19	4300 (-122)	NPP-1000 High Power Limit	
20	----	Scram Check Relay	K1
21	----	Scram Check Relay	K4
22	----	Scram Check Relay	K3

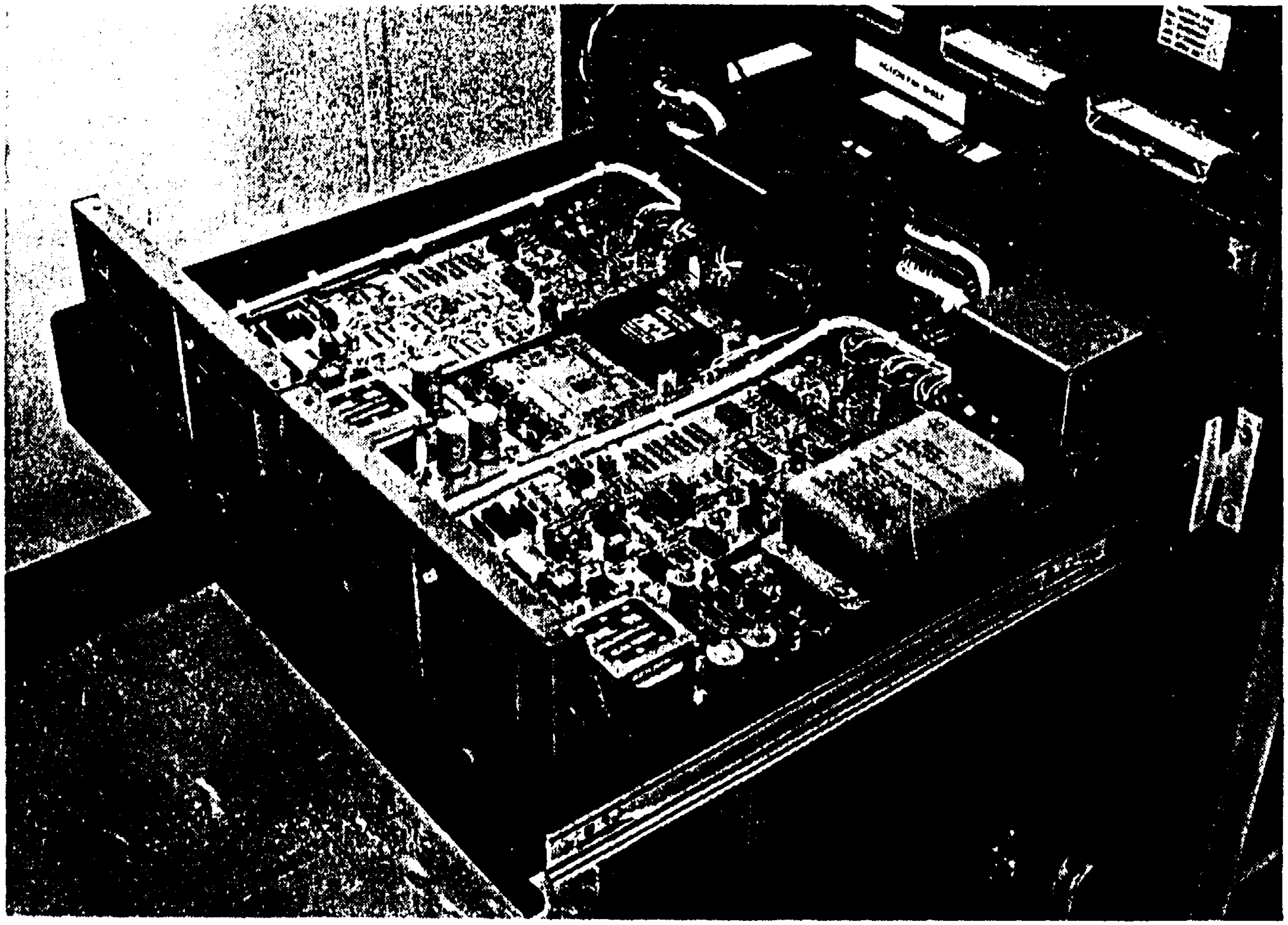


Figure 2-8. Components of Shelf Five in DAC.

IBM7532
the ~~BC-20~~ microcomputer). The Lab Master is a high speed module that is required for data collection in the pulsing mode, and also serves as a D/A converter for the translator input for servo (AUTO mode) operations.

7. Shelf Seven

IBM7532
Shelf seven in the DAC contains the ~~BC-20~~ (IBM PC/AT-compatible) microcomputer. The computer chassis consists of the following modules:

1. A central processing board (CPU) board consists of an Intel CPU board with 512 K on-board RAM and a battery backed real-time clock. It is configured to boot up from the disk drive when power is turned on.

2. A RAM expansion board adds 640 K of RAM. The board includes two RS232 serial input/output ports, and one printer port (not used). Serial port 1 connects to the DIS064 digital input scanner (shelf three) and serial port 2 connects to the NM-1000.

3. A Lab Master high speed analog input/output board, which connects directly to the Lab Master daughter board on shelf six. When the system is in pulse mode, the mother board receives high-speed analog input data from the pulse ion chamber via the NPP-1000 and the daughter board. In the servo configuration, the Lab Master generates a ±5 V analog signal from the digital output from the servo controller software. This signal is used to drive the stepping motor translators.

4. Two IC/network boards (the second network boards, which are high-speed, token-passing, local area network boards that operate at one megabaud over standard IBM communication cables. The network boards handle communication between the DAC and the CSC. The two boards provide system redundancy; the computer automatically switches to the alternate board if the main board fails.

5. A watchdog board which acts as fail-safe hardware capable of shutting down the reactor in case of a computer malfunction. The board controls relay contacts in the SCRAM circuit such that if the DAC loses power or stops operating, the relay will be deenergized and the reactor SCRAMMED. The board has two individual outputs connected to redundant watchdog relays. The board also acts as a monitor of the DAC software by providing four completely independent outputs which constantly check for proper operation of four separate software modules.

6. The digital output module (DOM32) provides up to 32 digital outputs. Most of these outputs are used to control the RLY08 boards on shelves two and three.

7. Two AI016 analog input boards, which receive 16 analog inputs per board. These inputs are received from terminal blocks TB1 and TB2 at the bottom of the DAC cabinet.

2.1.3. Power Monitors and Safety Systems

The power monitors and safety systems on the ICS monitor the power from source level to full power (1200 kW) and the rate of power change (from -30 to +3 sec period) in the steady state modes. The NM-1000 operational channel provides power monitoring, scram and control (in automatic mode) functions for reactor power using an ex-core fission chamber, amplifiers and high voltage power supplies, a dedicated microcomputer, and interfaces with the DAC and CSC. Power is displayed on the high-resolution CRT monitor and is also hard-wired to an LED log power bargraph display. In addition, two independent power monitor and safety systems are provided to monitor the reactor and shut the reactor down (SCRAM) in the event of an overpower condition. These channels utilize GA's NP-1000 or NPP-1000 nuclear instrumentation system, with uncompensated ion chambers (UCIC), and interface with the CSC and DAC. Power is displayed on the high-resolution CRT monitor and is also hard-wired to the LED percent power bargraph displays. The pulse channel monitors the power level up to 2000 MW in the pulse mode. The DAC will collect the pulse channel data via the Lab Master module and transmit the data to the CSC for processing. Each of these channels is explained in more detail below.

2.1.3.1. NM-1000 Operational Channel

The NM-1000 (Figure 2.9) is an industrial neutron monitoring system which is used both in research reactors and in nuclear power plants. It utilizes a fission chamber for the neutron detector, pulse processing electronics and a microcomputer to process instrument readings. Outputs are routed to the DAC and then to the CSC for processing and display. The NM-1000 is contained in two NEMA enclosures, one for the amplifier and pulse processing electronics and the other to house the microprocessor assemblies. The enclosures are mounted on the wall of the reactor room adjacent to the DAC. A functional block diagram of the NM-1000 is shown in Fig. 2-10.

The amplifier assembly contains the following:

1. Modular plug-in type subassemblies for the pulse preamplifier electronics.

2. Bypass filter and RMS electronics for the wide range Campbell amplifier.

3. Signal conditioning circuits.

4. Low voltage power supplies.

5. Detector high-voltage supply.

6. Digital diagnostics and communication electronics.

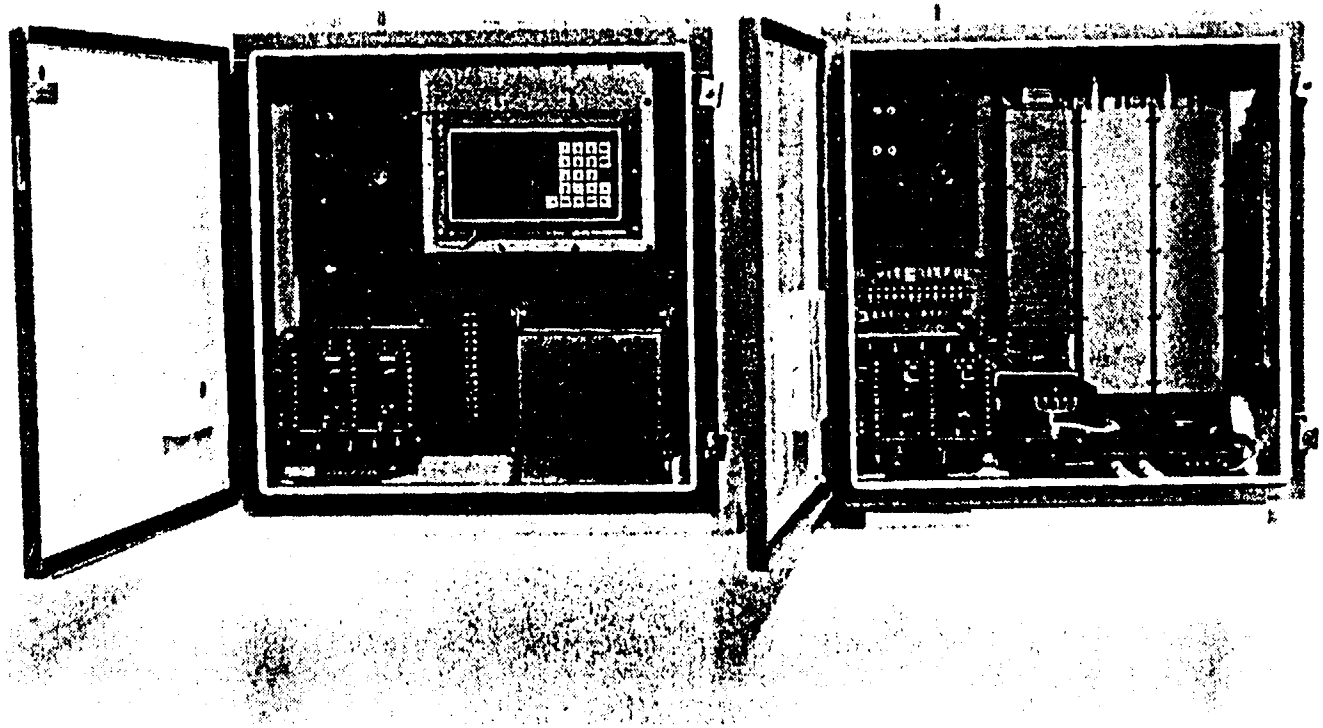
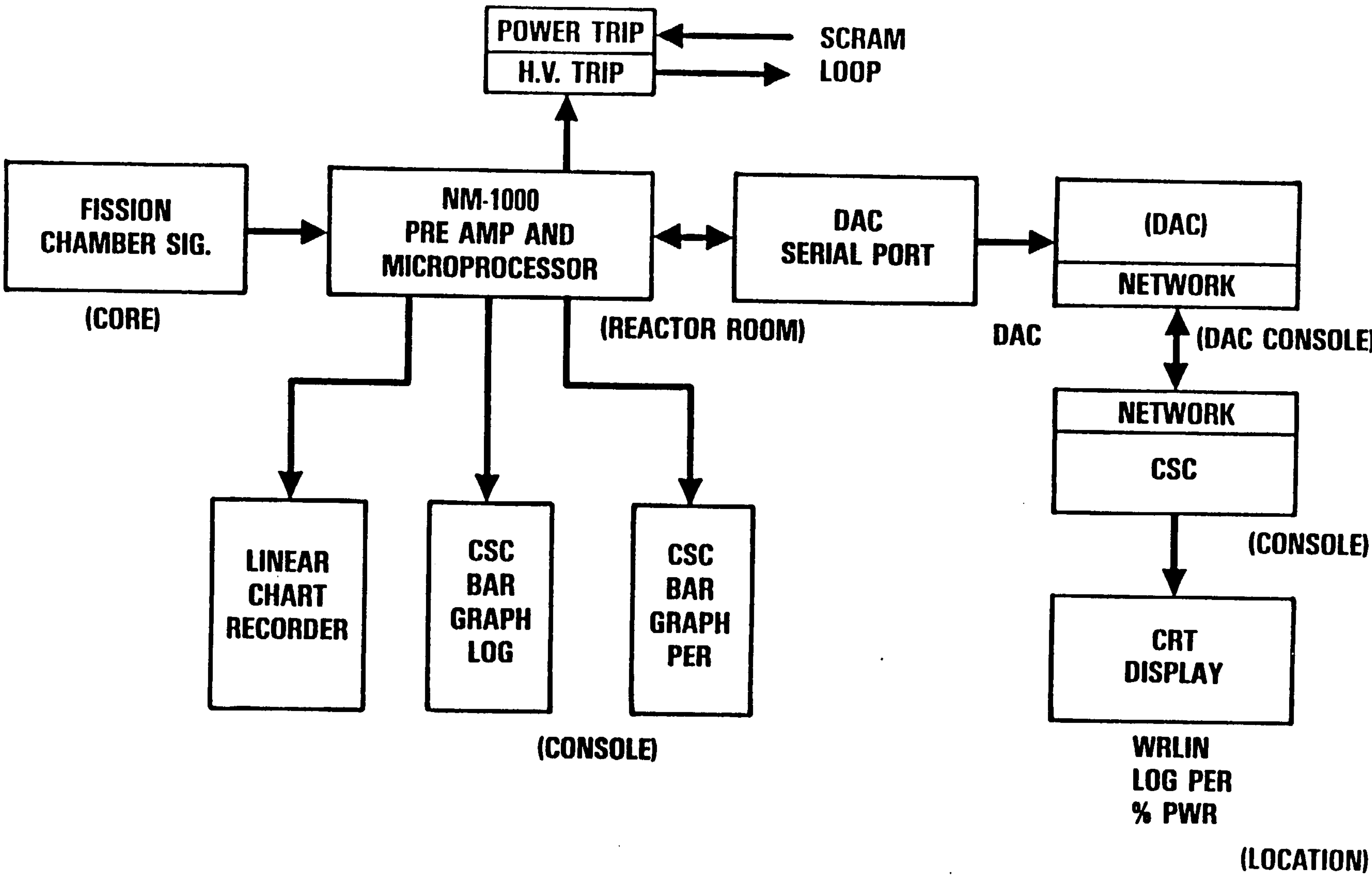


Figure 2-9. NM-1000 Power Monitor

OPERATIONAL POWER MONITOR FUNCTIONAL BLOCK DIAGRAM



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9-7-88

FIGURE 2.10

The processor assembly consists of:

1. Modular plug-in subassemblies for communication electronics between amplifier and processor).

2. Microprocessor.

3. Control/display module.

4. Low voltage power supplies.

5. Isolated 4 to 20 mA outputs.

6. Isolated rod withdrawal prohibit (RWP) outputs.

7. Isolated power/high voltage trip outputs.

8. Period trip contacts.

The NM-1000 as used on the TRIGA uses a standard 1.3 counts/sec/nv encased fission chamber to provide 10 decades of power indication - from shutdown (source) level to full power - hence it is also referred to as a wide-range power monitor. Both log power and linear power readouts are provided; the linear power indication is autoranging. A count rate or circuit is used to monitor power for six decades up from source level; the top four decades are monitored by a Campbelling circuit. When neutron flux levels become high enough so that the detector cannot be operated in the count rate mode (power proportional to the pulses from the detector), the Campbelling technique is used. This technique consists of electronically deriving a signal which is proportional to the root mean square of the inherent <u>current fluctuations</u> present in the fission chamber signal.

The amplifier/processor circuit employs designs which perform automatic on-line self diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system is calibrated and checked (including the testing of RWP trip points) prior to operation during the prestart checks. The checkout data is recorded for future use. The accuracy of the channels is ±3% of full-scale, RWP, and high power trip settings are repeated within 1% of full-scale input.

2.1.3.2. <u>NP-1000 Safety Channel</u>

The NP-1000 power safety channel is a complete linear percent power monitoring system housed within one compact enclosure in the DAC. A functional block diagram of the NP-1000 (safety channel 1) is given in Fig. 2-11. The NP-1000 enclosure contains the following:

1. Current-to-voltage conversion signal conditioning.

2. Power supply trip circuits.

3. Isolation devices.

4. Computer interface circuitry.

The power level trip circuit is hardwired into the SCRAM circuit and the isolated analog outputs are monitored by the CSC as well as being hardwired to a vertical LED percent power bargraph indicator on the right side of the CSC. The system uses an uncompensated ion chamber to detect neutrons.

2.1.3.3. NPP-1000 Safety/Pulse Channel

The NPP-1000 system is identical to the NP-1000, except that a pulse integrator circuit and nv peak circuit have been added to measure peak power and total energy in the pulse mode. The CSC automatically selects the proper gain setting for pulse or steady state mode, depending on the mode selected by the reactor operator. A functional block diagram of the NPP-1000 is given in Fig. 2-12.

2.1.3.4. Fuel Temperature Safety Channels

The fuel temperature safety channels are redundant, fail-safe monitors of fuel temperature and will scram the reactor if a trip limit of 500°C is reached. Both sensors are type K thermocouples resident in instrumented TRIGA fuel elements (Figure 2-13). These voltage inputs are optically isolated and converted to 0 to 20 mA output signals. Each current loop provides three important functions:

1. Input to a trip unit that scrams the reactor if the signal is too high;

2. Input to a vertical LED fuel temperature bargraph located on the reactor control console;

3. Input to the computer via the AI016 No. 2 analog input board.

Figure 2-14 is a block diagram showing the functional aspects of the fuel temperature monitors.

2.1.4 Control Rod Drive Switches and Circuits

To take the reactor to power, it is necessary to withdraw the control rods. This function is performed by using the UP switches on the CSC rod control panel.

The standard (rack-and-pinion) control rod drive circuit consists of the UP/DOWN switches on the CSC, and the UP/DOWN relays in the DAC. Pushbutton switches on the CSC rod control panel are arranged in four

SAFETY CHANNEL POWER MONITOR FUNCTIONAL BLOCK DIAGRAM

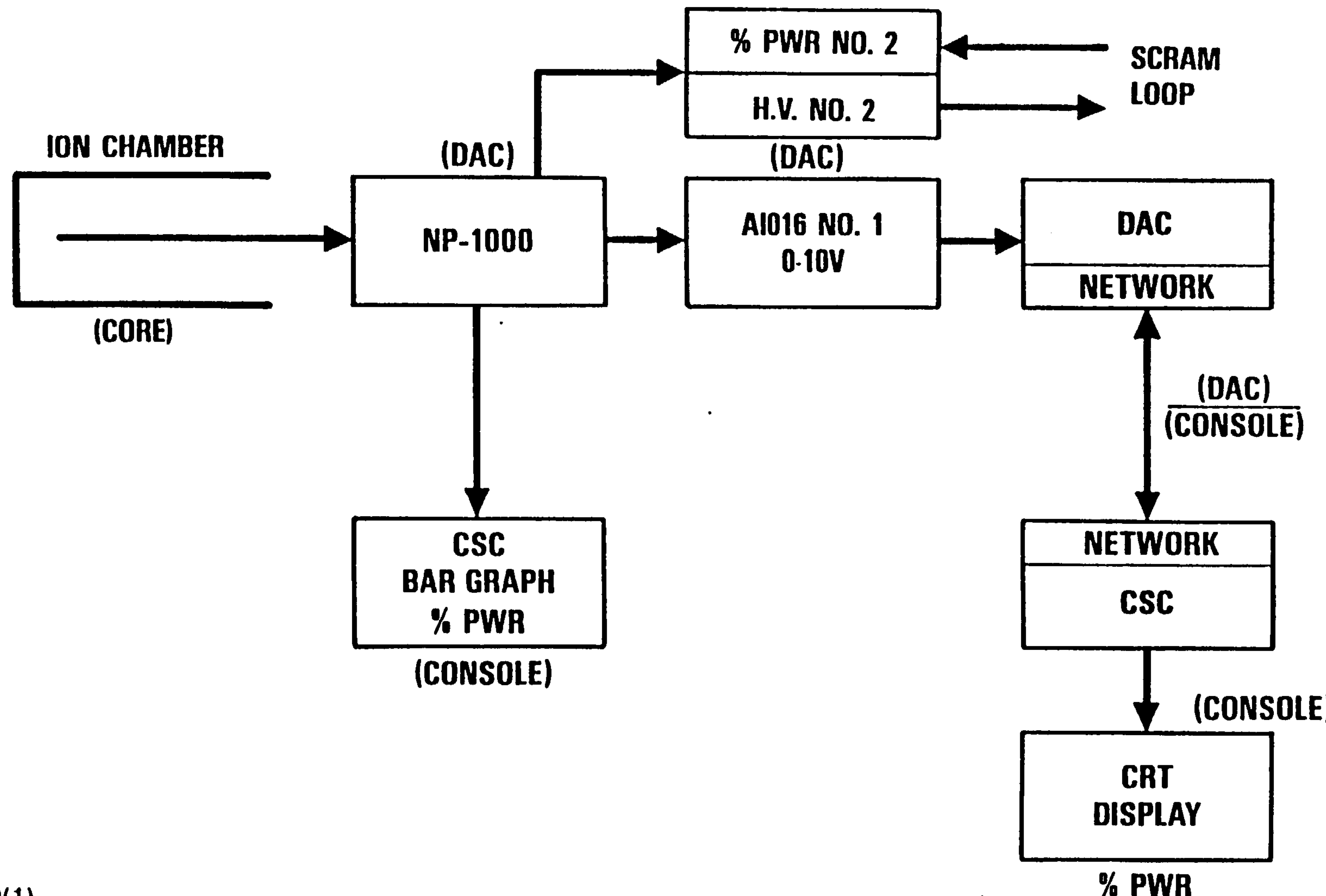

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FIGURE 2.11

SAFETY CHANNEL/PULSE POWER MONITOR FUNCTIONAL BLOCK DIAGRAM

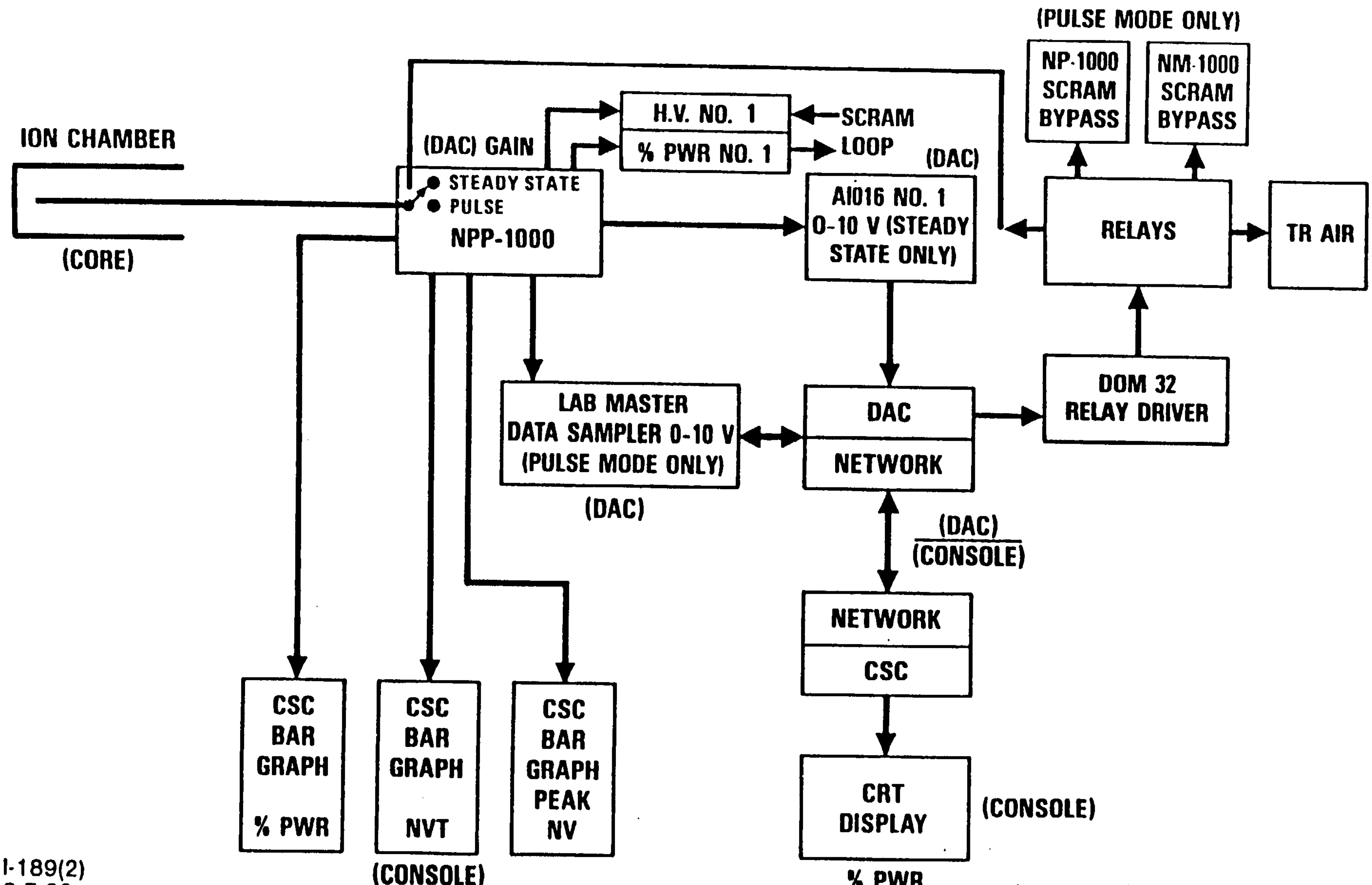

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FIGURE 2.12

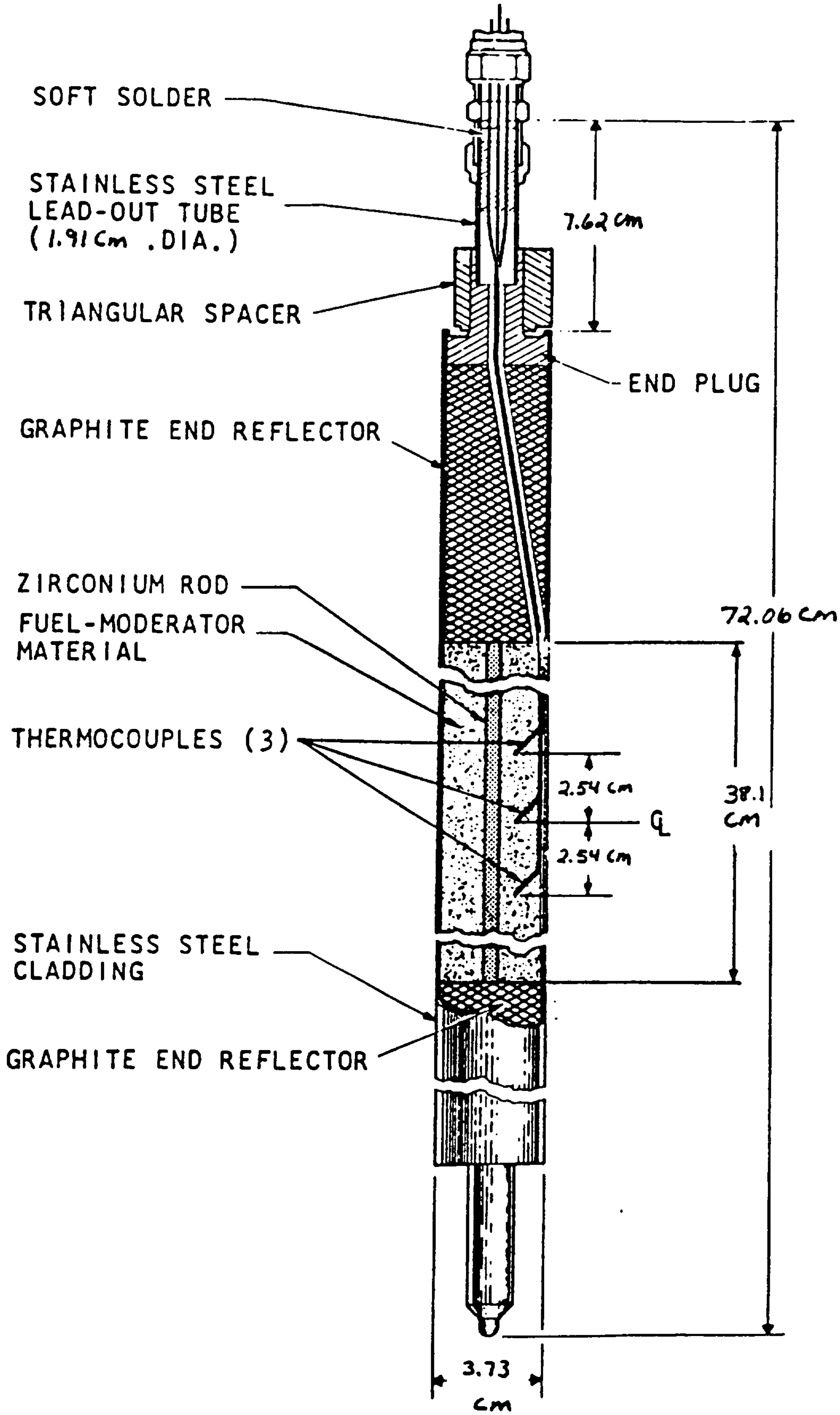

Figure 2-13. Thermocouple Fuel Element

FUEL TEMPERATURE MONITOR
FUNCTIONAL BLOCK DIAGRAM

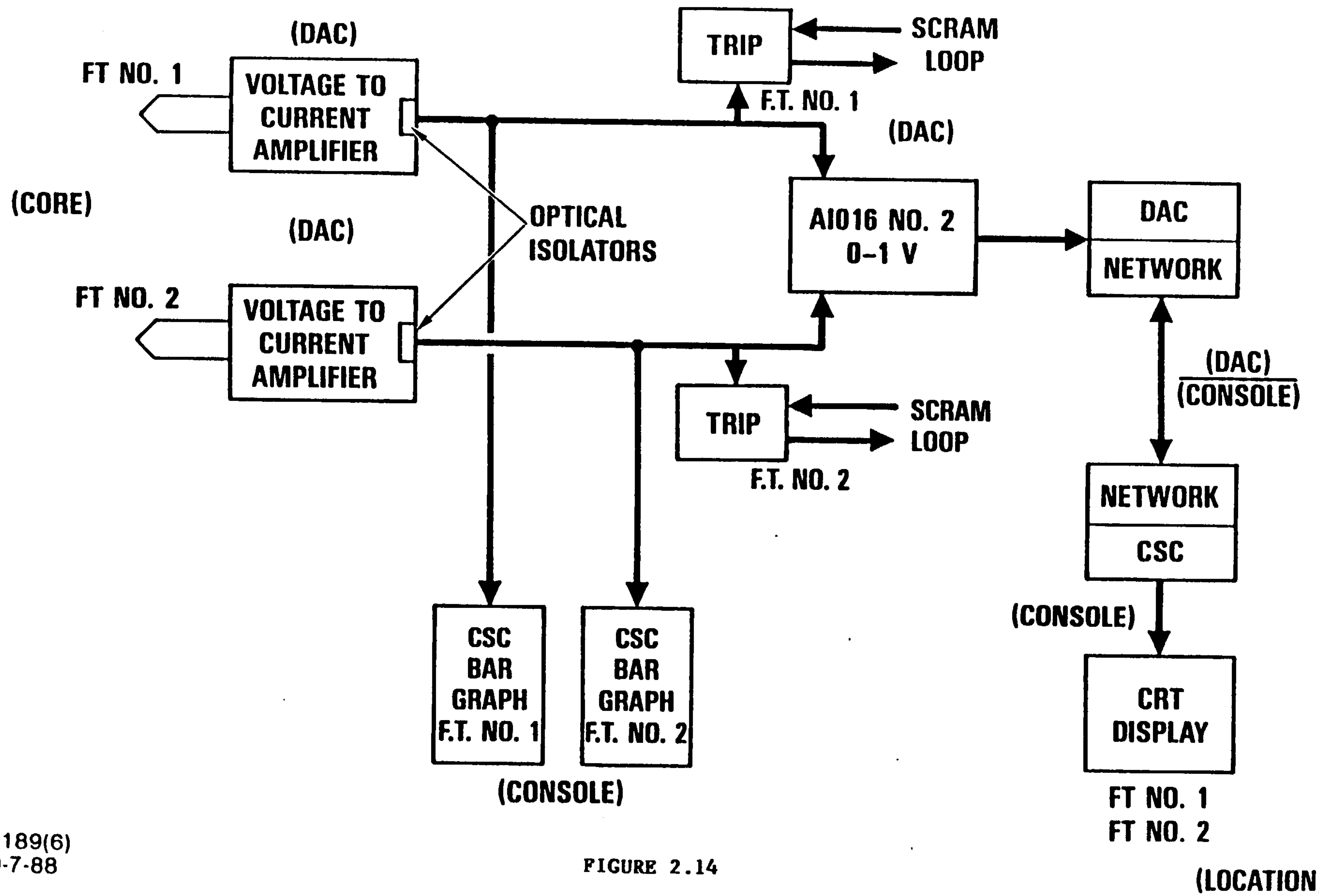

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FIGURE 2.14

vertical columns, one column for each control rod. Each row of switches contains a white DOWN switch, a white UP switch, and a yellow pushbutton MAGNET switch.

Indication of the control rod position on the high resolution CRT is as follows:

GREY control rod. A GREY colored rod indicates that the control rod is at its lower limit.

MAGENTA control rod. A MAGENTA rod indicates that the control rod and rod drive are at its upper limit.

GREEN control rod. A GREEN control rod indicates that the rod and drive are between the upper and lower limits.

MAGNET box. The YELLOW color in the box between each control rod and rod drive graphic on the high resolution CRT indicates that the electromagnet power is ON. The magnet power circuit is energized by a key switch located on the left side of the control panel. The absence of the yellow color, or a BLACK indication, signifies the absence of magnet current.

When the MAGNET pushbuttons are depressed, magnet current is interrupted and the YELLOW color in the MAGNET box will be eliminated. If a control rod is above the down limit, the rod falls back into the core. Releasing the button closes the magnet circuit and magnet current is reset.

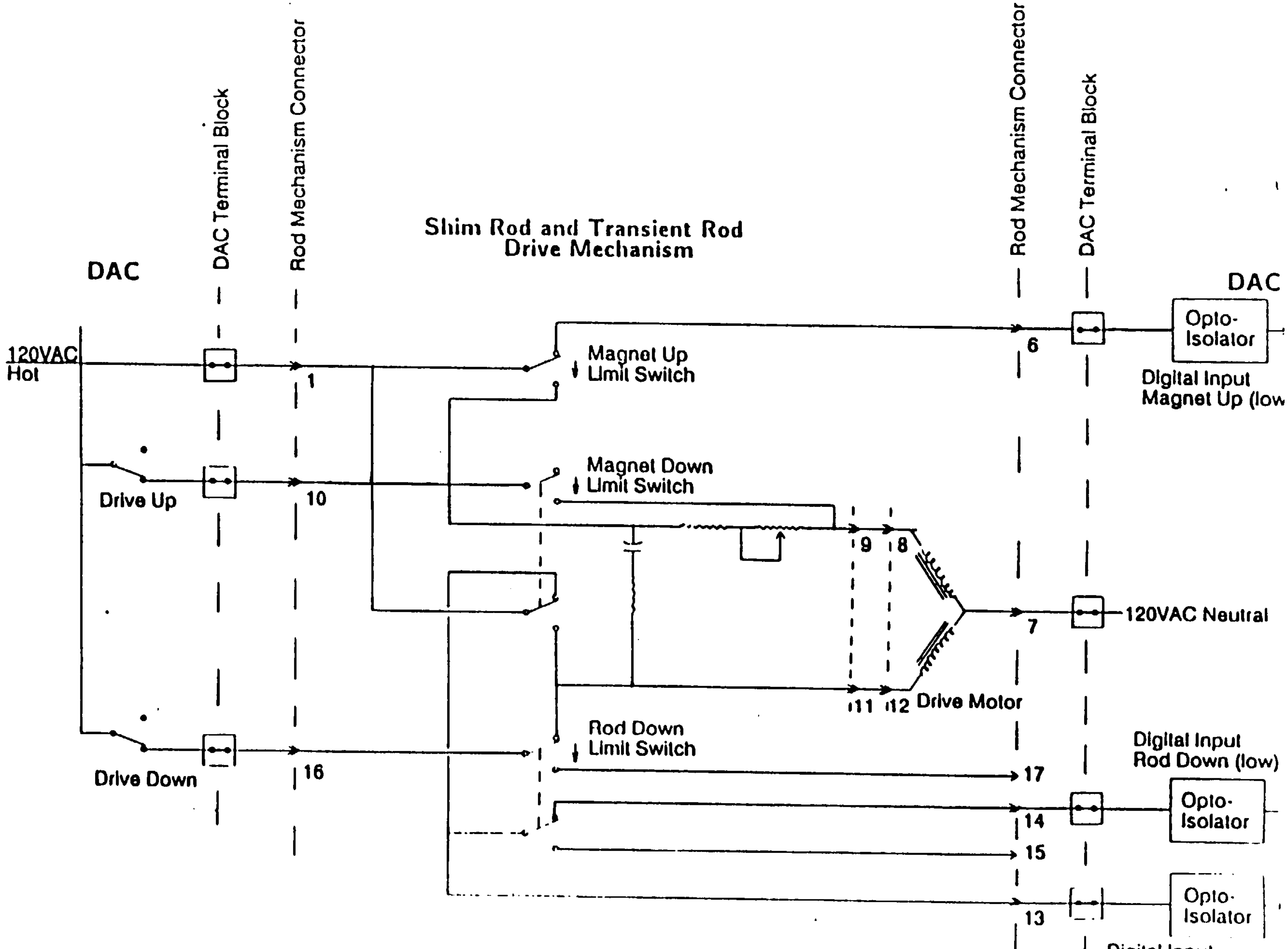
The UP/DOWN pushbutton drive relays in the DAC which remove the short circuit across a motor phase-shifting capacitor (Figure 2-15) and apply power to the motor in the corect phase to drive it either up or down.

Software interlocks in the UP pushbutton circuit make this circuit inoperative under certain conditions. A minimum source interlock relay in the NM-1000 prohibits rod withdrawal in the absence of a set minimum source level count. Also, any time the console is in transient mode of operation, the standard motor driven rods cannot be raised.

Second, withdrawal of more than one rod at a time is prevented. Software interlocks inhibit all rod UP drive relays in the DAC anytime two or more rod UP pushbuttons are pressed on the CSC rod control panel. All UP buttons must be released before any rod withdrawal is permitted.

Table 2.2 lists some possible combinations of control rod indicators and the significance of each combination.

In automatic and square wave modes, the regulating rod drive is removed from manual control. Software control is then exercised over rod motion using a PID algorithm (section 2.1.5), to drive the rod either up or down based on a comparison of the reactor power with the power demand set by means of the thumbwheel switches on the CSC mode control panel.



Shim Rod and Transient Rod Drive Mechanism
DAC
DAC Terminal Block
Rod Mechanism Connector
Rod Mechanism Connector
DAC Terminal Block
DAC
120VAC Hot
Drive Up
Drive Down
Magnet Up Limit Switch
Magnet Down Limit Switch
Rod Down Limit Switch
Drive Motor
1
10
16
6
9
8
7
11
12
17
14
15
13
Opto-Isolator
Digital Input Magnet Up (low
120VAC Neutral
Digital Input Rod Down (low)
Opto-Isolator
Opto-Isolator
Digital Input Magnet Down (low)

FIGURE 2.15

Table 2.2

CONTROL ROD AND DRIVE SWITCH COMBINATIONS

INDICATION: ROD COLOR	INDICATION: MAGNET BOX	CONTROL ROD AND DRIVE	SIGNIFICANCE
NORMAL CONDITIONS			
MAGENTA	YELLOW	Rod and drive completely withdrawn, magnet making contact	Rod insertion permissible
GREY	YELLOW	Rod and drive at their lower limits, magnet making contact	Rod withdrawal permissible
GREEN	YELLOW	Rod and drive off either limit, magnet making contact	Either rod insertion or withdrawal permissible
GREY	BLACK	After scram, control rod down	No rod withdrawal permitted until rod drive has reached its lower limit
ABNORMAL CONDITIONS			
GREEN	BLACK	Drive between limits, rod down, no magnet contact or magnet current	Rod-down switch not functioning properly
MAGENTA	BLACK	Drive completely up, rod down, no magnet contact or magnet current	Rod-down switch not functioning properly
GREEN	YELLOW	After dropping individual rod (scram) normal operation would give GREY, YELLOW	Rod has stuck above lower limit, or rod-down switch not operating properly

2.1.5. Servo Controller Description

The servo controller, in the automatic mode, controls the reactor power automatically to the demand power level selected by the operator. Thumb-wheel switches are provided on the mode control panel for the desired power selection. The servo controller will track and stabilize reactor power through the utilization of a proportional-integral-derivative (PID) algorithm. The servo controller system utilizes the latest digital computer technology coupled with extensively developed software. This is in comparison to previous TRIGA consoles, which used an analog computer control rod motion in the servo mode.

Reactor power level and changes are measured by an analog/digital input from the NM-1000 channel. The PID algorithm in the DAC then responds to this input by comparing it to the operator set demand power level on the CSC, by movement of the regulating control rod, which is powered by a precise translator/stepping motor drive. This drive will be driven up or down automatically to control the power level to the demand power level setting.

The drive mechanism for the servo control rod drive is an electric stepping-motor-actuated linear drive equipped with a magnetic coupler and rod position potentiometer.

2.1.6. SCRAM Circuit

The SCRAM circuit is shown in Fig. 2-16. Most of the scram circuit components are located in the DAC. **The circuit is completely hardwired and does not in any way depend on the computer in the console (CSC) or in the DAC, nor on any software.** Additionally, watchdog timer circuits in both the CSC and DAC would initiate a scram should there be a failure in any one of four control software modules. If not reset properly by any one of four software modules, the watchdog relays will deenergize, opening the SCRAM circuit and cause a reactor SCRAM.

2.1.6.1. Monitoring Components

The SCRAM circuit is monitored for two conditions: power supply output voltage and shorts to ground. The voltage output is monitored by an Action Pak unit (see Table 2-1). Insulation from the chassis ground is monitored by another Action Pak serving as a ground fault detector, which detects and reports shorts to ground (the entire scram circuit is isolated from ground). In each case, a relay contact closure output is provided to the DAC computer. A fault in either case will be indicated on the High Resolution CRT and Reactor Status CRT.

2.1.6.2. Control Inputs

The following are control inputs to the SCRAM circuit in the DAC:

1. **The NP-1000 and NPP-1000 Safety Channels.** The reactor power is monitored by these two channels using two independent ion chambers (UCIC). If the reactor power exceeds the trip

Notes:

1. The NM1000 is not at this time a safety channel. The NM1000 scrams are shown for illustration. If applicable these scrams supplement but do not replace the safety channels.
2. A low pool level scram is in each side of the circuit.
3. External scrams are for future use.

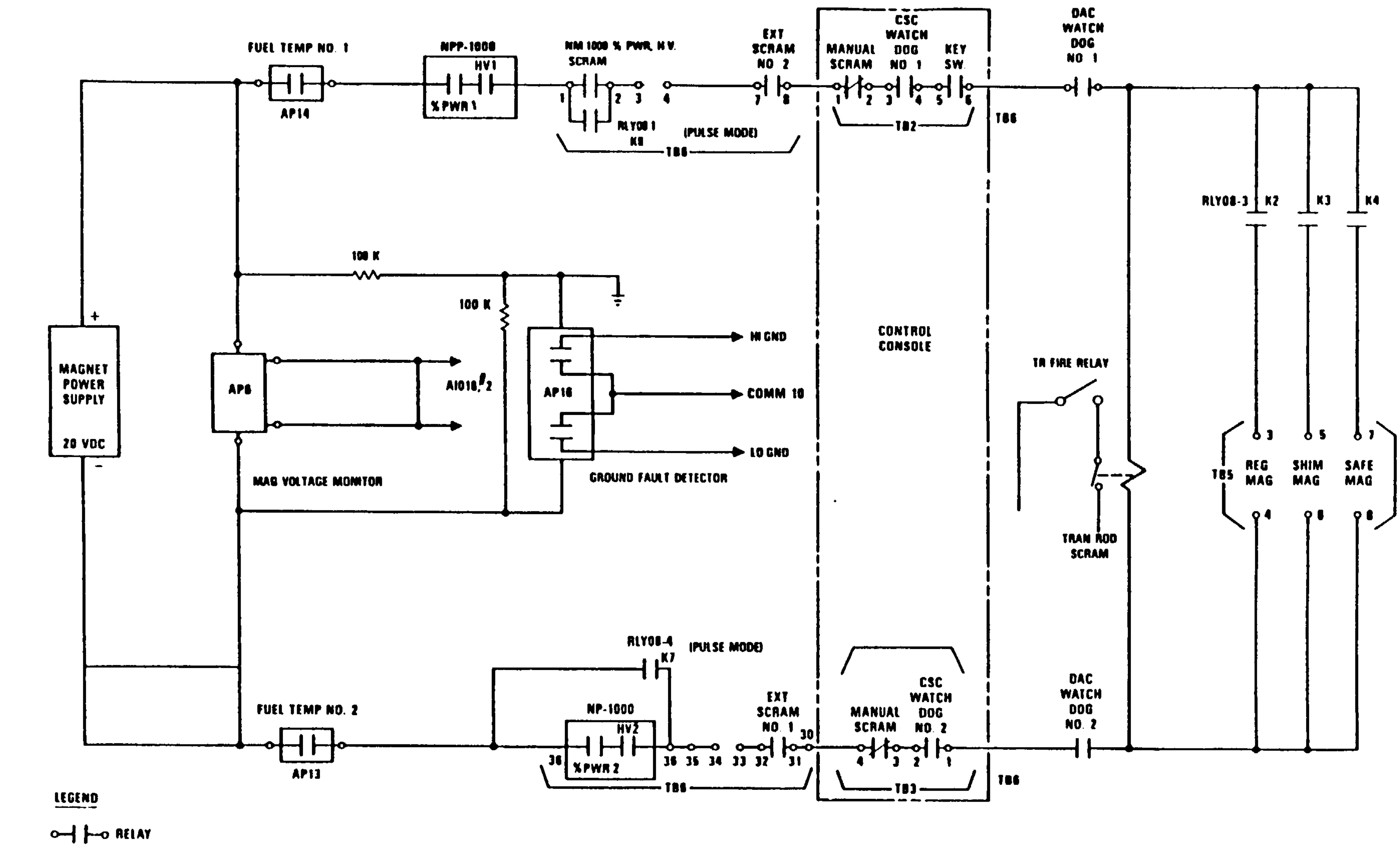


Figure 2-16. Hardwired Scram Circuit for ICS.

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setpoint (~~275~~ 1.050 kW or ~~110%~~ 105 of licensed power), the unit will open its relay contact in the hardwired scram circuit, close its contact to the DAC computer digital input, scramming the reactor. A SCRAM message will be displayed on the CSC.

Also installed in the NP and NPP enclosures are high voltage
Figure 2-16. Hardwired Scram Circuit for ICS.
power supplies for the associated detectors. Trip circuits in each of these devices monitor for a decrease in high voltage. At approximately a 20% decrease this trip will scram the reactor. 10%

2. **Fuel Temperature.** The reactor fuel temperature is monitored using two type K (chromel/alumel) thermocouple inputs. Each thermocouple input is conditioned by an Action Pak module, and monitors for high level alarm limits. If a temperature channel exceeds the limit setpoint or the thermocouple shorts, the unit will open its scram relay contact in the SCRAM circuit and close its contact to the DAC computer digital input, scramming the reactor. A SCRAM message will be displayed on the CSC.

3. **External Scrams.** There is one set of external SCRAM contacts in both the supply and return trains. These contacts are used for any external SCRAMs that may be required e.g., Two sets of contacts are required for each external scram wired into the scram circuit, one for the hardwired relays, and a second as a DIS064 input for scram indication on the CSC.

4. **CSC Manual SCRAM Switch.** The SCRAM circuit is connected to the CSC and the DAC. The RED SCRAM push button on the CSC has three sets of contacts. One set is in the supply train and one set is in the return train. A third set provides digital input to the csc computer. Pressing the scram button opens both sets of contacts in the scram circuit and closes the contacts to the CSC computer digital input.

5. **CSC Watchdog.** The CSC computer includes a watchdog timer board. The watchdog board has four independent timers, all of which must be continuously retriggered at least every 10 seconds by the operating software modules to prevent timeout.

   If a monitored software module fails or hangs up, the associated timer will timeout and both watchdog relays will be deenergized, opening their contacts in the SCRAM circuit. Also, if the CSC computer recognizes a SCRAM condition, either internally or externally, it will command the watchdog to reset and stop all timers. This causes an immediate SCRAM.

6. **CSC Magnet Power Key Switch.** The key switch on the CSC provides a secure method of preventing inadvertent startup of the

reactor. It has three positions: "OFF" interrupts the magnet power and forces a SCRAM condition; "ON" enables magnet power; and "RESET" enables magnet power and signals the CSC to reset from a SCRAM condition. The switch returns to the "ON" position when released. As the switch is moved from one position to another, the switch position is signaled via digital inputs to the CSC. Switching from "ON" to "RESET" will also scram the reactor should the reactor be operating.

7. **DAC Watchdog.** A watchdog board is included in the DAC computer also. The DAC watchdog is identical to the CSC watchdog and operates in the same manner.

2.1.6.3. Operating Outputs

Outputs controlled by the SCRAM circuit include the electromagnets for the regulating, shim I, and shim II rods, and the solenoid controlling the air supply to the transient rod. If the voltage from the power supply is adequate and all of the input control conditions are normal, the SCRAM interlock relay in the transient rod solenoid circuit will be energized and the solenoid control enabled.

Voltage is also supplied to the normal open contacts of the magnet control relays. These relays are controlled by the DAC computer as directed by the CSC computer. A closed contact will energize the associated electromagnet. If the rod is in contact with the magnet, it will follow the motion of the drive mechanism.

If, at any time, one or more of the input control conditions become abnormal, its SCRAM contacts will open. This will remove current from all rod electromagnets and deenergize the solenoid, thereby dropping all control rods into the core. The SCRAM condition must be cleared and reset before operations can be resumed.

2.1.7. Modes of Operation

The TRIGA ICS permits three standard operating modes: manual, automatic, and pulse.

The manual and automatic reactor control modes are used for reactor operations from source level to 100% of licensed power. These two modes are used for manual reactor startup, change in power level, and steady state operation. The pulse mode generates high-power levels for very short periods of time.

Manual rod control (Figure 2-17) is accomplished through the use of push buttons on the rod control panel. The top row of push buttons (magnet) is used to interrupt the current to the rod drive magnet. If the rod is above the down limit, the rod will fall back into the core and the magnet will automatically drive to the down limit, where it again contacts the armature.

The middle row of push button (UP) and the bottom row (DOWN) are used to position the control rods. Depressing these push buttons causes the control rod to move in the direction indicated. Several interlocks prevent the movement of the rods in the up direction under conditions such as the following:

1. Scrams not reset.

2. Magnet not coupled to armature.

3. Source level below minimum count. 1.2cps

4. Two UP switches depressed at the same time.

5. Mode switch in the pulse position.

6. Magnet current not enabled.

There is no interlock inhibiting the DOWN direction of the control rods.

Automatic (servo) power control can be obtained by switching from manual operation to automatic operation. All the instrumentation, safety, and interlock circuitry described above applies and is in operation in this mode. However, the servoed rod (regulating rod) is controlled automatically in response to the power level and period signal. The reactor power level is compared with the demand level set by the operator and is used to bring the reactor power to the demand level on a fixed preset period of 6.5 seconds. The purpose of this feature is to automatically maintain the preset power level during long-term power runs. Figure 2-18 is a functional block diagram of the auto-mode operation of the reactor.

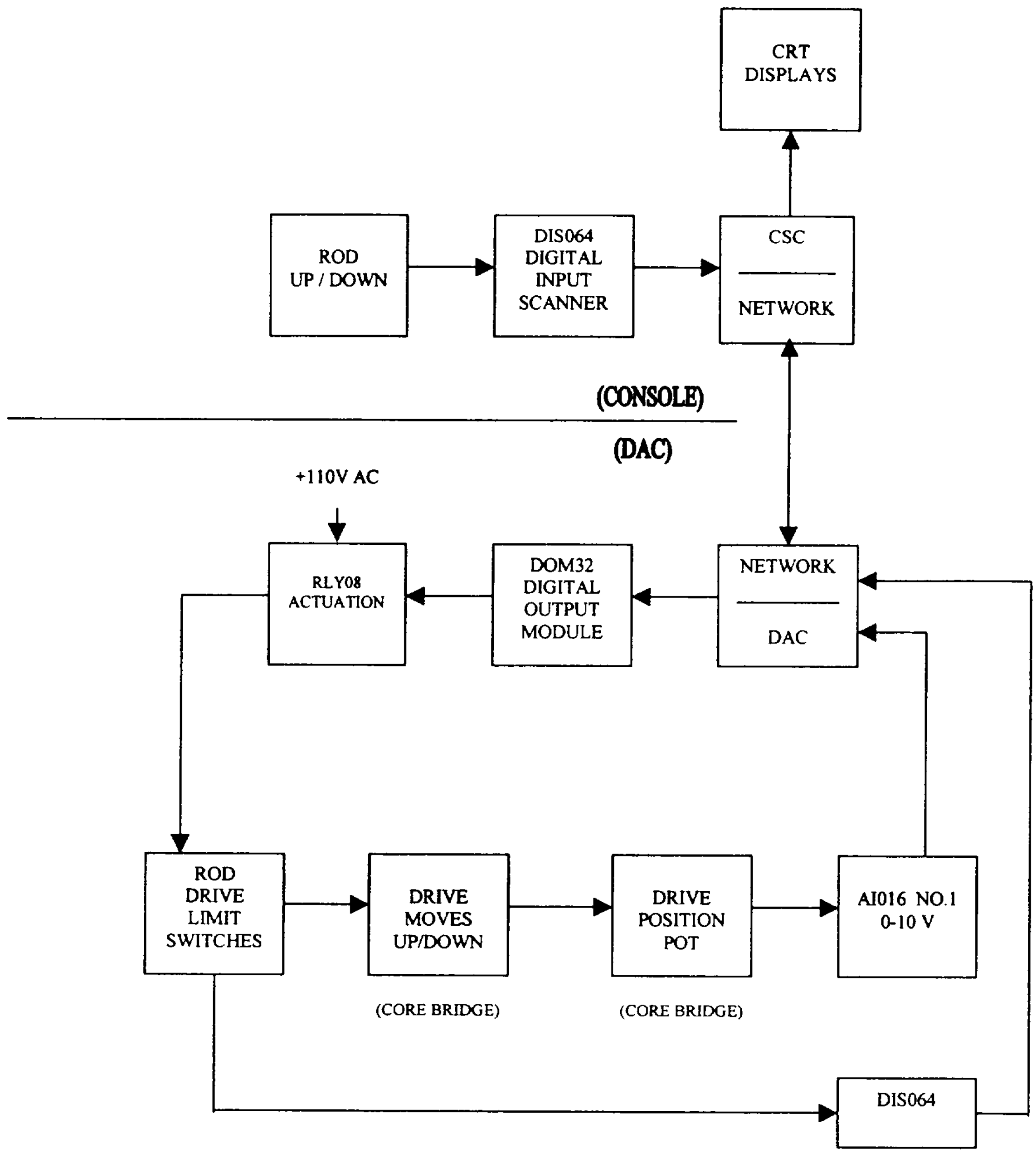
Reactor control in the pulsing mode (Figure 2-19) consists of establishing criticality at a power level below 1 kW in the steady state mode. This is accomplished by the use of the standard motor driven control rods, leaving the transient rod fully inserted. Selection of pulse mode on the CSC mode control panel causes several instrumentation changes in the DAC. First, both the NP-1000 and NM-1000 scrams are bypassed. Second, the gain of the NPP-1000 is changed to 2000~~0~~ MW full scale. Third, the nvt and nv peak circuits are enabled in the NPP-1000.

2.1.8 Auxiliary Monitoring Channels

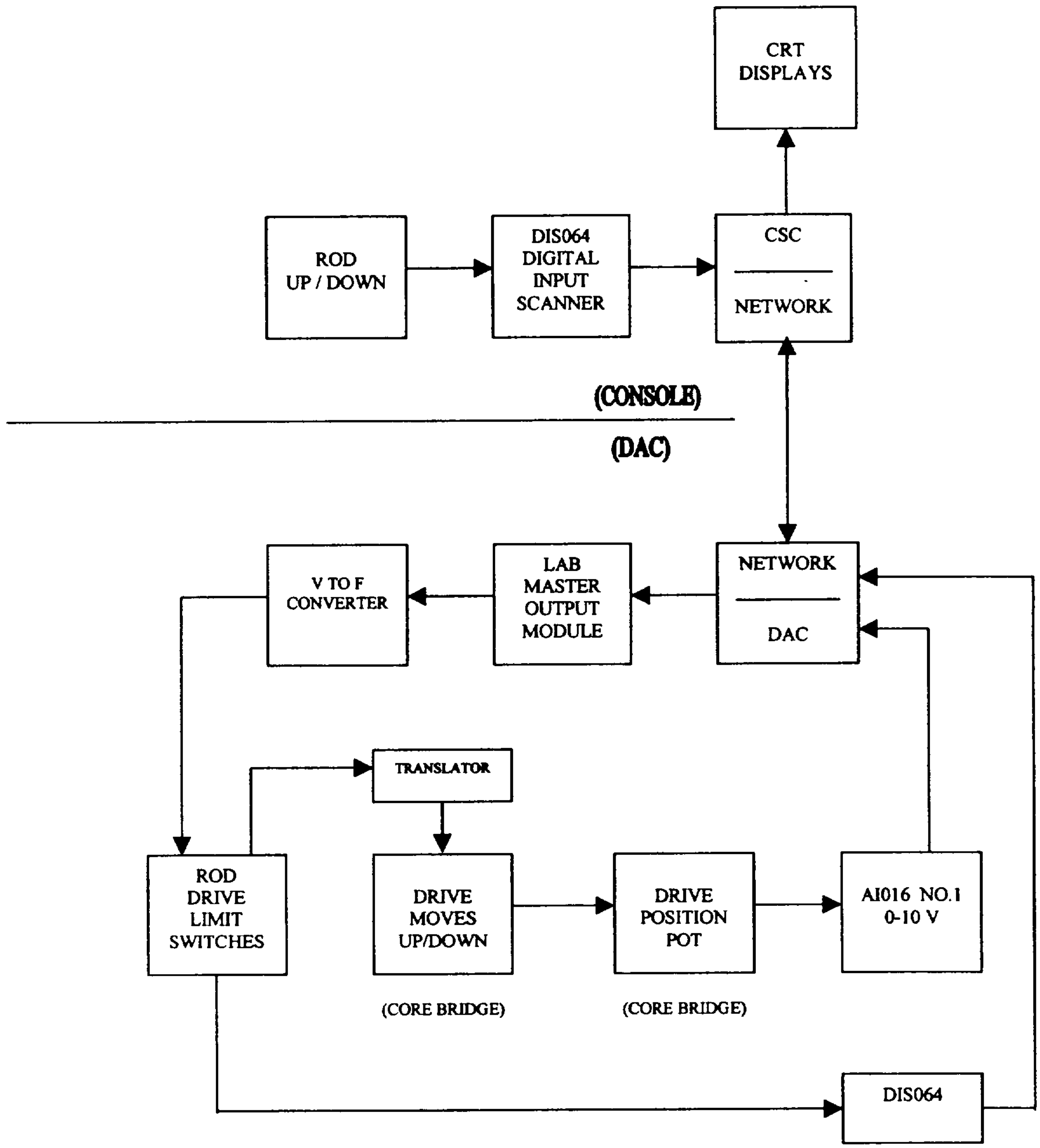
2.1.8.1 Temperature Monitoring Channel

The TRIGA coolant temperature is measured with a probe in the reactor tank and indicated on both CRT displays on the CSC. A platinum resistance temperature detector (RTD) is used as the temperature sensing element (Figure 2-20). The sensor is connected to an Action Pak located in the DAC.

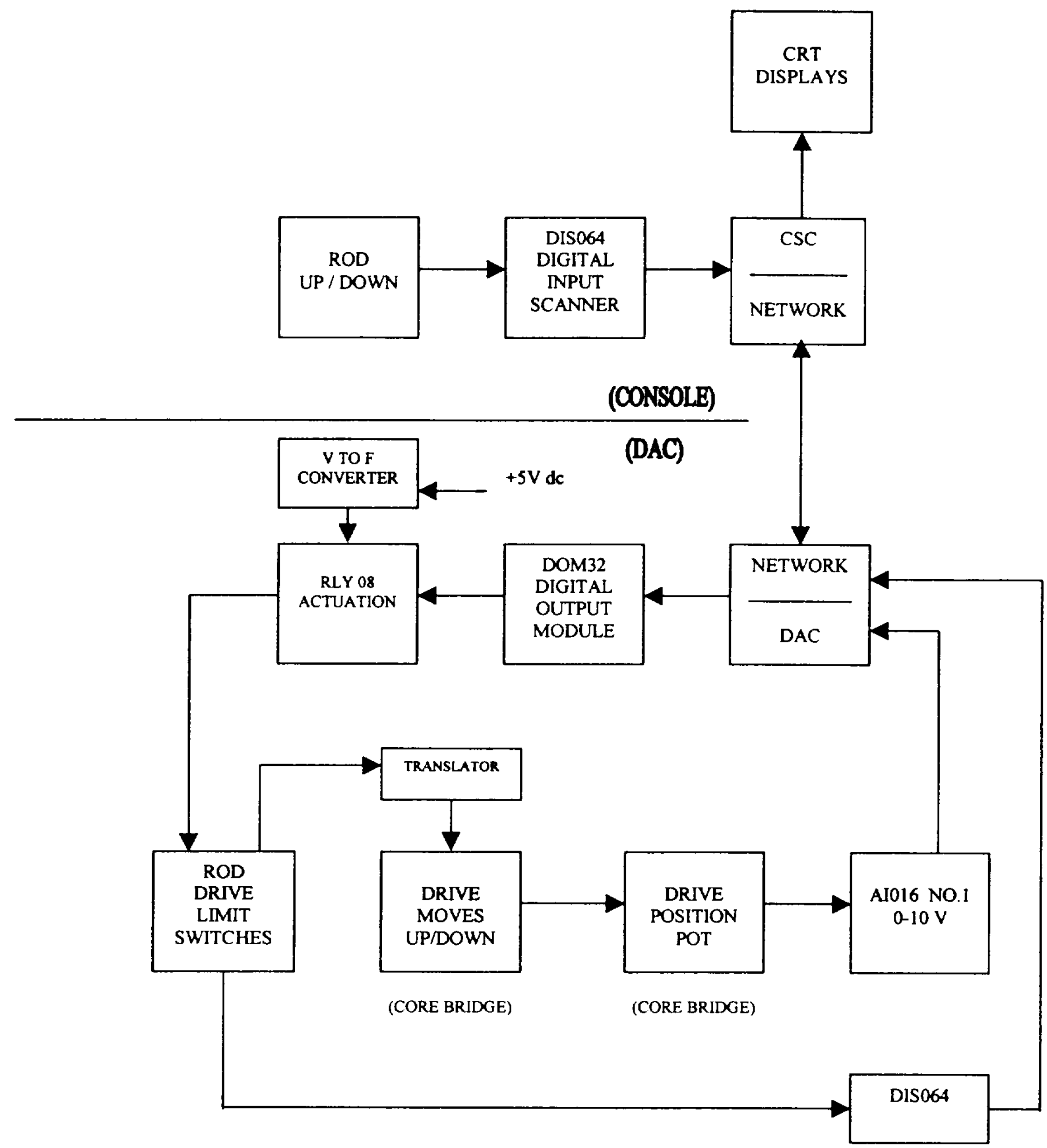
TRANSIENT ROD
MANUAL MODE FUNCTIONAL BLOCK DIAGRAM



REGULATING ROD
MANUAL MODE FUNCTIONAL BLOCK DIAGRAM



SHIM I & II CONTROL RODS
MANUAL MODE FUNCTIONAL BLOCK DIAGRAM



GENERAL ATOMICS

STEADY-STATE STANDARD AND TRANS ROD DRIVE FUNCTIONAL BLOCK DIAGRAM

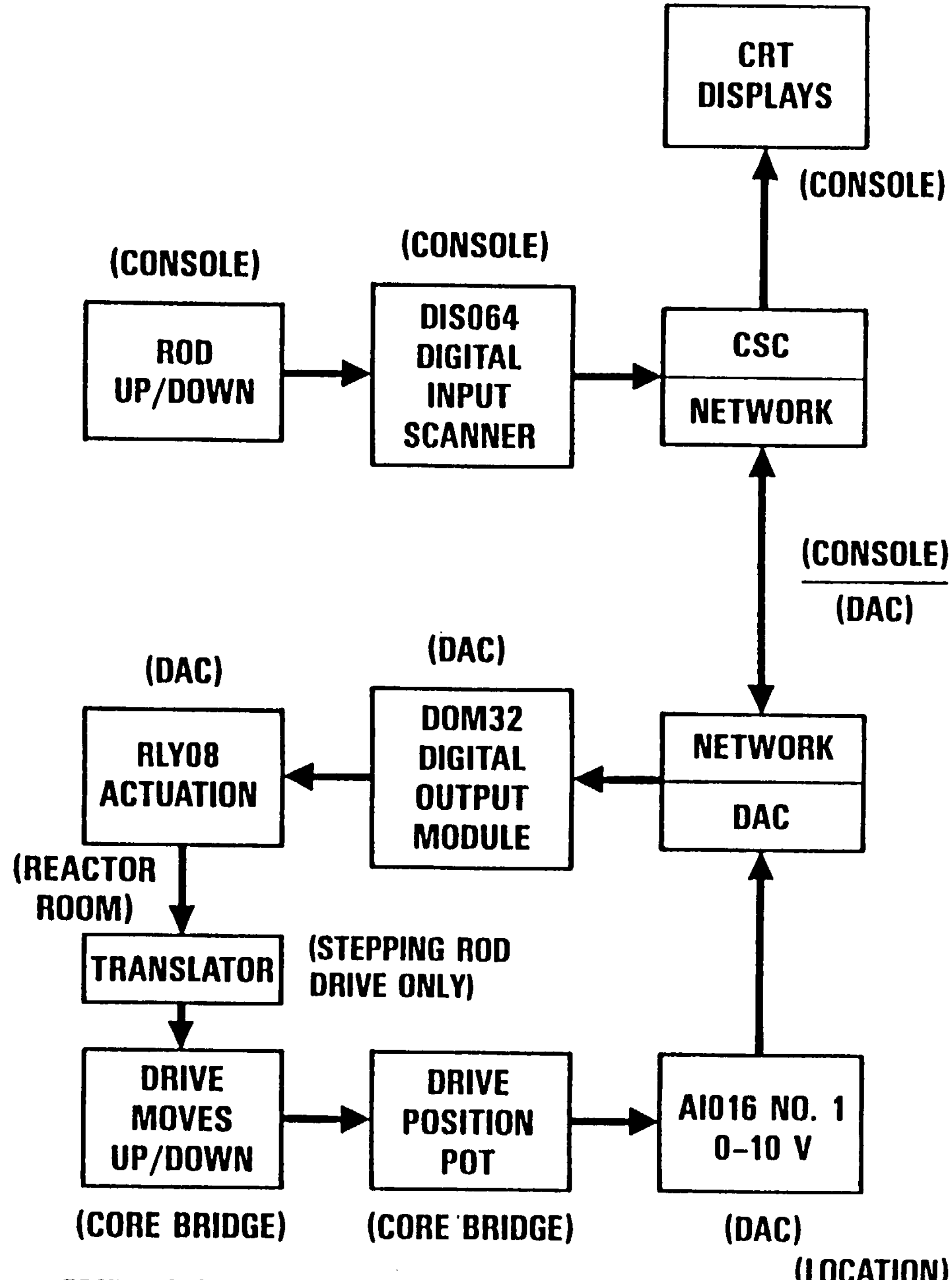


FIGURE 2-17

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GENERAL ATOMICS

AUTO MODE FUNCTIONAL BLOCK DIAGRAM

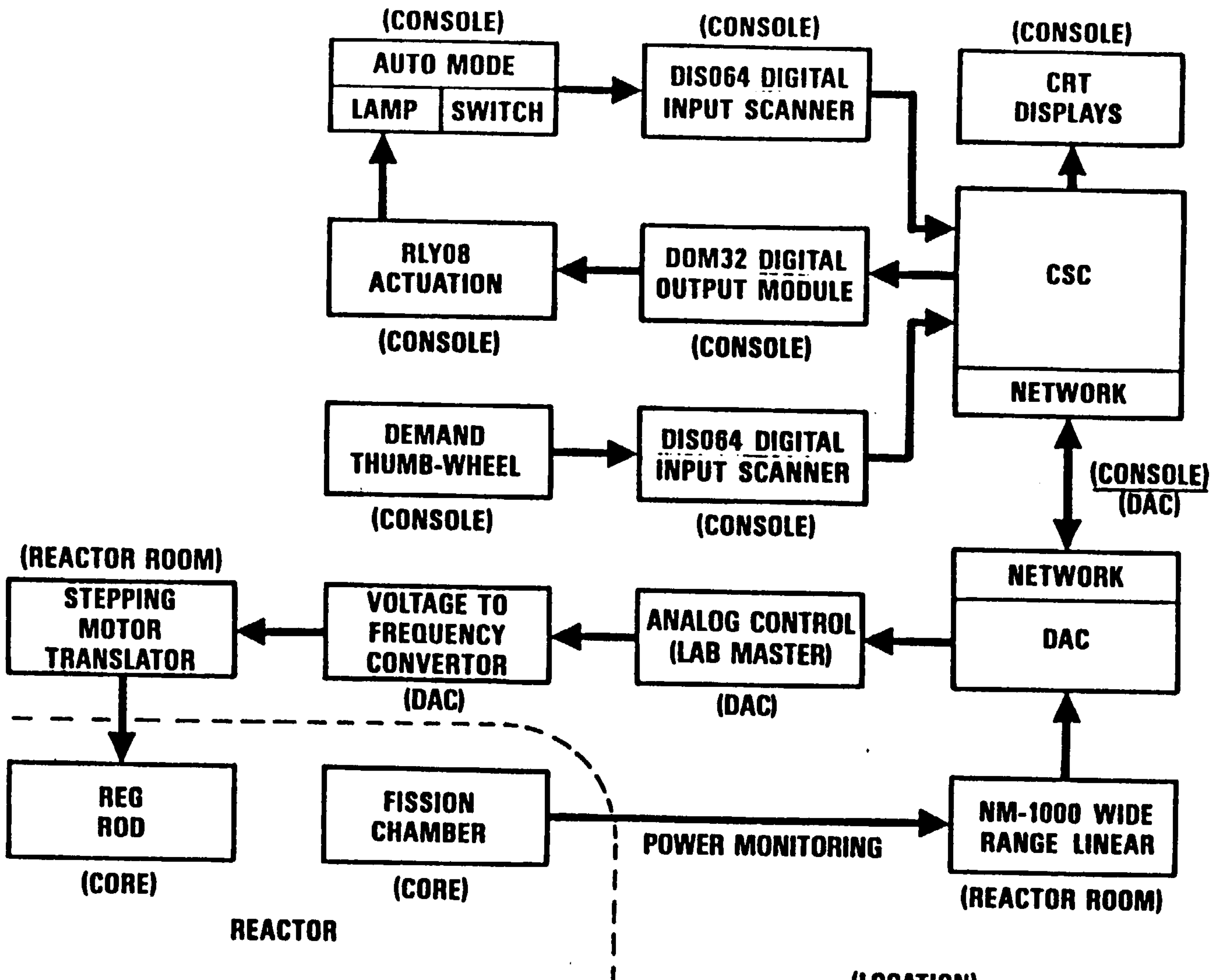

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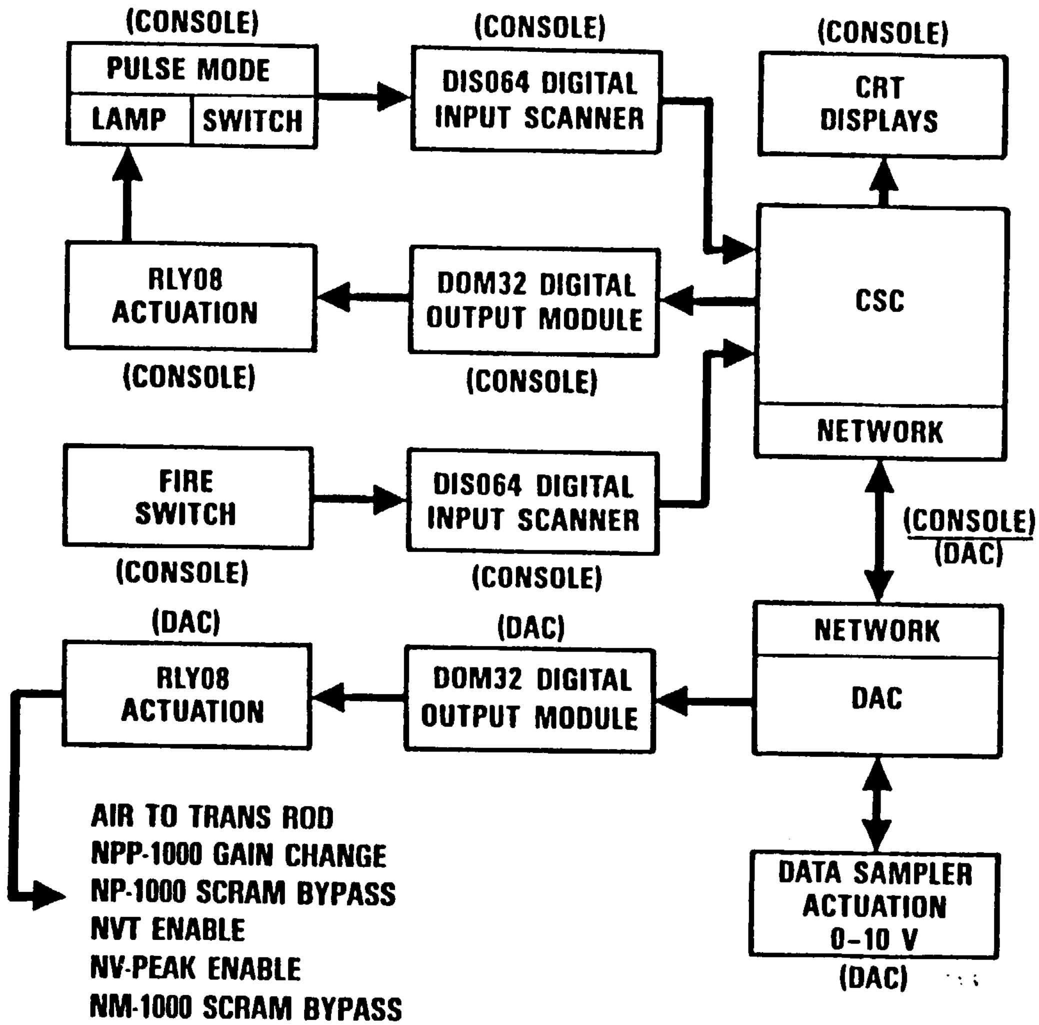
FIGURE 2-18

PULSE MODE FUNCTIONAL BLOCK DIAGRAM



(LOCATION)

FIGURE 2-19

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A second temperature probe used is a thermistor type inserted in the reactor tank. This temperature is read out directly on the console.

1.8.2 Area Radiation Monitor

a ?

Geiger-Muller (GM) radiation monitors are located at numerous points in the vicinity of the reactor shield. Monitored locations include areas near beam tubes, the pool surface, the reactor control room, and one moveable unit is available for use as conditions warrent. The units consist of a GM probe, a local alarm/readout near the probe, and a remote alarm/readout in the reactor control room on the auxilary panel. An area radiation level is also displayed on the status window of the CSC.

1.8.3 Conductivity Monitoring Channel

The electrical conductivity of the TRIGA reactor tank water, i.e., the ability of the water to carry an electric current is measured using a titanium-electrode conductivity cell, a conductivity probe selector switch, and a Wheatstone bridge circuit coupled to a tunable readout. Conductivity probes are installed in the purification system loop before and after the resin tank alloing monitoring of both reactor tank water and resin function. A local multi-sensor selector switch allows selection of either probe for readout on the monitor. The monitor consists of a tuning dial and two LED's which illuminate to indicate if the conductivity is above or below the dial setting. The conductivity probes may also be switched to provide a signal to the DAC for remote conductivity display on the CSC status window.

1.8.4 Coolant System

Numerous coolant system parameter are monitored to provide sufficient information to evaluate system performance. All parameters (except primary coolant flow rate) are displayed locally near the measurement point. Key parameters, including:

a. Differential pressure (primary vs. secondary)
b. Primary and secondary coolant flow
c. Primary coolant heat exchanger supply and discharge temperature
d. Pool water level

also provide signals to the DAC for remote display on the CSC status window in the control room.

Differential pressure is monitored by a differential pressure monitor. The unit consists of a local display, an adjustable alarm setpoint, and an alarm contact closure line wired to the DAC. The alarm setpoint is adjusted to provide an alarm whenever the pressure on the secondary (chilled water) side approaches or falls below the pressure on the primary (pool water) side. A normal positive pressure differential is maintained to prevent potentially radioactive pool water from leaking into the site chilled water system in the event of a heat exchange tube failure. Numerous pressure gauges installed in both the primary and secondary flow path in the coolant treatment room provide additional system performance information.

The primary coolant flow sensor consists of a probe type flow measuring element penetrating the coolant pipe. The secondary flow sensor consists of a venturie. Differential pressure signals from both sensors are connected to differential pressure transmitters which provide a 4-20 ma signal to the DAC for display in the CSC status window. A local flow gauge also indicates secondary coolant flowrate in the coolant treatment room.

Two resistance temperature detectors (RTD's) are mounted in stainless steel thermal wells in the primary coolant flow loop upstream and downstream of the heat exchanger. Lead wires from the RTD's are fed to the DAC for temperature display on the CSC status window. Numerous additional thermometers mounted in both the primary and secondary loops provide additional system performance information.

The pool level sensor consists of a stainless steel float type system suspended directly into the pool. Four float - magnetic read switch stations provide a contact state change signal to the DAC. Two float stations at the same level provide a scram signal when the water becomes abnormally low. Separate single float stations provide a signal when the pool level travels beyond the normal low and high operating levels.

3. TRIGA MARK I INSTRUMENTATION AND CONTROL SYSTEM SOFTWARE

The software for the computer based TRIGA Mark I reactor ICS is divided into fifteen different processes. Each one performs a specific function or functions and is essential for system performance. Eleven of the fifteen processes are associated with the CSC while the remaining four operate on the DAC. When operated as a whole, the software utilizes these processes in such a way to present the operator with an instrumentation and control system that is essentially real-time.

Each of the 15 processes is described briefly as to their function or functions. In order to better understand the description of each process, flow diagrams (Figures 3-1 and 3-2) for the CSC and DAC respectively show how each software process interacts with essential hardware components as well as with other software processes.

3.1. CSC Processes

1. SC (State Controller)

   The SC is a major process in the CSC and performs six functions:

   a. Receives input command from the reactor control console (RCC) keyboard via the KEY process.

   b. Receives rod control and mode control panel commands via the ZACK process.

   c. Initializing during "power on" via the BOOT process.

   d. Accepts system generated commands from the DISP process [see 3.1(3)] and the DAC command processor (DCP) process [see 3.2(3)].

CSC

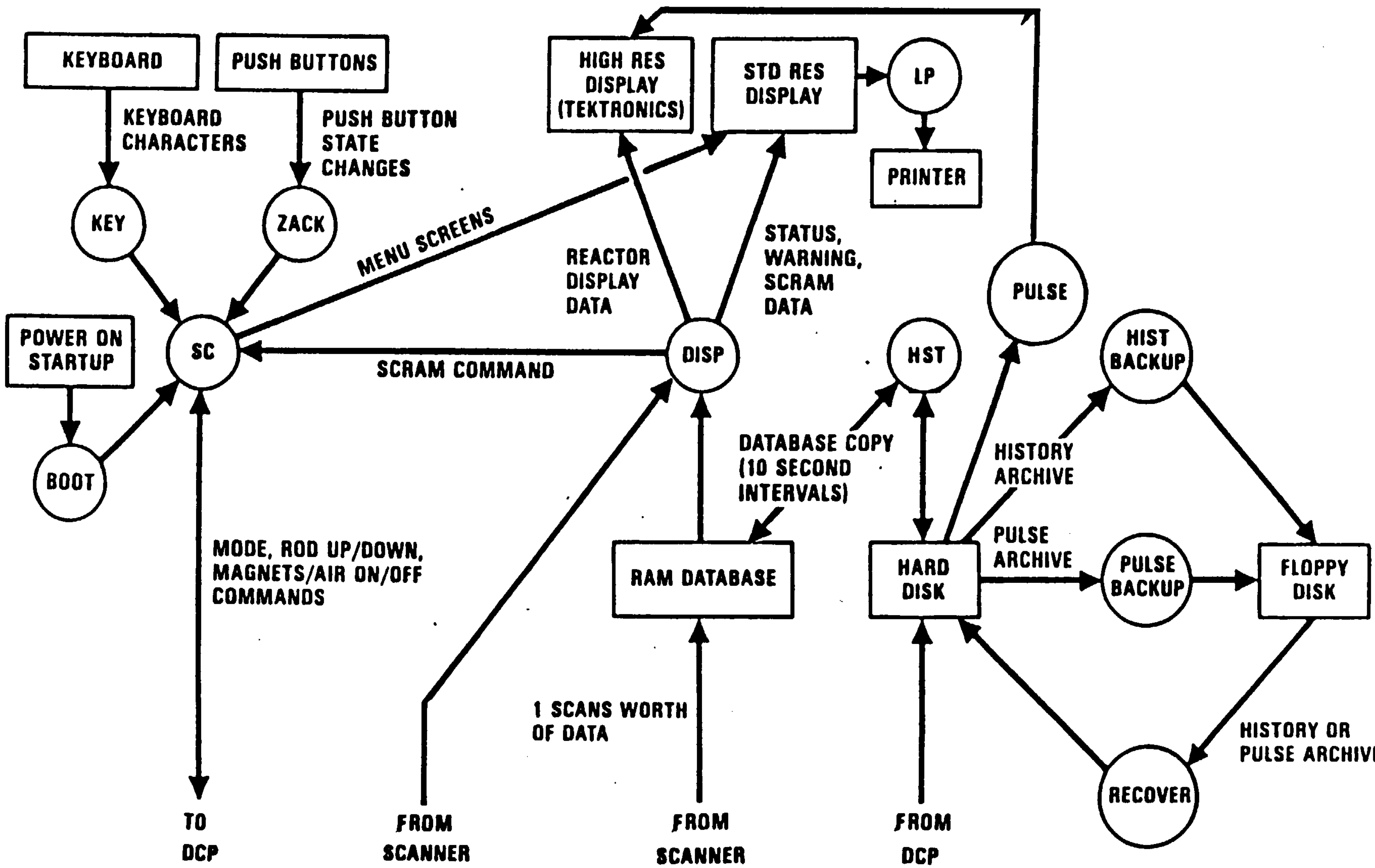

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Figure 3-1

GENERAL ATOMICS

DAC

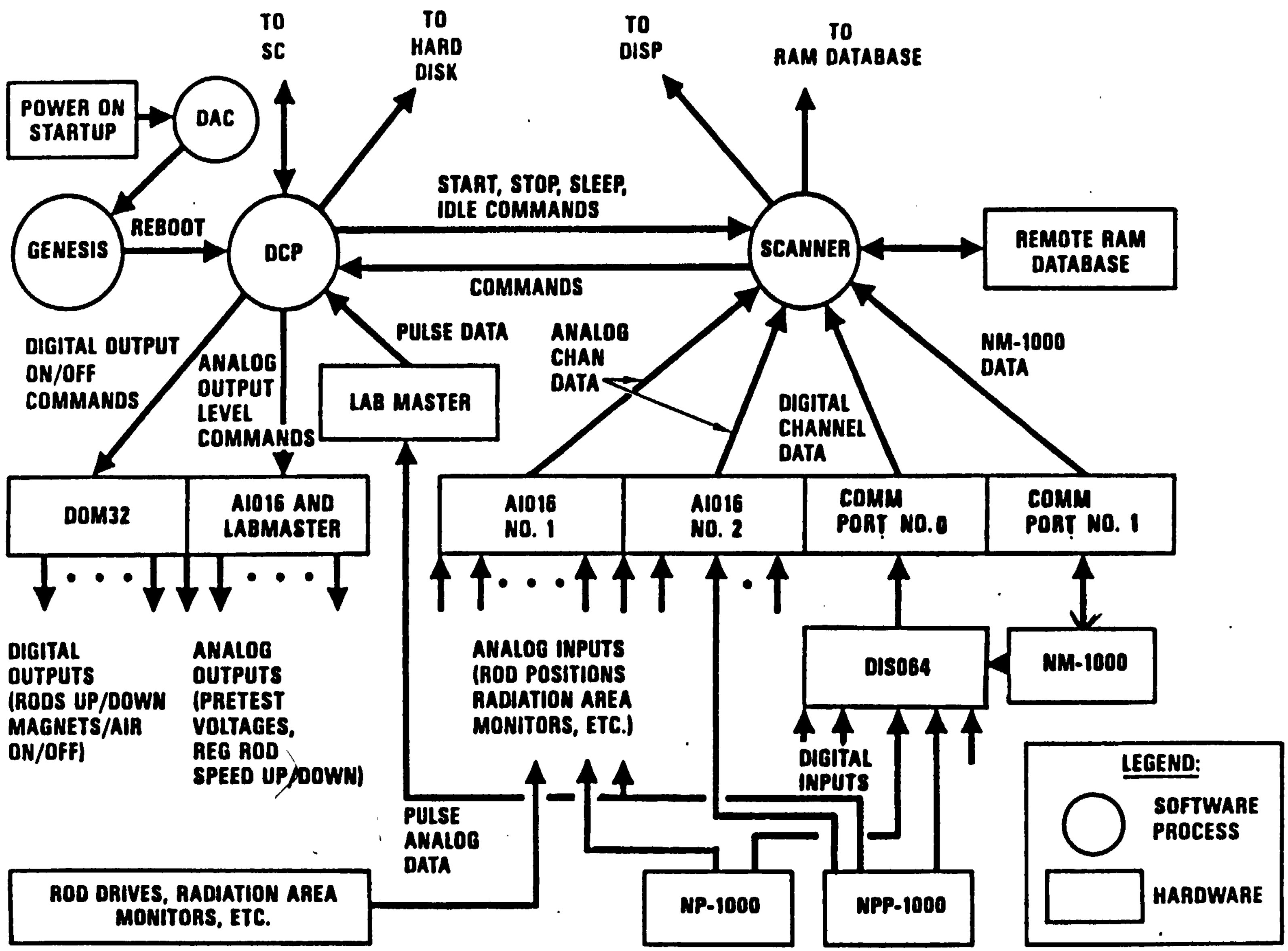


Figure 3-2

e. Transmits commands to the DCP process.

f. Provides menu screen information to the standard resolution display on the RCC.

2. BOOT

Initialization of the system database when ac power is applied is the sole function of the BOOT process.

3. DISP (DISPlay/animator)

Four functions are performed by the DISP process:

a. Provides real-time animation of the reactor rods, rod drive positions, and various bar graphs representing power and period on the high resolution display.

b. Updates the status, warning, and scram displays on the standard resolution screen.

c. Commands the SC to go to the scram state when a scram is detected in the database.

d. Receives new data commands from the SCANNER process [see 3.2(4)] located in the DAC.

4. KEY (KEYboard monitor)

When a key on the keyboard is depressed, the KEY processes sole function is to tell the SC process that a key has been depressed.

5. ZACK (ZACK DIS064 monitor)

The ZACK process, like the KEY process, indicates to the SC process that push button on the rod control or mode control panel has been depressed.

6. HST (HiSTory data logger and playback)

The HST process has two functions:

a. Write information from the database into the hard disk.

b. Read information from the hard disk into the database. In the steady state mode at 2 sec intervals (configurable interval), the information displayed on both the high and standard resolution screens is recorded on hard disk. This process continues to 5 min after the reactor is scrammed. It will also record the current screens at all scrams and warnings. The HST process, in the playback mode, will display this logged information under either automatic or manual control.

7. PULSE (PULSE graphics playback)

The PULSE process has the one function of displaying a pulse selected from the 10 pulses stored on the hard disk. The display is shown on the high resolution screen. Coordinate axis scaling can be changed to enhance resolution. The pulse data on the hard disk comes directly from the DCP process [see 3.2(3)].

8. LP (Line Printer)

LP process is another single function process. It takes data from standard resolution screen and formats the data for printer input.

9. HIST BACKUP (HISTory archive BACKUP)

The HIST BACKUP process allows hard disk history archive data to be written onto floppy disks.

10. PULSE BACKUP (PULSE archive BACKUP)

    The PULSE BACKUP process allows hard disk pulse archive data to be written onto floppy disks.

11. RECOVER (restore history or pulse archive)

    The RECOVER process allows restoration from floppy disk of either history or pulse archive data back onto the hard disk.

3.2. DAC Processes

1. DAC (DAC startup)

   The DAC process is needed only to initialize the GENESIS process. It has no other function.

2. GENESIS (real startup and network/scanner monitor)

   The GENESIS process performs two important functions:

   a. AC power on initialization of the DAC software.
   b. General system diagnostics.

   For example, before it boots the DCP process, it checks for proper operation of the communication network between the DAC and CSC. During normal operation the GENESIS process periodically checks operation of the network, DAC, and CSC. If improper operation is detected it will scram the reactor and initiate a DAC reboot sequence.

3. DCP (DAC Command Processor)

The DCP process is the major process in the DAC. It performs five functions:

a. The process communicates with the CSCs SC process through which reactor rods are moved and mode changes are implement. In short, it's through this process that the control system commands reactor hardware devices by the use of analog and relay contact closure signals.

b. The process keeps track of which mode the system is in.

c. Another function is the high speed acquisition and transfer of pulse data directly to the CSCs hard disk. This method of data gathering uses the DCP process since it is faster than the typical SCANNER process.

d. The DCP sends start/stop commands to the SCANNER process for proper system operations.

e. The DCP process contains the control system pretest mode software.

4. SCANNER (digital, analog, and NM-1000 input scanner)

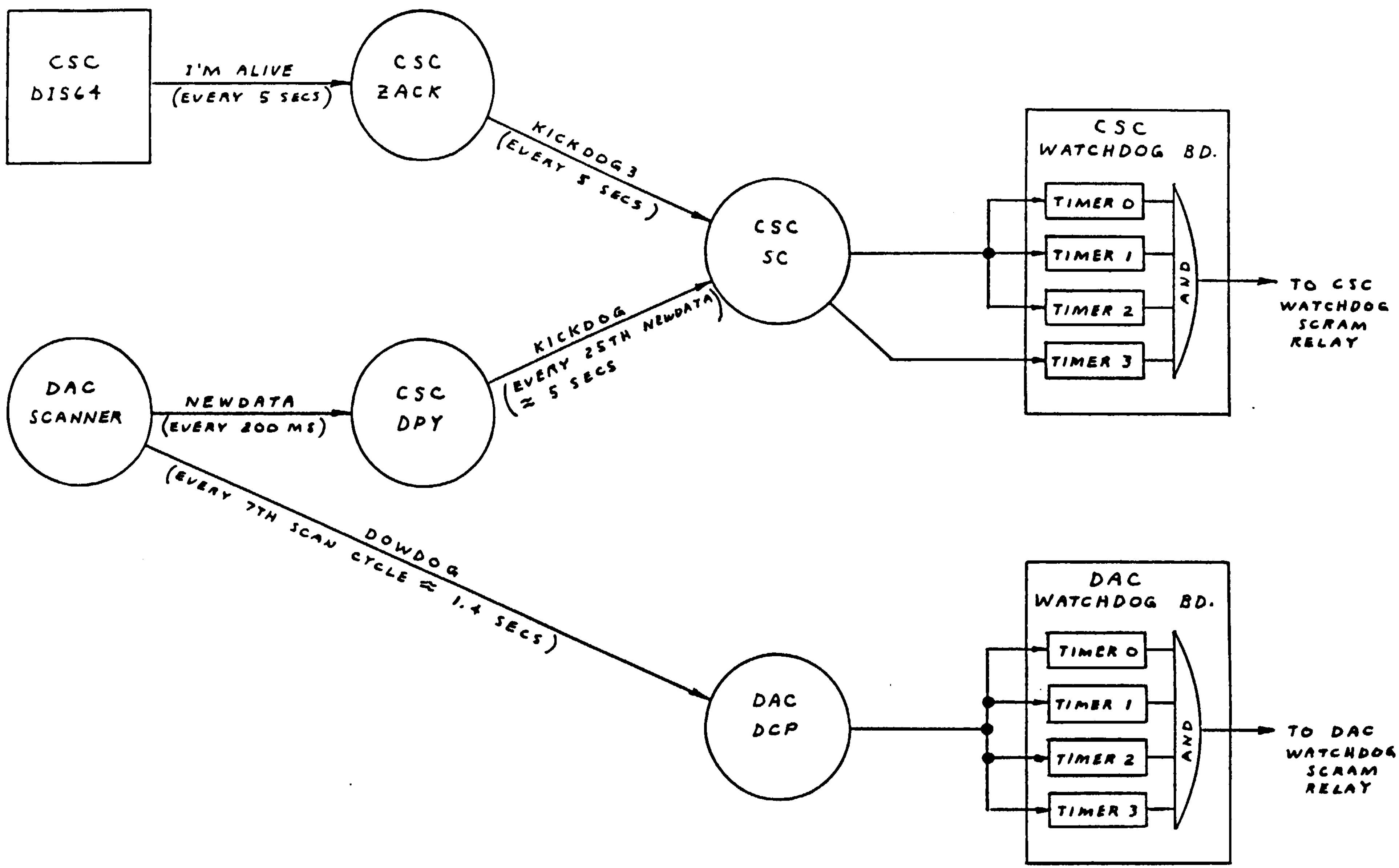
The SCANNER process handles all control system inputs other than pulse data as mentioned in 3.2(3) These include analog, digital, and RS232. It:

a. Continually scans these inputs approximately every 200 msec.

b. It transfers this data to the CSC and a local (DAC) database.

c. It performs preconfigured alarm setpoint checks on all incoming data and sends notification of any alarms to the CSCs DISP process.

d. It performs the auto mode PID algorithm. This algorithm calculates rod speed and direction each scan cycle and adjusts the speed and direction of the rod accordingly.



CSC/DAC WATCHDOG OPERATION

TRIGA® Reactors

APPENDIX A

CONTROL CONSOLE OPERATOR'S MANUAL

GENERAL ATOMICS

General Atomics TRIGA Operator's Manual
Second Edition September 1988

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CHAPTER 7 HISTORY DATA LOGGING AND PLAYBACK

FIGURES

Chapter 1

The Control Console

INTRODUCTION

The TRIGA reactor system is operated from the Control Console. This console includes:

- Standard video display
- High resolution graphics display
- Keyboard
- Mode selection panel
- Rod control panel
- Bar graphs
- Printer
- Control System Computer (CSC)

The standard video display shows tabular data about the status of various reactor components, warning conditions and reactor Scram condition(s). The high resolution display shows a graphical representation of reactor rods and drives, bar graphs of reactor power level and rate of change, and indication of fuel and water temperature. This display can also be used to show actual pulse shape and other characteristics.

The mode selection panel allows you to select the mode in which the reactor will operate. These modes include Manual Mode (Steady State), Auto Mode, Pulse Mode, and Square Wave Mode. The rod control panel allows you to directly control rod magnets, and to move rod drives up or down manually. The rod control panel also allows you to apply air pressure to the transient rod.

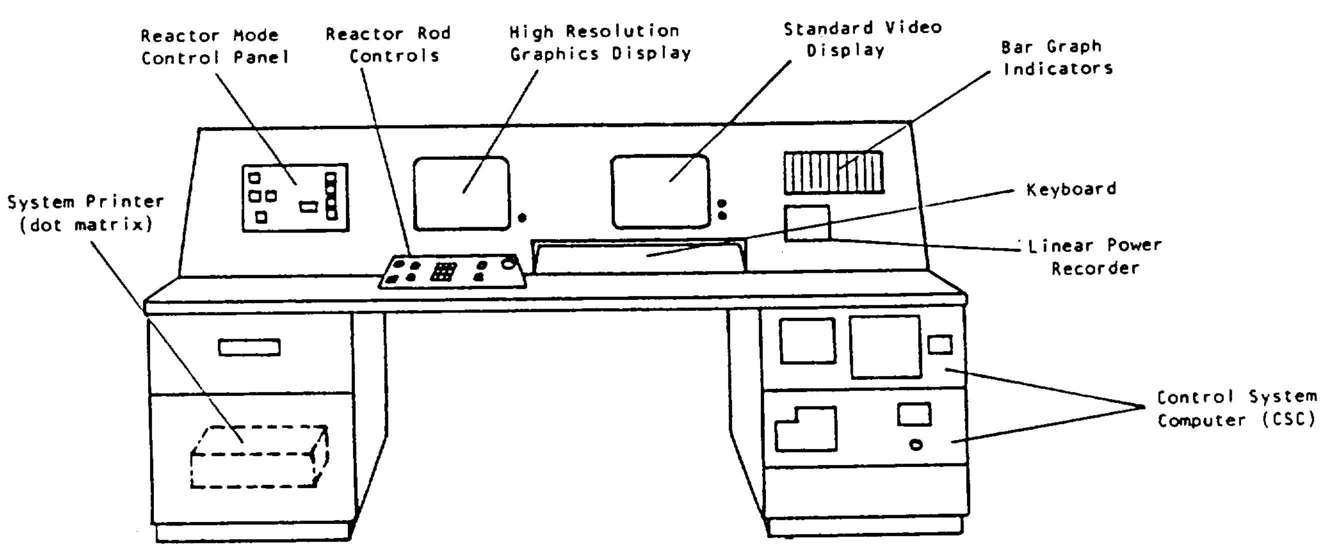

Figure 1.1 Control Console Layout

KEYBOARD

The following operations are always available by pressing the appropriate function key (F1 - F10) on the keyboard:

F1	NOT USED
F2	STANDARD WINDOW PRINT. Press this key to print a copy of the standard video display on the dot-matrix printer.
F3	PULSE DISPLAY. Press this key to initiate the menu sequence for displaying pulse data acquired by the computer during a Pulse Mode sequence.
F4	HISTORY PLAYBACK AND FILE ARCHIVES UTILITY
F5	OPERATOR LOGIN/LOGOUT. Press this key to display the operator login/logout menu. if you are logging on, the system will prompt you to enter a 6-letter password.
F6	ACCUMULATED OPERATOR TIME AND MEGAWATT HOURS. Press this key to get a display of the current elapsed time since you logged on to the system, and the accumulated megawatt-hours since you logged on.
F7	NOT USED
F8	NOT USED
F9	NOT USED
F10	NOT USED
SPACE	STATUS, WARNING, SCRAM WINDOW SELECT. Each time you press the SPACE bar, the standard video display will rotate to the next window.

VIDEO DISPLAYS

There are two video display screens on the CSC Console. The display on the right is a standard color graphics display. The display on the left is a special high resolution color graphics display.

Standard Display

The standard display screen is used to display operating information about the reactor, in tabular form. There are three active display windows: Status, Warning, Scram. You may switch between the three display windows with the SPACE bar. Figure 1.2 shows a typical Status Window display.

```
***************************************TEXAS STATUS WINDOW*****************************************
*                                                      *                                          *
* Primary Coolant Flow              151 GPM            *  Control Room RAM            2.00 mR     *
* Secondary Coolant Flow            125 GPM            *  Pool Access RAM            11.00 mR     *
* Heat Exhange Pressure              OK                *  Area 1 RAM                 10.00 mR     *
*                                                      *  Area 2-3 RAM               10.00 mR     *
* Pool Water Inlet Temp             24.6 C             *  Area 4-5 RAM               10.00 mR     *
* Pool Water OUtlet Temp            26.3 C             *  Spare RAM                  10.00 mR     *
* Pool Water Temp                   23.4 C             *                                          *
* Demin Outlet Condtvty              .10 µmo           *  Rx Rm Particulate Mon      3.2e+2 CPM   *
* Pool Water Level Lo                OK                *                                          *
*                                                      *  Stack Argon 41 Monitor     5.2e+1 CPM   *
* Min Source Interlock               No                *                                          *
* Power > 1 kW Interlock             No                *  Beam Port 1 thru 5         Closed       *
*                                                      *  Beam Port 2 Tangential     Closed       *
* Current Pulse Number               235               *  Beam Port 3 Radial Fst     Closed       *
*                                                      *  Beam Port 4 RAdial Slw     Closed       *
*                                                      *  Beam Port 5 thru 1         Closed       *
*                                                      *                                          *
*                                                      *  Rx Rm Door                 Closed       *
*                                                      *                                          *
*                                                      *  Rx Rm Neg Air Pressure       OK         *
*                                                      *                                          *
*                                                      *                                          *
***************************************************************************************************
```

Figure 1.2 Typical Status Window Display

High Resolution Display

The video display on the left side of the CSC console is a high resolution color graphics display. This display screen is used for two purposes. During normal reactor operation, this screen shows a graphic representation of the reactor. Following a pulse mode operation, a graph of the reactor power readings taken during pulse mode may be displayed.

Reactor Graphic Display

Figure 1.3 shows a typical reactor animation display. Please refer to this figure as you read this section.

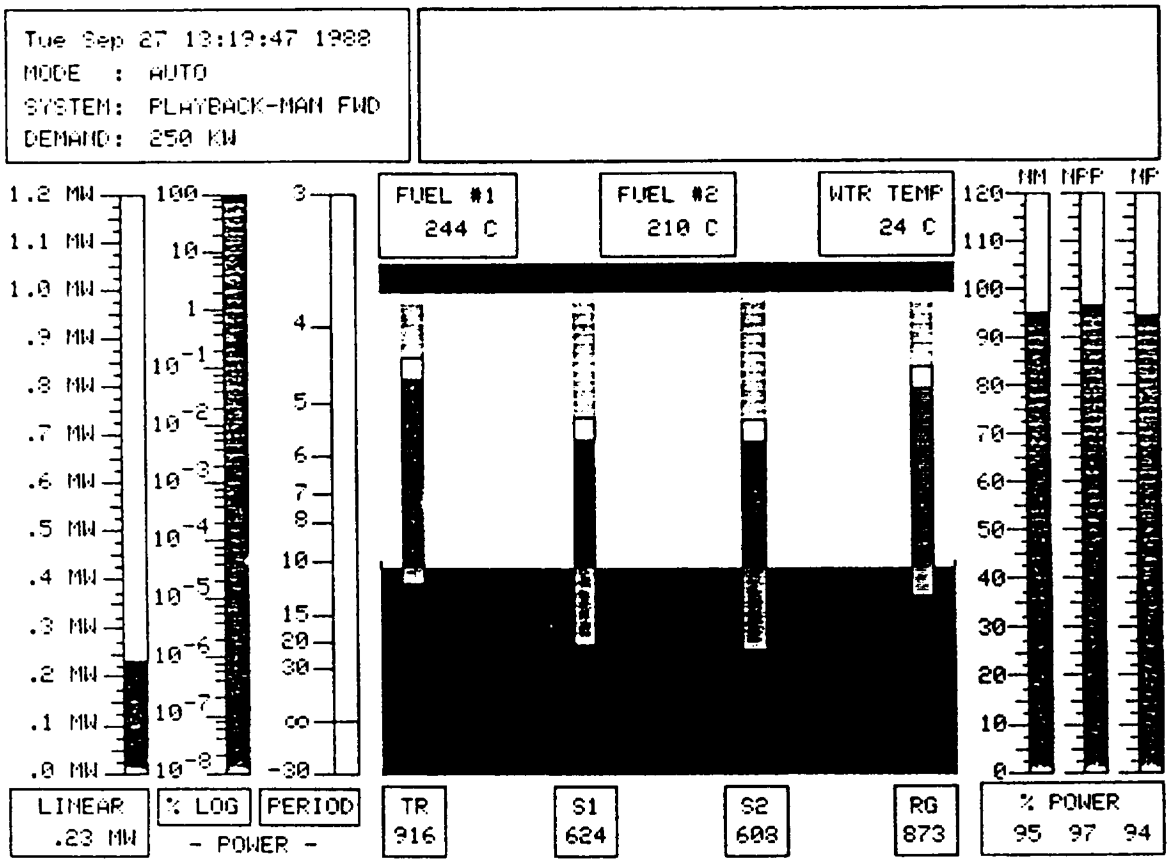

Figure 1.3 Typical Reactor Animation Display

The box in the upper right corner of the reactor display will always show the following information:

Date and Time
Control System Mode
Real Time or Playback Operation
Demand Power Level

The box in the upper right corner will display warning or scram messages. A message will be displayed until you press the ACK (acknowledge) button the control console. Several such messages may be queued, waiting for acknowledgement. These will display in turn as you press the ACK button. There

are typically five bar graph displays giving reactor power information. On the left side of the screen are scales for:

LINEAR POWER	This bar graph shows the current reactor power level as a percentage of maximum power, on a linear scale.
% LOG POWER	This bar graph shows the current reactor power level as a percentage of maximum power, on a logarithmic scale.
PERIOD (RATE OF CHANGE)	This bar shows the rate of change of the reactor power although somewhat indirectly. Period is proportional to 1/Rate of Change. If reactor power were steady, the rate of change would equal zero, the period would equal infinity. The greater the rate of change becomes, the lesser the period becomes.
% LINEAR POWER	Two or three bar graphs on the right side of the screen also indicate the current reactor power. These two graphs use a linear scale. The two graphs are redundant because they display the same information derived from independent power sensors.

Above the animated display of the reactor are three small indicator boxes. These boxes indicate the current temperature of the reactor fuel and the coolant water in the reactor.

Below each rod position in the animated display is a small box that indicates the current position of the rod drive mechanism. The scale for the position readout ranges from 0 to 999. The position is 0 if the rod drive is all the way down; the position is 999 if the rod drive is all the way up. These are adjustable and the operator may see a small offset reading when the rod drive is all the way down.

If the rod is all the way down, its color will be gray. If it is all the way up, the color will be magenta. The rod color will be green anywhere between fully down or fully up.

In the center of the screen is an animated display of the reactor. This display will show the current positions of the rods and the rod drive mechanisms. For each shim rod and the regulator rod, the small square box at the bottom of the drive mechanism indicates the status of the rod magnet power. If the box is yellow, then magnet power is ON. If the box is black, then magnet power is OFF. For the transient rod, this box indicates the status of the air pressure to the rod drive mlechanism. If the box is yellow, then air pressure is ON. If the box is black, then the air pressure is OFF.

Pulse Graphic Display

The high resolution graphics screen may also be used to display a graph of the reactor power readings acquired during pulse mode. In pulse mode, 5000 power readings are taken in a period of 1/2 second (1000 microseconds per reading). A typical reactor pulse display is shown in Figure 1.4 below. See Chapter 7, Pulse Mode for detailed information.

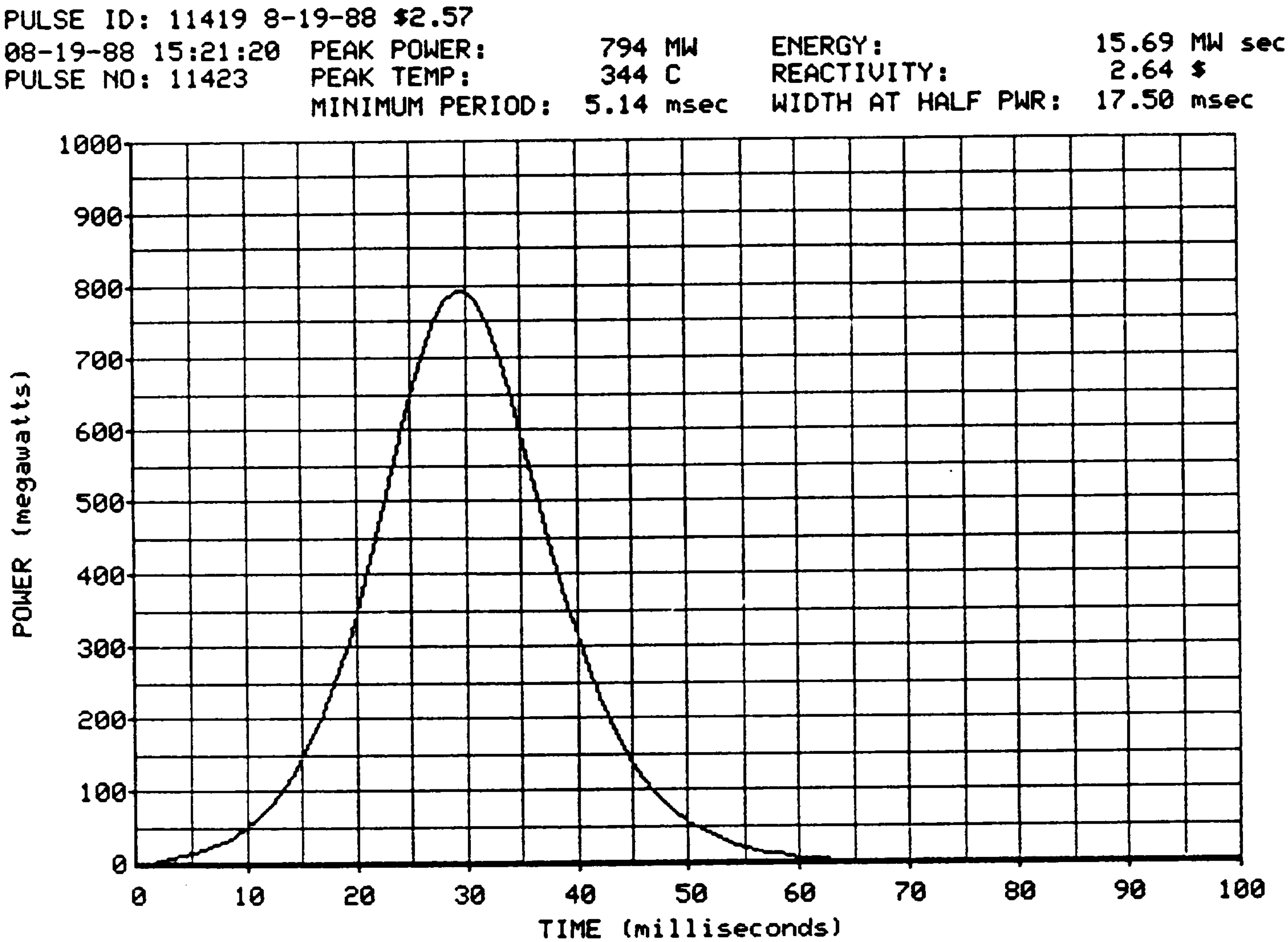

Figure 1.4 Typical Pulse Graphic Display

BAR GRAPH INDICATORS

There are eight bar graph indicators on the right side of the control console. These indicators are LED displays which are hardwired to sensors in the reactor system. They provide redundant readouts of the reactor state, which are independent of the video displays and system software. These indicators are:

% POWER	Two or three bar graph displays show the reactor power, as a percentage of maximum power. The scales for these indicators are linear.
LOG POWER	This indicator shows the current reactor power, in percent, displayed on a logarithmic scale.
PERIOD	This indicator shows the current rate of change of the reactor power. The units are seconds.

TEMPERATURE (°C)	Two indicators provide redundant readouts of the reactor fuel temperature. The units are degrees Celsius.
NV and NVT	One indicator each used during pulse mode showing pulse peak power and pulse energy. Units are MW and MWT (sec).

MANUAL CONTROLS

There are two groups of manual controls on the control console. On the left side of the console is the REACTOR MODE panel. This panel is shown below in Figure 1.5. The REACTOR MODE panel includes four pushbutton switches for selecting the reactor operation mode:

Pulse Mode
Manual Mode (Steady State)
Auto Mode
Square Wave Mode

This panel also includes the POWER DEMAND thumbwheel switch. This switch allows you to enter the power level which the system will maintain while it is in Auto Mode. This thumbwheel switch has no effect when the system is not in Auto Mode.

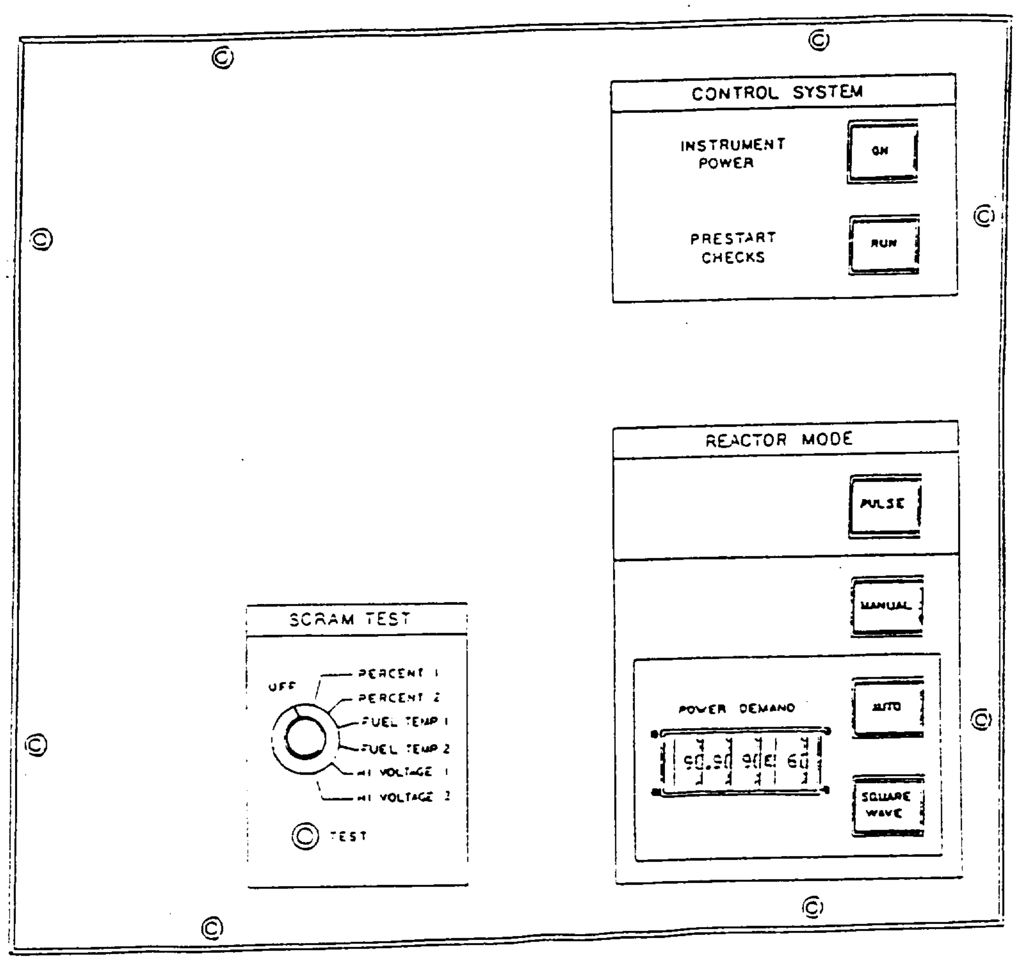

Figure 1.5 REACTOR MODE Panel

Directly beneath the video displays is another control panel, the ROD CONTROL panel. This panel, shown in Figure 1.6, allows you to control the reactor rods manually.

In the upper left corner is the MAGNET POWER keyswitch. If this switch is OFF, then all power is removed from the rod drive magnets. This will cause all rods to drop into the reactor core, therefore shutting the reactor down (SCRAMMING the reactor). This switch is momentarily turned to the RSET position to restart the reactor control system software. The switch will remain in the ON position during reactor operation.

The transient rod is controlled by air pressure, which keeps the rod in contact with the rod drive mechanism. The transient rod drive mechanism thus acts as a stop. Press the AIR button to turn the air pressure OFF. Press the FIRE bottom to turn air pressure ON (air pressure is only enabled if certain conditions exist).

The MAGNET buttons control the magnet power for each shim rod and the regulating rod. Press the MAGNET button to turn magnet power OFF, and therefore drop the rod into the reactor core. The magnet power will return when you release the button. Press the UP or DOWN buttons to move the control rods manually.

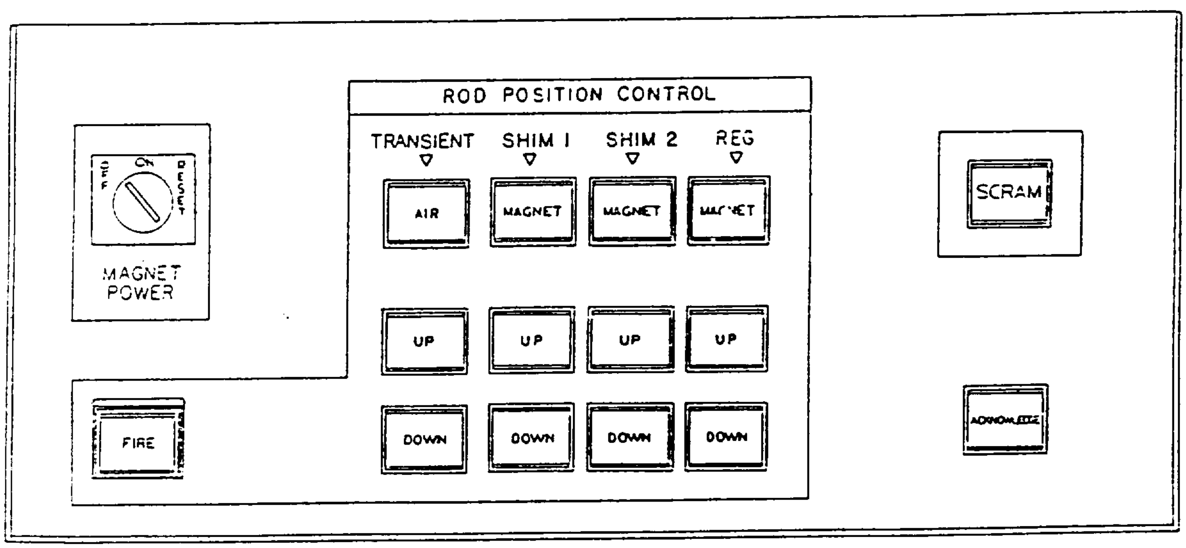

Figure 1.6 Rod Control Panel

Chapter 2

System Startup and Prestart Checks

NORMAL SYSTEM STARTUP

This is the procedure for starting up the TRIGA reactor control system.

1. Power up the system by pressing the console Instrument Power "ON" button located on the Mode panel. It will take about two minutes before the reactor core display is drawn on the high resolution screen. Meanwhile, the following messages, listed here in abbreviated form, will appear on the standard screen:

   a. ACTION INSTRUMENTS 1986 V2.30
   Equipment:
   (List of hardware items to be checked - memory, keyboard, etc.)

   b. Bootstrap Version 1.14 7/11/86 11:19:23 --- etc.
   enter device name ----- etc.

   c. Using 40 disk buffers

   phys mem = 1024K
   avail mem = 813K
   user mem = 729K
   etc.

   d. IC-DOS - DFS 3.0 - PROTECTED MODE

   Checking file system - etc.

   Phases 1 through 5

   e. BOOTING CSC SOFTWARE

   The reactor core will be drawn after item

   "dispmain : Beginning dpy process"

   After the high resolution screen display is drawn, if any of the safety channels or systems are not in a normal operating range, the system will display a message in the SCRAM/WARNING box on the high resolution display and in the SCRAM/WARNING windows on the standard display. Acknowledge these messages by pressing the acknowledge button.

2. Log in by pressing the F5 key to access the operator log in function. You will get the display screen which prompts you to enter your operator password.

3. Turn the key to the RESET position. The system will enter the Steady State (Manual) Mode.

NORMAL SYSTEM SHUTDOWN

This is the procedure for a normal shutdown of the TRIGA control system. Using the console keyboard,

1. Press the "Alt" and the "0" keys at the same time. The status screen will go blank.
2. Now press the "Ctrl" and the "C" keys at the same time. The status screen will display the message "USER REQUESTED EXIT!"
3. Then press the "Ctrl," "Alt," and "Del" keys at the same time.
4. The following message will appear on the standard screen:

   "CSC_# Starting system shutdown strike a key in five seconds to abort reboot 5...4...3.."

5. Strike any key to abort reboot. The following message will be added to the screen:

   "You may now shut off power to your computer or you may strike any key when ready to reboot"

6. By pressing the Instrument Power "ON" button, 120 V ac switched power to the control system is off.

ABNORMAL SYSTEM SHUTDOWN

Any shutdown that does not follow the steps outlined above is considered abnormal. The results will involve increased system startup time and possibly loss of data. Abnormal shutdowns will occur now and then and it is <u>not</u> a serious event. The hardware and software have been designed to accommodate abnormal shutdowns.

Startup after an abnormal shutdown is the same as the normal system startup. Instead of taking about two minutes, however, it will take about four minutes. This is due to more extensive memory checks and data transfers.

RUNNING PRESTART CHECKS WITHOUT STARTUP

You can command the system to run the Prestart Checks without continuing on to start up the reactor. This feature is convenient for running maintenance checks when you do not wish to operate the reactor. It also allows maintenance checks when you do not wish to operate the reactor. It also allows maintenance checks to be run by personnel who are not logged on to the system.

This is the procedure for running the Prestart Checks only, without starting the reactor:

1. The system must first be in the SCRAM mode.
2. Press the PRESTART CHECKS RUN button on the console.
3. The system will now run the Prestart Checks sequence. If this sequence completes successfully, then the system will return to the SCRAM mode.

PRESTART MODE

The Prestart mode is the mode in which the TRIGA control system runs diagnostic tests on various system devices. Prestart tests are limited to those devices which can cause the reactor to SCRAM.

The system will enter the Prestart mode if the system is in the scram state and you perform the following:

Press the PRESTART CHECKS RUN button.

After the checks, the system will return to the SCRAM mode. This feature allows maintenance personnel to run tests on the system without operating the reactor. This manual test feature is also useful for producing daily system status reports on the system printer.

DEVICES TESTED DURING PRESTART CHECKS

The Prestart mode tests only those devices which can cause the reactor to SCRAM. The following tests are performed during the Prestart mode.

1. Initiating the Prestart Mode

   The Prestart mode is initiated by pressing the PRESTART CHECK RUN button on the CSC Control Console while the system is SCRAMMED. An operator need not be logged in nor the key installed. This allows the test to be run by nonoperator personnel.

   When the button is pressed, the PRESTART CHECK RUN light is illuminated, the standard resolution screen is cleared and the message "TRIGA PRESTART CHECKS" is displayed introducing the checks.

2. DAC Software Control

   The Prestart software comprises a portion of the "dcp" process which runs in the DAC. When the Prestart checks are initiated by the CSC "sc" process, the CSC keyboard input and standard resolution screen output are redirected from the CSC software to the DAC software. That is, the Prestart software communicates across the network with these devices. Upon completion of the Prestart mode, these devices are restored to the CSC software.

3. Fuel Temperature Scram Circuits Test

   At the start of this test, the DAC Prestart relays are activated. This disconnects the thermocouple Action Paks from the Fuel Temperature Alarm Action Paks and connects the AI016 #1 channel 0 analog output to the Alarm Action Paks.

   The software then outputs a test voltage on the analog output channel to test the Alarm Action Pak operation. The first output test voltage corresponds to 95% of the test temperature configured by channel "FT_SCRM_LMT" in the configuration file. The corresponding temperature is printed on the screen. This temperature should be below the Action Pak alarm limit and the alarm should not be activated. The "SCRM_FT1_HI" digital input from the Action Pak is tested for an OFF condition and an OK/FAILED result is printed. The test is repeated for the second and third fuel temperature channel as a function of how many fuel temperature channels are configured by the "NUM_FUEL_PTS" configuration point. The digital inputs checked are "SCRM_FT2_HI" and "SCRM_FT3_HI," respectively.

Next, a test voltage corresponding to 100% of the test temperature is output on the analog output. This temperature should be above the Fuel Temperature Alarm Action Pak limit and the alarm should be activated. The corresponding digital input is tested for an ON condition and an OK/FAILED result printed. The test is repeated for the second and/or third fuel temperature channels.

4. Testing NM1000

Calibration modes 2 through 5 of the NM1000 are sequentially tested for correct power level outputs. Channel 51 of the NM1000 is programmed to the appropriate mode and the corresponding power level is read from channel 12. The power level is then compared with the configured test level and is deemed OK if it falls between 95% and 105% of the configured value. The configured values are stored in the following configuration channels:

Mode	Channel Tag
2	CTR_MID_PWR
3	CTR_HI_PWR
4	CMB_LOW_PWR
5	CMB_HI_PWR

If the power level is out-of-limit, the message "% Power = x.xxxExx, should be y.yyyEyy" is displayed.

At the end of the calibration tests, the NNM1000 mode is reset to mode 0, the normal mode.

Next, the NM1000 period trip is tested.* Since the mode has just been changed from mode 5 to mode 0, a momentary negative period should occur and the period trip should be OFF. The mode is then changed to mode 5 which should cause a momentary high positive period and activate the period trip. Both the OFF and ON trip conditions are checked by reading the "PER_HI" digital input.

With the mode currently in mode 5, the NM1000 % Power High Trip should be ON.* The mode is then reset to mode 0 and the % Power High Trip should turn OFF. The ON and OFF conditions are checked by reading the "SCRM_%P1_HI" digital input.

The source level trip is tested next. The operator is instructed to remove the neutron source from the reactor core and hit the spacebar key when ready to continue. The operator should allow enough time for the NM1000 power to drop below the source trip limit before continuing. The operator is then instructed to restore the neutron source and hit the spacebar key when ready to continue. The source trip condition is checked by reading the "RWP1" digital input.

*These trips are optional and may not be functional in all installations.

5. Testing NPP1000

To test the NPP1000's High % Power SCRAM trip and High Voltage SCRAM trip, the NPP1000's "Test" input is activated by one of the DAC's Pretest Relays. Since the Pretest relays were turned on during the Fuel Temperature tests, the ON status of the two NPP1000 inputs is checked first. The inputs are checked by reading the "SCRAM_%P2_HI" and "SCRAM_NPP_HV" digital inputs, respectively.

Activating the NPP1000's test input should also cause the NPP1000 to output a power level of 111%. The power level is checked by reading the AI016 #1 "NPP_PLIN_PWR" analog input and determining if the reading falls between 110% and 112%. To determine the upper and lower limits, the configuration channel "MAX_%PWR_150" is read to determine if full scale is 120% or 150%.

The Prestart relays are then turned off and the High % Power SCRAM trip and the High Voltage SCRAM trip are tested for their OFF condition.

6. Testing NP1000

The number of NPP/NP1000s is checked by reading the "NUM_NP1000" configuration channel. If the number is less than 2, this test is not performed. If the test is performed, it is identical to the NPP1000 test except the input channels tested are: "SCRAM_%P2_HI," "SCRAM_NPP_HV," and "NPP_PLIN_PWR," respectively.

7. Testing DAC Watchdog

The DAC Watchdog board is designed to toggle some of its register bits when accessed in a unique way. For this feature to work, the board must be installed, configured to the correct I/O address, and operating correctly. This feature is tested by the Board Alive Test and an OK/FAILED result is displayed.

While in the Prestart mode, the "scanner" process has been halted and therefore the DAC Watchdog board is not receiving refresh commands. Since we have been in the Prestart mode for some time now, the four Watchdog timers will be timed out. This condition is tested by the Timed Out Test. Next, the four timers are refreshed (triggered) and their refreshed condition is tested.

8. Continue or Abort?

Any time a check fails, the message "Continue or abort? (c/a) ..." is output to the screen. By entering a "c" or an "a," the operator can choose to continue or abort further checking. If "c" is selected, the work "Continued" is displayed and checking is resumed. Correspondingly, if an "a" is selected, "Aborted" is displayed and further checking is aborted. The message "*** ABORTING ALL FURTHER TESTING ***" is then displayed.

9. Prestart Passed/Failed

At the conclusion of the Prestart checks, the message "*** PRESTART PASSED ***" or "*** PRESTART FAILED ***" is displayed. If any check failed, the failed message is displayed. Otherwise, the passed message is displayed.

10. Print Copies of Test Results

Finally, the operator is queried for how many hard copies of the test results are to be printed on the printer. A number between 0 and 9 may be entered. Upon entry of the digit, the requested number of printouts is initiated, the Prestart "Run" light is extinguished, the "scanner" is restarted and the Status screen is restored to the CSC low resolution monitor.

DEVICES NOT TESTED DURING PRESTART CHECKS

Several devices are not tested during the Prestart mode. These devices are described below.

Low/Hi Water Level

The DAC has no control over the Water Level input. Therefore, this input cannot be tested automatically by the system.

However, this input may be tested manually. Actuate the Water Level limit switch. You should see the Low/Hi Water Level message on the CSC console.

External SCRAM Circuits

The DAC has no control over the two External SCRAM circuits. Therefore, these circuits cannot be tested automatically by the system.

However, these inputs may be tested manually. Open each of the External SCRAM circuits. You should see the External SCRAM message on the CSC console.

Magnet Power Keyswitch

The OFF position of the Magnet Power keyswitch must be tested manually.

Console Manual SCRAM Button

The red manual SCRAM button must be tested manually.

Reactor Room Manual SCRAM Button (Optional)

The red manual SCRAM button in the Reactor Room must be tested manually.

Chapter 3

SCRAM Mode

This chapter lists and describes the various conditions which can cause the reactor to shut down, or SCRAM. In most cases, the reactor is SCRAMmed by hardware monitoring devices, and the computer control system is then notified.

SCRAM - Fuel Temp #1 Hi*
SCRAM - Fuel Temp #2 Hi*

There are two fuel temperature monitors. Each monitor is connected to a hardware limit alarm module (Action Pak). The limit alarm will trip a relay causing the reactor to SCRAM if the fuel temperature rises to a preset value - 500°C in most systems.

SCRAM - NPP1000 HI VOLTAGE LO
SCRAM - NP1000 HI VOLTAGE LO

The NP/NPP1000 monitors the high voltage power supply to the ion chamber. This voltage must stay above 700 volts to ensure valid readings from the ion chamber. If the voltage drops below a preset value, the reactor is automatically SCRAMmed.

SCRAM - % Power #1 Hi (NP)
SCRAM - % Power #2 HI (NPP)

The NP1000 and NPP1000 monitors reactor power on a real-time basis. If the reactor power exceeds 110% of rated power, they will automatically SCRAM the reactor and provide a contact closure to the computer control system.

SCRAM - External #1
SCRAM - External #2

There are two external digital switch closure inputs. They are provided as spares for adding SCRAM inputs in the future.

SCRAM - CONSOLE MANUAL

You can cause a manual SCRAM at any time by pushing the red SCRAM button on the Control Console.

SCRAM - PLEASE LOG IN

The system will automatically SCRAM two minutes after you log off the system, if a new operator has not logged on.

SCRAM - Key Switch Off

The reactor will SCRAM whenever you turn the Magnet Power keyswitch to the OFF position.

SCRAM - Net Fault, Please Reboot

This SCRAM condition is caused by a loss of communications between the Data Acquisition Computer (DAC), the

*Fuel temperature scrams are optional and may not be applicable for all installations.

Control System Computer (CSC), because of a hardware failure.

SCRAM - CSC DIS064 Timeout

The DIS064 is a digital input scanner board. The DIS064 that is located in the CSC monitors all the control inputs (buttons, switches) on the control console. The DIS064 sends an "I'm alive" signal to the CSC every five seconds. A loss of this signal is a DIS064 Timeout. It indicates that control inputs from the console have been lost. The CSC will SCRAM the reactor if this occurs.

SCRAM - DAC DIS064 Timeout

The DIS064 is a digital input scanner board. The DIS064 that is located in the DAC monitors a variety of relays and limit alarms connected to sensors in the reactor area. The DIS064 sends an "I'm alive" signal to the DAC every five seconds. If this signal is lost, the DAC will SCRAM the reactor.

SCRAM - Data Base Timeout

The scanner is a software module that runs in the Data Acquisition Computer (DAC). The scanner periodically scans various device inputs to update the system data base. This SCRAM condition occurs if that scanning process stops functioning properly.

SCRAM - NM1000 Comm Flt*

This condition indicates that an error in communications with the NM1000 has occurred.

SCRAM - DOM32 Fault

The DOM32 is a digital output board. The outputs from the DOM32 control the rod drives and magnets, among other things. This fault condition indicates that an error has occurred in communications between the computer and the DOm32, and that computer control of the rods and rod drives has been lost.

SCRAM - AIO16 #1 Fault
SCRAM - AIO16 #2 Fault

The two AIO16 boards receive all the analog input signals in the system. If either of these boards stops functioning properly, this SCRAM condition occurs.

SCRAM - CSC Watchdog Fault
SCRAM - CSC Watchdog Timeout
SCRAM - DAS Watchdog Fault
SCRAM - DAC Watchdog Timeout

The watchdog timer boards are located in both the DAC and the CSC. Each watchdog board acts as a "dead man's switch." Each watchdog board should receive periodic reset ("I'm OK") signals from various software modules. If any of these signals stops coming, then the watchdog board will automatically time out and SCRAM the system.

SCRAM - NM1000 Data Error*

This condition indicates that the NM1000 data length and/or check sum has an error.

SCRAM - Reactor Room Manual*

You can cause a manual SCRAM at any time by pushing the large red button on the DAC assembly in the Reactor Room.

---

*Optional.

Chapter 4
Manual Mode (Steady State)

INTRODUCTION

The Manual Mode is also called the Steady State Mode. When the system is in Manual Mode, you control all rods, rod drives, and magnet currents with the pushbuttons on the CSC Console. Therefore, you can manually control the power output of the reactor, increasing reactor power by raising rods and decreasing power by lowering rods.

When the system is in Manual Mode, the computer takes no active control over the reactor rods, as long as the reactor power stays below the maximum allowable level. If you take an action that would cause the reactor to exceed the maximum allowable power level, then the power safety channels will SCRAM the reactor (i.e., shut the reactor down by dropping all the rods).

ENTERING MANUAL MODE

The system will enter the Steady State Mode when any one of the following conditions becomes true:

1. When the reactor operator applies magnet power by momentarily turning the keyswitch to RESET.
2. The operator pushes the MAN button on the console while the system is in any of the following states: Square Wave Ready, Pulse Ready, Auto Mode.
3. The Square Wave Ramp Up state does not reach the demand power level and times out.

TRANSIENT ROD CONTROL

When the system is in Manual Mode, you can manually move the transient rod up or down. If the transient rod is all the way down (at the bottom of the reactor core), you can turn the air on, without lifting the transient rod.

MOVING THE TRANSIENT ROD

This is the procedure for moving the transient rod manually:

1. Move the transient rod drive all the way down. Once the transient rod is all the way down, it will activate a switch indicating that it is at the bottom.
2. Press the FIRE button on the CSC console. This will turn on the air pressure to the transient rod drive mechanism. The transient rod will move about 1/2 inch, until it contacts the drive mechanism.

3. Press the UP/DOWN buttons for the transient rod. This will cause the rod drive to move up or down. The air pressure beneath the transient rod will hold it in contact with the rod drive.

4. If you press the AIR button, the air pressure will be removed and the transient rod will drop to the bottom of the reactor core. You cannot turn on air pressure to the transient rod again until you have moved the rod drive down to its bottom limit.

MOVING THE ROD DRIVE ONLY

You can position the transient rod drive mechanism up or down without moving the transient rod itself. This is the procedure.

1. Press the AIR button to remove air pressure from the transient rod. The transient rod drive mechanism will automatically wind down to its lowest position.

2. You can now move the transient rod drive mechanism with the UP and DOWN buttons. The transient rod itself will remain on the bottom of the reactor since the air pressure is off. Note also that the FIRE button will be inhibited as long as the drive mechanism is above its bottom limit.

SHIM ROD CONTROL

There is a magnet button for each shim rod. These MAGNET buttons are located on the CSC console. Each MAGNET button functions as a momentary switch. This means that when you press a MAGNET button, magnet power will be turned OFF as long as you hold the bottom. The magnet power will be turned on again when you release the button.

A shim rod drive can move its associated rod only when it is in contact with the rod and its magnet power is on. If you press the MAGNET button, the rod will drop to the bottom of the reactor. Note also that the rod drive will automatically wind down to make contact with the rod again.

The normal condition is that the rod drive is in contact with the rod, and magnet power is on. When this condition is true, then you can move the rod up or down manually by pressing the UP/DOWN buttons (for that particular rod) on the CSC console. The rod will continue moving while you hold the button down. It will stop moving when you release the button.

The rod will move at a constant speed, fixed by the mechanics of the drive mechanism. The rod will move at a fairly slow speed. It will take approximately one minute for the rod to move from the bottom to the top.

REGULATING ROD CONTROL

The regulating rod has MAGNET, UP and DOWN buttons which function the same as described for the shim rods above. The regulating rod drive has a variable speed motor, while the shim rod drives have constant speed motors. This variable speed feature is only available to the computer when the regulating rod is under automatic control. When you control the regulating rod manually, the rod drive will always move at maximum speed.

Chapter 5

Auto Mode and Square Wave Mode

INTRODUCTION

The TRIGA control system, when placed in Auto Mode, will control the positions of the regulating rod automatically to maintain a specific power level. This demand power level is taken from the setting of the thumbwheel switch on the CSC console.

In Auto Mode, the shim rods are under manual control, while the computer controls the regulating rod according to a PID algorithm. The regulating rod has a range of 0 to 100% if rod travel.

The regulating rod is always under computer control and the remaining rods are always under manual control.

ENTERING AND EXITING AUTO MODE

You can transfer the system to Auto Mode from the Manual Mode (also called Steady State Mode). You do this by simply pressing the AUTO button on the CSC console.

The system will take the following actions when its enters Auto Mode;

1. Disable manual control of the regulating rod. The regulating rod is always under computer control in Auto Mode.
2. Inhibit the FIRE button if the transient rod drive is not fully down.
3. Report any change in the demand power level, as indicated by the thumbwheel switch on the control console.

The system can transfer from Auto Mode to either Steady State Mode or SCRAM Mode.

You transfer the system to Steady State Mode by pressing the MAN button on the CSC console. The system will transfer to SCRAM Mode upon the occurrence of any SCRAM condition (including pressing the red SCRAM button on the console).

SETTING DEMAND POWER

You set the demand power level with a thumbwheel switch. This switch is labeled POWER DEMAND and it is located on the left side of the control console, on the REACTOR MODE panel.

Enter the desired demand power level as a four-digit number, in exponential format. The digit to the right of the "E" specifies the power of 10 which will be multiplied by the first three digits. The units will be watts.

Examples:

Demand Power	Switch setting
1 kilowatt	001 E 3
200 watts	020 E1
560 kilowatts	560 E 3
1 megawatt	001 E 6
100 watts	100 E 0
1 megawatt	100 E 4

SHIM ROD CONTROL

Control of the shim rods is limited to three states: stopped, up, or down. The speed of shim rod movement is not controllable. A shim rod moves at a constant speed, fixed by the characteristics of the rod drive system.

REGULATING ROD CONTROL

The regulating rod is always under computer control when the system is in Auto Mode. The regulating rod drive may be controlled by any of four control modes. The particular control mode at a given time is a function of the reactor power level versus the demand power and the regulating rod position.

The regulating control modes are:

Proportional Control (P)
Derivative Control (D)
Proportional-Derivative Control (PD)
Proportional-Integral-Derivative Control (PID)

TRANSIENT ROD CONTROL

The transient rod is always under manual control when the system is in Auto Mode. You cannot fire the transient rod upward rapidly, as in Pulse Mode. However, you can apply air to the transient rod, and then move the rod up and down manually, as you would move a shim rod.

Note

The transient rod drive mechanism must be fully down before the system will also allow you to apply air pressure to the transient rod.

Once you have applied air to the transient rod, then you may control it manually with the UP and DOWN buttons to balance the power profile of the reactor.

AUTO MODE CONTROL EXAMPLES

Here are two examples of the sequence of events that the system takes when it enters the Auto Mode. These two cases represent the most common situations.

Case 1: Reactor Power is Below Demand Power

Let's start with these assumptions:

1. The reactor power level is below the demand power level.
2. The system is in the MANUAL Mode.

When you press the AUTO button, the system enters the Auto Mode. The computer will control the position of the regulating rod, according to a Proportional-Integral-Derivative (PID) algorithm. The computer will try to drive the reactor power to match the demand power, by moving the regulating rod. We have assumed the reactor power is below the demand power. Therefore, the system will move the regulating rod UP.

Case 2: Reactor Power is Above Demand Power

Let's start with these assumptions:

1. The reactor power level is above the demand power level.
2. The system is in the MANUAL Mode.

When you press the AUTO button, the system enters the Auto Mode. The regulating rod is under automatic control, according to a PID algorithm. We began by assuming that reactor power is above the demand power level. Therefore, the computer is not moving the regulating rod downward.

SQUARE WAVE MODE

The Square Wave Mode allows you to bring the reactor to a steady high power level in the shortest amount of time. After the power has rapidly risen to a certain limit, then the system will automatically transfer to Auto Mode. Once in Auto Mode, the system will maintain a constant power level, according to a Proportional-Integral-Derivative (PID) algorithm.

In order to bring the power to a preset level in the shortest time, the transient rod must be fired up to a certain position to produce enough reactivity in the core. After the transient rod has been fired, the system will transfer to Auto Mode to hold the power constant.

A successful Square Wave Mode operation consists of three sequential states:

Square Wave Ready State
Square Wave Ramp Up State
Automatic State

All of the following conditions must exist for the system to enter the Square Wave Ready State.

1. The system is in the Steady State Mode.
2. The SQUARE WAVE button on the console has been pushed.
3. The reactor power is less than 1 kW.

4. The reactor power is steady. "Steady" means that the rate of change of power is greater than a ±26 second period.

5. Air pressure is not applied to the transient rod drive mechanism.

6. The transient rod is fully down.

7. All shim rods and the regulating rod must be above the down limit.

Once all the conditions above have been met, the system will transfer to the Square Wave Ready State. The system will illuminate the SQUARE WAVE button on the console. It will also enable the transient rod FIRE button on the console.

The system is now waiting for you to press the FIRE button to bring reactor power up. Note that at this point you can still adjust the transient rod drive position manually.

When you press the FIRE button, the system will transfer to the Square Wave Ramp Up State and apply air pressure to the transient rod drive mechanism. The air pressure will force the transient rod upward, producing a sudden increase in power. This power increase is called square wave ramp-up.

The NM1000 provides power readings five times per second. If the power level reaches the demand power level (as set by the thumbwheel switch) within ten seconds, then the system will transfer to Auto State. If the demand power level is not reached within ten seconds, then a "Power Level Not Reached Timeout" error will occur. The system will transfer back to Steady State Mode and will display an error message at the CSC console.

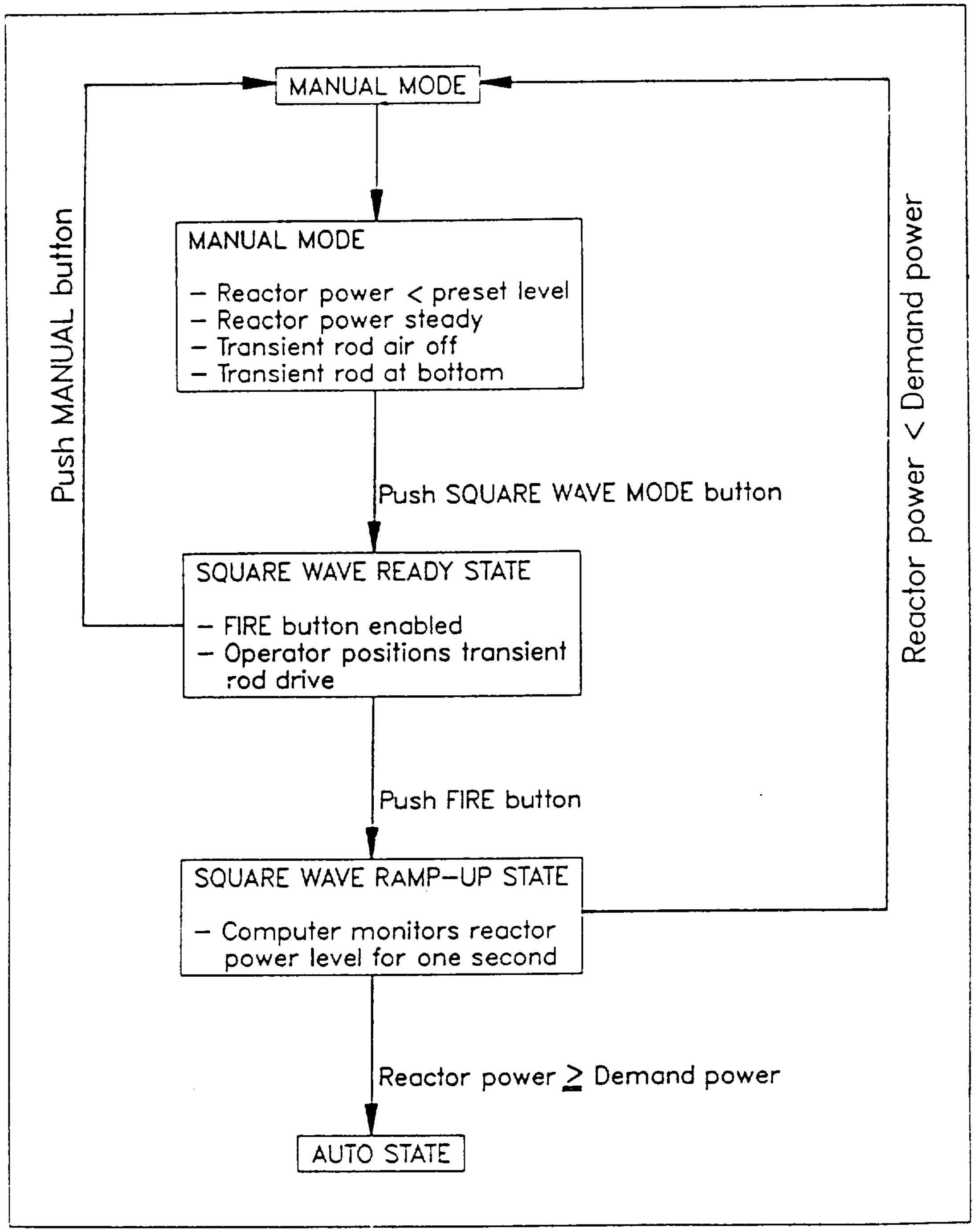

Figure 5.1 Flow Diagram for Square Wave Mode

Chapter 6
Pulse Mode

PULSE MODE

The Pulse Mode allows you to produce a very high power, short duration pulse from the reactor. Typical pulse power levels reached by a 250 kW reactor are 1500 to 2000 MW. You accomplish this pulse effect by firing the transient rod upward with compressed air.

During Pulse Mode, the system will taken 5000 power level readings over a 1/2 second period and store those readings on the hard disk. Once the data has been collected, you can view various calculated values of the data on the standard video display. You can also view a graph of the pulse on the high resolution video display (if you first SCRAM the reactor).

Recordings of the last ten pulse recordings are archived on the hard disk.

PREREQUISITE CONDITIONS

The following conditions must all exist before the system can enter the Pulse Mode:

1. The system is in the Steady State Mode.
2. Reactor power is less than 1 kW.
3. The reactor power is steady. "Steady" means that the rate of change of power is greater than a ± 26 second period.
4. Air is not applied to the transient rod drive.
5. The transient rod is all the way down.

If any of these conditions is not true, then the system will remain in the Steady State Mode. An error message will also be displayed at the CSC console.

OPERATOR ACTIONS

Assuming that all the required conditions (listed above) have been met, the system will enter the Pulse Mode when you do the following:

1. Position the transient rod drive to adjust how high the transient rod will go when you fire it.
2. Press the PULSE button on the CSC console.

3. You will now see this prompt on the standard video display:

   Enter Pulse ID String

   Type in any identifying name that you would like to give to the pulse. Make the same something descriptive, that you can recall easily later on. Once you have typed in the pulse name, press the ENTER key.

4. Press the FIRE button on the CSC console. This will apply air pressure beneath the transient rod, causing the rod to move upward rapidly. The rod will stop when it comes into contact with the rod drive mechanism and will then fall back down into the reactor after a preset time.

This rapid firing of the transient rod up out of the reactor will cause the reactor to produce a very high, very short-lived power pulse. The power pulse will reach its peak and decay on the order of milli-seconds.

While the transient rod is being fired upward, the computer switches into a high-speed data acquisition mode to record the reactor power pulse. The computer will record 5000 pcwer readings during a 1/2 second period. The system will drop the transient rod and return to SCRAM Mode once this pulse data acquisition period is over.

You will see a tabular summary of the pulse data on the standard video screen about 4 to 5 seconds after you fire the transient rod. A sample pulse data display is shown below in Figure 6.1. Once you have viewed this data, press SPACEBAR to return the display to the normal status window display.

```
                    PULSE DATA
                    ----------

Identification ......... 11250 $2.57 1-19-88
Number ................. 205
Timestamp .............. Tue Jan 19 14:32:37 1988

Peak Temperature ....... 346 C
Peak Pulse Power ....... 806 MW

Total Energy ........... 16.557 MW secs
Width at Half Power .... 18.200 msec

Minimum Pulse Period ... 5.332 msec
Reactivity ............. 2.580 $



  *** Hit the Spacebar to Continue ***
```

Figure 6.1 Sample Pulse Data Tubular Display

If you wish to see a graph of the pulse on the high resolution graphics display, you must first SCRAM the reactor by pushing the red manual SCRAM button. Since the data that appears in the summary screen also appears on the graphic display, the descriptions of the various pulse measurement terms are given in the section below, Displaying Pulse Data.

DISPLAYING PULSE DATA

Here is the procedure for displaying a graph of a pulse on the high resolution color graphic display.

1. You must SCRAM tne reactor before you can display a pulse on the graphics display.

2. Press the F3 key (Pulse Display) on the CSC console keyboard. You will see a screen display similar to Figure 6.2, showing a list of the last ten pulses recorded. Type in the number (0-9) of the pulse you wish to display. Then press the ENTER key.

PULSE SELECT MENU

SELECT	PULSE #	DESCRIPTION
0	20	8-14-88 $ 2.50
1	19	8-13-88 $ 2.56
2	18	8-12-88 $ 2.00
3	17	8-12-88 $ 2.57
4	16	8-11-88 $ 2.50
5	15	8-11-88 $ 2.57
6	14	8-11-88 $ 2.60
7	13	8-11-88 $ 2.21
8	12	8-10-88 $ 2.57
9	11	8-10-88 $ 2.25

ENTER YOUR SELECTION (0-9) ----------> ______
(HIT RETURN TO EXIT)

Figure 6.2 Pulse Select Screen Menu

3. Now the system will prompt you to select the format for the pulse display, graphic or nongraphic. If you select the nongraphic display, you will see a tabular display similar to that in Figure 6.1 above. If you select the graphic display, you will see an x-y graph of the pulse. Select the graphic display type for this example.

4. You will next see the screen display in Figure 6.3, giving you the opportunity to change the scaling parameters of the graph. Normally, you will just skip over these fields by pressing ENTER.

You can shift the graph to the left by typing in an X-axis offset. You can specify an offset of 0 to 450 milliseconds, in increments of 50 milliseconds.

You can change the range of the X-axis from 50 to 500 milliseconds, in increments 50 milliseconds.

You can change the range of the Y-axis from 0 to 2000 megawatts in increments of 500 megawatts.

PULSE SCALING

ITEM	X-AXIS OFFSET	X-AXIS MAX	Y-AXIS MAX
SCALING	0	100	1200
UNITS	MS	MS	MW
MAX VALUE	450	500	2400
RESOLUTION	50	50	400

DO YOU WISH TO ALTER PULSE SCALING (Y/N/ENTER) ---> ___

Figure 6.3 Pulse Scaling Data Menu

5. Specify the resolution of the graphic display, as shown in Figure 6.4. The pulse graph will come on the screen fastest if you choose low resolution. It will take the longest time to come up if you choose very high resolution. The pulse display will appear on the high resolution video display as an x-y graph, as shown in Figure 6.5.

PULSE GRAPHICS RESOLUTION

SELECT	RESOLUTION	CURRENT
1	LOW	
2	MEDIUM	<--------
3	HIGH	
4	VERY HIGH	

ENTER 1 TO 4 ---> ______

Figure 6.4 Pulse Graphics Resolution Menu

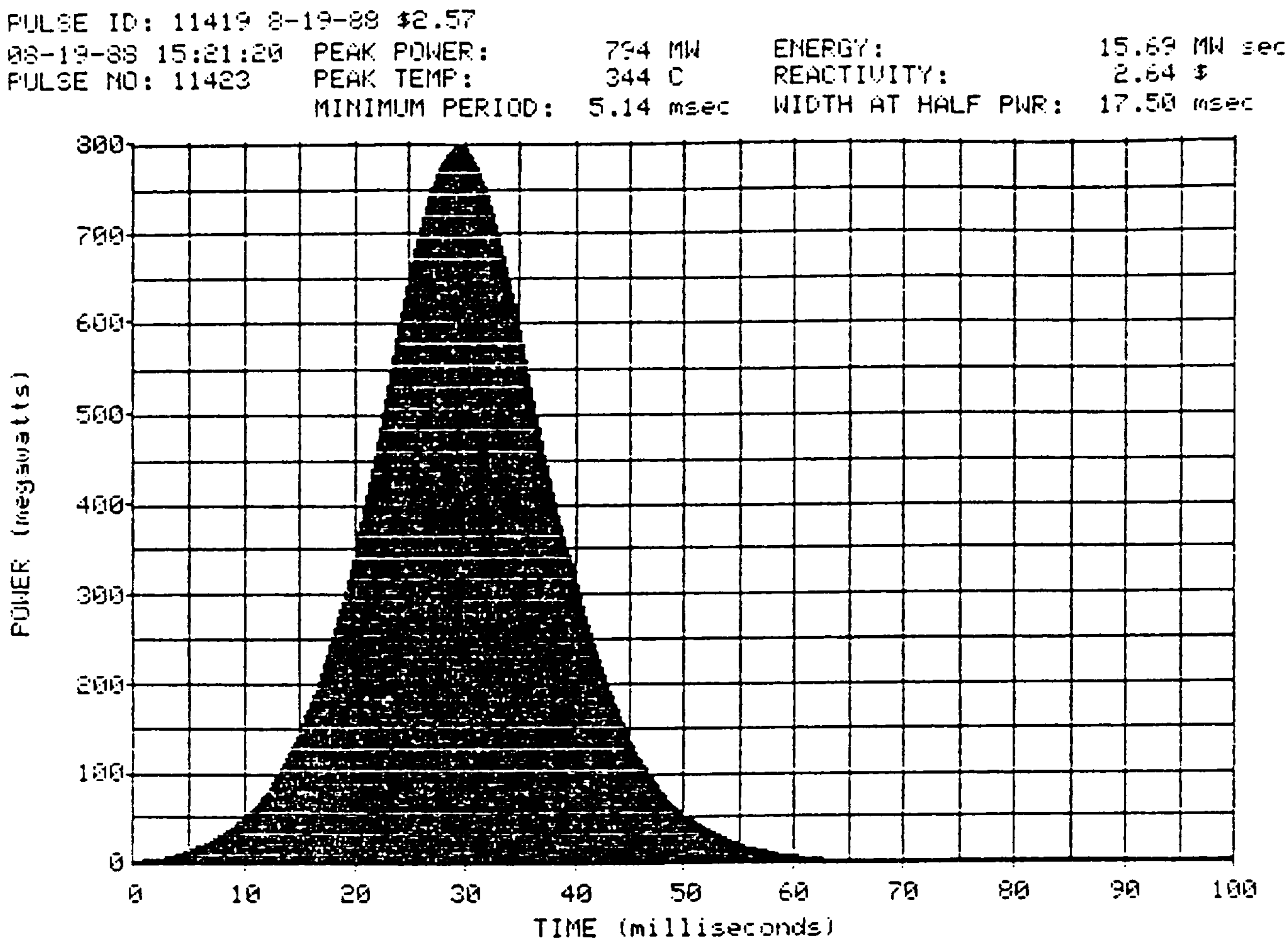

Figure 6.5 Pulse Data Graphic Display

Note that the various numeric values of the data are displayed at the top of the graph. These pulse parameters are explained below:

Pulse Identity	This is the name you gave to the pulse before you fired the transient rod.
Pulse Number	This is a sequential number assigned to the pulse by the computer. Pulse numbers vary from 1 to 30,000. The sequence will reset to 1 once the maximum count has been reached.
Pulse Timestamp	Date and time the transient rod was fired.
Peak Temperature	This is the highest fuel temperature reading (in °C) taken during the four seconds following the pulse peak. The peak temperature normally occurs about two seconds after the pulse peak.

Peak Pulse Power This is the highest digital value (in megawatts) of the 5000 power level readings.

Total Energy This is the total energy produced during the pulse period. It is calculated as an integration (i.e., the area under the pulse curve). The unit is measurement is megawatt-seconds).

Width at Half Power This is the width of the pulse at 1/2 the peak power reading. The unit of measurement is milliseconds (ms).

Minimum Pulse Period The minimum pulse period is a measure of the pulse at 1/2 the peak power reading. The unit of measurement is milliseconds (ms).

The minimum pulse period is defined as:

$$T_{\min} = (100x10^{-6})/In(P_{m+1}/P_m)$$

P_{m+1} and P_m are adjacent power readings in MW such that $(P_{m+1} - P_m)$ is the maximum change in power among the 5000 readings taken during the pulse. (100×10^{-6}) is the sampling interval of 100 microseconds.

Pulse Reactivity The reactivity parameter is a function of the minimum pulse period and of the reactor fuel type. The equation for calculating reactivity is:

Reactivity = (Fuel Type/Minimum Pulse Period) * (1/0.0070) + 1

Fuel Type = $20 * 10^{-6}$ sec (FLIP fuel)

Fuel Type = $45 * 10^{-6}$ sec (standard fuel)

Fuel Type = $69 * 10^{-6}$ sec (aluminum clad, graphite, reflector core)

Chapter 7

History Data Logging and Playback

HISTORY LOGGING

Whenever the reactor is in operation, a log of events and device states is automatically recorded on the hard disk. This data history logging begins whenever the system enters the Manual Mode (Steady State). Logging continues for five minutes following a system SCRAM, then terminates automatically.

Data is recorded approximately every 30 seconds. Also, any time there is a change in the Alarm Window, the Warning Window or the SCRAM Window, that event is also recorded, regardless of time.

HISTORY PLAYBACK

History playback takes the form of replaying "snapshots" of the two video display screens in sequence as they were recorded. Thus, you will see during playback the rod movements and bar graph displays just as they occurred during real time operation.

You can play back the recorded history either manually or automatically. When you play back the data manually, you will step through each recorded frame manually, by pressing a key on the keyboard. When you play back the recorded history automatically, the data will be displayed automatically in sequence, at a variable speed.

Manual Playback

You must first SCRAM the reactor before you can play back recorded history data. Press F4 on the console keyboard to access the History Data Archive Utility menu, as shown in Figure 7.1.

```
Historic Data Archive Utility

Enter the number corresponding to your choice:

1) Enter History Playback Mode
2) Back-up Archive Data to Floppy Disks
3) Restore Archive Data from Floppy Disks
4) Terminate 5 Minute Post-Scram Data Logging
5) Exit
```

Figure 7.1 Historic Data Archive Utility Menu

Acknowledge and clear any warnings by pressing the ACK button on the console. Then stop the post-SCRAM logging, if it has been less than five minutes since you scrammed the reactor.

Select Option 1 from the menu to play back the recorded log data. You will see the mode change to PLAYBACK-MAN FWD. This indicates that the system is in Manual Playback Mode, and that the direction of data playback will be "forward." You have various options available by pressing these keys on the keyboard:

F (lower case)	Play back data in the FORWARD direction, from the current frame toward the end of the history file. One frame per step.
Shift-F (upper case)	FORWARD playback, every sixth frame.
CTRL-F	FORWARD playback, every 60th frame.
ALT-F	FORWARD playback, every 360th frame.
R (lower case)	Play back data in the REVERSE direction, from the current frame toward the beginning of the history file.
Shift-R (upper case)	REVERSE playback, every sixth frame.
CTRL-R (upper case)	REVERSE playback, every 60th frame.
ALT-R	REVERSE playback, every 360th frame.
E (lower case)	MOVE TO END of archive file.

B (lower case) MOVE TO BEGINNING of archive file.

M MOVE TO MIDDLE of the archive file.

A (lower case) AUTO playback, one frame per second.

If you try to move beyond the beginning or end of the history file, then you will hear a beep each time you press the key.

Auto Playback

If you select Auto Playback mode, the system will display the recorded data history frames sequentially, at one frame per second. There are three playback speeds available. You can control the playback rate from the keyboard with these two keys:

Exiting Playback Mode

You cannot restart the system while it is in the Playback mode. Press the F4 key to exit from the Playback mode.

ARCHIVING HISTORY OR PULSE DATA

This group of utilities allows you to write history data log files and pulse data files to diskette (floppy disk), and to restore those files to the hard disk.

The system must be SCRAMmed and you must be logged on as an operator before you can access the data archiving utilities.

Backup to Diskette

Press the F4 key on the keyboard to select the Historic Data Archive Utility menu. Select option 2, "Back-up Archive Data to Floppy Disks." You will then see this prompt:

ENTER the number corresponding to your choice:

1. BACKUP pulse archive data.
2. BACKUP history archive data.

"Pulse Archive Data" consists of the data readings for the last ten Pulse Mode operations. "History Archive Data" consists of the data from the periodic logging of device states and console displays during normal operation (Manual or Auto Mode).

This is the prompt you will see after you have selected the type of data you want to archive:

ENTER the number corresponding to your choice:

1. 360 K Byte Diskette.

2. 1.2 M Byte Diskette.

SELECT the TYPE OF DISKETTE you are using. You will then see this prompt:

INSERT A FORMATTED DISKETTE IN DRIVE A:

STRIKE ANY KEY WHEN READY

Drive A is the diskette drive on the Control System Computer, which is located on the right side of the control console. You should use a 5-1/4 in. diskette, formatted from MS-DOS.

Now you will be prompted to give a name to the diskette:

You should select a name that will be meaningful to label the diskette, something which will aid you in remembering the contents of the diskette, or indexing it in your files.

After you have typed in a name for the diskette, press ENTER. The system will then copy the data files to the diskette, and display a message on the screen when it has completed.

Restore from Diskette

Press F4 to access the History Data Archive Utility screen menu. Select item 3, "Restore Archive Data . ." If an archive file already exists on the hard disk, it will be overwritten by the file from the diskette. You will get the following warning message to prevent accidental loss of archive data from the hard disk:

WARNING: The old archive file will be destroyed
(do you wish to continue? [Y/N])

If you choose Y, the system will copy the data archive file from the diskette to the hard disk. It will display a message when it has completed the operation.

If you choose N, then the operation will be aborted. Data files on the hard disk will not be altered.

OPERATION SUPPORT SYSTEMS

Table of Contents

1.0 REACTOR WATER SYSTEMS

Reactor water systems consist of the reactor pool, a system for water purification and a cooling system. Containment of water in the reactor pool is a fundamental requirement. This requirement assures convection flow for reactor operation and provides a radiation shield for personnel protection. Leakage of any pool water is also of interest to control potential contamination by any radionuclides present in the water.

The basic parameter that characterizes the reactor pool condition is the pool water level. Other system conditions of importance are the water purity and pool temperature.

The water purification and cooling systems for the TRIGA reactor serve several functions, as follows:

Purification System

a. Maintains low conductivity of the water to minimize corrosion of all reactor components, particularly the fuel elements.

b. Reduces radioactivity in the water by removing nearly all particulate and soluble impurities.

c. Maintains the optical clarity of the water.

Cooling System

a. Provides heat dissipation for the heat generated in the reactor.

b. Supplies a flow through the diffuser to increase the N-16 decay time.

All components in the water system are made of materials compatible with the aluminum in the reactor system.

1.1 Water Purification System

The skid-mounted purification system, as shown in Fig. 1-1 consists principally of a pump, flowmeter, fiber cartridge filter, and mixed-bed type demineralizer. Pipes valves and probes for measuring the conductivity of the water complete the system. Pool water from the reactor tank enters the system through a surface skimmer or a subsurface inlet. Water reenters the pool through a return line at the pool surface.

The surface skimmer collects foreign particles that float on the surface of the water in the tank. Water at the surface of the tank flows over the top of the floating portion of the skimmer, which collects these particles. Larger particles collect on the plastic screen in the floating part of the skimmer and the smaller particles pass through the skimmer and collect in the purification system.

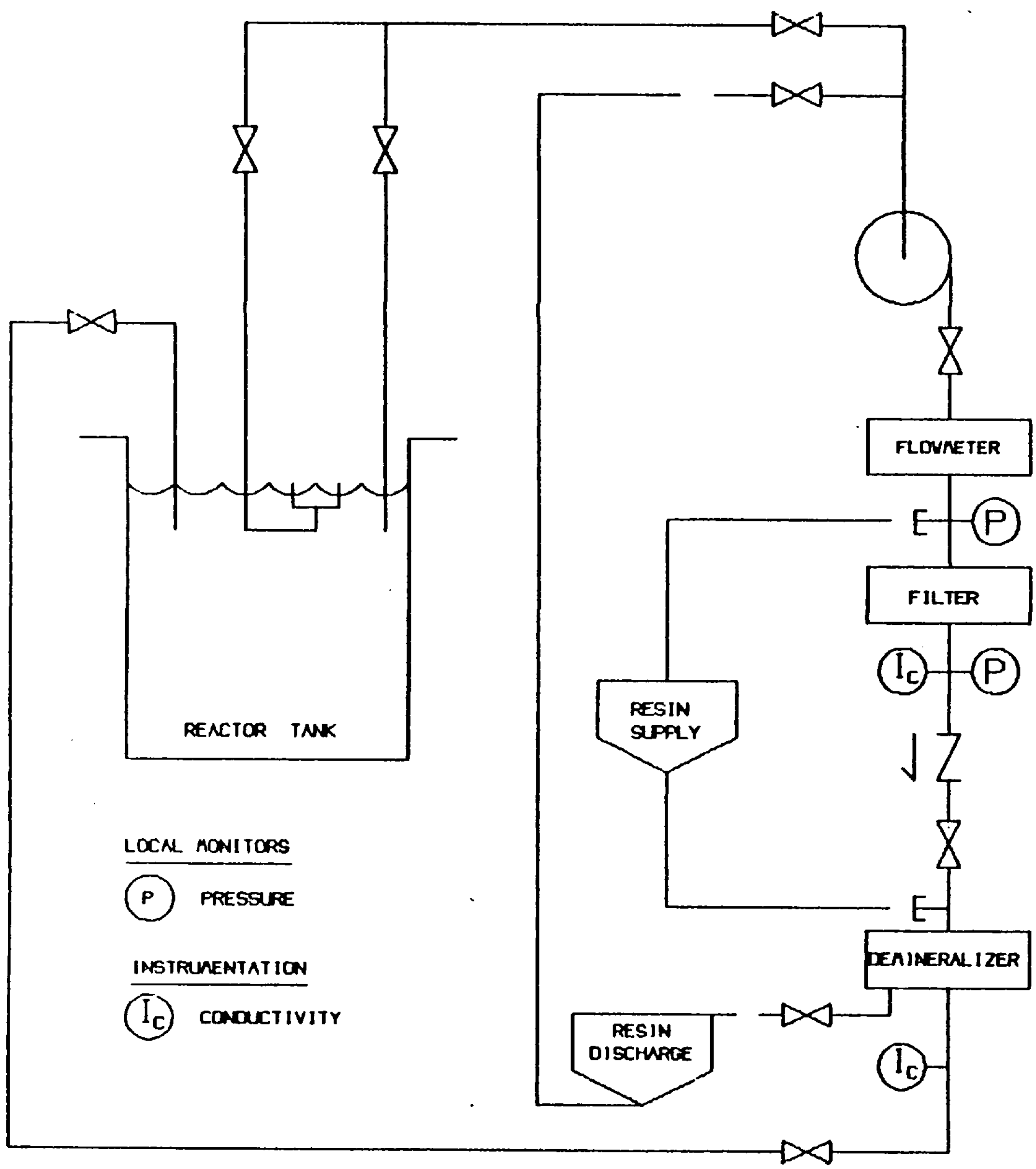

Figure 1-1. Reactor water purification system schematic

1.1.1 Purification System Pump

The water system pump is a centrifugal type. The pump unit is driven by a directly coupled induction motor, and will self prime. All parts in contact with the water are stainless steel. Mechanical seals prevent leakage of water at the pump shaft. Motor power is 3/4 horsepower single phase 208 volts.

1.1.2 Demineralizer

The prime function of the demineralizer is to remove soluble impurities from the water in order to maintain the conductivity of the water at a sufficiently low level to prevent corrosion of the reactor components. The demineralizer units used with TRIGA reactors are of the mixed-bed type, which removes both positive and negative ions from the circulating water. The positive ions are replaced by hydroxyl (OH) ions and the negative ions by hydrogen (H) ions. The OH and H ions combine to form water. Consequently, and contaminants in the water are concentrated on the resin and replaced by pure water. Each demineralizer unit contains an intimate mixture of anion resin and cation resin.

Resin deterioration is primarily by ion exchange. However, temperatures in excess of 120°F will also effect resin performance. Deposits of organic mateials may also effect resin exchange capacity. Replacement of the resin is to meet the specifications and instructions of the purification system manufacturer. This information is found in the appropriate operation and maintenance manual. Resin volume is 3 cubic feet (85 liters).

1.1.3 Filter

The filter element is a fiber cartridge that removes insoluble particulate matter from the reactor water system. The cartridges are replaced when they become clogged, rather than being back-flushed and reused. Filtration assists the control of corrosion processes and neutron activation by removal of particulates from the water system. In addition, by removal of solid particles the Filter improves the optical clarity of the water in the reactor tank and extends the operating life of the deminieralizer resin. A filter specification of at least 20 micron should be the rating limit. Other particle size ratings such as 5 or 10 micron would be acceptable replacements.

1.1.4 Pipes and Valves

All pipes, valves and fittings are schedule 80 plastic pipe products of polyvinyl chloride (PVC). Nominal pipe size of the inlet and outlet lines between the reactor pool and the purification skid is 1½ inches.

1.1.5 Flowmeter

An inline flowmeter on the purification skid monitors water flow rate through the resin bed. Flowmeter range is 8 - 20 gallons per minute (30-80 liters per minute).

1.1.6 Conductivity Probes

There are two conductivity probes in the water system. One probe, located upstream from the demineralizer, measures the conductivity of the water leaving the reactor pool; the other probe, located downstream from the demineralizer, measures the conductivity of the water as it leaves the demineralizer. These probes indicate whether the demineralizer is operating properly or the status of resin depletion. Each conductivity probe consists of a titanium electrode conductivity cell with plastic flow shield and pipe thread fitting for installation in the water system. The cells connect to a conductivity monitor on the purification skid. A thermistor for temperature compensation in each conductivity cell normalizes the conductivity to a reference temperature of 25°C.

1.1.7 Pressure Gauges

A pressure gauge on each side of the filter measures the differential pressure drop across the filter. The pressure drop provides indication of the filter effectiveness and will indicate the need for replacement of the filter cartridges.

1.1.8 Remote sensors

Several remote sensors monitor purification system parameters as part of the reactor instrumentation system. These remote sensors are pool water level, pool water temperature, and purification system conductivity. The pool system sensors are a switch for pool level and termistor for pool temperature. A signal from the conductivity meter on the purification skip to the reactor instrumentation system provides a remote indication of water purity conditions.

1.2 Primary Cooling System

The primary cooling system (see Fig. 1.2) consists of a pump, heat exchanger, temperature probes, flow meter, associated valves and piping, and N-16 diffuser.

Since oxygen is present in the reactor pool water, there is N-16 production in the reactor pool through the (n,p) reaction. However, the transport time from the reactor core to the surface of the pool permits much of the N-16 with its 7-sec half-life to decay before it reaches the surface. This decay time will increase by means of a diffuser nozzle that is a short distance above the top grid plate, and which directs a

portion of the cooling water downward over the core. Adjustment of the flow through the diffuser nozzle is by a valve near the surface of the pool.

1.2.1 Primary Cooling Pump

The primary cooling system pump is a centrifugal type. Parts in contact with water are made of stainless steel. Pump shaft seals are mechanical seals to control water leakage. Motor power is 7.5 horsepower, three phase 460 volt.

1.2.2 Heat Exchanger

A heat exchanger provides for the removal of 1000 kW of heat from the reactor pool water. All parts of the heat exchanger in contact with the reactor pool primary water are made of stainless steel. The water in the secondary cooling circuit flows through six passes on the shell side of the heat exchanger. A central chilling station supplies the heat sink for the secondary cooling capacity.

1.2.3 Secondary Cooling Loop

Maintenance and operation of the secondary chill water loop is part of the physical plant. A blending station for the heat exchanger controls the local temperatures and pressures of the chill water loop. Components of the blending station are a mixing valve, isolation valves, throttle valve, check valve, pipe, pump, and instrumentation. Components are of carbon steel construction. A 10 horsepower motor controls operation of the blending station.

1.2.4 Piping and Valves

All piping and fittings in the system are of aluminum alloy. The main water circuit piping to the pool is schedule 40 pipe with a nominal diameter of 4 inches. Pipe sections of the system connect with a groove, coupling and gasket system. Valves are ball type of stainless steel materials.

1.2.5 Temperature Probes

Temperature control of the secondary loop and local indication of the primary loop inlet and outlet temperatures depend on several temperature sensors. The controller sensor is a well type device. Both sensors for local indication in the primary pool, water loop are metallic conductor type thermometers.

Temperature probes for remote indication measure the temperature of the water before and after passing through the heat exchanger. Each probe is a resistance thermometer of the tip-sensitive type. To ensure long term stability, a small coil

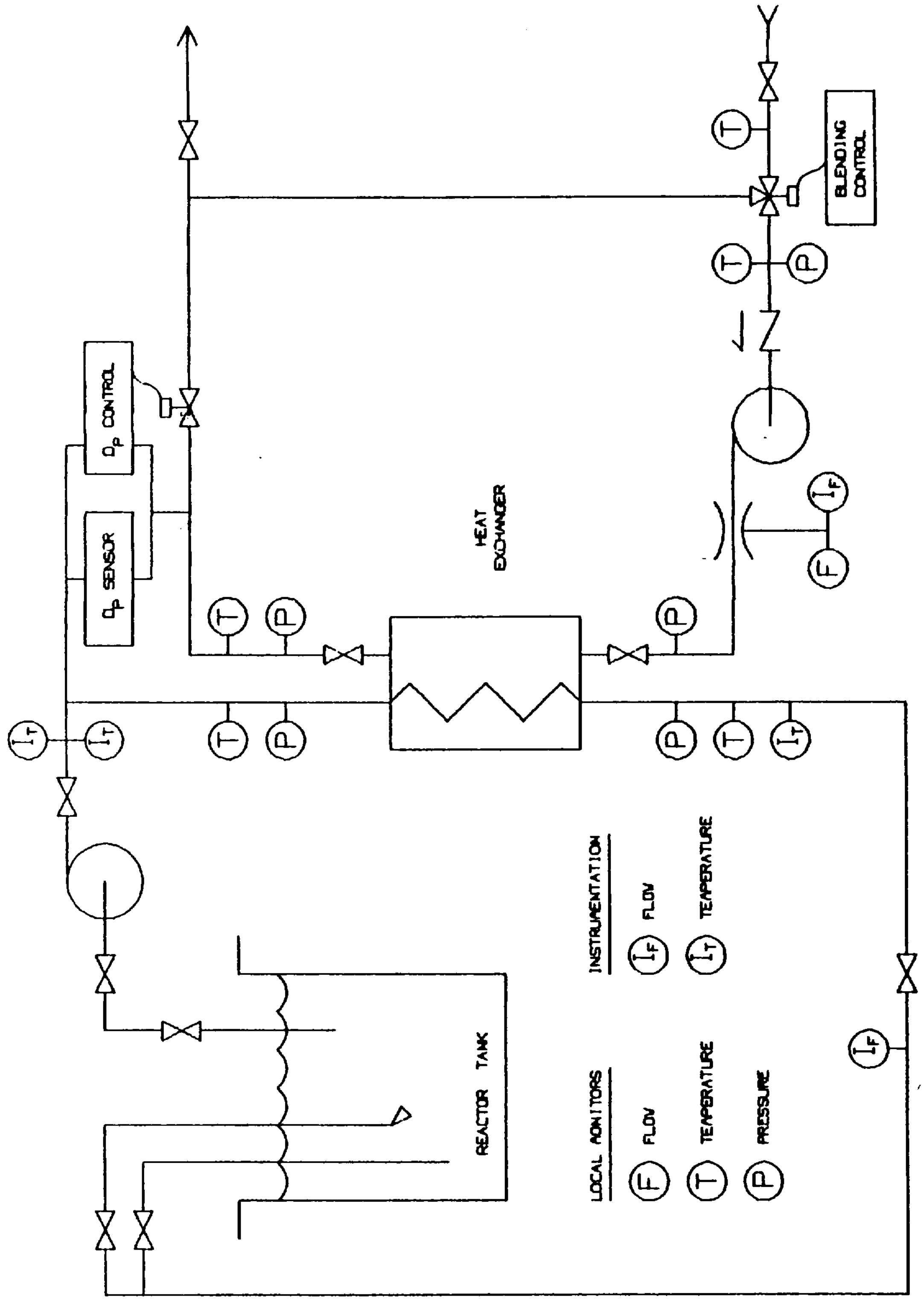

Figure 1-2. Reactor water cooling system - primary loop

of nickel alloy wire with a large temperature coefficient of resistivity is the temperature-sensitive element. The resistance of the coil changes as the temperature changes, thus giving the temperature indication. Remote readout goes to the reactor instrumentation system.

1.2.6 Pressure Control

Differential heat exchanger pressure control maintains a barrier against leakage of any radionuclides into the secondary chill water loop. Pressure control of a throttle valve in the secondary loop will maintain a set pressure differential at the heat exchanger. The two measurement points are the maximum pressure on the pool water side (primary inlet) and the minimum pressure on the chill water side (secondary outlet). The differential set point is about 5 psid. An alarm signal will provide indication of a loss of pressure differential (<4 psid) to the reactor instrumentation system.

Other pressure sensors with venturi orifices or pressure plates will provide measurement of flow within the primary and secondary loops. Square root signal processors convert the signal pressure differentials to signals that are proportional to the flow rate. A local readout indicates secondary loop flow. Remote indication to the reactor instrumentation system provides readouts for both the secondary and primary flow loops.

2.0 Air Confinement Systems

Air confinement systems are those structures and equipment that control the flow of air through the reactor bay. Components of the air confinment system are the structural features system and the air ventilation systems.

Structural features include walls, floor, roof, doors and other boundary penetrations. Air leakage control at door entryways, construction joints and boundary penetrations is by seals or sealents. Complete confinement such as a containment system with virtually no leakage is not the goal.

Ventilation equipment for the room includes two systems. One system provides HVAC functions and controls air exchange rates in the reactor bay. Another system provides an exhaust pathway to control release of gaseous effluents from the reactor pool area during both normal and emergency conditions. Isolation dampers in each ventilation system restrict the free movement of air in the air ducts if the air fans are not operable.

The air confinement systems in the reactor bay serve several functions, as follows:

HVAC System

a. Provides heating and cooling for personnel comfort and equipment requirements.
b. Controls room air humidity.
c. Maintains a pressure gradient between the reactor bay and other areas so that leakage is into the reactor bay.
d. Controls economic operation of the unit by recirculation of room air.
e. Controls the air exchange rate of the room by exhaust of all room air.

Argon Purge System

a. Controls the amount of argon-41 present in the reactor bay area during reactor operation.
b. Provides for collection, filtration, and measurement of other potential radionuclide hazards in reactor operation areas.
c. Allows control by initial isolation and subsequent dilution of the release of radionuclides that might be present in accident situations.

2.1 Heat, Ventilation, and Air Conditioning

A single HVAC unit controls air temperature and humidity in the reactor bay. The unit consists of supply fan, return fan, tempering coils, distribution ducts, isolation dampers, exhaust stack, and control system. System design capacity is 12,000 cfm. A schematic of the confinement system is shown in Figure 2 - 0.

2.1.1 Supply and Return Air Fans

Two fans, one for supply air and one for return air control the movement of air through the system. The fans are belt driven cabinet fans. Horsepower sizes of the fans are 7.5.

2.1.2 Tempering Coils

Both hot and cold water coils control the energy for room air comfort. The respective sources for hot and cold water are a boiler in the local laboratory building and a chiller at a remote central station.

2.1.3 Ventilation Duct

Air discharge into the reactor bay is through two circular ducts that run along opposite walls. Branch ducts with exit diffusers disperse the air at several elevations in the room. Return air ducts consist of two rectangualr return ducts as wall returns and two circular return points on the opposite wall near the floor.

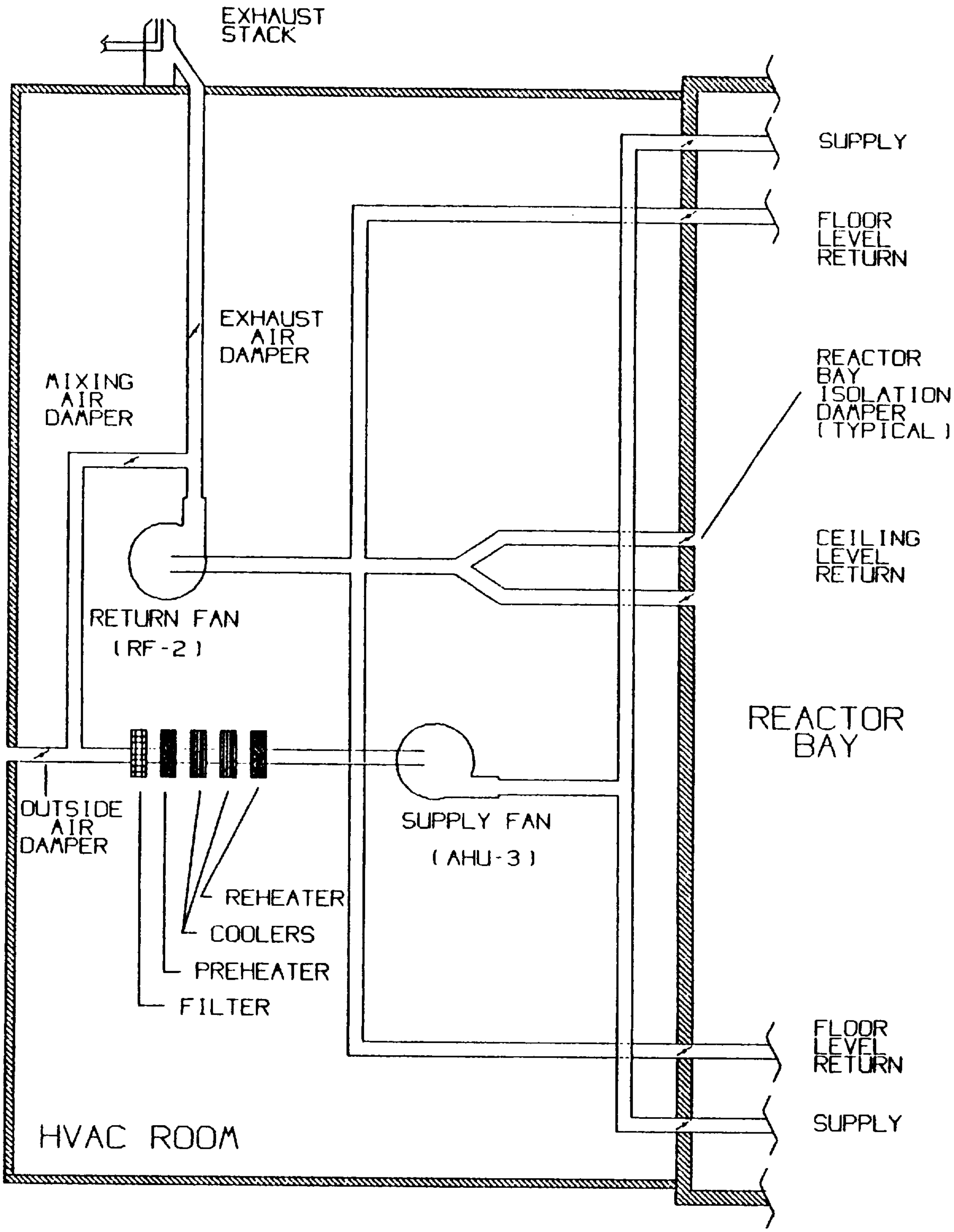

Figure 2.0 - Reactor Air Confinement

2.1.4 Isolation Dampers

Special dampers in supply air and in return air ducts will perform the function of room isolation. Each damper will fail to the shut position on loss of power. Switches on the dampers provide indication of the damper positions. An electrical signal from a relay closure will close the isolation dampers and initiate shutdown of the HVAC fans. A radioactivity setpoint will initiate the isolation signal.

2.1.5 Exhaust Stack

Room air exhaust from the reactor bay is through a stack on the building roof. Exhaust air will consist of partial exhaust by recirculation of air during periods in which the reactor is not in operation. Otherwise a complete exhaust of the room will exchange the room air at least twice per hour. The exhaust velocity will be 4800 fpm (24 mps) with an effective release height of 21 feet (6.5 meters) above the building roof elevation.

2.1.6 Control System

Flow control dampers, thermostats, pressure controllers, temperature controllers and safety switches are all parts of the control system. Basic system control depends on a frequency drive for the air handling fans. Through the control set points for temperature and pressure these drives determine the air flow rates and differential pressures between the reactor bay and other areas. The set point is .06 inches (0.15 cm) of water relative to the ambient atmospheric conditions. A differential value of .02 inches (0.05 cm) is the set point relative to building areas. Key control system features available at a panel in the reactor control room are low vs. high volume operation, isolation damper status and differential pressures.

2.2 Argon Purge System

The argon purge system controls the release and accumulation of argon-41. Argon-41 is a radionuclide that occurs by neutron irradiation of air. The system consists of an exhaust fan, filter fan, pipe, valves, controls and exhaust stack. System design capacity is 1035 cfm. A schematic of the argon purge system is shown in Figure 2 - 2.

2.2.1 Purge Fan

The purge fan is a centrifugal belt driven fan with an 11.5 horsepower motor. Location of the fan is on the roof in the vicinity of the exhaust stack. The fan induces a low pressure in all purge system lines to draw air into the system.

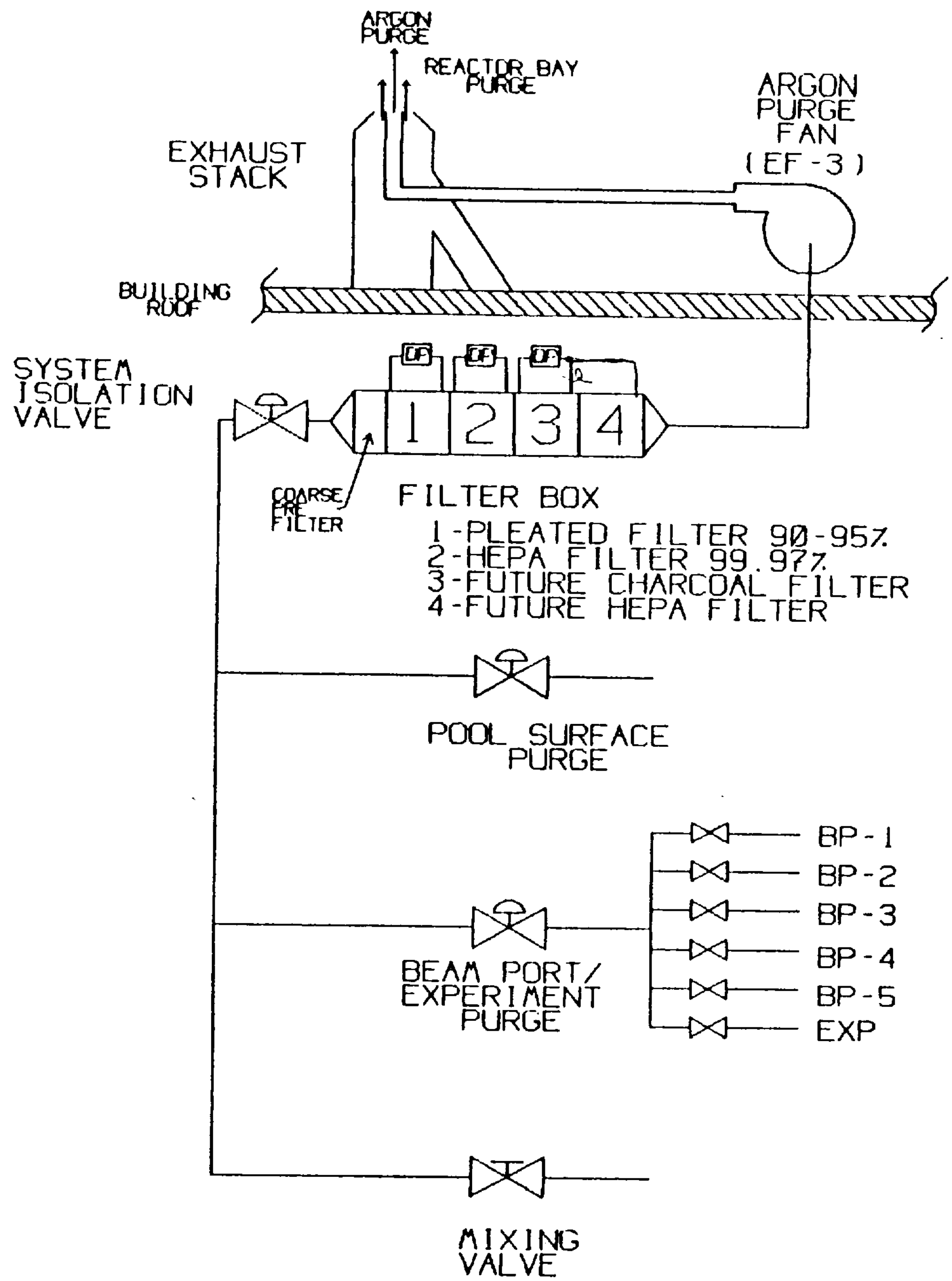

Figure 2.1 - Argon-41 Air Purge

2.2.2 Filters

A filter unit to remove particulates contain several filter types. A prefilter preceedes the main filters to remove airborne dust and debris. After the prefilter is a pleat type filter of 90-95% efficiency and then a high efficiency praticulate absorber of 99.97% efficiency. The first filter is to extend the life of the HEPA unit. Differential pressure measurements with magnehelic type gauges monitor the pressure drops across the filter sections to indicate filter condition. Another section of the filter unit is available to add other filter types such as carbon filters.

2.2.3 Pipe and Valves

Purge system pipe consists of a main section and three subsections for collection of air. The main system is 8 inch pipe with an 8 inch isolation valve. Mixing of room air through a 6 inch and 4 inch, with pneumatic actuators, determine the system air suction points. One is the reactor pool surface and the other is the beam tube manifold. Each beam tube has a separate valve that requires manual operation. Two sample supply lines and one return line complete the system to provide a method to monitor the effluent for radioactive nuclides.

2.2.4 Exhaust Stack

The argon purge system exhaust location is at the concentric center of the reactor bay exhaust stack. Mixing of the two exhausts which should have similar velocities occurs at the release point of the stack.

2.2.5 Control Features

Control features at the reactor control room panel are a fan operation with valve isolation and control position of two source valves. Status indicators monitor stack exhaust velocity, fan operation and isolation valve position.

Manual operation of an alarm circuit at the panel provides an alarm signal for evacuation of the reactor bay in the event of an emergency.

3.0 Experimental and Irradiation Facilities

Several experimental facilities are available in the reactor. A rotary specimen rack for production of long-lived radioisotopes is located in a well in the reflector assembly. A pneumatically operated "rabbit" transfer system, which penetrates the reactor core lattice, provides for production of short-lived radioisotopes. At the center of the core lattice is a central thimble that makes possible the extraction of a highly collimated beam of radiation or insertion of small samples into the region of maximum flux. Two radial, one tangential, and one thru beam port tube are available for neutron beam experiments. Beam tubes and the space in the water around the reflector can also be used for irradiation experiments.

3.1 Rotary Specimen Rack Facility

The rotary specimen rack is designed for experiments when one or more samples are to be irradiated for a significant length of time. The facility consists of the following components:

Rotary specimen rack (RSR)

Specimen removal tube

Drive and indicator assembly

Tube and shaft assembly

Specimen lifting assemblies

3.1.1 Rotary Specimen Rack

The rotary specimen rack assembly located in a circular well in the reflector assembly is shown schematically and in a photograph in Figs. 3-1 and 3-2, respectively. It consists of a ring-shaped, seal-welded aluminum housing containing an aluminum rack mounted on special bearings. The rack supports 40 evenly spaced tubular aluminum containers that serve as receptacles for specimen containers. Each receptcle has an inside diameter of 1.2 in. (31.7 mm) and a height of 10.8 in. (27.4 cm) and can hold two specimen containers.

The rack can be rotated around the core for the insertion or removal of samples. An alignment disk in the drive and indicator assembly allow allignment of each sample tube under the specimen removal tube. The internal rack may be rotated manually or by motor from the drive mechanism on the reactor bridge. Motion is transmitted to the rack through a drive shaft.

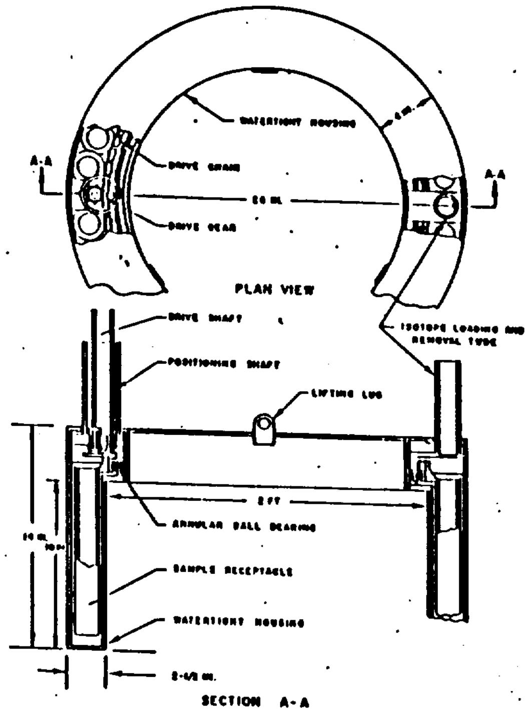

Fig. 3-1. Rotary specimen rack assembly (schematic)

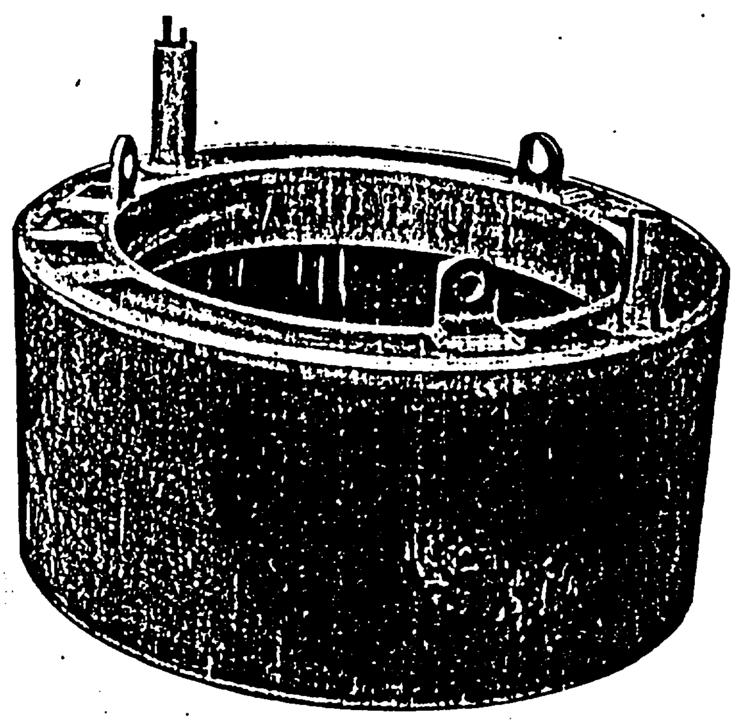

Fig. 3-2. Rotary specimen rack assembly

3.1.2 Specimen Removal Tube

The specimen removal tube, located 180 degrees from the tube and shaft assembly, terminates at the top of the reactor in a funnel located just below one of the top plates of the reactor bridge. The tube has an internal diameter of 1.3 in. (33.9 mm) and an axial offset of approximately 18 in. (45.7 cm) between the top and bottom of the tube to avoid direct-beam radiation from the core. Loading and unloading of the 40 specimen positions in the rack take place through this tube.

3.1.3 Tube and Shaft Assembly

The tube and shaft assembly consists of a straight aluminum tube enclosing the drive. It connects the rotary rack housing with the drive and indictor assembly on the reactor bridge at the top of the reactor tank. Since this tube is in a straight line from the reflector, shielding is provided by 5 ft. (1.5 m) of polyethylene in the tubing.

3.1.4 Drive and Indicator Assembly

The drive and indicator assembly is mounted on the reactor bridge. It has an indictor dial with 40 divisions (one for each rack position), a crank for rotating the specimen rack, a motorized gear train, and a locking rod handle. The motorized drive permits continuous rotation at about one revolution per minute. It consists of a fractional horsepower motor, a worm gear, and a slip clutch located inside the drive-and-indicator assembly box. Use of the motor assures a uniform average flux to all samples in the rack.

3.1.5 Specimen Lifting Assembly

Two specimen lifting assemblies are available for insertion and removal of specimen containers from the RSR.

One assembly shown in Fig. 3-3, is a standard fishing pole specifically modified for this purpose. The fishing pole enables the operator to keep isotopes at a safe distance and provides maximum flexibility during handling.

The other assembly shown in Fig. 3-4 consists of a hand crank reel mounted to a support arm attached to the reactor bridge. The support arm is capable of rotating in the horizontal plane allowing alignment above the unloading tube and subsequent swivel to a sample transport container.

Both systems utilize an electric cable attached to the reel. This cable serves as both a hoisting cable for the specimen container and a power conductor for actuating the specimen pickup tool. The pickup assembly is a small solenoid-operated, scissors-like device that fits into the upper end of the specimen container. The pickup solenoid is actuated from a button on the reel.

Actuation of the pickup assembly solenoid is normally required only when releasing a container from the mechanism; the container can be installed by manually pushing it over the scissors.

3.1.6 RSR Specimen Container

All materials placed into the RSR must be put into a special sample container such as that shown in Fig. 1-5. The container consists of a cylindrical body and screw cap molded from polyethylene. The cap is formed to fit the pickup tool which is used for loading and removing the containers from the rotary rack.

The polyethylene container is suitable only for experiments of short duration. An optional aluminum container is recommended for use in longer-term experiments and should always be used when fissionable or other heat producing materials are irradiated. This aluminum container has the same approximate dimensions and capacity as the polyethylene specimen container.

The containers may be used individually, or two container bodies may be screwed together and used as a single unit, to simplify handling. Either way the rotary rack will accommodate two containers in each of the forty positions.

<u>Container Dimensions</u>

Overall length-single	5-1/2 in. (14 cm)
Overall length-double	9-3/16 in. (23.3 cm)
Outside diameter (max)	1-1/4 in. (3.2 cm)
Inside diameter	15/16 in. (0.94 cm)
Usable inside height	3-1/2 in. (8.9 cm)
Total inside volume	2.4 $in.^3$ (40 cc)

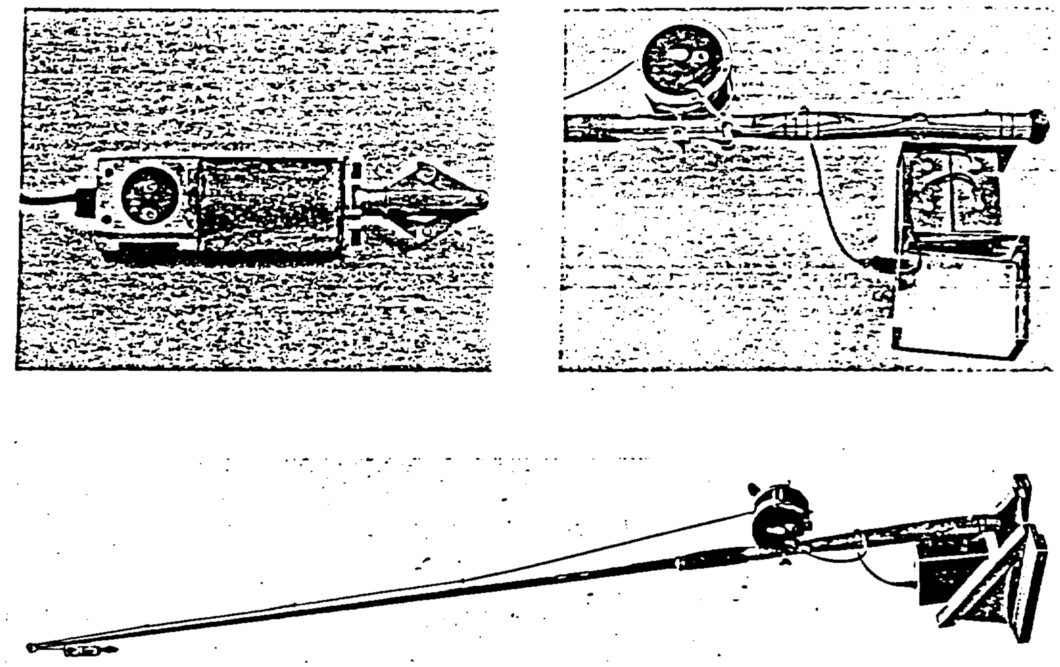

Fig. 3-3. Fishing pole specimen lifting assembly

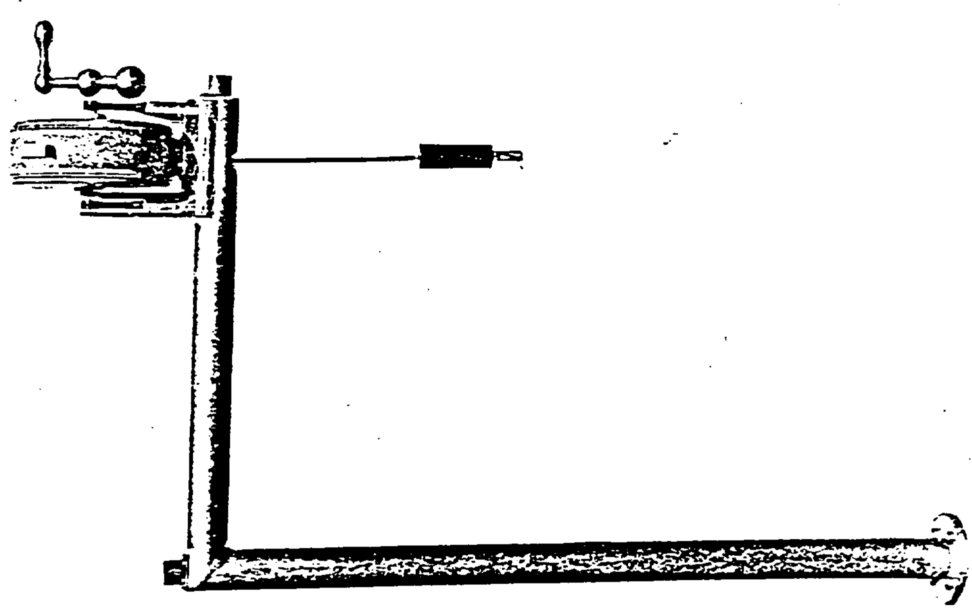

Fig. 3-4. Typical reactor bridge mounted specimen-lifting assembly

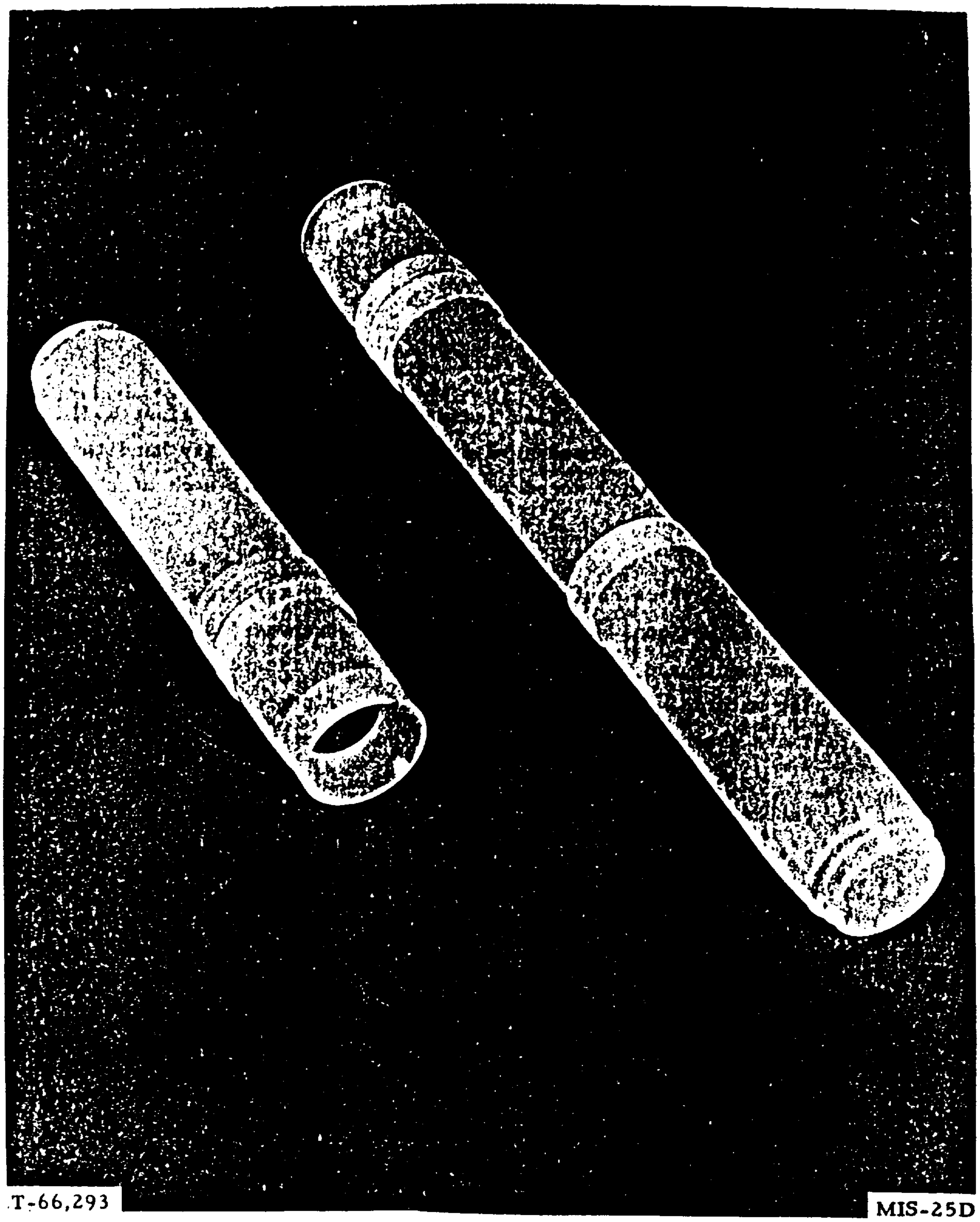

Fig. 3-5. Plastic irradiation specimen containers

3.2 Pneumatic Transfer System

Production of very short-lived radioisotopes is accomplished by a pnuematic transfer system, which rapidly conveys a specimen to and from the reactor core. The transfer system is a recirculating loop type with CO_2 purge to minimize the production of Argon 41. Primary components of the system include a blower, filter, solenoid valve, control box, and associated tubing, as indicated in Fig. 3-6. An optional specimen transfer box allows the sample movement or received from any one of several locations. Tubing from the blower extends to both the terminus assembly and the receiver-sender unit of the pneumatic transfer system. The blower provides a pressure differential for the injection or ejection of the specimen by means of a vacuum. Sample movement is achieved by operating solenoid valves which change air flow direction in the loop.

System operation is controlled by both the reactor operator and the experimenter. The reactor operator must authorize each sample insertion by the experimenter. Local control of the system is by the experimenter utilizing a control station near the sample send-receive station. The system may be operated in either a manual or automatic mode. In the manual mode, the experimenter must manually initiate sample insertion and removal from the reactor core. In the automatic mode, the sample is automatically ejected from the core after a preset exposure time.

All samples inserted into the transfer system must be placed inside a special polyethylene capsule or rabbit shown in Fig. 3-7. The capsule is placed into the sender-receiver station inverted, so that when it is injected into the core it comes to rest in an upright position approximately at the midplane of the core.

	Container Inside Dimensions
Diameter	2/3 in. (1.7 cm)
Length	4-1/2 in. (11.4 cm)
Volume	1.6 $in.^3$ (27 cm^3)

3.3 Central Thimble

The central thimble in the center of the core provides space for the irradiation of samples at the point of maximum flux. It also provides a highly collimated beam of neutrons and gamma radiation when the water is pneumatically expelled from the section of the thimble above the core.

The thimble is an aluminum tube with an inside diameter of 1.33 in. (3.38 cm). It extends from the top of the reactor tank through the central hole in the top and bottom grid plates and terminates in a plug below the bottom grid plate. Four 1/4 in. (6.3 mm) diameter holes are drilled in the tube at the top of the active lattice. These holes are used to empty the water from the tube for experimental purposes. The water is expelled by forcing it down and out with air pressure. The air is supplied by a hose attached to a special fitting at the top of the tube. The location of the holes prevents expulsion of the water from the section of the tube located within the active lattice.

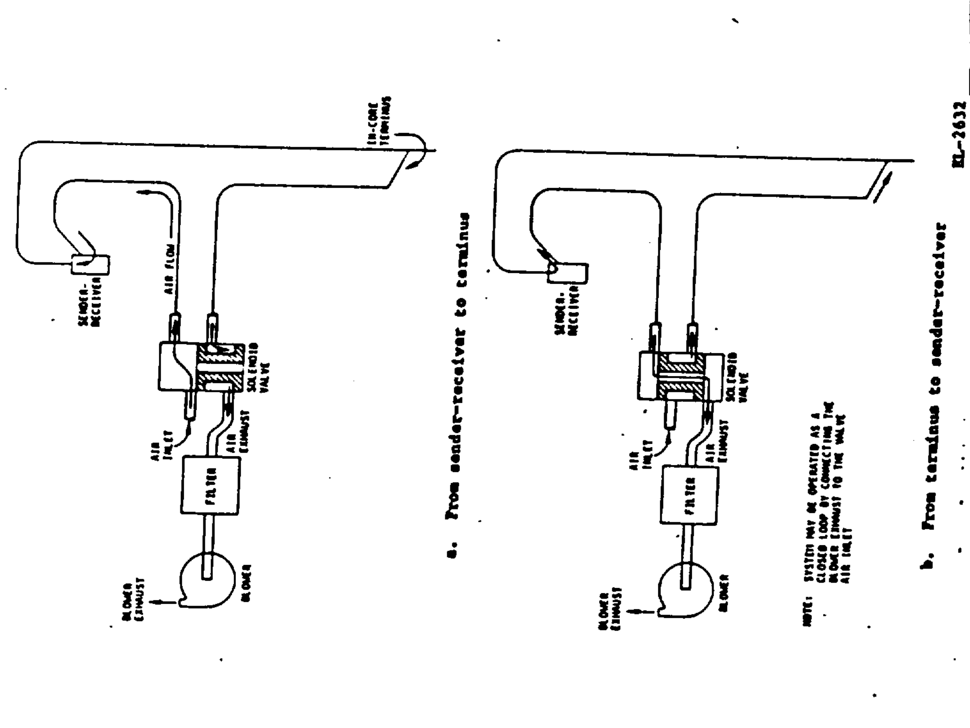

Fig. 3-6. Transfer of rabbit between terminus and sender-receiver

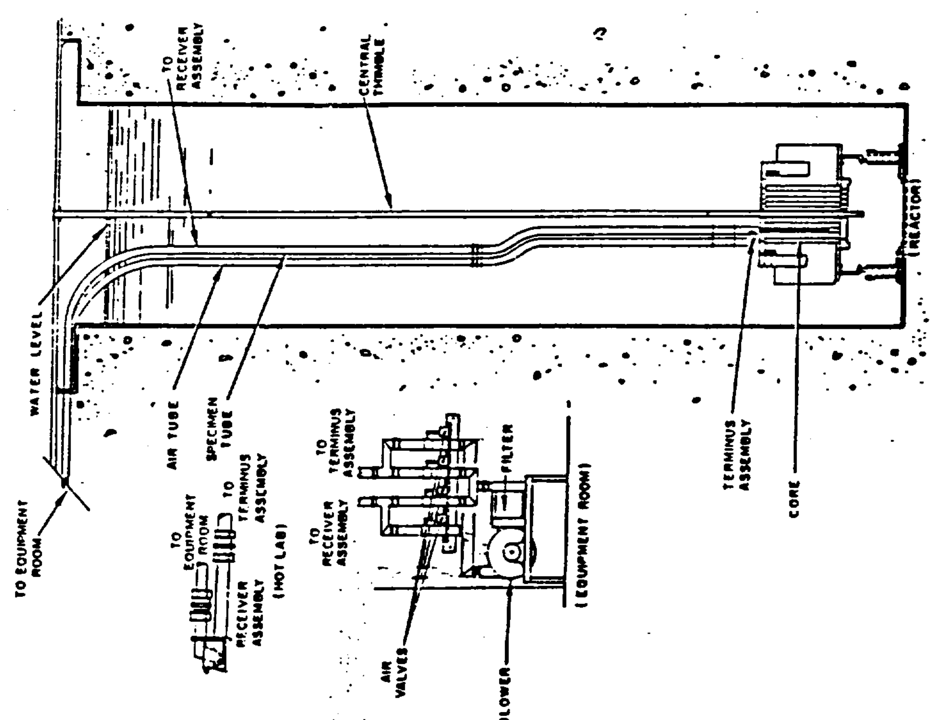

Fig. 3-7. Pneumatic transfer system schematic

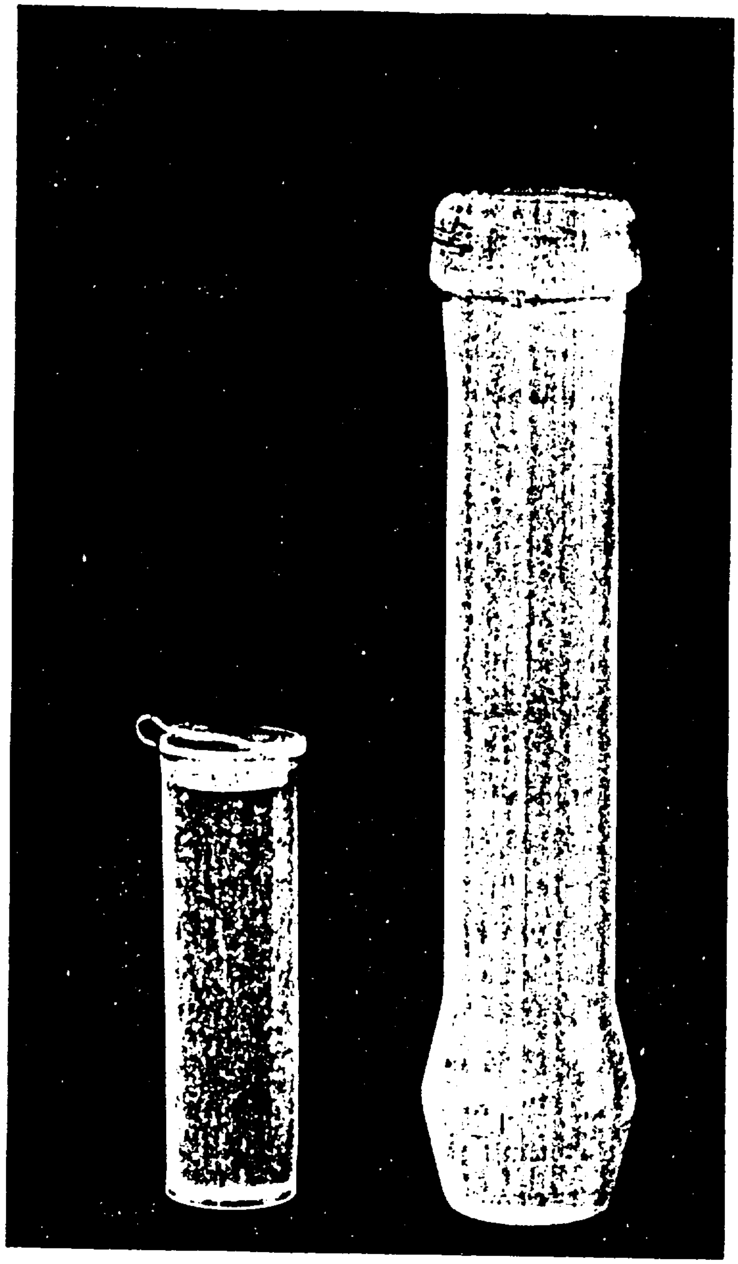

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Fig. 3-8. Specimen Rabbit used in Pneumatic Transfer System

3.4 Beam Port Facilities

The beam ports provide tubular penetrations through the concrete shield and the reactor tank water, making beams of neutrons and gamma radiation available for a variety of experiments. They also provide an irradiation facility for large (up to 6 inches in diameter) specimens in a region close to the core.

There are five 6-in.-diameter (15.2 cm) beam ports divided into three categories as follows:

3.4.1 Thru Port

Beam ports 1 and 5 are connected together end to end to result in one thru tube beam port. The tubes penetrate the concrete shield and are each coupled to opposite ends of a tube extending from the reflctor with a flexible bellows. The tube passes through the reflector tangential to the core and provides an open path allowing experiments to be inserted at one end of the thru tube and removed from the other.

3.4.2 Tangential Beam Port

This beam port, BP-2, is oriented tangentially to the outer edge of the core and penetrates the concrete shield structure and the reactor water, terminating at the outer edge of the reflector. A hole is drilled in the graphite tangential to the outer edge of the core.

3.4.3 Radial Beam Ports

There are two radial beam ports, each of which penetrates the concrete shield structure and the reactor water. The second radial beam port, BP-3, pierces the graphite reflector and terminates at the inner edge of the reflector. This beam port permits access to a position adjacent to the core. One radial port, BP-4, terminates at the outer edge of the reflector.

A step is incorporated into each beam port to prevent radiation streaming through the gap between the beam tube and shielding plug. Additional steps are incorporated into two beam tubes to provide space for divergence of the beam. The inner section of each beam tube is an aluminum pipe 6 inches in diameter. The outer sections of beam ports 1, 2, and 4 are formed of a stainless steel pipe 8 inches in diameter. Beam ports 3 and 5 outer sections are composed of 8, 12, and 16 inch diameter sections joined to form a diverging cavity. A "shadow" shield is incorporated into the concrete structure around beam ports 3 and 5. The shield consists of a _____ inch thick by _____ inch thick lead ring placed around the 16 inch pipe.

Special shielding is used in the beam tubes to reduce radiation outside the concrete shield to a safe level when the beam port is not in use. The shielding is provided in four sections as follows:

1. An inner shield plug.

2. An outer shield plug.

3. A lead-filled shutter.

4. A lead-lined plate and cover.

A typical inner shield plug, shown in Fig. 3-9 is about 48 inches in length. The shielding is formed by graphite, lead and steel. Rollers are provided to facilitate the insertion and removal of the inner shield plug. To help guide the plug over the step in the beam tube during insertion, the inner end of the plug is cone-shaped. A threaded hole is provided in the outer steel end for attaching the beam-tube plug-handling tool. A special spacer disk is used in the beam tubes to align and support the handling tool in the beam tube sections.

A typical, outer wooden shield plug is shown in Fig. 3-9. A handle on the outer end of this plug is provided for manual handling. Button-like protrusions are installed to reduce friction during the insertion and removal. The plug is equipped with an electrical circuit consisting of a position switch mounted in the front of the plug and an electrical connector at the rear of the plug. The switch can be actuated only by the inner plug when the inner plug is installed in the beam tube. The connector is inserted in an outlet box mounted in the beam tube to complete the circuit to the control console. A display on the console for each beam port indicates if the plugs are in place.

The lead-filled shutter and lead-lined plate provide limited gamma shielding when the plugs are removed. The shutter is contained in a rectangular steel housing recessed in the outer surface of the concrete shield. The shutter is operated by a removable push rod on the face of the shield structure. Insertion of the rod through a valve assembly on the side of the shutter system allows manipulation of the shutter without removal of shutter coverplates. The shutter housing is covered by a steel plate lined with 1-1/4 in. (3.2 cm) of lead for additional shielding. The plate and cover are equipped with a rubber gasket. The plate is normally left bolted in place at all times while a small circular cover is removed when the beam port is in use. The cover can be bolted shut and sealed to prevent loss of shielding water if the beam tube should develop a serious leak.

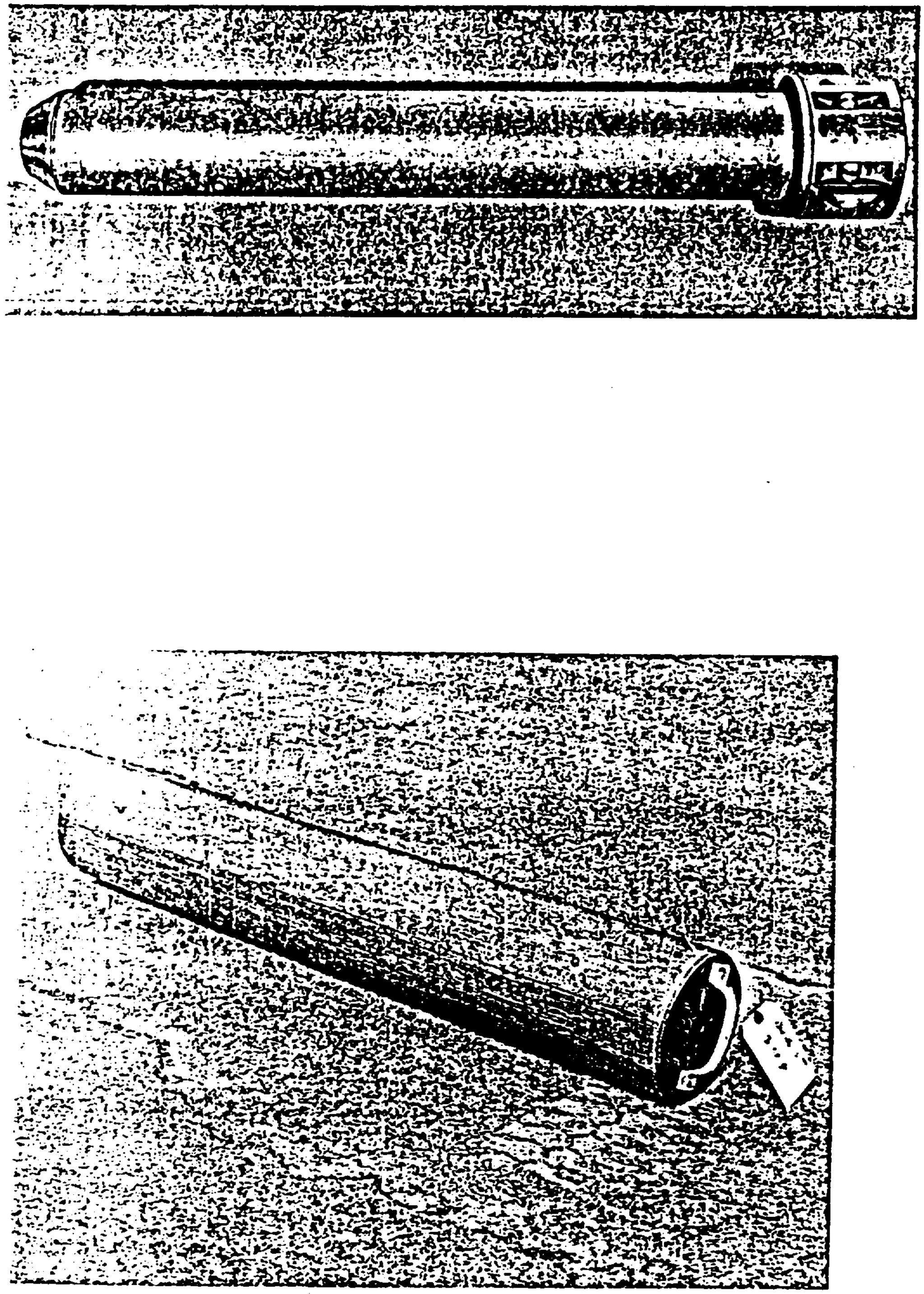

Fig. 3-9. Typical inner and outer shield plug

4.0 Equipment, Accessories, Tools and Other Items

4.1 Utility Equipment

4.1.1 Pool Light Assembly

Illumination of the reactor core is provided by three waterproof lights in the reactor tank. Each light assembly consists of a 300-W light enclosed in a waterproof aluminum housing. An aluminum pipe, which extends from the housing to the top of the reactor tank, forms a support for the housing and a waterproof conduit for the electrical wiring. The lights are supported from the aluminum angle at the top fo the tank by an aluminum clamp assembly.

4.1.2 Bay Bridge Crane

A bridge crane assembly consisting of a twin beam bridge, a trolley, and a 5 ton hoist is available for movement of heavy materials in the reactor bay. The moveable bridge is supported by two wall mounted rails on opposite sides of the room near the ceiling. The hoist is mounted on a moveable trolley supported by the bridge allowing access to all areas of the reactor bay. A catwalk platform with handrail supported by the bridge is available for use in activities.

The bridge crane is operated from a moveable pendant suspended from the bridge structure. The trolley and bridge are driven by electric motors coupled to a gear drive and brake assembly. The hoist is equipped with both a mechanical brake and an electric brake. All drive motors are reversible and two speed allowing precise load placement.

4.1.3 Fuel Element Storage Racks

Fuel storage racks each capable of holding 6 fuel elements, are located underwater along the walls of the reactor tank to provide temporary storage of fuel-moderator, graphite dummy elements, or other experiment components. Two 3/8 in. diameter, aluminum rods are used to suspend each storage rack. These rods are fastened at their lower ends to the rack and at the upper ends to brackets on the tank lip angle. If desired two storage racks can be stacked, one vertically above the other with a slight horizontal offset to allow access to the elements in the lower rack.

Circular storage racks with 19 element capacity are available for storage of fuel elements in the fuel storage wells or for additional storage space in the pool. These racks may be supported by either cable or rod. The floor storage wells are 16 feet deep by 10 inch diameter with provisions for water circulation lines. Closure of the wells is by a 6" thick lead shield plug and locakable stainless steel bar.

4.1.4 Pool Stirrer

The pool stirrer is used to mix pool water during reactor power calibration. Pool mixing produces a more uniform pool temperature by reducing pool thermal layering and thermal columns due to water bouyancy differences. The pool stirrer consists of an electric motor driving a stainless steel shaft and impeller. The assembly is clamped to an approriate support at the pool surface with the impeller submerged in the pool.

4.2 Fuel Element Equipment

4.2.1 Fuel Element Handling Tool

The fuel element handling tool, shown in Fig. 4-1, is used for repositioning the fuel-moderator and graphite dummy elements and the neutron source holder. Made of stainless steel, this tool consists of a grapple mechanism, a weight, a handle and grapple release, and a flexible control cable.

4.2.2 Fuel Element Inspection Tool

The fuel element inspection tool is used to accurately inspect a pulsing fuel element for longitudinal growth and changes of element bow. Strain gauges mounted on the tool allow for simultaneous measurement of both bow and length.

The upper support plate of the tool is bolted to the aluminum angle at the top of the reactor tank and extends downward approximately 10 ft. (3.7 m) into the tank, permitting the inspection of an irradiated fuel element and providing approximately 7 ft (2.7 m) of shielding water over the element. All parts of the tool to be in contact with water are either aluminum or stainless steel. The aluminum support tube structure has holes at the bottom and the top to allow shield water to fill the interior of the pipe.

The length of an element is measured utilizing a strain gauge attached to an arm supporting a small roller which rides on the top of the triangular space at the top of the element. The bow of an element is measured utilizing a strain gauge attached to an arm supporting a small roller which rides on the surface of the element cladding near the center of the active area.

A swinging arm assembly supports the strain gauge mounting arms. The arm is swung back and an element is placed on the support bearing at the bottom of the tool. The arm is then swung into place, engaging the element and the strain gauge roller assemblies, and latched in place. A rubber roller engages the element allowing element rotation for average length and bow measurement.

4.2.3 Fuel Reference Element

A standard element is furnished with the inspection tool. It is a solid piece of aluminum with the same dimensions and the same top and bottom end fixtures as those on a regular fuel element. The top and the side of the element have small steps machined to allow

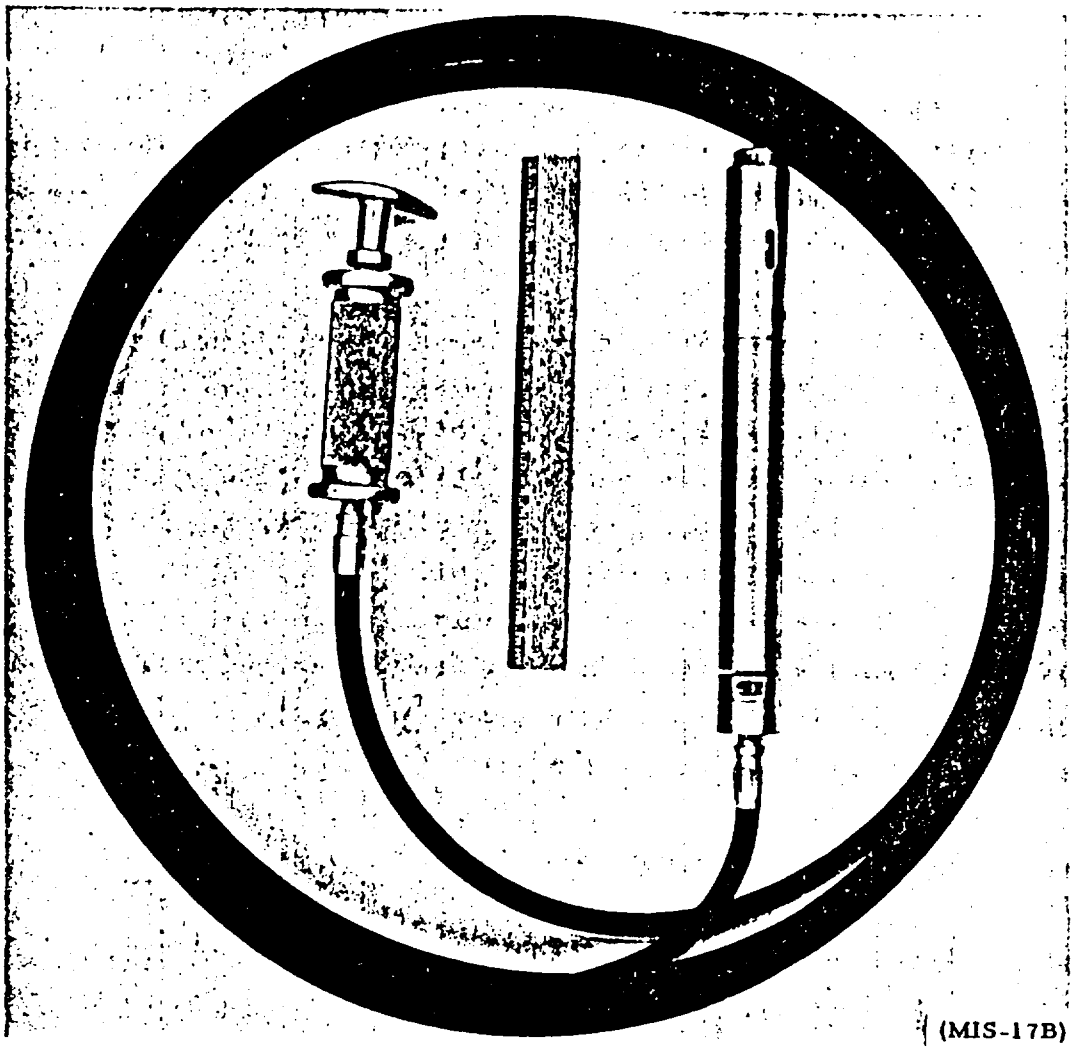

Fig. 4-1. Typical fuel element handlling tool

calibration of the strain gauge amplifiers. The length step are 0, +0.050, +0.100, and +0.150 inch with respect to the design length of 25.125 inches. The bow steps are 0, -0.010, -0.020, -0.030, -0.040, -0.050, -0.060, and -0.070 inch from a zero bow condition.

4.2.4 Fuel Element Supports

Special support for fuel elements are required in locations where the lower grid plate has cutouts for alternate location at control rods. All of the cutouts not containing control rods must have a support spacer installed. The spacer which sets on the safety plate is provided to support a fuel element in the proper core position. Failure to install this support will allow the fuel element to pass out of the core through the grid hole.

4.2.5 Fuel Element Adapter

A special adapter is required on fuel which has only a straight pin on the bottom of the element. This adapter is required to support the elements at the correct height in reactor and allow convection flow of coolout through the support grid plate. The adapters are installed on the elements over the existing bottom pin. The adapter design prevents the adapter from dropping off or being removed after initial installation.

4.3 Core Grid Components

4.3.1 Graphite Dummy Elements

Graphite dummy elements occupy the grid positions not filled by fuel-moderator elements and other core components. These elements are of the same dimensions and construction as the fuel-moderator elements, but are filled entirely with graphite and are aluminum clad. Each graphite dummy element weighs 2.8 pounds (1.3 kg), and is anodized after it is assembled.

4.3.2 Source Holder

The source holder shown in Fig. 4-2 is an anodized aluminum rod assembly, with a cavity to contain the neutron source. The dimensions of this assembly permit it to be installed in any of the fuel locations in the core, but it generally occupies one of the outermost positions. A 0.093-inch (2.4-mm) diameter hole is drilled through the top end fitting, permitting a long stainless steel wire to be looped through for ease of handling. The source holder is cylindrical, with a small shoulder at the upper end. This shoulder supports the assembly on the upper grid plate, the rod itself extending down into the core region. The rod clears the lower grid plate by about 1/2 inch (12.7 mm). The neutron source is contained in a cavity in the lower portion of the rod assembly and is located approximately at the vertical center of the core.

4.3.3 Grid Plate Center Cutout

4.3.4 Control Rod Grid Tube Tool

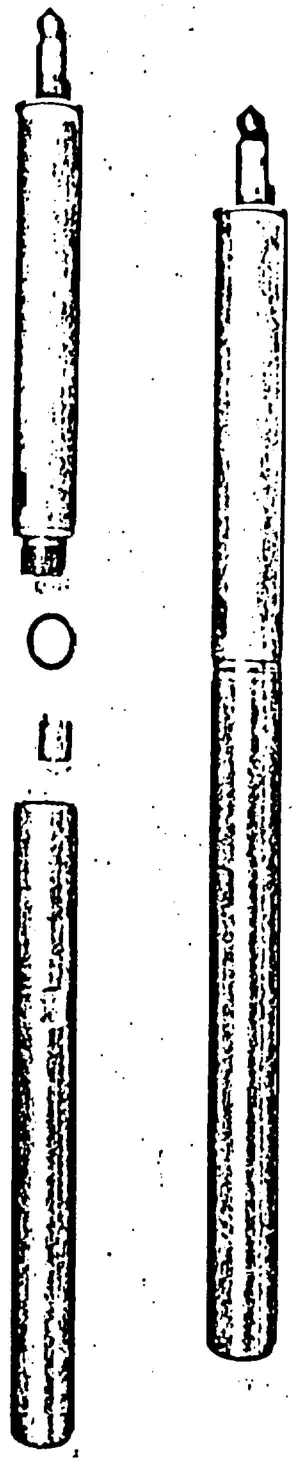

Fig. 4-2. Source holder

4.4 Beam-Port Shield-Plug Handling Equipment

The beam-port shield-plug handling system is designed to insure personnel safety during insertion and removal of the beam-tube inner shield plug by providing necessary shielding from the very intense gamma radiaiton streaming from the tubes while the reactor is shut down. This system consists of the following items:

1. Cask and carriage.

2. Removal tools.

4.4.1 Plug Cask

The cask and carriage, shown in the photographs of Fig. 4-3, is a portable lead-shielded container that holds the beam-tube inner shield plug during insertion and removal. The cask rests in yokes, which are a part of the carriage assembly. The carriage assembly is on wheels and can be moved about to position the cask at any beam port. Vertical movement of the cask is provided by the power-operated scissors lift mechanism of the carriage. The cask and carriage assembly weighs approximately ~~2,000~~ 4,000 lbs. (900 kg).

4.4.2 Plug Removal Tools

A handling rod with two different diameter end adapter plates is used to remove and install the inner beam tube plugs. The end plates must be choosen to match the size of the outer end of the plug being removed. A support plate must be installed in the bottom of the large diameter beam ports to support the inner shield plug in the outer larger diameter areas of the access tube.

4.5 Irradiation Holders

4.5.1 RSR Rabbits

RSR rabbits are specially designed polyethylene or aluminum containers in which all samples irradiated in the rotary specimen rack are encapsulated.

4.5.2 PNT Rabbits

PNT rabbits are specially designed polyethylene containers in which all samples irradiated in the pneumatic transfer system are encapsulated.

Fig. 4-3. Cask and carriage

Chapter V

Chapter V

RADIOLOGICAL SAFETY

In the TRIGA Reactor Facility, as in any reactor facility or laboratory conducting research with sources of ionizing radiation, suitable precautions must be taken against exposure of personnel to radiation and contamination hazards.

Health physics operations at the facility encompass such activities as environmental surveys, personnel monitoring, bio-assays, and routine monitoring operations. This chapter, however, is limited to a discussion of radiological safety activities directly associated with TRIGA reactor operations. The discussion is divided into five sections:

1. Radiological safety terminology and regulations.
2. Potential radiological hazards at the TRIGA Reactor Facility.
3. Health physics instrumentation.
4. Radiological safety equipment and procedures.
5. Decontamination and waste disposal procedures.

Although the health physics group is responsible for administering and supporting radiological safety procedures, operating personnel are primarily responsible for radiologically safe operation of the Facility.

5.1. RADIOLOGICAL SAFETY TERMINOLOGY

A glossary of radiological safety terminology has been provided in Appendix to assist the reader in acquainting himself with some of the terms and concepts used in describing some of the health physics techniques and equipment employed at the TRIGA Reactor Facility. This glossary should be used as a reference in studying the materials contained in this chapter.

5.2. RADIOLOGICAL SAFETY REGULATIONS

In operating the TRIGA Mark I reactor, the primary consideration must always be safety.

Both NRC and State of Texas regulations are used as guides for the conduct of health physics operations at the facility . These regulations* describe in detail:

1. Personnel exposure limitations and record requirements.
2. Permissible concentrations of radioisotopes in air and water.
3. Radiation warning signs and symbols.
4. Waste-disposal requirements.

Reprints of both NRC and State of Texas Radiological Safety Regulations are available.

5.3. POTENTIAL RADIOLOGICAL HAZARDS

The operation of the TRIGA reactor does not in itself constitute a radiological hazard. However, some potentially hazardous sources are generated as a result of its operation. Operating personnel must be cognizant of these sources and the manner in which they should be treated. A discussion of these potential hazards is the burden of this section.

5.3.1. Fuel Elements

Irradiated fuel elements are the strongest source of radiation within the reactor. When the reactor is at power, a fuel element is a source of intense gamma and neutron radiation. After shutdown, fission-product decay in the fuel continues to make the element a strong beta-gamma source, although the beta radiation is, for the most part, absorbed in the fuel cladding. Calculations indicate that if the reactor is operated at 100 kw for several hours the equilibrium activity associated with one of these

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*Title 10, Chapter 1, Part 20, Code of Federal Regulations, "Standards for Protection against Radiation"; "Texas Regulations for Control of Radiation".

elements is approximately 10^4 curies at the time of shutdown. The calculated dose rate in air from a single fuel element after prolonged operation at 100 kw and at a distance of 6 ft is approximately 10^3 r/hr at the time of shutdown.

In operating the TRIGA reactor it is often necessary to transfer fuel from the reactor tank to storage. Because of the radiation level associated with irradiated fuel-moderator elements, elements are normally kept under water for shielding. If an element must be removed from the reactor tank, it is placed in a shielded transfer cask to reduce radiation levels to tolerable limits.

In addition to the external radiation hazards present during fuel transfer operations, loose oxides and other irradiated particulates on the surface of the fuel-element cladding constitute a potential contamination hazard. This tactile contamination can be transferred to handling equipment and eventually to personnel if improper handling techniques are employed.

Contamination problems associated with transfer of a damaged fuel element are cause for even greater concern. Care must be taken to contain any liquid that drips from the fuel element or the transfer cask, personnel should wear protective clothing, and a continuous air monitor should be used to detect any particulate or gaseous activity that may become airborne.

Equipment used in the transfer of a ruptured fuel element must be carefully handled to avoid spreading the contamination.

5.3.2. Fuel-element Cladding Failure

Should activity be detected with no experiments in the reactor, it is likely that the activity originates in the core, either from volatile

contaminants located there or from a break in the cladding of a fuel element. It has been found that most of the fission-product activity released as a consequence of a cladding rupture is trapped in the reactor coolant.

Experience has also shown that a fuel-cladding rupture may result in the release of some airborne activity in the immediate vicinity of the reactor tank. Such a release would be detected by the continuous air monitor located near the tank. A cladding defect occurring during a reactor transient will result in a sudden increase of the air-monitor counting rate. The magnitude of the increase will depend on the size of the failure, but even a very small break will cause the air-monitor counting rate to increase by several thousand counts per minute. During steady-state operation, the air-monitor counting rate will increase more gradually.

The airborne activity detected consists predominantly of rare-gas fission products. Their daughter isotopes will collect on the air-monitor filter paper. Initially, 18-min Rb^{88} (daughter of 2.8-hr Kr^{88}) will predominate. Following the decay of rubidium, Sr^{88}, Cs^{139}, and Ba^{139} will be encountered.

5.3.3. Isotopes and In-core Experiments

Several potential radiation and contamination hazards are associated with the production of radioisotopes or the irradiation of in-core experiments.

Calculations indicate that the reactor is capable of producing an equilibrium concentration of radioisotopes of approximately 800 curies in the rotary specimen rack if the reactor is operated at 100 kw. The operating time required to produce an equilibrium concentration of 800 curies is a function of the specific isotope being irradiated.

With the 800 curies equally distributed in 40 sample positions, the maximum amount of activity which can be withdrawn at one time from the rotary specimen rack is approximately 20 curies. A 20-curie source constitutes an <u>intense</u> source of radiation. Therefore, extreme care must be used in handling radioisotopes during their removal from the rotary specimen rack.

Positive control measures are required to prevent unnecessary personnel exposures or contamination during isotope-handling operations. These measures may include the use of remote-handling tongs, absorbent paper matting, portable radiation shields, etc.

In addition to the isotopes themselves being made highly radioactive, the containers used to protect the isotopes during their irradiation will also become radioactive. Plastic sample containers are used when the radiation exposure to the container will be small enough not to produce radiation damage in the plastic. The principal advantage of plastic containers is the small amount of induced activity associated with the container after its removal from the reactor. Activation of the plastic is minimized by the low neutron activation cross section of plastic. Aluminum containers are used where the exposure accumulated by a container during a single or series of irradiations would be high. <u>Aluminum containers must be used when fissionable materials are irradiated</u>. The principal disadvantage of using aluminum containers is the amount of induced activity produced in the container even during very short exposure periods. Table 5.1 compares the radiation levels associated with plastic and aluminum containers having equal radiation exposures.

Table 5.1

A COMPARISON OF INDUCED RADIOACTIVITY IN ALUMINUM AND PLASTIC SAMPLE CONTAINERS
(For an exposure of 15 min at a reactor power level of 250 kw)

Material	Distance (cm)	Decay Time (minutes)	Intensity (mr/hr)
Aluminum	50	5	6000
Plastic	50	5	1.5
Aluminum	50	21	750
Plastic	50	21	Less than 1.0

In-core experiments constitute both a radiation and a contamination hazard. Activation of experimental components can result in a source of intense radiation when removed from the reactor. Activated material attached to or contained in the component becomes a source of tactile contamination. Here again, extreme caution must be exercised in handling in-core components following their irradiation.

5.3.4. Reactor Coolant

Impurities in the reactor cooling water will be made radioactive when the reactor is operated. Typical of impurities which are activated if they enter the reactor coolant are sodium (Na^{24}) and chlorine (Cl^{38}). These elements are common contaminants on the surfaces of objects placed in the reactor that have not been cleaned following handling.

Radioactive sodium is also generated by a (n, α) reaction in aluminum ($Al^{27} + n \rightarrow Na^{24} + \alpha$).

Other contaminants which may normally be found in the coolant after extended operation of the reactor include Fe^{55}, Fe^{59}, Cr^{51}, and Zn^{65}. The presence of fission products in the reactor coolant could signify either a leak in an isotope-production container, in an experimental assembly containing fissionable material, or possibly a ruptured fuel element.

Should impurities develop in the coolant water and become activated, they can be removed to a great extent by circulating the water through the demineralizer in the water-treatment system. This, however, is done at the expense of increasing the radioactivity fixed on the ion-exchange resin. Because the ion-exchange resin in the demineralizer concentrates radioactivity in excess of the concentration in the pool water, servicing the demineralizer should always be done using strict contamination control procedures.

Natural radioactive decay will also reduce the radioactivity levels of contaminated water.

If the coolant water should become excessively contaminated, it could be pumped to the liquid-waste tank. The need for such action has never

arisen in General Atomic's operating experience with the TRIGA.

5.3.5. Loss of Cooling Water

The reactor tanks have been designed to minimize the possibility of loss of water. In all cases there are two watertight barriers. The TRIGA has an aluminum tank as the inner barrier. The aluminum tank is located inside a thick concrete liner. A foundation drainage system below the shield structure provides for effluent control if a leak should occur.

Loss of water will cause shutdown of the reactor. Experiments with the subcritical assembly indicate that the reactivity worth of the water is on the order of 10% (more than $12.00). Consequently, were the reactor operating during a catastrophic event in which cooling water was completely lost, the reactor would shut down when the water level dropped to a few inches below the upper grid plate.

The concrete tank must rupture before a large fraction of the reactor tank water can escape. While this possibility is believed to be exceedingly remote, a calculation has been performed to evaluate the radiological hazard associated with such an accident, under the condition that the reactor has

been operating for a long period of time at 1500kw prior to loss of all of the shielding water. Table 5.2 shows the calculated radiation dose rate at two different locations. The first location is 23ft above the unshielded reactor core at the top of the core tank; the second is at the TRIGA Reactor Facility pool level and at a location subjected to scattered radiation from a ceiling assumed to be of thick concrete and assumed to be above the top of the reactor shield. This ceiling maximizes the reflected radiation dose. Normal roof structures would produce considerably less back-scattering. Time is measured from the conclusion of operation.

Table 5.2

CALCULATED RADIATION DOSE RATE
FOR TANK-RUPTURE ACCIDENT
(1000 hours of operation)

Time	Direct Radiation (r/hr)	Scattered Radiation (r/hr)
1 min	4.9×10^3	4.6
1 day	6.8×10^2	0.64
1 week	2.8×10^2	0.26
1 month	1.0×10^2	0.10

Table 5.2 shows that if an individual did not expose himself to the core directly, he could work for approximately 2 hours at the top of the shield tank after one day without being exposed to radiation in excess of a quarterly occupational dose. This would permit sufficient time to view the interior of the shield tank with a mirror and to make emergency repairs.

The direct radiation from the unshielded core would be highly collimated by the shield structure and therefore would not normally give rise to a public hazard.

5.3.6. Beam Tubes

Beams of radiation streaming from the reactor core through any straight passages penetrating the shielding water also constitute radiation

hazards. Several types of access tubes may pass through the water from the reactor core--an unshielded isotope removal tube, beam collimating tubes, etc.

Several types of radiation with broad energy spectra are emitted from the core through these tubes. Both fast and slow neutrons radiate from tubes when the reactor is operating. Gamma radiation persists after reactor shutdown, but it is more intense during reactor operation. Beta radiation emitted from the core is usually masked by the accompanying gamma radiation.

The relative proportion of the various types and intensities of radiation in a beam streaming from the reactor may be varied by using selective absorbers or moderators in the beam path. A relatively unmodified radiation beam will be composed of fast neutrons, slow neutrons, and gamma rays of about equal biologically effective intensity.

5.3.7. Production of Radioactive Gases

Nitrogen-16 and argon-41, produced by irradiation of the coolant water and air trapped in beam tubes and irradiation facilities, are the only radioactive gases generated during TRIGA reactor operation that might constitute a potential radiological hazard. As will be seen in this section, they are not present during operation in such quantity as to create an active hazard.

Nitrogen-16. Even though significant quantities of 7-sec N^{16} are produced by the irradiation of the reactor coolant, the transport time of the nitrogen from the reactor core to the surface of the shielding water--measured to be 42 sec when the reactor is operating at 100 kw, which corresponds to six half lives of N^{16}--provides a large attenuation factor. Experiments with the TRIGA Mark I indicate that N^{16} makes a negligible contribution to the small residual radiation flux at the top of the reactor shielding water during operation.

Argon-41. Experiments have shown that in regions of the reactor containing air in which the neutron flux is the same as that in the rotary specimen rack, an equilibrium concentration of 3 $\mu c/cm^3$ of 109-min A^{41}

is established after the reactor has operated at 100 kw for several hours. Under these conditions the equilibrium content of A^{41} in the whole of the rotary specimen rack is calculated to be 100 mc; and in a 6-in.-diameter beam tube, less than 10 mc. After A^{41} has been purged from these cavities, it takes another several hours to re-establish equilibrium concentration.

The analysis of the hazards associated with an exposure to this radioactive argon has been based on the NRC maximum permissible concentration (MPC) for continuous exposure. Even during extensive operation of the TRIGA reactor no argon concentrations in excess of 20% of the quoted MPC have been detected. Moreover, the transfer of argon from the rotary specimen rack to the reactor room is inhibited by the long, slender access tube.

5.3.8. Sample Activation

Sampler activation is a routine hazard associated with the normal operation activities. An analysis of the possible radiation levels, radiation damage and sample hazards is a requirement prior to sample irradiation. Handling procedures that include encapsulation requirements, leak tests and monitoring are for the purpose of safe sample manipulation. A review is made of each experiment application to determine that the safety hazards are acceptable.

5.3.9. Other Sources

The startup source emits neutrons at a rather steady rate whether or not the reactor is operating. Fission products are produced in fission counters. Thermocouples, connectors, and stray bits of hardware, along with brass, stainless steel, and other materials, become sources of long-lived radioactivity when irradiated by thermal neutrons. Equipment used in conducting an experiment may become radioactive if it is placed near the reactor core. Thermocouple leads and their installation generally contain elements in which long-lived radioactivity is induced. Tubing used as conduits for wires, coolants, and other circulating fluids is subject to becoming radioactive under neutron bombardment, depending upon the

material from which it is made. Deposits of welding flux which bear sodium can cause serious trouble, especially if the sodium goes into solution. Accessories used near the reactor core may be expected to become radioactive. Clamps, harnesses, and mountings are likely to contain elements in which significant radioactivity may be induced by neutrons. Foils and wires used to monitor neutron flux are made radioactive in the process, as may be impurities in selective filters that are sometimes used to enclose such monitors. Tools used to disassemble components and capsules and to manipulate sources are subject to acquiring contamination upon contact. Floors, furniture, and apparatus may become contaminated by the spillage of sources or the dripping of fluids, e.g., water from the reactor pool. Wrappings used to protect more valuable things from tactile contamination become contaminated themselves. Radioactive dust, vapors, and gases can pollute the air unless they are confined to the point of generation.

5.4. HEALTH PHYSICS INSTRUMENTATION

Man's senses cannot detect the presence of radiation or contamination. Detection and measurement of radioactive materials require the use of special radiation-sensitive instruments. Even these instruments do not detect the presence of radiation directly; rather, they detect the ionization produced by the radiation as it interacts with matter.

Health physics monitoring instrumentation employed at the TRIGA Reactor Facility to detect and measure ionization radiation may be divided into three general categories: (1) portable survey instruments, (2) continuous environment monitors, and (3) personnel-monitoring devices.

5.4.1. Portable Survey Instruments

Portable survey instruments are used to detect the presence of alpha, beta, gamma, and neutron radiation. Specifically, they indicate the radiation intensity with respect to the source's position, e.g., 200 mr/hr at 1 ft. In some cases, e.g., of GM detector models,

a single instrument can be used to detect several types of radiation. Survey instruments must be rugged, simple in operation, versatile, and, most important, reliable.

5.4.2. Continuous Environmental Monitors

Continuous environmental monitors (Figs. 5.5 through 5.10) are employed at the TRIGA Reactor Facility to detect and measure gamma activity near the reactor tank, airborne radioactive particulates, and radioactive materials suspended in the reactor coolant. Unlike survey instruments, which measure radiation intensities, i.e., milliroentgens per hour, roentgens per second, etc., continuous environmental monitors measure concentrations of activity, e.g., curies per milliliter or curies per cubic centimeter. In some instances a monitoring system may include a recorder or meter to indicate rates of concentration buildup and activity decay. Depending on the particular detector considered, environmental monitors can detect sources of alpha, beta, and gamma radiation. Continuous air and water monitors used at the TRIGA Reactor Facility are usually "fixed-position" detectors lacking the portability of survey meters. However, special monitors may be employed to augment the coverage provided by the fixed environmental monitors.

5.4.3. Personnel-monitoring Devices

The primary personnel-monitoring instrument used at the Facility is the pocket ionization chamber. These instruments are worn in pairs (at the same location on the body) for the evaluation of gamma, X-ray, and neutron dosage. Ion chambers with walls thin enough to measure beta rays can be made, but generally they are not necessary for personnel monitoring. Pocket ion chambers sensitive to thermal neutrons alone are available where exposure to thermal neutrons may be significant. Film badges are also worn for the measurement of fast-neutron and gamma dosage. Other film badges, sensitive to beta and gamma radiation, are worn in conjunction with pocket ion chambers.

devices used at the Facility.

5.5. REACTOR AND ENVIRONMENTAL SURVEYS

The performance of reactor and environmental surveys is an integral part of the radiological safety program at the TRIGA Reactor Facility.

5.5.1. Reactor Surveys

Reactor surveys are conducted to provide the reactor operating staff with up-to-date information on the existence, location, and significance of active or potential radiation and/or contamination hazards at the Facility.

In most instances reactor surveys are performed using the portable survey instrumentation described above.

5.5.2. Environmental Surveys

Environmental surveys involve the analysis of soil, vegetation, water, and air samples. Environmental surveys are performed in order to evaluate the effectiveness of off-site radiation and contamination detection and control methods.

5.6. RADIOLOGICAL SAFETY PROCEDURES AND EQUIPMENT

The manipulation of radioactive or contaminated materials should never be considered a "routine" operation. Each operation should be considered in the light of the peculiar problems it presents. In general, it is best to develop protective techniques to fit a particular operation rather than to attempt to establish a set of all-inclusive procedures. There are, however, certain general rules applicable to radiological safety. These rules, listed below, are based on one fundamental precept:.

NO EXPOSURE IS THE PREFERRED EXPOSURE!

1. Every portable radiation source (e.g., neutron and gamma

calibration sources) must be assigned to some individual who will be responsible for it.

2. No one is to be permitted access to a radioactivity-contaminated area or is to be exposed to radiation dose rates in excess of 2 mrem/hr without being apprised of it in advance.
3. Personnel are not to undertake work with radioactive materials or radiation generators without a knowledge in advance of the exposure which they will incur and the means they have available to control this exposure.
4. All radioactive material and contamination is to be confined to zones with adequate boundaries and suitable portal facilities that will prevent dispersal beyond the zone.
5. Protective clothing, such as coats, overalls, shoes, overshoes, caps, and gloves, will be worn by each person as necessary to prevent contamination of his skin.
6. Contaminated garments will not be worn away from the scene of the contaminating work and will be replaced with clean garments when the generation of severe contamination has subsided.
7. Personnel and equipment must be checked for radioactive contamination with appropriate instrumentation prior to departure from a contamination control zone.

The applicability of these rules depends primarily on whether the radioactive materials to be handled represent a potential external hazard or a potential internal hazard. In the case of an external hazard, e.g., the gamma rays and neutrons emitted from the Sb-Be startup source, safe handling procedures incorporating the use of shields, controlled exposure times, and varied source-personnel geometries can be employed to effectively reduce or eliminate personnel exposures. Where the material to be manipulated constitutes a potential internal hazard, e.g., radioactive dusts, vapors, liquids, etc., safety techniques must be developed using special anticontamination equipment. Depending on the situation, one or more pieces of anticontamination equipment might be required.

Several rules specificly for control of radioactive contamination follow.

1. Food containers will not be used to hold radioactive materials.
2. No refrigerator used for radioactive samples will be used for food, and vice versa.
3. No personal effects such as purses, combs, or cosmetics will be set down in contamination control zones.
4. At the close of each work period the face, hands, hair, and clothes of each person who has worked with radioactive material will be inspected for contamination by a competent person equipped with suitable instrumentation.
5. Do not use harsh chemicals to decontaminate the skin. If a surgical scrub is inadequate, consult a health physics surveyor.
6. Smoking and eating will not be permitted in laboratories or workrooms in which unsealed radioactive materials are used or stored.
7. Pipetting of radioactive solutions by mouth is forbidden.

5.7. DECONTAMINATION AND RADIOACTIVE-WASTE DISPOSAL

5.7.1. Decontamination

Decontamination is a recourse where contamination control fails. It always produces more contaminated material and hence magnifies the volume of waste requiring disposal.

Decontamination is selective erosion; Processes are chosen to remove a maximum of the contaminant and a minimum of the base substance. Both the nature of the contaminants and the surface contaminated determine the ease with which contamination accrues and with which it may be removed.

Oftentimes, waiting for radioactive decay is a practical expedient for short-lived contaminants.

Particulate contaminants may be removed to some degree with air movement. Vacuum cleaners control the contaminant better than air jets.

Adhesives are practical in applications where they are able to bind the particulates while remaining strippable from the base. Brushing may also dislodge particulates; its effectiveness is enhanced when done in conjunction with vacuum cleaning.

Contaminants bound in an oil film coating a surface may be freed by using detergents or degreasers with fluid flushing. Steam cleaning is often effective. However, if the contaminant goes into solution, it may recontaminate by chemical reaction with the surface.

Contaminants plated or chemically bound to a surface may be removed if a solvent for the contaminant can be found that does not seriously attack the surface. Sometimes there is no choice but to remove some of the surface with the contaminant.

Porous materials like wood and concrete are most practically freed of radioactive contamination by cutting (grinding) away the material into which the contaminant has penetrated.

Skin should not be treated with strong chemicals. A surgical scrub, using a good soap in running warm water and brushing lightly (so the bristles don't bend tangent to the surface) thoroughly for eight minutes should reduce hand contamination to the point at which the radiation will be less injurious than an acid or alkali wash. If appreciable body contamination remains following a surgical scrub, recommendations regarding further treatment should be obtained from a health physicist.

5.7.2. Waste Disposal

No radioactive material should be disposed of via the sewer or ordinary trash.

It is desirable to keep wastes of high specific activity in a concentrated form for collection, storage, and preparation for disposal. Wastes should be collected in small, suitably shielded vessels in the laboratory of origin. Polyethylene bottles of one-half gallon capacity or less are suitable for most liquids and pails with tight covers for solids of

modest dimensions. Waste-collection vessels must be appropriately marked with a magenta radiation emblem and the words "Caution--Radioactive Material."

When containers are filled, or when a primary collection has been completed and it is desirable to remove the major wastes from a laboratory, the individual responsible for the radioactive material should notify Health Physics for waste-disposal service.

Appendix

RADIOLOGICAL SAFETY TERMINOLOGY

GENERAL

Radiological: pertaining to the study and use of radioactive materials and radiation-generating apparatus.

Radioactivity: that property of a substance which causes it to emit ionizing radiation. This property is the spontaneous transmutation of the atoms of the substance.

Radiation: the propagation of energy through space radially from the point of origin. The types of radiation of concern in radiological safety are those which generate ionization as they dissipate their energy in the matter through which they pass.

Gamma and X-ray Radiation: electromagnetic quanta of wavelengths less than ultraviolet, travelling with the speed of light, and each conveying energy proportional to their frequency.

Beta Radiation: energetic electrons.

Alpha Radiation: energetic helium nuclei.

Neutron Radiation: a neutron is a chargeless particle of mass similar to that of the hydrogen ion. Fast neutrons convey energy from their source by virtue of the velocity at which their mass travels. Slow neutrons may be termed "thermal" neutrons when their kinetic energy is equivalent to that of the thermal motion of the atoms among which they diffuse; thermal neutrons no longer convey significant energy from their origin but are capable of releasing energy as a consequence of the transmutation which occurs when they are captured by absorber nuclei.

Contamination: an impurity which pollutes or adulterates another substance. In radiological safety, contamination refers to the radioactive materials which are the sources of ionizing radiations.

MEASUREMENTS OF RADIATION LEVELS

Dose: The radiation delivered per unit volume to a specified area, volume, or to the whole body.

Dose Rate: Radiation dose delivered per unit time.

Activity: The basic unit of radioactivity is the historical curie or practical becquerel. A curie (c) is defined as that amount of radioactive material that decays at a rate of 3.7×10^{10} disintegrations per second. A becquerel is equivalent to 1 disintegration per second.

Roentgen: A term used to demote X-ray and γ-ray dose only. By definition, one roentgen (r) is that amount of radiation that produces 83 ergs/g of dry air or 1 electrostatic unit of charge of either sign per cubic centimeter of dry air at STP. One milliroentgen is 1/1000 roentgen.

Roentgen Absorbed Dose (rad): A term used to denote the energy (from any type of radiation) absorbed in any material. By definition, one rad signifies the absorption of 100 ergs/g of material. It is nearly equivalent to the amount of energy deposited in mixed body tissue by one roentgen of X- or γ-radiation.

Roentgen Equivalent, Man (rem): A measure of the biological effect produced by any type of radiation. By definition, one rem is the quantity of radiation (of any type) which produces the same biological effects in man as those resulting from the absorption of one roentgen of X- or γ-radiation.

Relative Biological Effectiveness (RBE): A factor that compares, for the same energy absorbed per unit volume, the biological effect of the various types of radiation.

Quality Factor: A factor that improves the qualitative expression that relates the radiation human impact to the tissue dose.

Approximate (most conservative) values of dose factors:

X-ray	1
Gamma	1
Beta	1
Neutron (thermal)	2-5
Neutron (fast)	10
Alpha	20

Relationship of rem to r and rad:

rem = r x RBE = rad x RBE.
rem = r x QF = rad x QF

ACCEPTABLE LEVELS OF RADIATION

Maximum Permissible Concentration (MPC): The greatest quantity of contamination permitted in a medium in a given circumstance.

TABLE 22.1—*Summary of recommendations*[a,b]

A. Occupational exposures (annual)[c]		
1. Effective dose equivalent limit (stochastic effects)	50 mSv	(5 rem)
2. Dose equivalent limits for tissues and organs (nonstochastic effects)		
a. Lens of eye	150 mSv	(15 rem)
b. All others (*e.g.*, red bone marrow, breast, lung, gonads, skin and extremities)	500 mSv	(50 rem)
3. Guidance: Cumulative exposure	10 mSv × age	(1 rem × age in years)
B. Planned special occupational exposure, effective dose equivalent limit[c]	see Section 15	
C. Guidance for emergency occupational exposure[c]	see Section 16	
D. Public exposures (annual)		
1. Effective dose equivalent limit, continuous or frequent exposure[c]	1 mSv	(0.1 rem)
2. Effective dose equivalent limit, infrequent exposure[c]	5 mSv	(0.5 rem)
3. Remedial action recommended when:		
a. Effective dose equivalent[d]	>5 mSv	(>0.5 rem)
b. Exposure to radon and its decay products	>0.007 Jhm^{-3}	(>2 WLM)
4. Dose equivalent limits for lens of eye, skin and extremities[c]	50 mSv	(5 rem)
E. Education and training exposures (annual)[c]		
1. Effective dose equivalent limit	1 mSv	(0.1 rem)
2. Dose equivalent limit for lens of eye, skin and extremities	50 mSv	(5 rem)
F. Embryo-fetus exposures[c]		
1. Total dose equivalent limit	5 mSv	(0.5 rem)
2. Dose equivalent limit in a month	0.5 mSv	(0.05 rem)
G. Negligible Individual Risk Level (annual)[c]		
Effective dose equivalent per source or practice	0.01 mSv	(0.001 rem)

[a] Excluding medical exposures.

[b] See Table 4.1 for recommendations on $\bar{Q}$.

[c] Sum of external and internal exposures.

[d] Including background but excluding internal exposures.

Important contributions to future modifications will be made by studies concerned with relative biological effectiveness at low doses, the shape of dose response curves at low doses, their dependence on dose rate and considerations of the many agents that can modify the carcinogenic response. Additional information on human risks of

radiation exposure and the projection models to be used with them and on weighting factors for use in radiation protection can be expected to become available in the future. The NCRP will follow developments in these areas closely and use them to form the basis of new or revised recommendations when necessary.

NCRP Radiation Dose Limits

Category	Value
Maximum permissible dose equivalent for occupational exposure	
Combined whole-body occupational exposure	5 rems in any one year
Retrospective annual limit	10 to 15 rems in any one year
Long-term accumulation	(Age - 18) X 5 rems
Skin	15 rems in any one year
Hands	75 rems in any one year (25/quarter)
Forearms	30 rems in any one year (10/quarter)
Other organs, tissues, and organ systems	15 rems in any one year (5/quarter)
Fertile women (with respect to fetus)	0.5 rem in gestation period
Dose limits for the public, or occasionally exposed individuals	
Individual or occasional	0.5 rem in any one year
Students	0.1 rem in any one year
Population-group dose limits* (Total exposure from "Man-made radiation above and in addition to natural background which averages about 0.1 rem per year)	
Genetic	0.17 rem average per year
Somatic	0.17 rem average per year
Emergency dose limits (less urgent)	
Individual	25 rems
Hands and forearms	100 rems, total
Family of radioactive patients	
Individual (under age 45)	0.5 rem in any one year
Individual (over age 45)	5 rems in any one year

*Average for population group is intended to ensure that no individual exposure exceeds 0.5 rem/year.

§20.1003 Definitions.

As used in this part:

Absorbed dose means the energy imparted by ionizing radiation per unit mass of irradiated material. The units of absorbed dose are the rad and the gray (Gy).

Act means the Atomic Energy Act of 1954 (42 U.S.C. 2011 et seq.), as amended.

Activity is the rate of disintegration (transformation) or decay of radioactive material. The units of activity are the curie (Ci) and the becquerel (Bq).

Adult means an individual 18 or more years of age.

Airborne radioactive material means radioactive material dispersed in the air in the form of dusts, fumes, particulates, mists, vapors, or gases.

Airborne radioactivity area means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations --

(1) In excess of the derived air concentrations (DACs) specified in appendix B, to §§20.1001 - 20.2401, or

(2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake (ALI) or 12 DAC-hours.

ALARA (acronym for "as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

Annual limit on intake (ALI) means the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue. (ALI values for intake by ingestion and by inhalation of selected radionuclides are given in Table 1, Columns 1 and 2, of appendix B to §§20.1001 - 20.2401).

Background radiation means radiation from cosmic sources; naturally occurring radioactive material, including radon (except as decay product of source or special nuclear material); and global fallout as it exists in the environment from the testing of nuclear explosive devices or from past nuclear accidents such as Chernobyl that contribute to background radiation and are not under the control of the licensee. "*Background radiation*" does not include radiation from source, byproduct, or special nuclear materials regulated by the Commission.

Bioassay (radiobioassay) means the determination of kinds, quantities or concentrations, and, in some cases, the locations of radioactive material in the human body, whether by direct measurement (in vivo counting) or by analysis and evaluation of materials excreted or removed from the human body.

Byproduct material means --

(1) Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or utilizing special nuclear material; and

(2) The tailings or wastes produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore bodies depleted by these solution extraction operations do not constitute "byproduct material" within this definition.

Class (or *lung class* or *inhalation class*) means a classification scheme for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials are classified as D, W, or Y, which applies to a range of clearance half-times: for Class D (Days) of less than 10 days, for Class W (Weeks) from 10 to 100 days, and for Class Y (Years) of greater than 100 days.

Collective dose is the sum of the individual doses received in a given period of time by a specified population from exposure to a specified source of radiation.

Commission means the Nuclear Regulatory Commission or its duly authorized representatives.

Committed dose equivalent ($H_{T,50}$) means the dose equivalent to organs or tissues of reference (T) that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

Committed effective dose equivalent ($H_{E,50}$) is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues $\left(H_{E,50} = \sum W_T H_{T,50}\right)$.

Constraint (dose constraint) means a value above which specified licensee actions are required.

Controlled area means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

Critical Group means the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances.

Declared pregnant woman means a woman who has voluntarily informed the licensee, in writing, of her pregnancy and the estimated date of conception. The declaration remains in effect until the declared pregnant woman withdraws the declaration in writing or is no longer pregnant.

Decommission means to remove a facility or site safely from service and reduce residual radioactivity to a level that permits--

(1) Release of the property for unrestricted use and termination of the license; or

(2) Release of the property under restricted conditions and termination of the license.

Deep-dose equivalent (H_d), which applies to external whole-body exposure, is the dose equivalent at a tissue depth of 1 cm (1000 mg/cm^2).

Department means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95 - 91, 91 Stat. 565, 42 U.S.C. 7101 et seq.) to the extent that the Department, or its duly authorized representatives, exercises functions formerly vested in the U.S. Atomic Energy Commission, its Chairman, members, officers, and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to sections 104 (b), (c), and (d) of the Energy Reorganization Act of 1974 (Pub. L. 93 - 438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to section 301(a) of the Department of Energy Organization Act (Pub. L. 95 - 91, 91 Stat 565 at 577 - 578, 42 U.S.C. 7151).

Derived air concentration (DAC) means the concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work (inhalation rate 1.2 cubic meters of air per hour), results in an intake of one ALI. DAC values are given in Table 1, Column 3, of appendix B to §§20.1001 - 20.2401.

Derived air concentration-hour (DAC-hour) is the product of the concentration of radioactive material in air (expressed as a fraction or multiple of the derived air concentration for each radionuclide) and the time of exposure to that radionuclide, in hours. A licensee may take 2,000 DAC-hours to represent one ALI, equivalent to a committed effective dose equivalent of 5 rems (0.05 Sv).

Distinguishable from background means that the detectable concentration of a radionuclide is statistically different from the background concentration of this radionuclide in the vicinity of the site or, in the case of structures, in similar materials using adequate measurement technology, survey, and statistical techniques.

Dose or *radiation dose* is a generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as defined in other paragraphs of this section.

Dose equivalent (H_T) means the product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The units of dose equivalent are the rem and sievert (Sv).

Dosimetry processor means an individual or organization that processes and evaluates individual monitoring equipment in order to determine the radiation dose delivered to the equipment.

Embryo/fetus means the developing human organism from conception until the time of birth.

Entrance or access point means any location through which an individual could gain access to radiation areas or to radioactive materials. This includes entry or exit portals of sufficient size to permit human entry, irrespective of their intended use.

Exposure means being exposed to ionizing radiation or to radioactive material.

External dose means that portion of the dose equivalent received from radiation sources outside the body.

Extremity means hand, elbow, arm below the elbow, foot, knee, or leg below the knee.

Eye dose equivalent applies to the external exposure of the lens of the eye and is taken as the dose equivalent at a tissue depth of 0.3 centimeter (300 mg/cm^2).

Generally applicable environmental radiation standards means standards issued by the Environmental Protection Agency (EPA) under the authority of the Atomic Energy Act of 1954, as amended, that impose limits on radiation exposures or levels, or concentrations or quantities of radioactive material, in the general environment outside the boundaries of locations under the control of persons possessing or using radioactive material.

Government agency means any executive department, commission, independent establishment, corporation wholly or partly owned by the United States of America, which is an instrumentality of the United States, or any board, bureau, division, service, office, officer, authority, administration, or other establishment in the executive branch of the Government.

Gray [See §20.1004].

High radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

Individual means any human being.

Individual monitoring means --

(1) The assessment of dose equivalent by the use of devices designed to be worn by an individual;

(2) The assessment of committed effective dose equivalent by bioassay (see Bioassay) or by determination of the time-weighted air concentrations to which an individual has been exposed, i.e., DAC-hours; or

(3) The assessment of dose equivalent by the use of survey data.

Individual monitoring devices (individual monitoring equipment) means devices designed to be worn by a single individual for the assessment of dose equivalent such as film badges, thermoluminescence dosimeters (TLDs), pocket ionization chambers, and personal ("lapel") air sampling devices.

Internal dose means that portion of the dose equivalent received from radioactive material taken into the body.

Lens dose equivalent (LDE) applies to the external exposure of the lens of the eye and is taken as the

dose equivalent at a tissue depth of 0.3 centimeter (300 mg/cm^2).

License means a license issued under the regulations in parts 30 through 36, 39, 40, 50, 60, 61, 70, or 72 of this chapter.

Licensed material means source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license issued by the Commission.

Licensee means the holder of a license.

Limits (dose limits) means the permissible upper bounds of radiation doses.

Lost or missing licensed material means licensed material whose location is unknown. It includes material that has been shipped but has not reached its destination and whose location cannot be readily traced in the transportation system.

Member of the public means any individual except when that individual is receiving an occupational dose.

Minor means an individual less than 18 years of age.

Monitoring (radiation monitoring, radiation protection monitoring) means the measurement of radiation levels, concentrations, surface area concentrations or quantities of radioactive material and the use of the results of these measurements to evaluate potential exposures and doses.

Nonstochastic effect means health effects, the severity of which varies with the dose and for which a threshold is believed to exist. Radiation-induced cataract formation is an example of a nonstochastic effect (also called a deterministic effect).

NRC means the Nuclear Regulatory Commission or its duly authorized representatives.

Occupational dose means the dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include dose received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, from voluntary participation in medical research programs, or as a member of the public.

Person means --

(1) Any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, Government agency other than the Commission or the Department of Energy (except that the Department shall be considered a person within the meaning of the regulations in 10 CFR chapter I to the extent that its facilities and activities are subject to the licensing and related regulatory authority of the Commission under section 202 of the Energy Reorganization Act of 1974 (88 Stat. 1244), the Uranium Mill Tailings Radiation Control Act of 1978 (92 Stat. 3021), the Nuclear Waste Policy Act of 1982 (96 Stat. 2201), and section 3(b)(2) of the Low-Level Radioactive Waste

Policy Amendments Act of 1985 (99 Stat. 1842)), any State or any political subdivision of or any political entity within a State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and

(2) Any legal successor, representative, agent, or agency of the foregoing.

Planned special exposure means an infrequent exposure to radiation, separate from and in addition to the annual dose limits.

Public dose means the dose received by a member of the public from exposure to radiation or radioactive material released by a licensee, or to any other source of radiation under the control of a licensee. Public dose does not include occupational dose or doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, or from voluntary participation in medical research programs.

Quality Factor (Q) means the modifying factor (listed in tables 1004(b).1 and 1004(b).2 of §20.1004) that is used to derive dose equivalent from absorbed dose.

Quarter means a period of time equal to one-fourth of the year observed by the licensee (approximately 13 consective weeks), providing that the beginning of the first quarter in a year coincides with the starting date of the year and that no day is omitted or duplicated in consecutive quarters.

Rad (See §20.1004).

Radiation (ionizing radiation) means alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in this part, does not include non-ionizing radiation, such as radio- or microwaves, or visible, infrared, or ultraviolet light.

Radiation area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

Reference man means a hypothetical aggregation of human physical and physiological characteristics arrived at by international consensus. These characteristics may be used by researchers and public health workers to standardize results of experiments and to relate biological insult to a common base.

Rem (See §20.1004).

Residual radioactivity means radioactivity in structures, materials, soils, groundwater, and other media at a site resulting from activities under the licensee's control. This includes radioactivity from all licensed and unlicensed sources used by the licensee, but excludes background radiation. It also includes radioactive materials remaining at the site as a result of routine or accidental releases of radioactive material at the site and previous burials at the site, even if those burials were made in accordance with the provisions of 10 CFR part 20.

Respiratory protective device means an apparatus, such as a respirator, used to reduce the individual's

intake of airborne radioactive materials.

Restricted area means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

Sanitary sewerage means a system of public sewers for carrying off waste water and refuse, but excluding sewage treatment facilities, septic tanks, and leach fields owned or operated by the licensee.

Shallow-dose equivalent (HS), which applies to the external exposure of the skin or an extremity, is taken as the dose equivalent at a tissue depth of 0.007 centimeter (7 mg/cm^2) averaged over an area of 1 square centimeter.

Sievert (See §20.1004).

Site boundary means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

Source material means --

(1) Uranium or thorium or any combination of uranium and thorium in any physical or chemical form; or

(2) Ores that contain, by weight, one-twentieth of 1 percent (0.05 percent), or more, of uranium, thorium, or any combination of uranium and thorium. Source material does not include special nuclear material.

Special nuclear material means --

(1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Commission, pursuant to the provisions of section 51 of the Act, determines to be special nuclear material, but does not include source material; or

(2) Any material artificially enriched by any of the foregoing but does not include source material.

Stochastic effects means health effects that occur randomly and for which the probability of the effect occurring, rather than its severity, is assumed to be a linear function of dose without threshold. Hereditary effects and cancer incidence are examples of stochastic effects.

Survey means an evalulation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present.

Total Effective Dose Equivalent (TEDE) means the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.

Uranium fuel cycle means the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel to the extent that these activities directly support the production of electrical power for public use. Uranium fuel cycle does not include mining operations, operations at waste disposal sites, transportation of radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and byproduct materials from the cycle.

Very high radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates.

(Note: At very high doses received at high dose rates, units of absorbed dose (e.g., rads and grays) are appropriate, rather than units of dose equivalent (e.g., rems and sieverts)).

Week means 7 consecutive days starting on Sunday.

Weighting factor W_T, for an organ or tissue (T) is the proportion of the risk of stochastic effects resulting from irradiation of that organ or tissue to the total risk of stochastic effects when the whole body is irradiated uniformly. For calculating the effective dose equivalent,

Organ Dose Weighting Factors

Organ or tissue	W_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surface	0.03
Remainder	[1]0.30
Whole Body	[2]1.00

[1] 0.30 results from 0.06 for each of 5 "remainder" organs (excluding the skin and the lens of the eye) that receive the highest doses.

[2] For the purpose of weighting the external whole body dose (for adding it to the internal dose), a single weighting factor, w_T=1.0, has been specified. The use of other weighting factors for external exposure will be approved on a case-by-case basis until such time as specific guidance is issued.

Whole body means, for purposes of external exposure, head, trunk (including male gonads), arms above the elbow, or legs above the knee.

Working level (WL) is any combination of short-lived radon daughters (for radon-222: polonium-218, lead-214, bismuth-214, and polonium-214; and for radon-220: polonium-216, lead-212, bismuth-212, and polonium-212) in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha particle energy.

Working level month (WLM) means an exposure to 1 working level for 170 hours (2,000 working hours per year/12 months per year=approximately 170 hours per month).

Year means the period of time beginning in January used to determine compliance with the provisions of this part. The licensee may change the starting date of the year used to determine compliance by the licensee provided that the change is made at the beginning of the year and that no day is omitted or duplicated in consecutive years.

[56 FR 23391, May 21, 1991, as amended at 57 FR 57878, Dec. 8, 1992; 58 FR 7736, Feb. 9, 1993; 60 FR 36043, July 13, 1995; 60 FR 48625, Sept. 20, 1995; 61 FR 65127, Dec. 10, 1996; 62 FR 4133, Jan. 29, 1997; 62 FR 39807, July 21, 1997]

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§20.1004 Units of radiation dose.

(a) Definitions. As used in this part, the units of radiation dose are:

Gray (Gy) is the SI unit of absorbed dose. One gray is equal to an absorbed dose of 1 Joule/kilogram (100 rads).

Rad is the special unit of absorbed dose. One rad is equal to an absorbed dose of 100 ergs/gram or 0.01 joule/kilogram (0.01 gray).

Rem is the special unit of any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor (1 rem=0.01 sievert).

Sievert is the SI unit of any of the quantities expressed as dose equivalent. The dose equivalent in sieverts is equal to the absorbed dose in grays multiplied by the quality factor (1 Sv=100 rems).

(b) As used in this part, the quality factors for converting

Table 1004(b).1-Quality Factors and Absorbed Dose Equivalencies

Type of radiation	Quality factor (Q)	Absorbed dose equal to a unit dose equivalent[a]
X-, gamma, or beta radiation	1	1
Alpha particles, multiple-charged particles, fission fragments and heavy particles of unknown charge	20	0.05
Neutrons of unknown energy	10	0.1
High-energy protons	10	0.1

[a] absorbed dose in rad equal to 1 rem or the absorbed dose in gray equal.

(c) If it is more convenient to measure the neutron fluence rate than to determine the neutron dose equivalent rate in rems per hour or sieverts per hour, as provided in paragraph (b) of this section, 1 rem (0.01 Sv) of neutron radiation of unknown energies may, for purposes of the regulations in this part, be assumed to result from a total fluence of 25 million neutrons per square centimeter incident upon the body. If sufficient information exists to estimate the approximate energy distribution of the neutrons, the licensee may use the fluence rate per unit dose equivalent or the appropriate Q value from table 1004(b).2 to convert a measured tissue dose

	Neutron energy (MeV)	**Quality factor[a](Q)**	**Fluence per unit dose equivalent[b] (neutrons cm^{-2} rem^{-1})**
(thermal).....	2.5×10^{-8}	2	980×10^{6}
	1×10^{-7}	2	980×10^{6}

$1x10^{-6}$	2	$810x10^6$
$1x10^{-5}$	2	$810x10^6$
$1x10^{-4}$	2	$840x10^6$
$1x10^{-3}$	2	$980x10^6$
$1x10^{-2}$	2.5	$1010x10^6$
$1x10^{-1}$	7.5	$170x10^6$
$5x10^{-1}$	11	$39x10^6$
1	11	$27x10^6$
2.5	9	$29x10^6$
5	8	$23x10^6$
7	7	$24x10^6$
10	6.5	$24x10^6$
14	7.5	$17x10^6$
20	8	$16x10^6$
40	7	$14x10^6$
60	5.5	$16x10^6$
$1x10^2$	4	$20x10^6$
$2x10^2$	3.5	$19x10^6$
$3x10^2$	3.5	$16x10^6$
$4x10^2$	3.5	$14x10^2$

[a] Value of quality factor (Q) at the point where the dose equivalent is maximum in a 30-cm diameter cylinder tissue-equivalent phantom.

[b] Monoenergetic neutrons incident normally on a 30-cm diameter cylinder tissue-equivalent phantom.

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§20.1005 Units of radioactivity.

For the purposes of this part, activity is expressed in the special unit of curies (Ci) or in the SI unit of becquerels (Bq), or their multiples, or disintegrations (transformations) per unit of time.

(a) One becquerel=1 disintegration per second (s^{-1}).

(b) One curie=3.7×10^{10} disintegrations per second=3.7×10^{10} becquerels=2.22×10^{12} disintegrations per minute.

[56 FR 23391, May 21, 1991; 56 FR 61352, Dec. 3, 1991]

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§20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under §20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of --

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see §20.1206(e)(1)) and during the individual's lifetime (see §20.1206(e)(2)).

(c) The assigned deep-dose equivalent and shallow-dose equivalent must be for the part of the body receiving the highest exposure. The deep-dose equivalent, lens dose equivalent, and shallow-dose equivalent may be assessed from surveys or other radiation measurements for the purpose of demonstrating compliance with the occupational dose limits, if the individual monitoring device was not in the region of highest potential exposure, or the results of individual monitoring are unavailable.

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of appendix B to part 20 and may be used to determine the individual's dose (see §20.2106) and to demonstrate compliance with the occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity (see footnote 3 of appendix B to part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received while employed by any other person (see §20.2104(e)).

[56 FR 23396, May 21, 1991, as amended at 60 FR 20185, Apr. 25, 1995]

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§20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that--

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 millisievert) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with §35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

(c) A licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 mSv). The licensee or license applicant shall include the following information in this application:

(1) Demonstration of the need for and the expected duration of operations in excess of the limit in paragraph (a) of this section;

(2) The licensee's program to assess and control dose within the 0.5 rem (5 mSv) annual limit; and

(3) The procedures to be followed to maintain the dose as low as is reasonably achievable.

(d) In addition to the requirements of this part, a licensee subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR part 190 shall comply with those standards.

(e) The Commission may impose additional restrictions on radiation levels in unrestricted areas and on the total quantity of radionuclides that a licensee may release in effluents in order to restrict the collective dose.

[56 FR 23398, May 21, 1991, as amended at 60 FR 48625, Sept. 20, 1995; 62 FR 4133, Jan. 29, 1997]

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Chapter VI

Chapter VI

REACTOR PHYSICS

In addition to his knowledge of reactor components, instrumentation, and the mechanics of reactor operation, the TRIGA operator is expected to have an understanding of the basic principles of reactor physics applicable to the TRIGA Mark I. Material reviewing these basic principles is presented in this chapter.

6.1. REACTOR PHYSICS*

When a neutron is captured by a nucleus, the resulting compound nucleus acquires an excitation energy equal to the binding energy of the neutron plus whatever kinetic energy the neutron had immediately before its capture. This excitation energy is usually given up by the emission of one or two light particles--most commonly, gamma rays. For the heaviest nuclei, however, another mode of energy release is possible: this is the splitting of the compound nucleus into two approximately equal parts plus the emission of neutrons and gamma rays. This process is called fission.

The binding energy of an additional neutron is different for different

*This brief account is written for those who have had little experience with reactor physics. Complete accounts of the theory can be found in the standard treatises, two of which are S. Glasstone and M. C. Edlund, Elements of Nuclear Reactor Theory, D. Van Nostrand Company, Princeton, N.J., 1952; and A. M. Weinberg and E. P. Wigner, Physical Theory of Neutron Chain Reactors, University of Chicago Press, Chicago, 1958. Two useful references which treat other branches of nuclear engineering and give briefer accounts of the theory are S. Glasstone, Principles of Nuclear Reactor Engineering, D. Van Nostrand Company, Princeton, N.J., 1955; and H. Etherington (ed.), Nuclear Engineering Handbook, McGraw-Hill Book Company, New York, 1958. The numerical data in Tables 6.1 and 6.2 have been taken from the Nuclear Engineering Handbook.

target nuclei, being larger by 1.0 to 1.5 Mev for the "even-odd"* isotopes U^{233}, U^{235}, and Pu^{239} than it is for the "even-even" isotopes Th^{232} and U^{238}. In particular, the binding energy of an additional neutron for the even-odd isotopes is greater than the critical energy required to produce fission, but it is smaller for the even-even isotopes. As a result, fission can be produced by slow neutrons in U^{233}, U^{235}, and Pu^{239}, but can be produced only by fast neutrons in Th^{232} (neutron energy greater than 1.3 Mev) and U^{238} (neutron energy greater than 1.2 Mev).

Nuclear reactors depend for their operation upon the fact that for each neutron which produces a fission, two, three, or sometimes even more neutrons are produced in the fission reaction. Some of these neutrons will escape completely from the reactor, some will be captured in materials other than the fuel, and some will be captured by the fuel in reactions which do not lead to fission, e.g., (n, γ) reactions. If exactly one neutron produces a fission for each fission in the previous generation, then the fission rate, or the number of fissions per generation, will remain constant, and the reaction will be self-sustaining at a constant power level. The multiplication factor, k, is defined as

$$k = \frac{\text{number of fissions in one generation}}{\text{number of fissions in previous generation}} .$$

The condition, then, for a constant power level is that k equal 1. It is also clear that provision must be made for varying k above and below unity so that the reactor may be brought up to power after a shutdown or reduced in power during operation.

Neutron leakage from the surface of the reactor becomes relatively less important as the size of the system is increased, since the surface increases as the square of the distance and the volume increases as the cube. It is often convenient to introduce the infinite multiplication factor k_∞, defined as the value of k for an infinite system, or--more realistically--

*The term "even-odd" refers to the number of protons and neutrons in the nucleus of an atom.

for a system for which neutron leakage is negligible. For a finite system, the relationship between k and k_∞ will depend primarily upon two distances: the size of the reactor, R; and the neutron migration length, M, which is proportional to the average distance between the point at which a fission neutron is produced and the point at which it is absorbed. The relative neutron loss, $k_\infty - k$, increases with M and decreases with R. It can be shown that to a crude first approximation,

$$k_\infty - k = C \frac{M^2}{R^2} ,$$

provided $k_\infty - k$ is small compared to 1, where C is a numerical factor near unity which depends in part on the geometrical shape of the reactor. It is clear that a system for which k_∞ is only slightly larger than 1, such as one which uses natural uranium as a fuel, will need to be much larger than one for which k_∞ can be close to 2, such as one which uses enriched uranium.

For the fissionable materials used as fuels in reactors, three important characteristics are the following: ν, the average number of neutrons produced per fission; α, the ratio of the number of neutrons captured by the fuel in nonfission reactions to the number producing fission; and η, the average number of neutrons produced per neutron captured by the fuel. If n fissions are produced, then α n neutrons are captured in nonfission reactions, and $\eta = \nu/(1 + \alpha)$.

Table 6.1 gives the values of these quantities for several different fuels and thermal (0.025 ev) neutrons. Note that a relatively small enrichment of U^{235} in a mixture of U^{235} and U^{238} can produce a large change in η. Increasing the percentage of U^{235} from its value of 0.71% in natural uranium to 5% produces a change in η which is 80% of the change achieved by going to pure U^{235}.

The ratio α is calculated from the relative abundances and the cross sections (probabilities) for the various processes. Since the cross sections

Table 6. 1

NEUTRON REGENERATION FOR THERMAL NEUTRONS

Fuel	ν	α	η
U (natural)	2. 47	0. 837	1. 33
U (5% U^{235})	2. 47	0. 272	1. 94
U (20% U^{235})	2. 47	0. 202	2. 05
U^{235}	2. 47	0. 183	2. 09
U^{233}	2. 55	0. 132	2. 29
Pu^{239}	2. 91	0. 416	2. 02

for different processes vary with neutron energy in different ways, α and η are functions of the neutron energy. In particular, η is less than 1 for natural uranium except for energies near thermal. Since fission neutrons are produced with energies in the Mev range, they must be slowed down to thermal energies before the value of η given in Table 6. 1 applies. In pure natural uranium,the majority are absorbed before they slow down to thermal energies, and as a result the muliplication factor is less than unity and it is impossible to have a self-sustaining chain reaction in a mass of unmoderated natural uranium, regardless of its size. In U^{235} and Pu^{239}, the average value of α is less than 1, and that of η is greater than 1, throughout the entire energy range.

In natural uranium it is essential, and with enriched fuels it is usually desirable, to slow the neutrons down to thermal energies in some material other than the fuel. This moderator, as it is called, must have two properties: it must be efficient at slowing down the neutrons (i. e. , do it in as few collisions as possible) and it must be a poor absorber of neutrons. The first requirement indicates a light element, since the average relative energy loss per collision is approximately inversely proportional to the mass of the moderator nucleus. The second requirement eliminates such light elements as lithium and boron, and makes hydrogen unsuitable for use with natural uranium. Carbon and heavy water are the usual moderators for natural uranium, while water or other hydrogen-containing compounds are commonly used with enriched fuel.

Large reactors, especially those containing natural uranium, are often analyzed in terms of what is known as the " four-factor formula," which is

$$k_\infty = \eta\varepsilon\, pf \ ,$$

where η is the number of neutrons produced per neutron absorbed in the fuel; ε is the fast fission factor, essentially a correction factor to take into account the fact that some fissions will be produced by fast neutrons; p is the resonance escape probability, the probability that a neutron will escape capture while it is being slowed through the resonance energy range (approximately 1 to 100 ev) to thermal energy; and f is the thermal utilization, the fraction of the thermal neutrons which are absorbed in the fuel, as opposed to the total neutrons absorbed, including those absorbed in moderator, coolant, or other foreign material.

For a homogeneous mixture of natural uranium and graphite, both being powders, ε is about 1.0 and the product pf varies with the ratio of moderator to fuel, having broadly a maximum of about 0.6. We have, then, $k_\infty = 1.33 \times 1 \times 0.6 = 0.8$, which indicates the impossibility of using such a combination in a reactor. However, if the uranium is used in rods of the order of an inch or two in diameter which are embedded in a matrix of solid graphite (heterogeneous system), the resonance escape probability is increased and the product pf has a maximum of about 0.8. Since ε is about 1.03 in this case, k_∞ can be about 1.09 (the value for an actual reactor will be somewhat less than this because of losses in the cladding, structural materials, etc.) and it is possible to construct such a reactor, albeit a rather large one. The original Chicago reactor and the large Hanford reactors are of this type.

For reactors using enriched fuel, the four-factor formulation is not as useful and the design is done from first principles. This involves a solution (usually by machine calculations) of the Boltzmann diffusion equations to whatever degree of approximation is needed. No attempt will be

made to discuss these methods here, but descriptions can be found in the standard treatises.

So far, no mention has been made of one of the most important aspects of the fission process: the presence of delayed neutrons. The masses of the fission-product nuclei are generally far from the region of stability. A few of them move toward the stability region by emitting beta particles and then promptly emitting neutrons. The delay time of the beta decay is also the delay time of the delayed neutrons. Table 6.2 lists the properties of the six known groups of delayed neutrons emitted during the fission of U^{235}. The fractional yield (β_i) is the yield divided by ν (2.47 for U^{235}). It represents the fraction of all fission neutrons which are due to that particular group of delayed neutrons. The sum of the β_i's (equal to 0.00666 and represented by β) is the fraction of all the fission neutrons which are delayed. The mean life, $\bar{\tau}$, is the weighted average of the $\sum_i \tau_i \beta_i/\beta$ and is equal to 12.3 sec.

Table 6.2

DELAYED NEUTRONS FROM THERMAL FISSION OF U^{235}

Group	Probable Precursor	Mean Life (sec)	Yield (neutrons/fission)	Fractional Yield
1	Br^{87}	56	0.00063	0.00025
2	I^{137}	22	0.00351	0.00142
3	Br^{89}	4.5	0.00310	0.00125
4	I^{139}	1.5	0.00672	0.00272
5	As^{85}	0.43	0.00211	0.00085
6	Li^{9}	0.05	0.00043	0.00017
Total			0.01650	0.00666

Source: G. R. Keepin, et al., Phys. Rev., Vol. 107, 1957, pp. 1044-1049.

So far we have discussed only the steady-state behavior of a reactor. Let us suppose that a reactor operating at a constant power level (k = 1) suddenly had its multiplication factor changed by δk. If the mean lifetime

of the neutrons, which is also the time between generations, is represented by ℓ and there are n fissions per unit volume, we can write

$$\frac{dn}{dt} = \frac{\delta\, kn}{\ell} ,$$

which has the solution

$$n = n_o \exp\left(\frac{\delta k}{\ell}\right)t = n_o \exp(t/T) ,$$

where $T = \ell/\delta k$ and is called the reactor period or e-folding time. For fission neutrons in a natural uranium reactor, ℓ is about 0. 001 sec. This means then that for a δk of 0. 005, or one-half of one percent, the reactor period would be 0. 2 sec and the power level would increase by e^5, about 150-fold, in one second. This would make control of the reactor very difficult. Fortunately, the effect of the delayed neutrons is to raise ℓ to an effective value of about 0. 1 sec, provided δk is less than β. The reactor period then becomes 20 sec, and the reactor is easily controllable.

Using a time-dependent diffusion equation and taking into account the delayed neutrons, it is possible to derive the following equation:

$$\rho = \frac{k - 1}{k} = \frac{\ell}{Tk} + \sum_i \frac{\beta_i}{1 + T/\tau_i} .$$

The quantity $\rho = \delta k/k$ is called the reactivity, and k refers to k-effective.

This is the famous inhour equation (discussed in Section 6. 2) which gives the relationship between the reactivity and the reactor period in terms of the delayed neutron parameters and the prompt neutron lifetime. If m groups of delayed neutrons are considered and ρ is known, the equation is an algebraic equation of order m + 1 (in T). Of the m + 1 roots, m are negative and the corresponding terms in the solution for neutron flux damp out exponentially with time. The remaining root is algebraically the largest, and hence the corresponding term in the flux equation is dominant. The root will be negative if ρ is negative and positive if ρ is positive. The inhour

equation can be solved analytically or by using a type of analog computer known as a reactor simulator. Conversely, of course, if T is known, the equation can be used to find the corresponding reactivity.

There are two important limiting cases for which the equation is easily solved approximately.

Case I: Large positive or negative T. If T is large compared with $\bar{\tau}$, i. e., $T/\tau_i >> 1$, we can neglect the first term and the 1 in the denominators of the other terms. The equation then reduces to

$$\rho = \frac{\sum \beta_i \tau_i}{T} = \frac{\beta \bar{\tau}}{T} .$$

Using the values $\beta = 0.00666$ and $\bar{\tau} = 12.3$ sec for U^{235}, we have $\rho = 0.093/T$, which corresponds to the effective value of 0.1 sec for ℓ which was quoted above. The value of β will vary with the average energy of the neutrons producing fission. For example, in the TRIGA Mark I, with the average energy of the order of five time thermal, $\beta\bar{\tau}$ is 0.11 sec.

Case II: Small positive T. If T is much smaller than $\bar{\tau}$, i.e., $T/\tau_i << 1$, the equation can be written

$$\rho \doteq \frac{\ell}{Tk} + \beta ,$$

or

$$T = \frac{\ell}{k(\rho - \beta)} .$$

Since k is not too far from unity, this means that the period is essentially the mean lifetime of the prompt neutrons divided by the excess of ρ over β, i. e., by the excess of k over $1 + \beta$. The reason for this is easy to see. If k exceeds $1 + \beta$, then the reactor will be critical on prompt neutrons alone; the delayed neutrons will no longer affect the mean lifetime; and the period will be orders of magnitude smaller than when it is governed by the delayed neutrons. If $k = 1 + \beta$, the reactor is said to be prompt critical.

Obviously, the relative size of ρ compared with β is of very great importance in determining the reactor period and hence the ease of control.

This is the underlying reason for the definition of one of the standard units of reactivity: the dollar.

The reactivity of a system is one dollar if $\rho = \beta$.

One hundredth of a dollar is one cent. For the TRIGA Mark I, $\beta = 0.007$, so a two-dollar flash is one for which the reactivity introduced is 0.0146 $\delta k/k$, about 1.5%.

NOTE: The value of the dollar in terms of the fraction of neutrons varies with the kind of fuel.

The variation of reactivity with temperature is complex and must, in general, be treated individually for a specific reactor. Among other effects, both the migration length and the size of the reactor will change, as will the densities of the various substances and the cross sections for the various nuclear processes. If there is a resonance absorption of neutrons at an energy range near the operating range, the resonance will be broadened by the Doppler effect, with a resulting decrease in the resonance escape probability and a decrease in reactivity. Fortunately, most reactors have negative temperature coefficients, i.e., the multiplication factor decreases with increasing temperature. The unique properties of TRIGA which are responsible for its large prompt negative temperature coefficient are discussed more throughly in Technical Foundations of TRIGA.*

6.2. INHOUR EQUATION FOR THE TRIGA MARK I REACTOR

In order to standardize our terminology concerning measurements of reactivity in the TRIGA Mark I, a numerical statement of the inhour equation is shown in Fig. 6.1. Here the excess reactivity in dollars is plotted versus reactor period in seconds and as a function of ℓ/β from an IBM-650 computation based on the following equation:

$$\frac{\delta k}{\beta_{eff.}} = \frac{\ell}{\beta T} + \sum_{i=1}^{6} \frac{f_i}{1 + \lambda_i T} ,$$

*General Atomic, Report GA-471, August 27, 1958.

where ℓ = prompt generation time,

β = effective value of the dollar and is taken as 0.007 ,

f_i = fraction of delayed neutrons in the i^{th} group,

$\lambda_i = 1/\tau_i$ = decay constant for the i^{th} group.

The experimental data from transient tests were found to fit a value of $\ell/\beta = 0.011$ Thus, $\ell \sim 45$ μsec is the best estimate of the lifetime for $\beta = 0.007$. The delayed-neutron data have been derived from G. R. Keepin, et al.*

The physics of the TRIGA Mark I reactor have been studied in considerable detail on a subcritical mockup and on the operating reactor. In the succeeding sections, the more important aspects of these studies are discussed.

6.2.1. Critical Mass

The TRIGA Mark I reactor initially attained criticality with 54 fuel elements, or about 1.9 kg of U^{235}. The present fuel loading is approximately 80 fuel elements, i.e., about 2.9 kg of U^{235}.

6.2.2. Moderating Properties of Zirconium Hydride

Experiments performed by General Atomic personnel at the Brookhaven National Laboratory prior to the construction of the original TRIGA reactor, showed that zirconium hydride has very unusual moderating properties for slow neutrons. These experimental results can be explained by assuming that the hydrogen-atom lattice vibrations can be described by an Einstein model with a characteristic energy $h\nu$ = 0.130 ev. This description is consistent with the theory that the hydrogen atom occupies a lattice site at the center of a regular tetrahedron of zirconium atoms. The basic consequences of this model, which have been experimentally verified, are that

1. Neutrons with energies less than $h\nu$ cannot lose energy in collisions with zirconium hydride.

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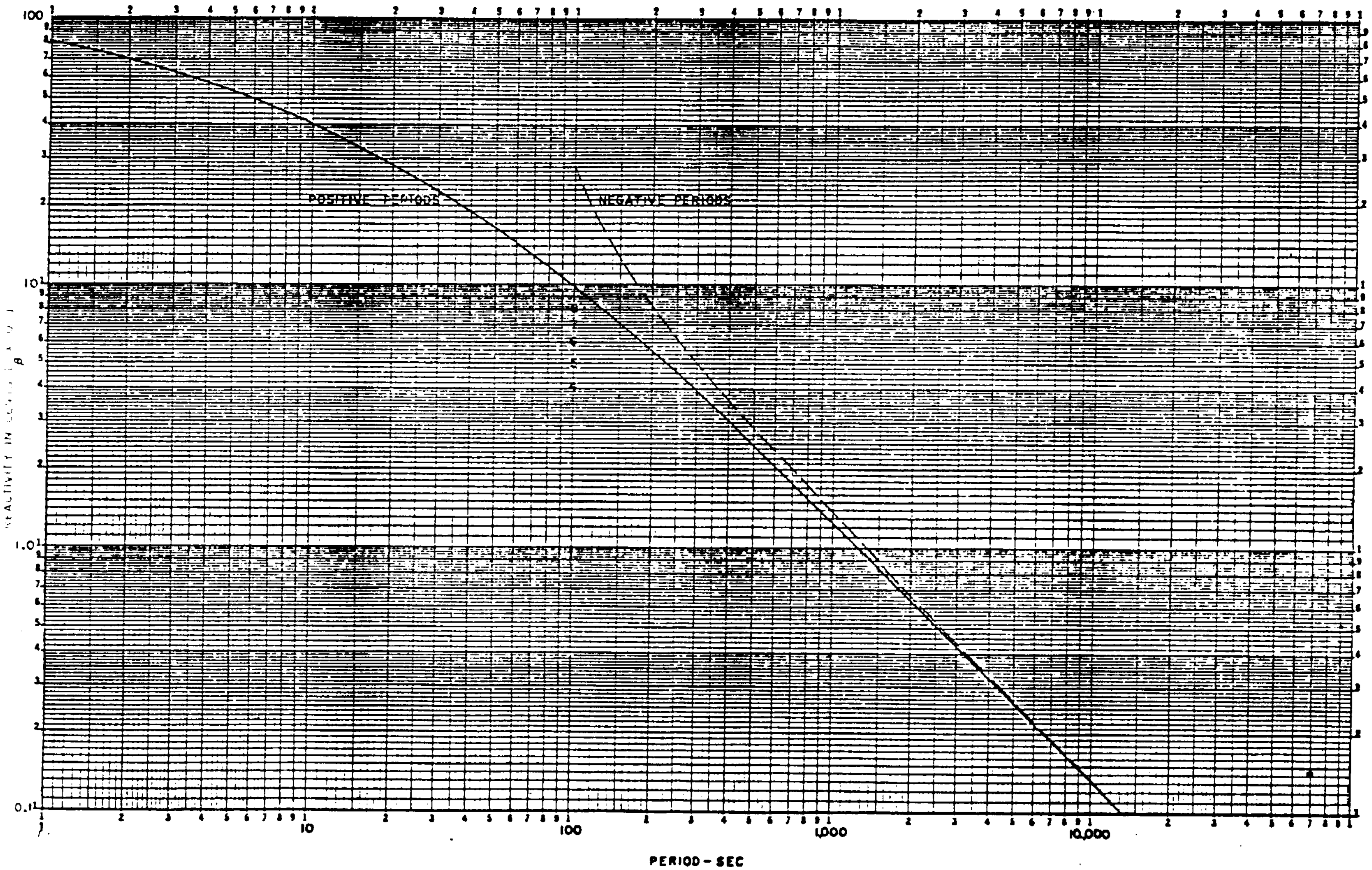
*Phys. Rev., Vol. 107, 1957, pp. 1044-1049.



Inhour curve

2. A slow neutron can gain energy $h\nu$ in a collision with zirconium hydride with a probability proportional to $\exp(-h\nu/kt)$, a probability which increases very rapidly with temperature.

Since $h\nu >> kt$, we find that zirconium hydride is not effective in thermalizing neutrons whose energies are less than $h\nu$ but that it can speed up neutrons already thermalized by water by transferring to them a quantum of energy $h\nu$.

6.2.3. Temperature Coefficients

A particularly large effort has gone into designing the reactor in such a way that an increase in the temperature of the fuel elements will result in a relatively large decrease in reactivity. This large prompt negative fuel-temperature coefficient results from the following effects:

1. The uranium in the fuel elements is approximately 20% U^{235} and 80% U^{238}. The capture resonances in U^{238} are Doppler-broadened by an increase in temperature. This causes a decrease in the energy self-shielding of the resonances and therefore an increase in the resonance capture probability. The temperature coefficient calculated for this effect is -2×10^{-5} $\delta k/k$ per $^{\circ}$C and is expected to be very nearly independent of temperature.
2. When the fuel temperature increases, the zirconium hydride temperature follows it essentially instantaneously. This increases the probability of a neutron's gaining an energy $h\nu$ from the lattice vibrations. The increased fraction of speeded-up neutrons results in an increased thermal leakage from the core and an increased <u>relative</u> number of neutron captures in the water. The temperature coefficient due to these effects is calculated to be about -1.0×10^{-4} $\delta k/k$ per $^{\circ}$C ±25% in the range from 20° to 220°C. This value assumes that the Mark I core contains four water-filled control-rod thimbles.

3. At normal temperatures, the reactivity should depend on hydride temperature as exp (-hν/kt). This implies a temperature coefficient which increases with increasing hydride temperature.

   This initial dependence is valid for a temperature of about 100°C. Experimental values of the prompt temperature coefficient have been obtained from steady-state reactivity compensation experiments on the TRIGA Mark I. These experiments indicate that the average prompt temperature coefficient is -1.2×10^{-4} δk/k per °C for fuel temperatures from 20° to 300°C.

6.2.4. Isothermal Temperature Coefficient

The isothermal temperature coefficient (bath coefficient) describes the change in reactivity associated with an equal rise in fuel and coolant temperature.

Measurements of reactivity changes as a function of temperature were made in the TRIGA Mark I reactor by heating the pool water and allowing temperature equilibrium to be established between the fuel and the pool water. No measurable changes were observed from 20° to 40°C; however, from 40° to 47°C, there was a $0.06 decrease in reactivity.

Reactivity effects associated with reactor water temperature will have little effect for the following reasons:

1. Under steady-state operation conditions, the large heat capacity of the pool will prevent rapid temperature changes.
2. The temperature of the core water does not change significantly during the prompt transient. The prompt-transient behavior of the reactor is determined primarily by changes in fuel temperature.

6.2.5. Void Coefficient

If water in the core is replaced with a very-low-density material, i.e., steam, gases, etc., a reactivity change will occur.

The void coefficient of reactivity is defined as the change in reactivity per percent void. The void coefficient is a function of position in the core,

water fraction, and water temperature. This quantity is useful in studying the effects on reactivity produced by the loss or formation of voids within the core, as could occur, for example, with the sudden flooding of an in-core experiment or the production of steam and/or radiolytic gas along the surface of a fuel element.

No detailed measurements of the void coefficient have been made with the reactor; however, in a subcritical mockup, measurements were made by displacing water with triangular pieces of styrofoam between fuel elements. These voids extended axially over the graphite reflector region. Measurements at 20°C showed a variation from -5×10^{-3} $\delta k/k$ per 1% void near the center of the core to -2×10^{-3} $\delta k/k$ per 1% void averaged over the reactor. The reactor configuration had a 2.5-in. water reflector.

Calculations indicate that the total reactivity value of the water in the TRIGA Mark I core is about 10%.

6.2.6. Reactivity Perturbations

Perturbations of reactivity resulting from physical changes in the core and reflector are of importance to the safety of the reactor.

The following reactivity changes have been measured for adding typical fuel elements, relative to the water-filled fuel location in an otherwise completely loaded core:

Ring Location	Worth of Element Compared to Water	
	\$	% $\delta k/k$
B	1.66	1.21
C	1.07	0.78
D	0.71	0.52
E	0.70	0.51
F	0.61	0.44

6.3. XENON POISONING IN THE TRIGA MARK I REACTOR

Time-dependent reactivity effects occur in any reactor as a result of fuel burnup and fission-product buildup. The fission product which

affects reactivity the most on a short-term basis is Xe^{135}, with its high thermal cross section of 2.87×10^6 barns. Calculations have been carried out for the TRIGA Mark I reactor, but calculations of xenon effect in the TRIGA Mark I indicate equilibrium reactivity losses as high as 1.9% $\delta k/k$ ($2.53) for 250 kw steady-state operation after about 30 hr continuous operation. Normally the reactor is not operated for long periods of time at high power, so xenon seldom represents a major operational problem. A family of curves illustrating the loss of reactivity as a function of xenon buildup in the TRIGA Mark I is shown in Fig. 6.2.

6.4. TRANSIENT EXPERIMENTS

A series of step reactivity insertions has been performed with the TRIGA Mark I to determine the transient characteristics of the reactor.* The parameters of interest include

1. Power as a function of time.
2. Fuel-element temperatures.
3. Reactor pulse shape.
4. Integrated energy per pulse.

Ion chambers and thermocouples were used to collect power-level and temperature data during each transient. These data were recorded on a fast 36-channel recording galvanometer. Analysis of the galvanometer traces provided information on pulse intensity (peak power), pulse shape, and fuel temperature.

Both uncompensated ion chambers and fission chambers have been found satisfactory for measurements during transients. Compensated chambers are not usable, since the apparent power is strongly dependent on the compensating voltage. The power registered by uncompensated chambers is believed to be accurate, since no change in the measured peak

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*A complete discussion of TRIGA transient behavior may be found in Technical Performance Report for The TRIGA Mark F Reactor, General Atomic Report GA-1727, October 10, 1960

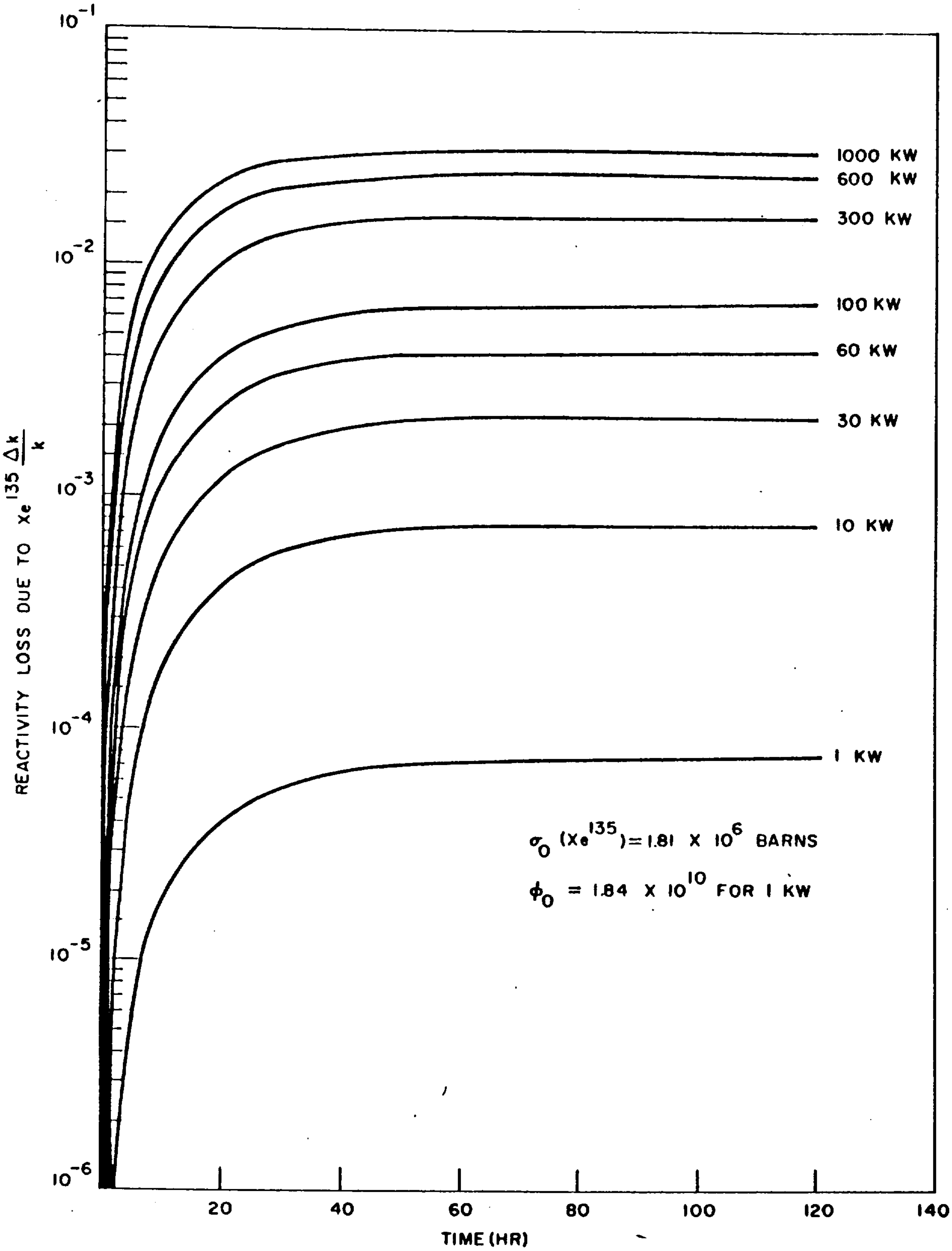


Fig. 6.2--Calculated reactivity loss due to Xe^{135} buildup

power was noticed when chamber sensitivity was varied by a factor of 30. Chambers are located several feet from the reactor core at a position at which the sensitivity is 2.0 to 5.0 × 10^{-7} amp/Mw.

Pulse shape has been examined both with ion chambers feeding directly into the recorder galvanometers and with a Keithly micromicroameter as preamplifier. No variation in response has been found for the less-sensitive ranges of the micromicroammeter.

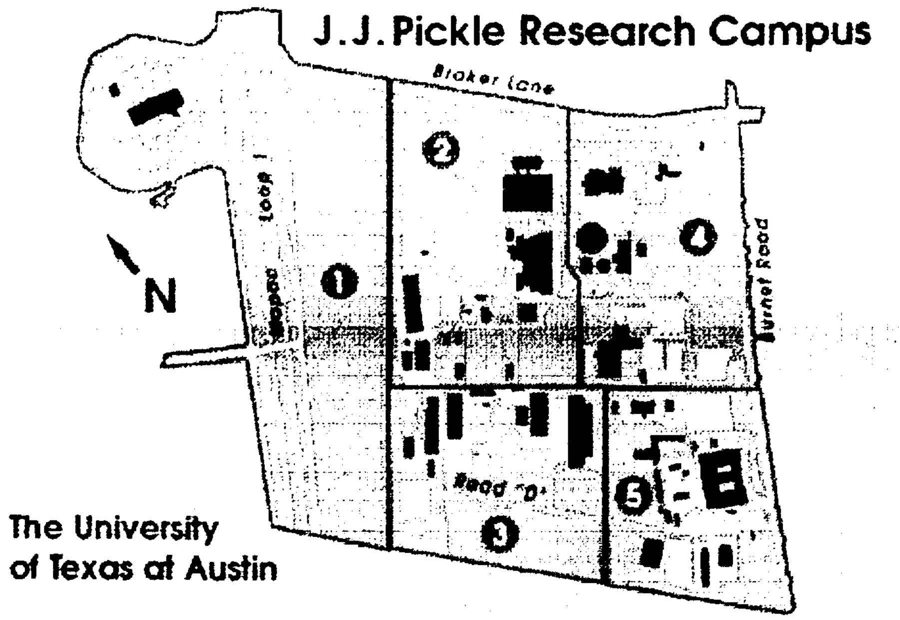
For each transient, the reactor was brought to criticality and held at a constant low power. The transient rod was adjusted so that the system reactivity rose above delayed critical by a predetermined amount when the pneumatic rod was driven out of the core.

The transient rod was removed from the core in less than 0.1 sec. After it was fully removed, the power of the reactor increased on the asymptotic period for several periods before an appreciable amount of power was generated. Thereafter, the temperature of the core increased rapidly. The reactor power increased to a maximum and was then observed to decrease with a period approximately equal to the asymptotic period. Following the prompt burst, the reactor power level decreased to the level measured in the quasi-equilibrium experiments. The major portion of the transient energy release was stored as thermal energy in the fuel elements during the transient, and it subsequently diffused slowly into the cooling water. For typical transient experiments, the reactor is scrammed one second after the burst to reduce the effects of delayed neutron heating.

The TRIGA Mark I is licensed for step reactivity insertions as large as $3.00; many such 16 Mw-sec/800 Mw peak power transients have been performed to demonstrate the safety of the system. In conducting transient studies, measured fuel temperatures were found to approach the $ZrH_{1.6}$ phase transition temperature (530°C).

In order to minimize metallurgical effects resulting from high local temperatures, routine pulsing reactivity insertions are restricted

to \$3.00 except for special experiments. Under these conditions, peak fuel-element temperatures will not exceed 530°C, even if the reactor is not scrammed following a pulse.



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PRC Maps : Overview Buildings

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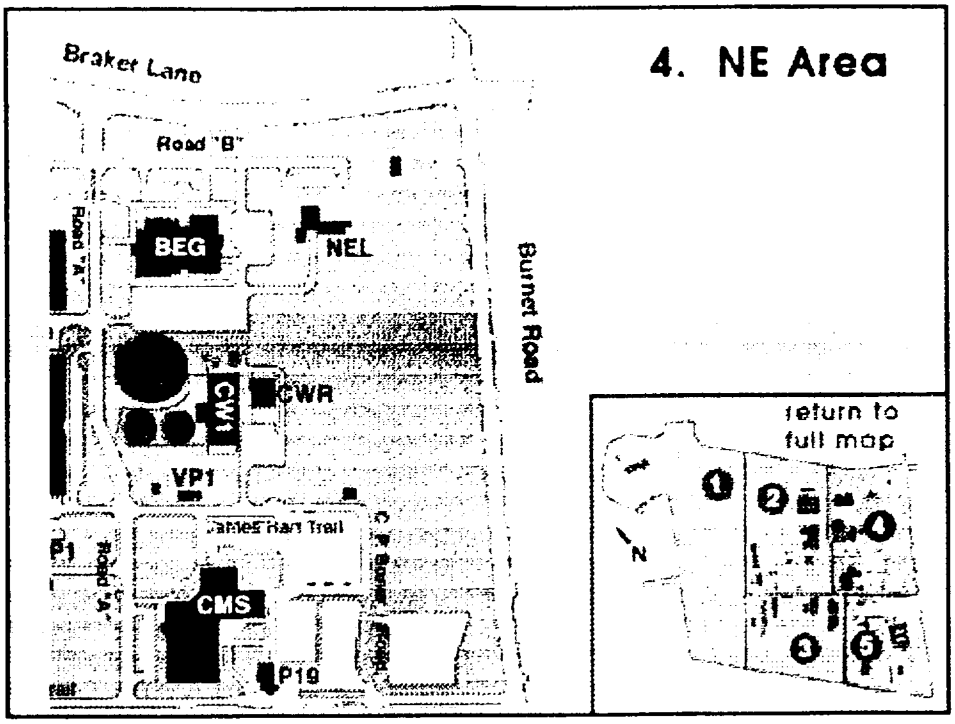

BEG	B.E.G. Research and Administration Building	CMS	The Commons
CW1	Physical Modeling Building	CWR	Center Water Resources Lab
NEL	Nuclear Engineering Teaching Lab	P19	PRC Custodial/Safety/Guards
VP1	Radiocarbon Lab		

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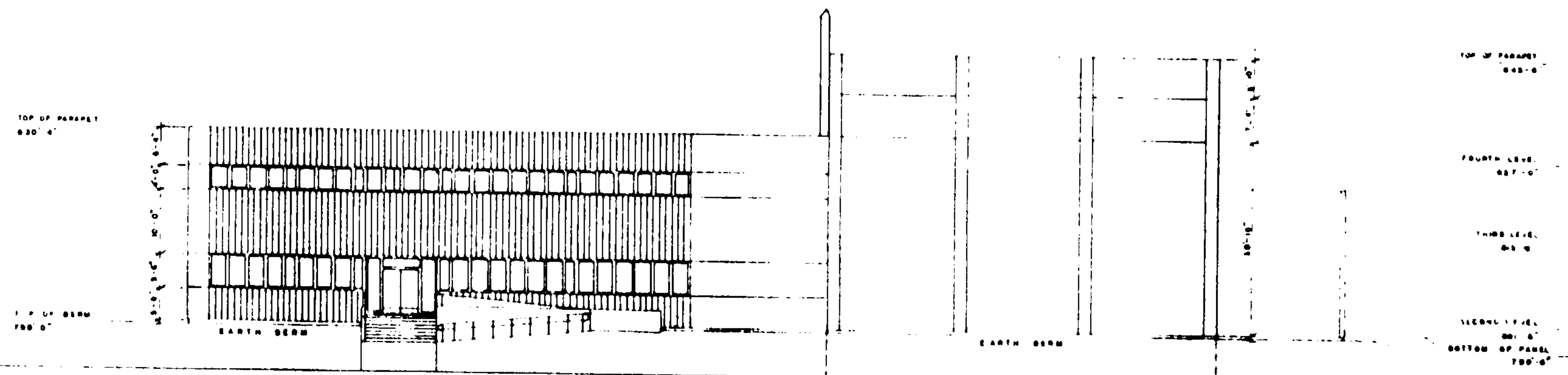
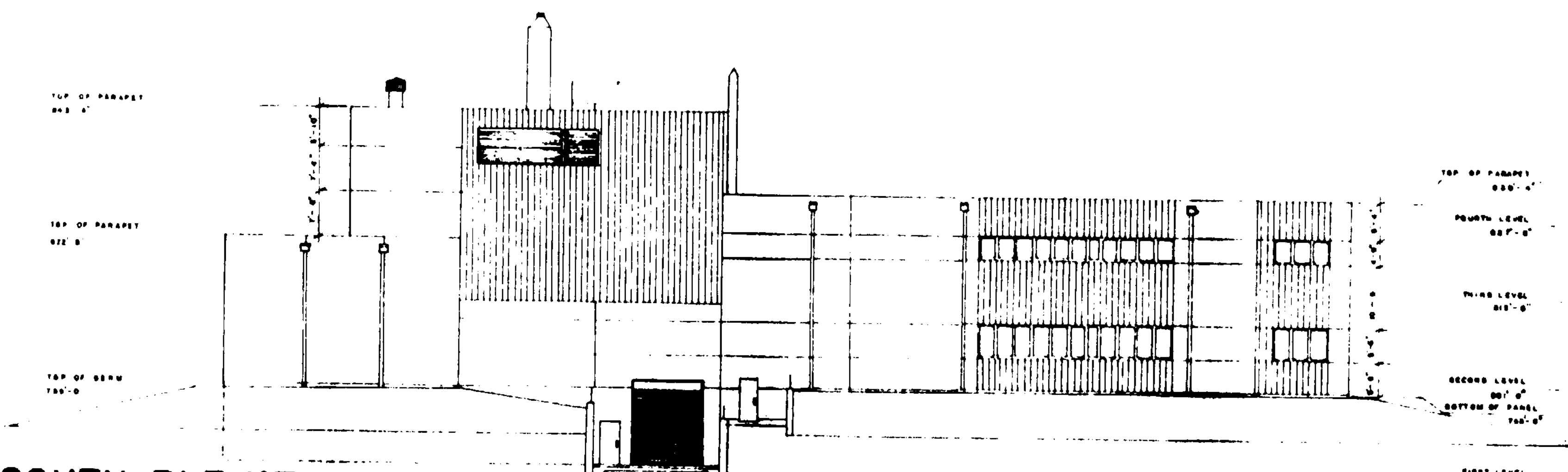


NORTH ELEVATION



SOUTH ELEVATION

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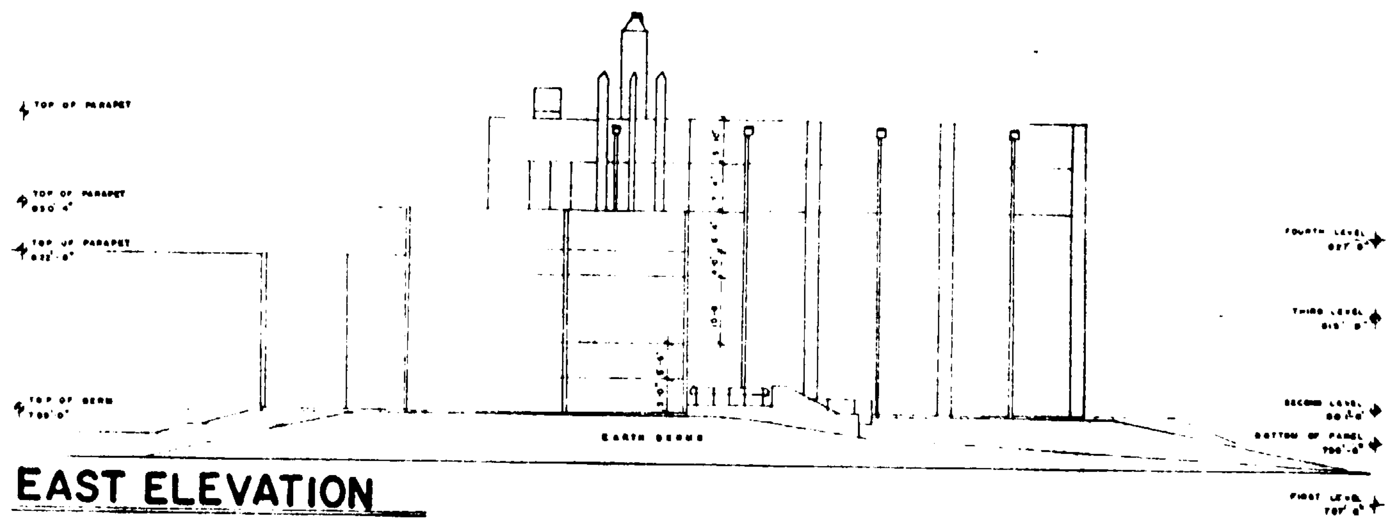
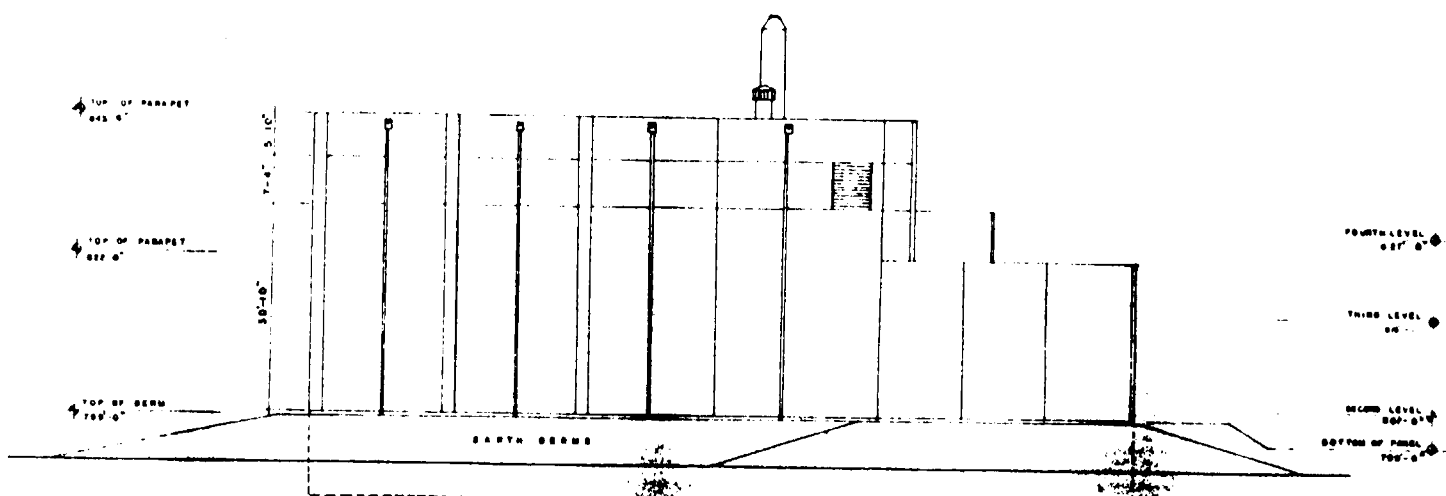


EAST ELEVATION



WEST ELEVATION

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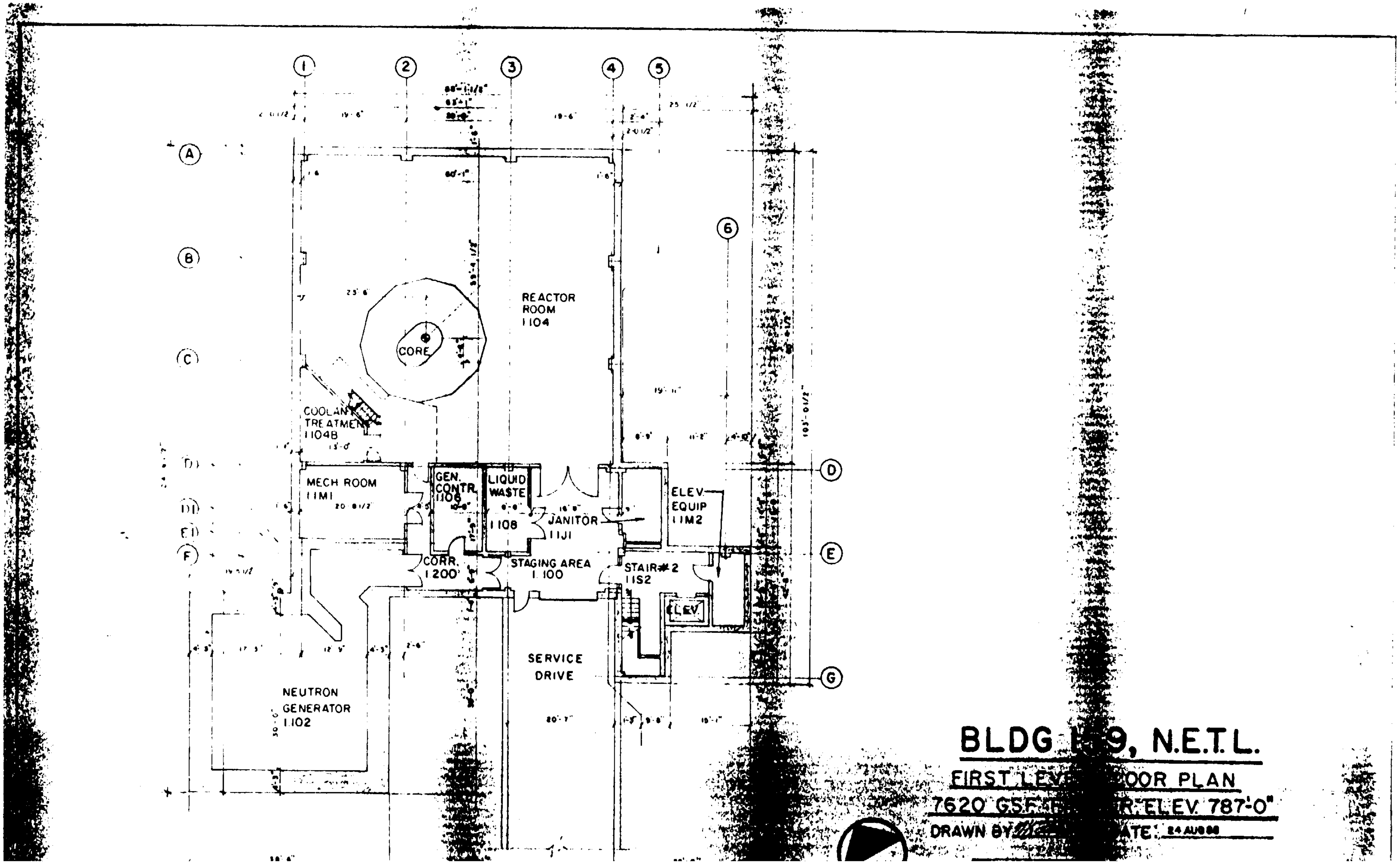

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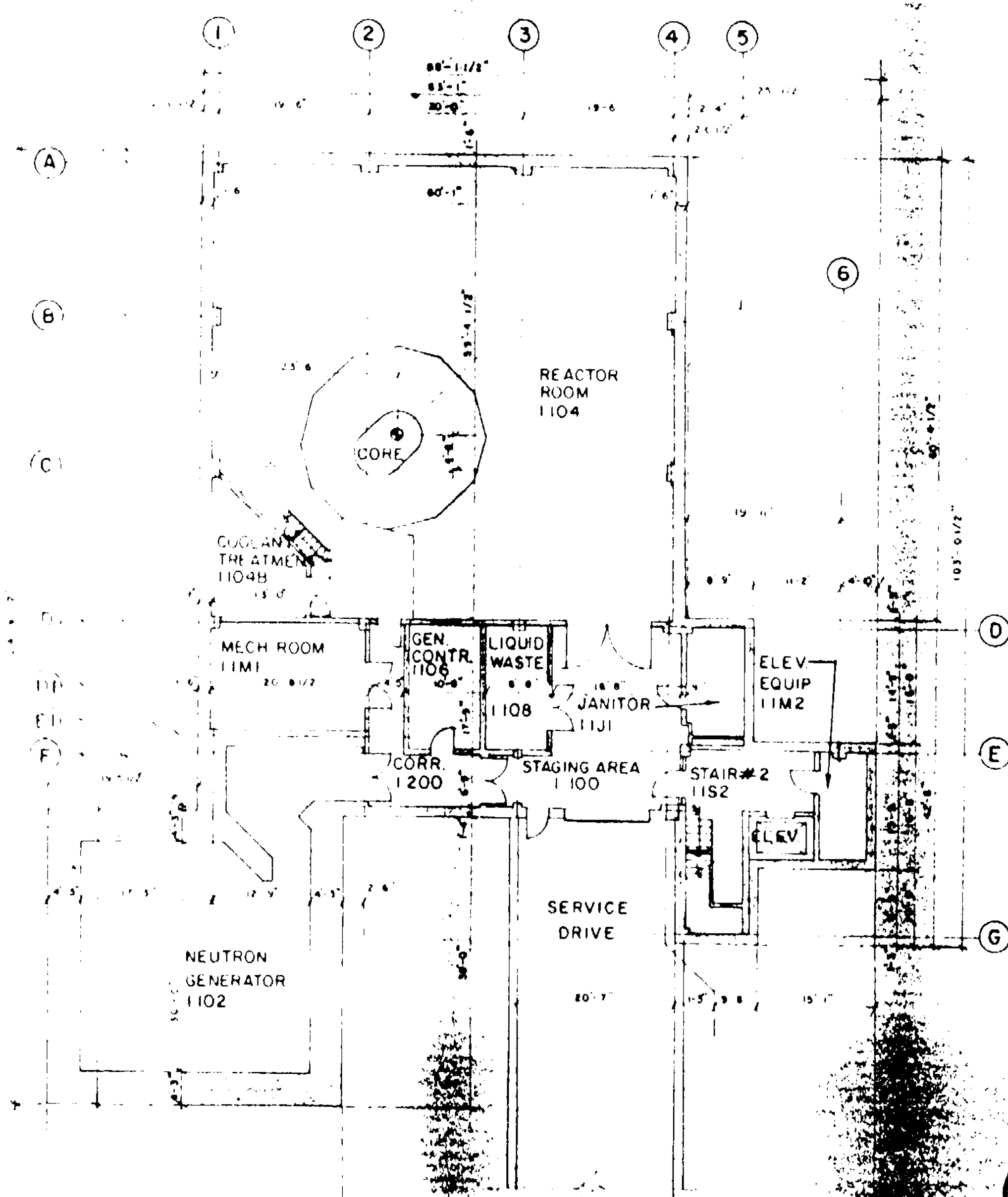

BLDG 159, N.E.T.L.

FIRST LEVEL FLOOR PLAN

7620 GSF FIN. FLR. ELEV. 787'-0"

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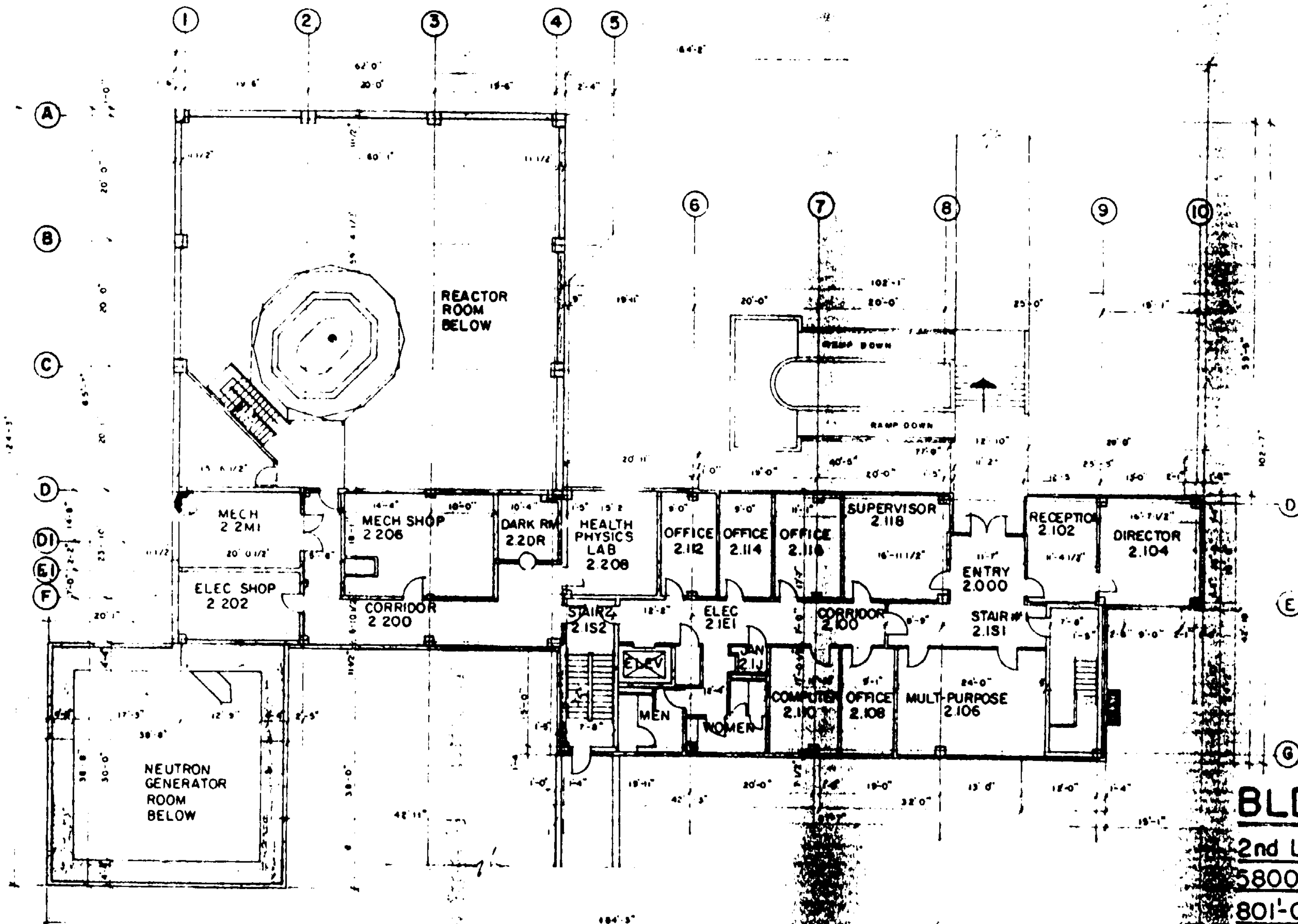

SCALE: 1/16"=1'-0"

BLDG 159, NETL

2nd LEVEL FLOOR PLAN

5800 G.S.F. FIN. FLR ELEV. 801'-0"

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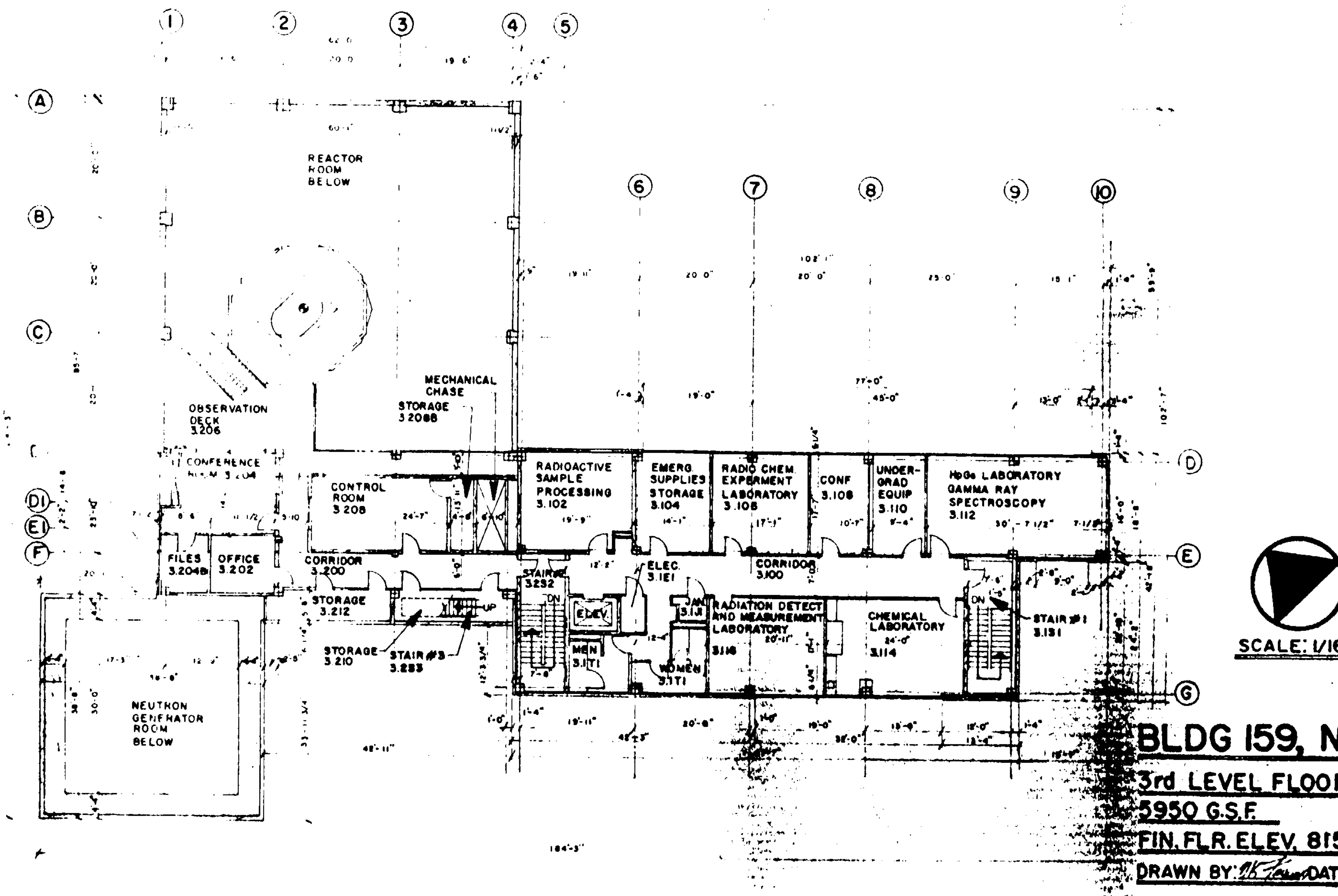

BLDG 159, NETL

3rd LEVEL FLOOR PLAN

5950 G.S.F.

FIN. FL.R. ELEV. 815'-0"

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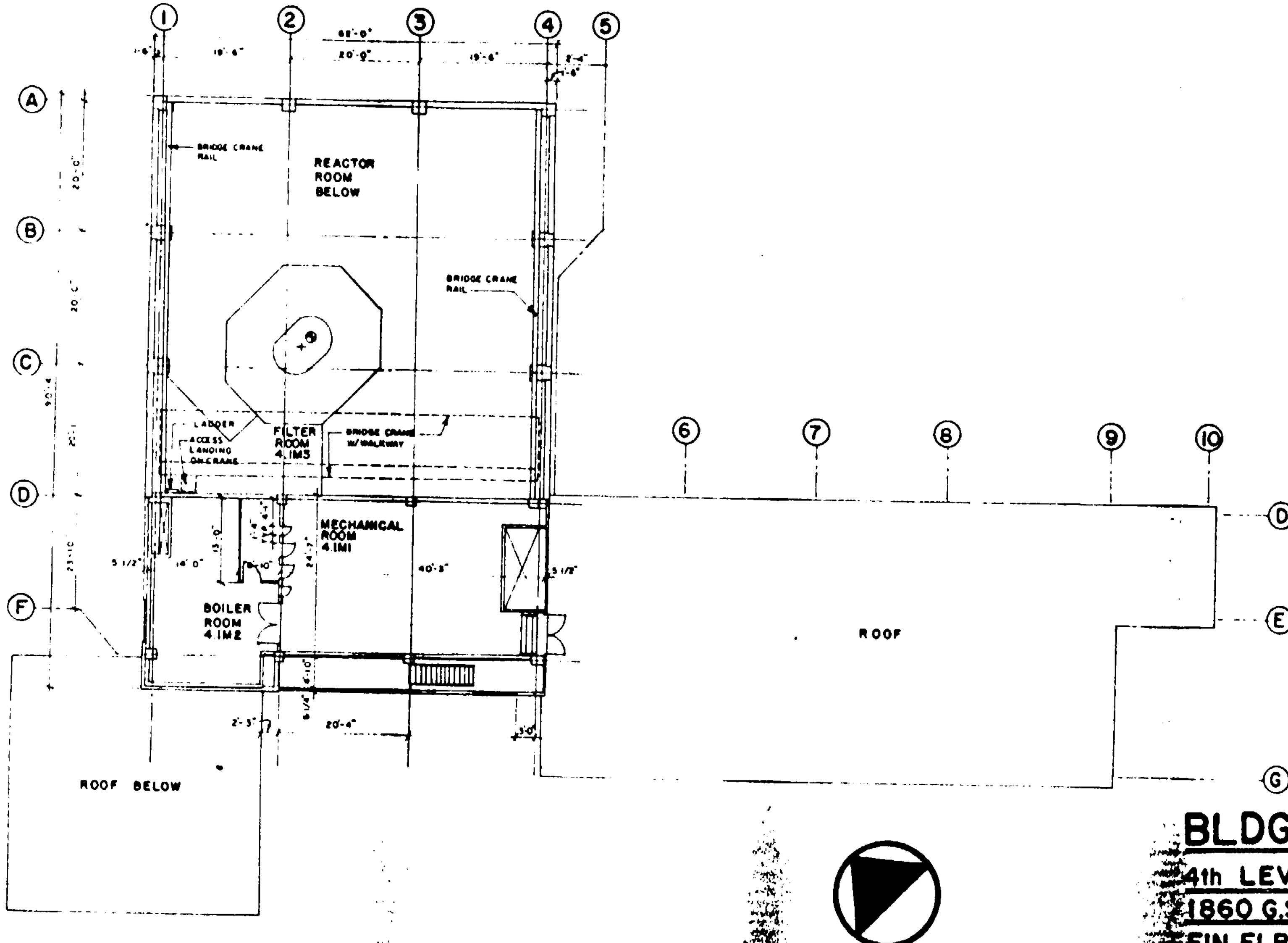
SCALE: 1/16"=1'-0"



BLDG 159, N.E.T.L.

4th LEVEL FLOOR PLAN

1860 G.S.F.

FIN. FLR ELEV. 827'-0"

SCALE: 1/16"=1'-0"

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Rules of Thumb

alpha particles

An alpha particle of at least 7.5 MeV energy is needed to penetrate the nominal protective layer of skin.

(0.07mm or $7 mg/cm^2$)

An estimate of the range of alpha particles in air depends on the particle energy.

0.6 cm/MeV	0 - 4 MeV in air
1.2 cm/MeV	4 - 8 MeV in air

beta particles

A beta particle of at least 70 keV energy is needed to penetrate the nominal protective layer of skin.

(0.07mm or $7 mg/cm^2$)

The approximate range for beta particles in any medium depends on the material density.

0.4 cm/keV	in air
0.2 cm/MeV	low density
0.1 cm/MeV	moderate density

gamma radiation

For most gamma rays the exposure rate in r/hr at one meter in air is given to within 20% by the source strength in Curies and the energy in MeV.

$$D(t) = 0.5 * E * C$$
$$= 500 \text{ mr/hr-MeV-Ci at 1 meter}$$

The dose rate to tissue in rads/hr in an infinite medium uniformly contaminated by a gamma emitter is approximately determined by the energy in MeV and the ratio of the concentration in $Curies/meter^3$ to the material density.

$$D(t) = 2.12 * E * c/\rho$$
$$= 1640 \text{ rad/hr-(MeV-Ci/m}^3\text{) in air}$$

One-half inch of lead provides approximately one half-value-layer for 1 MeV gammas.

Four inches of water provide approximately one half-value-layer for 1 MeV gammas.

neutron radiation

Attenuation of neutrons depends on material properties for energy moderation and particle absorption.

Hydrogenous materials are good neutron energy moderators.

Specific elements such as boron and cadmium are good neutron particle absorbers.

gamma radiation

Tenth-value layer:	0.1	0.5	1.0	2.0	MeV
graphite	9.1	15.7	21.3	30.5	cm
aluminium	5.6	10.2	14.0	19.8	cm
iron	0.9	3.6	4.8	6.9	cm
water	14.0	23.9	33.0	45.7	cm
concrete	6.1	11.4	15.7	22.4	cm
lead	0.1	1.3	3.0	4.6	cm

neutron radiation

Half-value layer:	0.5	1.0	2.0	5.0	MeV
polyethylene	1.5	2.0	2.5	4.1	cm
paraffin	1.5	2.0	2.5	4.1	cm
water	1.7	2.2	2.8	4.5	cm
concrete	3.2	4.2	5.3	8.6	cm
wet soil	4.1	5.4	6.8	11.1	cm
dry soil	5.6	7.5	9.4	15.4	cm

Reactor CONTROL

Period data - Stable period data curves

Define the neutron kinetics data

Delayed neutron fraction: $\beta := .007$

Prompt neutron lifetime: $l := 42 \cdot 10^{-6}$

Six group neutron precursor data: $j := 0, 1..5$ $\quad \lambda_j := \frac{\ln(2)}{\text{thalf}_j}$ $\quad \sum_j f_j = 1$

$f_j :=$	$\text{thalf}_j :=$
.038	54.5
.213	21.8
.188	6.00
.407	2.23
.128	.496
.026	.179

Average neutron lifetime: $\text{lavg} := (1-\beta)\cdot l + \sum_j \left[(f_j \cdot \beta) \cdot \left(\frac{1}{\lambda_j} + l \right) \right]$

$\text{lavg} = 0.089$

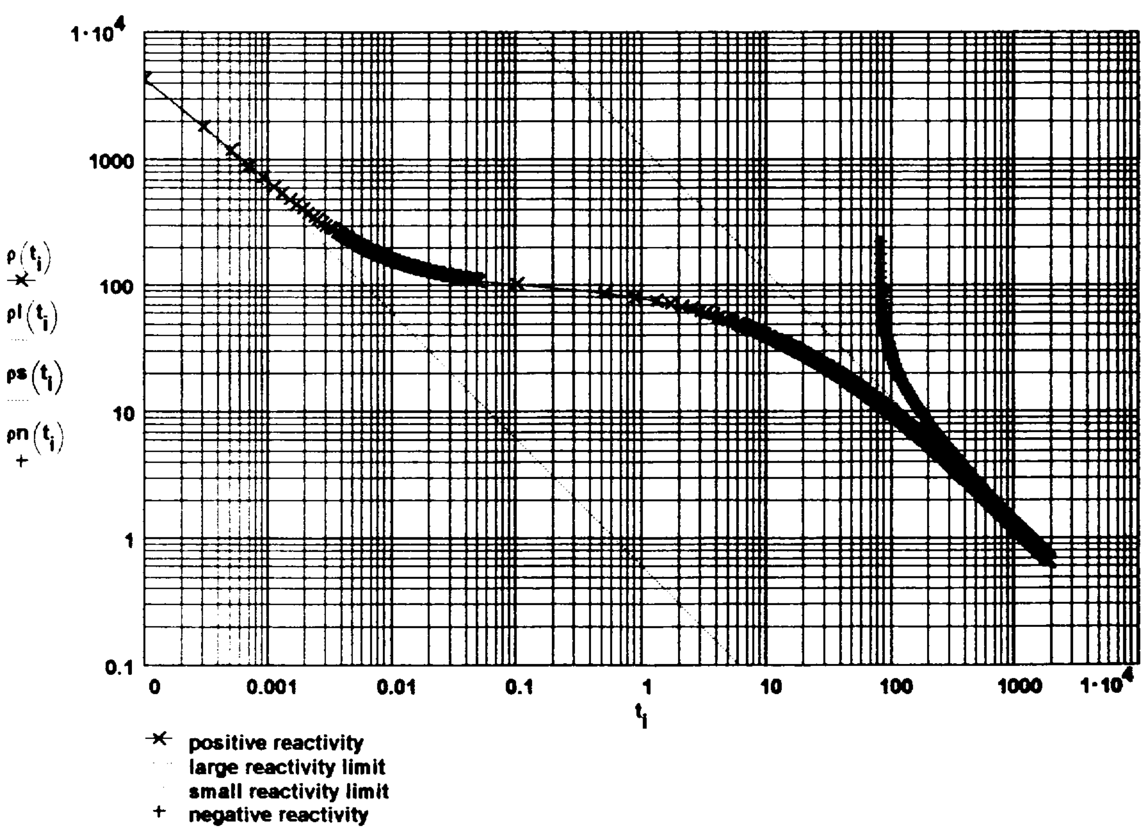
Define the reactivity equation: $\rho(t) := \frac{100 \cdot t}{t + l} \cdot \left[\left(\frac{l}{\beta \cdot t} \right) + \sum_j \frac{f_j}{1 + \lambda_j \cdot t} \right]$

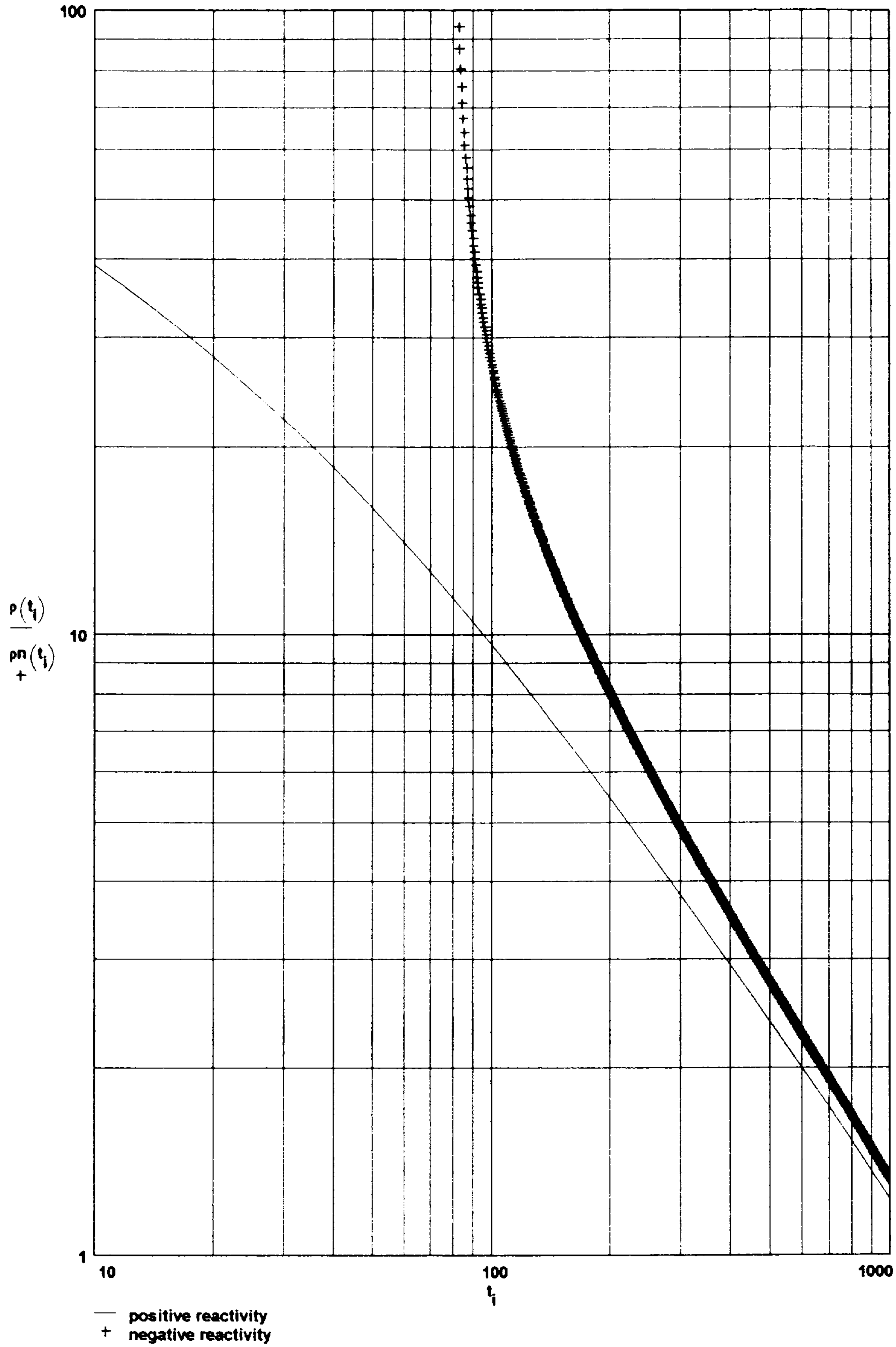
Negative reactivity: $\rho n(t) := \text{if}(t<80, 0.01, -\rho(-t))$ $\quad \rho n(80) = 235.51$

Large reactivity estimate: $\rho l(t) := \frac{100}{\beta} \cdot \frac{l}{t + l}$

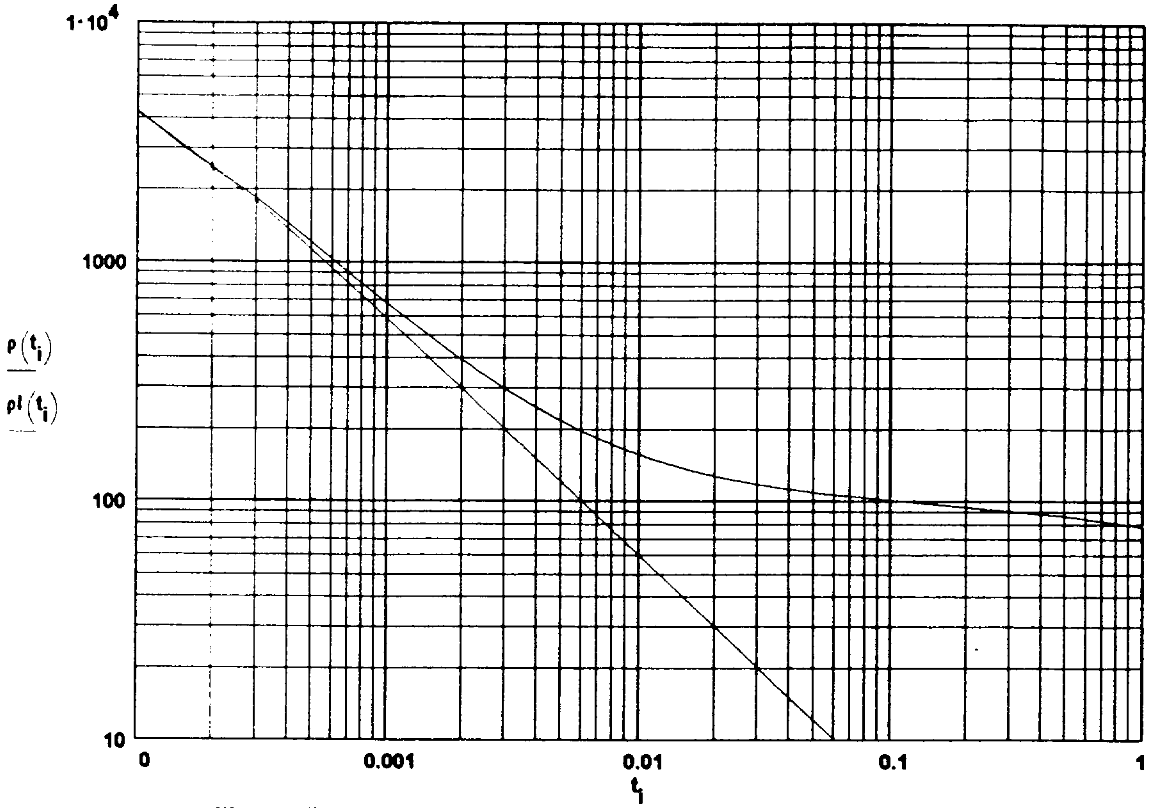
Small reactivity estimate: $\rho s(t) := \frac{100}{\beta} \cdot \frac{\text{lavg}}{t}$

Define the time data points: $i := 0, 2..10000$ $\quad t_i := \text{if}\left(i<500, \frac{i}{10000} + .0001, \frac{i - 500}{5} + 0.1 \right)$

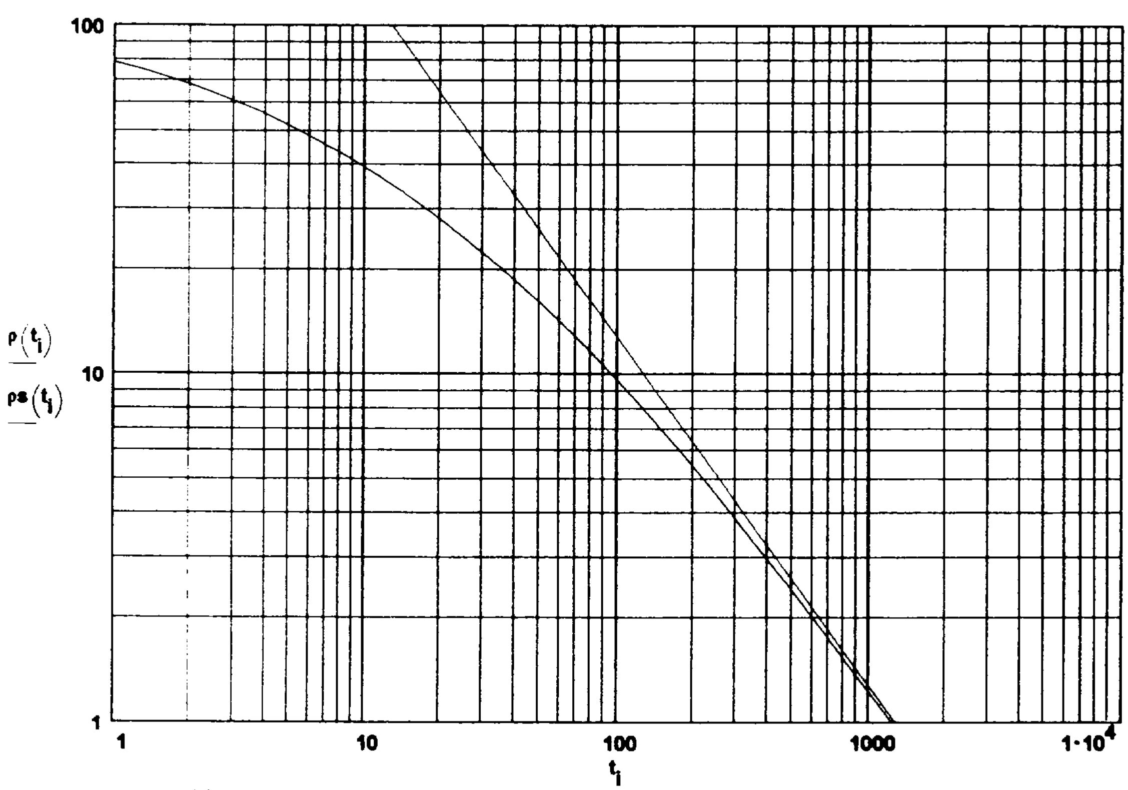




100
10
1
10
100
1000
$\frac{\rho(t_i)}{\rho n_+(t_i)}$
t_i
positive reactivity
+ negative reactivity

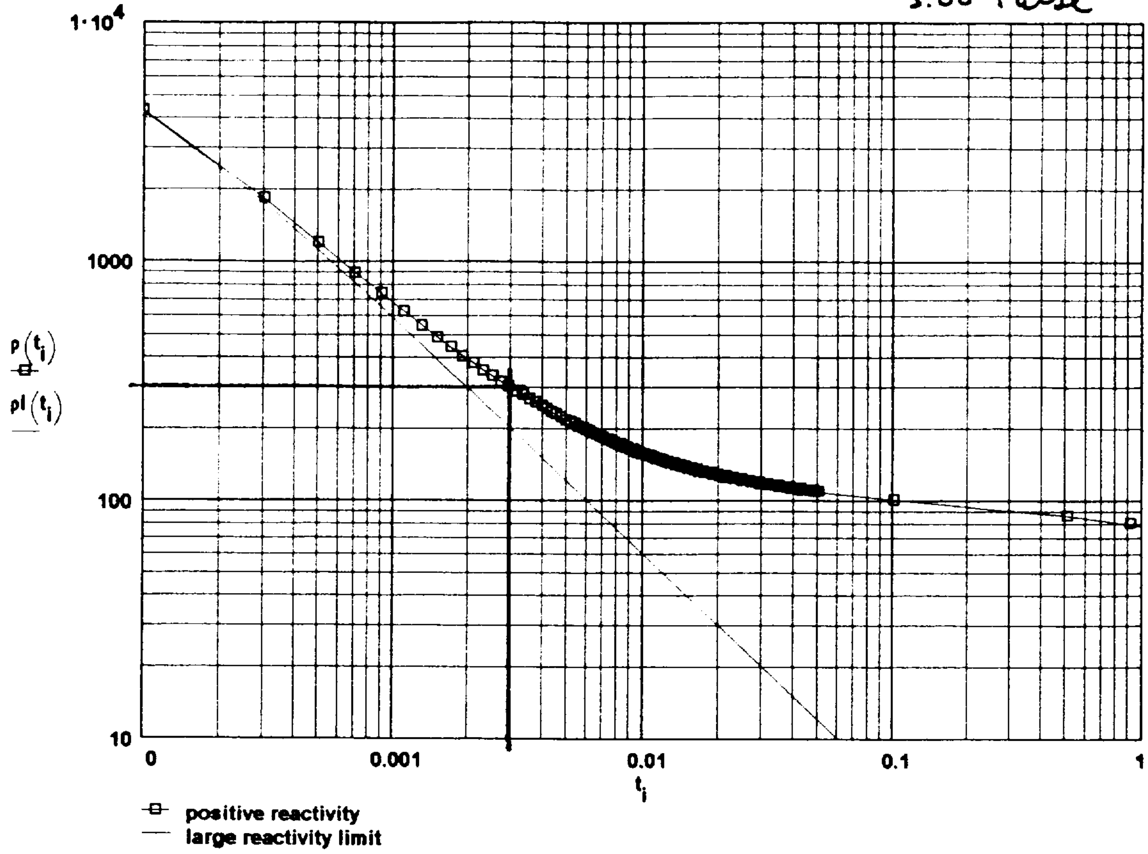


— positive reactivity
— large reactivity limit

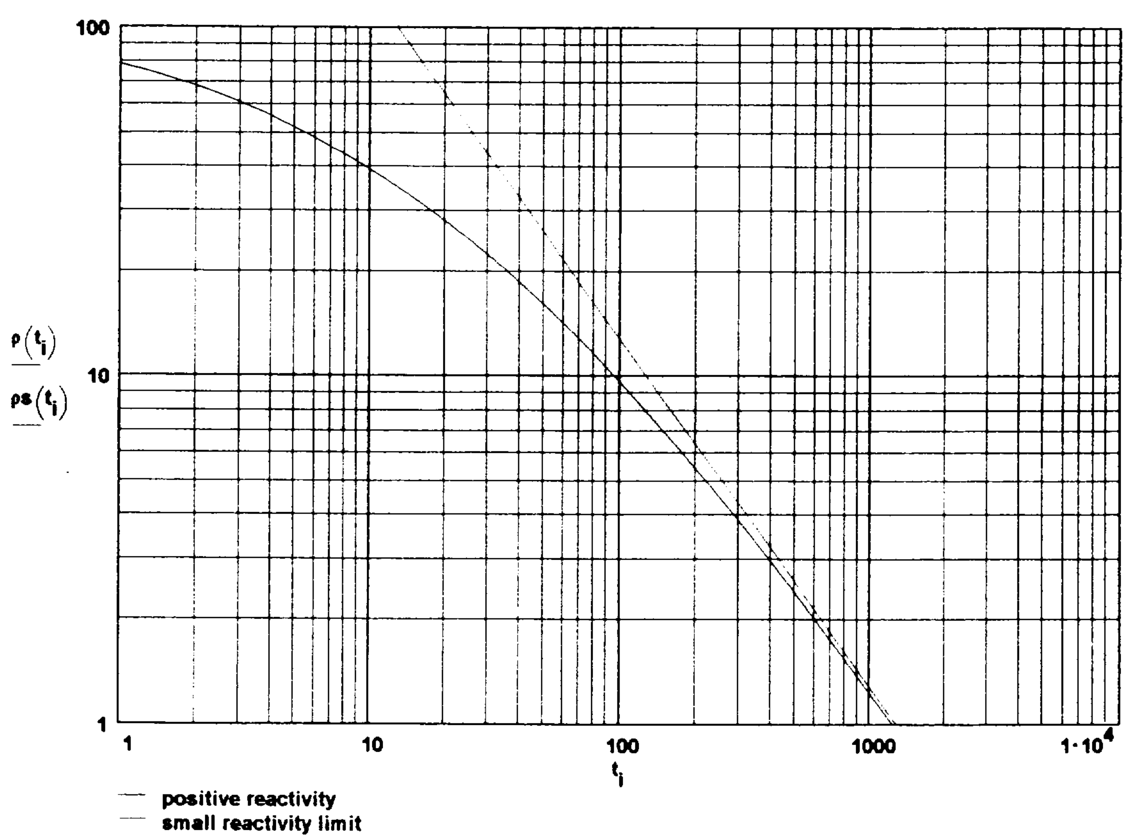


— positive reactivity
— small reactivity limit

#3.00 Pulse



1·10^4
1000
100
10
ρ(t_i)
ρl(t_i)
0
0.001
0.01
0.1
1
t_i
positive reactivity
large reactivity limit

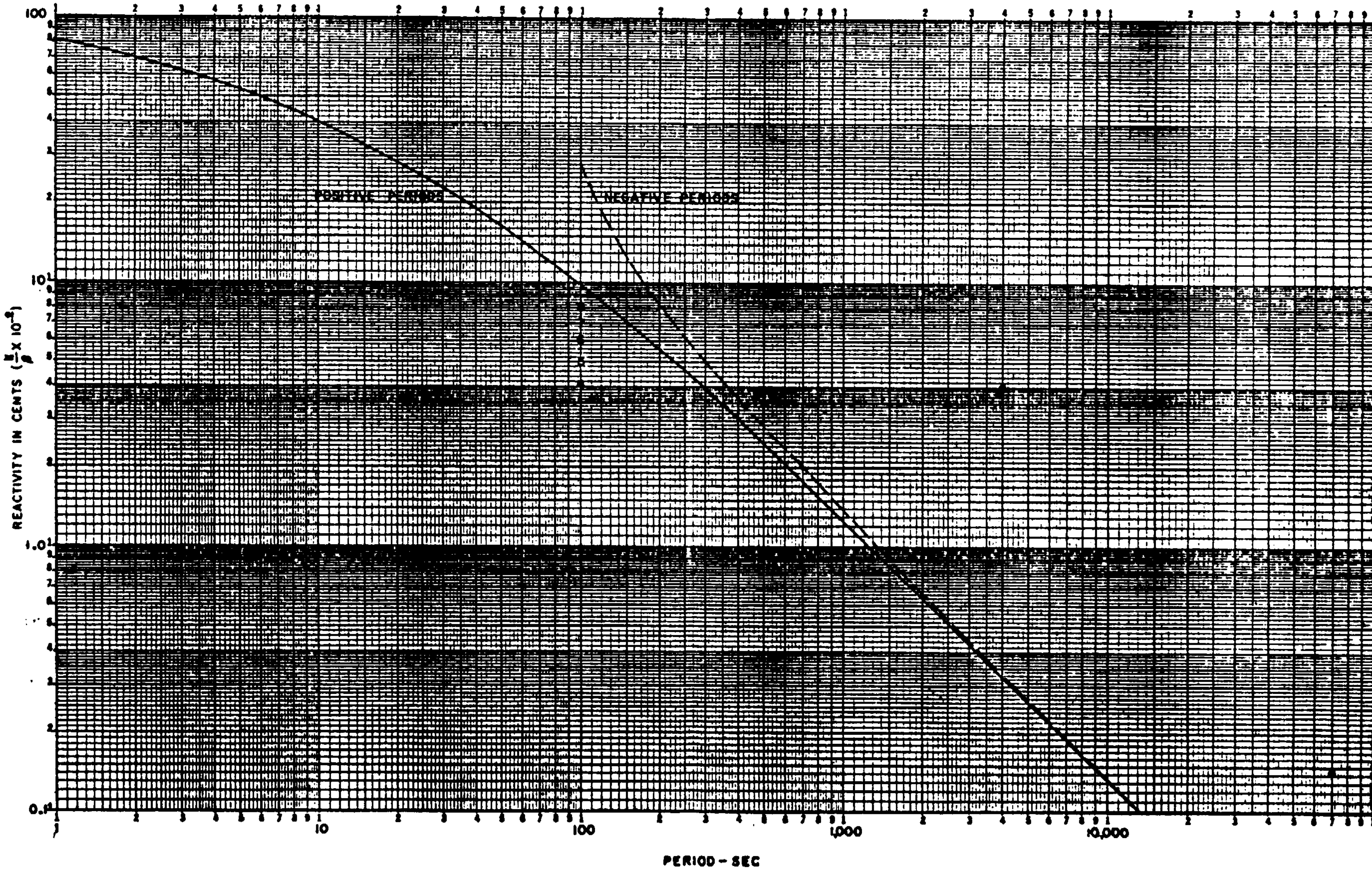


100
10
1
ρ(t_i)
ρs(t_i)
1
10
100
1000
1·10^4
t_i
positive reactivity
small reactivity limit

Period vs Reactivity

Period	Cents	Period	Cents	Period	Cents	Period	Cents
.5	87.36	10.5	38.28	21.0	27.12	41.0	18.20
1.0	78.77	11.0	37.48	22.0	26.44	42.0	17.92
1.5	72.75	11.5	36.72	23.0	25.79	43.0	17.65
2.0	68.07	12.0	36.00	24.0	25.18	44.0	17.38
2.5	64.23	12.5	35.32	25.0	24.61	45.0	17.12
3.0	60.9	13.0	34.67	26.0	24.06	46.0	16.88
3.5	58.20	13.5	34.04	27.0	23.54	47.0	16.63
4.0	55.75	14.0	33.45	28.0	23.04	48.0	16.40
4.5	53.59	14.5	32.87	29.0	22.57	49.0	16.17
5.0	51.64	15.0	32.33	30.0	22.12	50.0	15.95
5.5	49.89	15.5	31.80	31.0	21.68	51.0	15.74
6.0	48.29	16.0	31.29	32.0	21.27	52.0	15.53
6.5	46.82	16.5	30.81	33.0	20.87	53.0	15.33
7.0	45.47	17.0	30.34	34.0	20.49	54.0	15.13
7.5	44.23	17.5	29.89	35.0	20.13	55.0	14.94
8.0	43.07	18.0	29.45	36.0	19.78	56.0	14.76
8.5	41.98	18.5	29.03	37.0	19.44	57.0	14.58
9.0	40.97	19.0	28.62	38.0	19.11	58.0	14.40
9.5	40.02	19.5	28.23	39.0	18.80	59.0	14.23
10.0	39.13	20.0	27.85	40.0	18.49	60.0	14.06

Period	Cents	Period	Cents	Period	Cents	Period	Cents
62.0	13.73	105.0	9.24	210.0	5.17	425.0	2.72
64.0	13.43	110.0	8.90	220.0	4.96	450.0	2.57
66.0	13.13	115.0	8.59	230.0	4.77	475.0	2.45
68.0	12.85	120.0	8.30	240.0	4.60	500.0	2.33
70.0	12.58	125.0	8.03	250.0	4.43	525.0	2.22
72.0	12.32	130.0	7.78	260.0	4.28	550.0	2.13
74.0	12.08	135.0	7.54	270.0	4.13	575.0	2.04
76.0	11.84	140.0	7.31	280.0	4.00	600.0	1.95
78.0	11.61	145.0	7.10	290.0	3.87	625.0	1.88
80.0	11.39	150.0	6.90	300.0	3.76	650.0	1.81
82.0	11.18	155.0	6.72	310.0	3.65	675.0	1.74
84.0	10.98	160.0	6.54	320.0	3.54	700.0	1.68
86.0	10.79	165.0	6.37	330.0	3.44	725.0	1.62
88.0	10.60	170.0	6.21	340.0	3.35	750.0	1.57
90.0	10.42	175.0	6.06	350.0	3.26	775.0	1.52
92.0	10.24	180.0	5.91	360.0	3.18	800.0	1.47
94.0	10.07	185.0	5.78	370.0	3.10	825.0	1.43
96.0	9.91	190.0	5.64	380.0	3.01	850.0	1.39
98.0	9.75	195.0	5.52	390.0	2.95	875.0	1.35
100.0	9.60	200.0	5.40	400.0	2.88	900.0	1.31



Inhour curve

TRIGA
Reactivity Estimates

location	reference material	% δk/k
Ring A	fuel vs. water	4.00
Ring B	fuel vs. water	1.07
Ring C	fuel vs. water	0.85
Ring D	fuel vs. water	0.54
Ring E	fuel vs. water	0.36
Ring F	fuel vs. water	0.25
Ring G	fuel vs. water	0.19
3 elements (1D, 2E)	fuel vs. water	1.25
6 elements (6B)	fuel vs. water	?.??
dummy min.	graphite vs. water	+0.05
dummy max.	graphite vs. water	+0.20
thru tube	void vs. graphite	-0.45
piercing tube	void vs. graphite	-0.35
CTR	void vs. water	-0.50
PNT-A1	one poison sample	-?.??
PNT-G1	one poison sample	-0.16
RSR	poison 40 places	-0.40

POWER COEFFICIENT DATA
- Polynomial curve fit

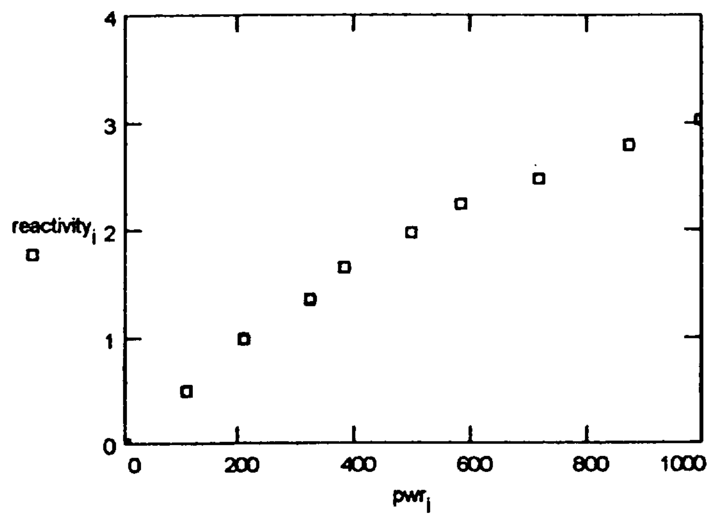
This document fits a linear or quadratic function to a set of power coefficient data.
The data is for the UT-TRIGA research reactor.
This is the initial core startup condition with 87 fuel elements.
Fuel temperature data is from two instrument elements.

Data was taken on 04/24/92.

First, define the data arrays:

$i := 0, 1 .. 9$

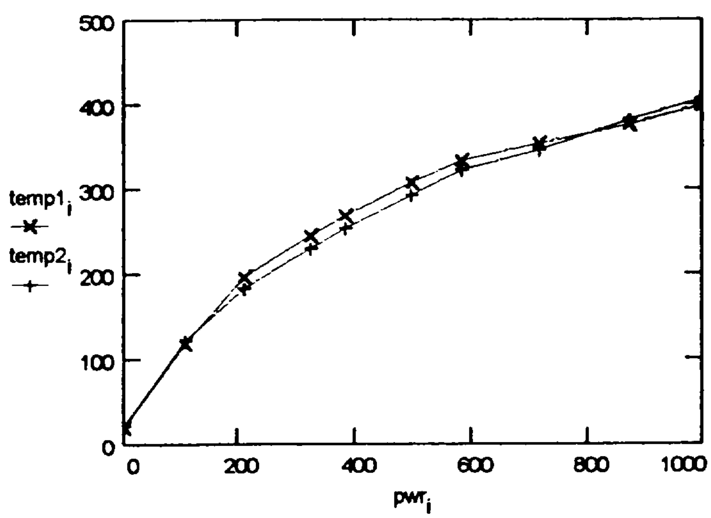
$pwr_i :=$	$reactivity_i :=$
0	0
109	0.495
210	0.973
326	1.353
383	1.640
497	1.964
583	2.240
717	2.470
871	2.786
996	3.022



$N := length(pwr)$

$N = 10$

$temp1_i :=$	$temp2_i :=$
21	23
118	123
197	183
245	229
268	254
308	293
334	323
353	346
375	382
398	405



Compute sample statistics:

$\mathrm{corr}(pwr, reactivity) = 0.9828$

Linear fit:

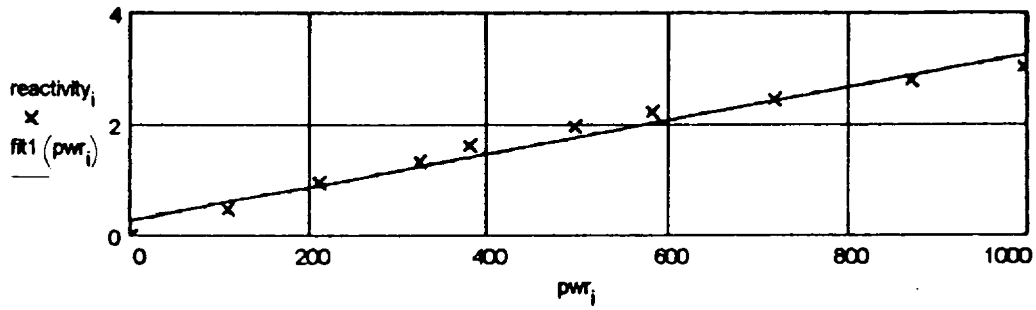
$m := \mathrm{slope}(pwr, reactivity) \qquad m = 0.003$

$b := \mathrm{intercept}(pwr, reactivity) \qquad b = 0.2916$

Linear equation: $\mathrm{fit1}(pwr) := m \cdot pwr + b$

Compute mean squared error:

$$SSE_L := \sum \overrightarrow{\left[(reactivity - \mathrm{fit1}(pwr))^2\right]} \qquad MSE_L := \frac{SSE_L}{N-2} \qquad MSE_L = 0.0379$$



Quadratic fit using matrix operations $\qquad i := 0 .. N-1$

Create second variable: pwr squared $\qquad pwr2 := \overrightarrow{(pwr^2)}$

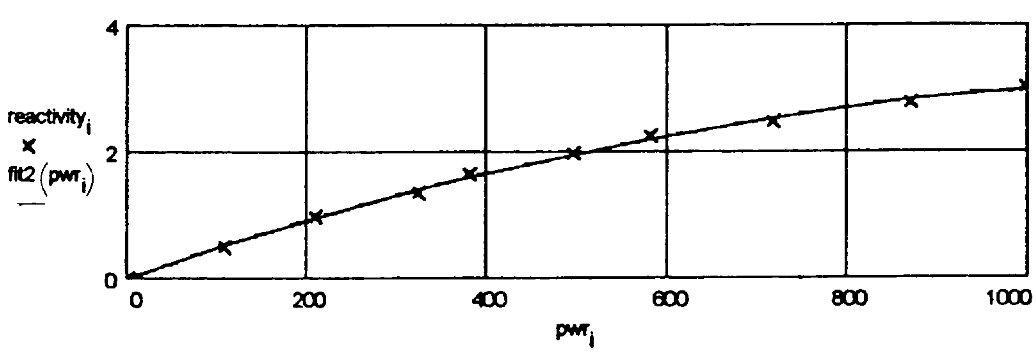
Create X matrix $\qquad X_{i,0} := 1 \qquad X^{<1>} := pwr \qquad X^{<2>} := pwr2$

$$b := \left(X^T \cdot X\right)^{-1} \cdot \left(X^T \cdot reactivity\right) \qquad b = \begin{pmatrix} 0.005 \\ 0.0049 \\ -1.8854 \cdot 10^{-6} \end{pmatrix}$$

Quadratic curve: $\qquad \mathrm{fit2}(pwr) := b_0 + b_1 \cdot pwr + b_2 \cdot pwr^2$

$$SSE_Q := \sum \overrightarrow{\left[(reactivity - \mathrm{fit2}(pwr))^2\right]} \qquad MSE_Q := \frac{SSE_Q}{N-3} \qquad MSE_Q = 0.0017$$

Graph the curve against the data:



UT-TRIGA Startup data 1992

TRIGA

Reactor Parameters

Average thermal flux per unit power	1×10^7
Peak to average flux ratio, center position	2.2
Fast to thermal flux ratio, center position	2.2
Axial peaking factor for flux or power	1.4
Effective delayed neutron fraction	.007
Prompt neutron lifetime (microseconds)	42

Delayed Neutron Groups

Neutron group	half-life	relative fraction
1	54.5	.038
2	21.8	.213
3	6.00	.188
4	2.23	.407
5	.496	.128
6	.179	.026

TRIGA
Neutron and Gammma Flux Estimates

The following table gives estimates of the thermal (.025 eV) and fast (>1 Mev) neutron fluxes and the gamma flux at the indicated locations at a unit power of 1 watt.

Location	Thermal Neutron Flux (n/cm^2-sec) per watt	Fast Neutron Flux (n/cm^2-sec) per watt	Gamma Flux (r/hr) per watt
Core center	4×10^7	2×10^7	3×10^2
Core edge	1×10^7	7×10^6	1×10^2
Reflector edge	2×10^6	2×10^5	2×10^{-1}
Center tube assembly	4×10^7	2×10^7	3×10^2
Pneumatic terminal	1×10^7	7×10^6	1×10^2
Rotary specimen rack	7×10^6	5×10^5	8×10^0
Beam-port reflector	8×10^6	5×10^6	9×10^{-1}
Beam-port shutter	8×10^3	5×10^3	2×10^{-1}

Maximum nominal power is 1.1 Megawatt.

TRIGA Reactor Core Burnup

Reference design burnup ~100 Megawatt-days $refBU = 100$

Reactor excess reactivity

Beginning of life $kBOL = 1.049$

End of life $kEOL = 1.030$

Rate of burnup

$$rate := \frac{kBOL - kEOL}{refBU} \qquad rate = 1.9 \cdot 10^{-4}$$

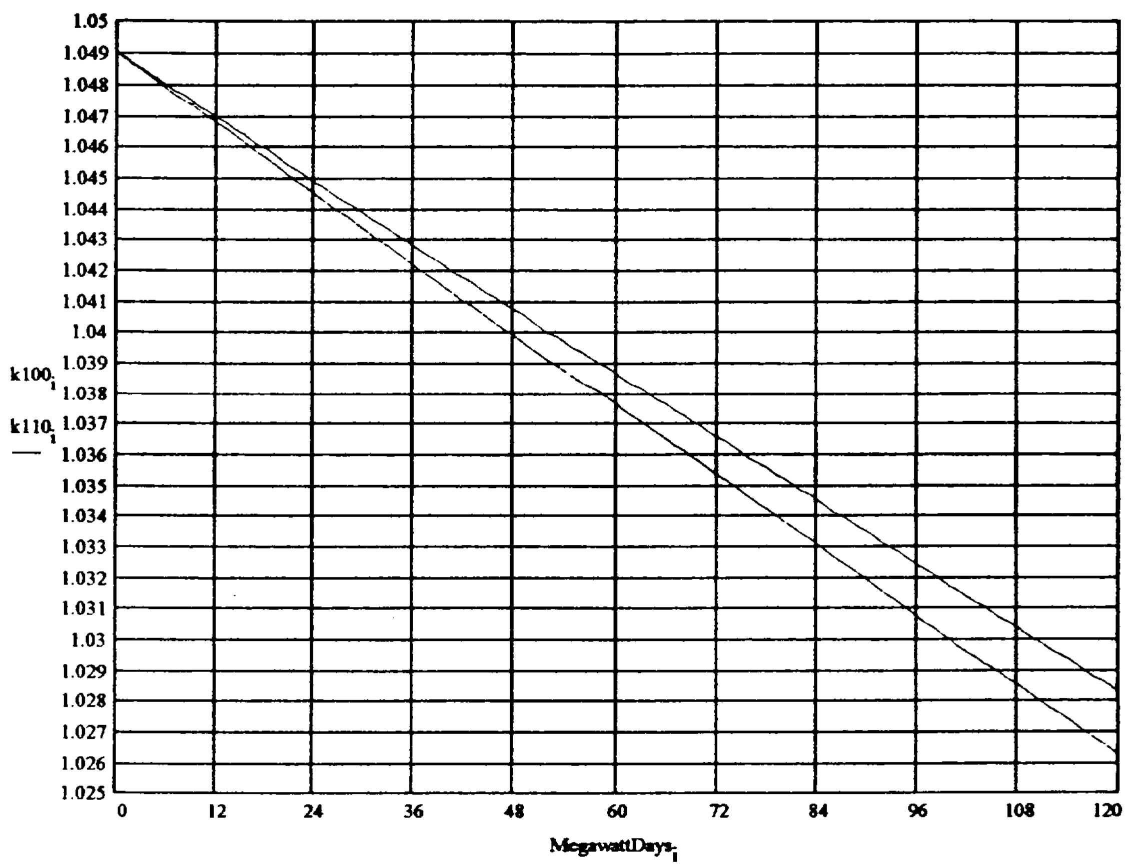
$$rate \cdot \left(100 \cdot 110^{-1}\right) = 1.727 \cdot 10^{-4}$$

$i := 0..120$

$MegawattDays_i := i$

Excess reactivity for total burnup of 100 megawatt-days $k100_i := kBOL - rate \cdot i$

Excess reactivity for total burnup of 110 megawatt-days $k110_i := kBOL - 100 \cdot 110^{-1} \cdot rate \cdot i$



UT-TRIGA Annual Data

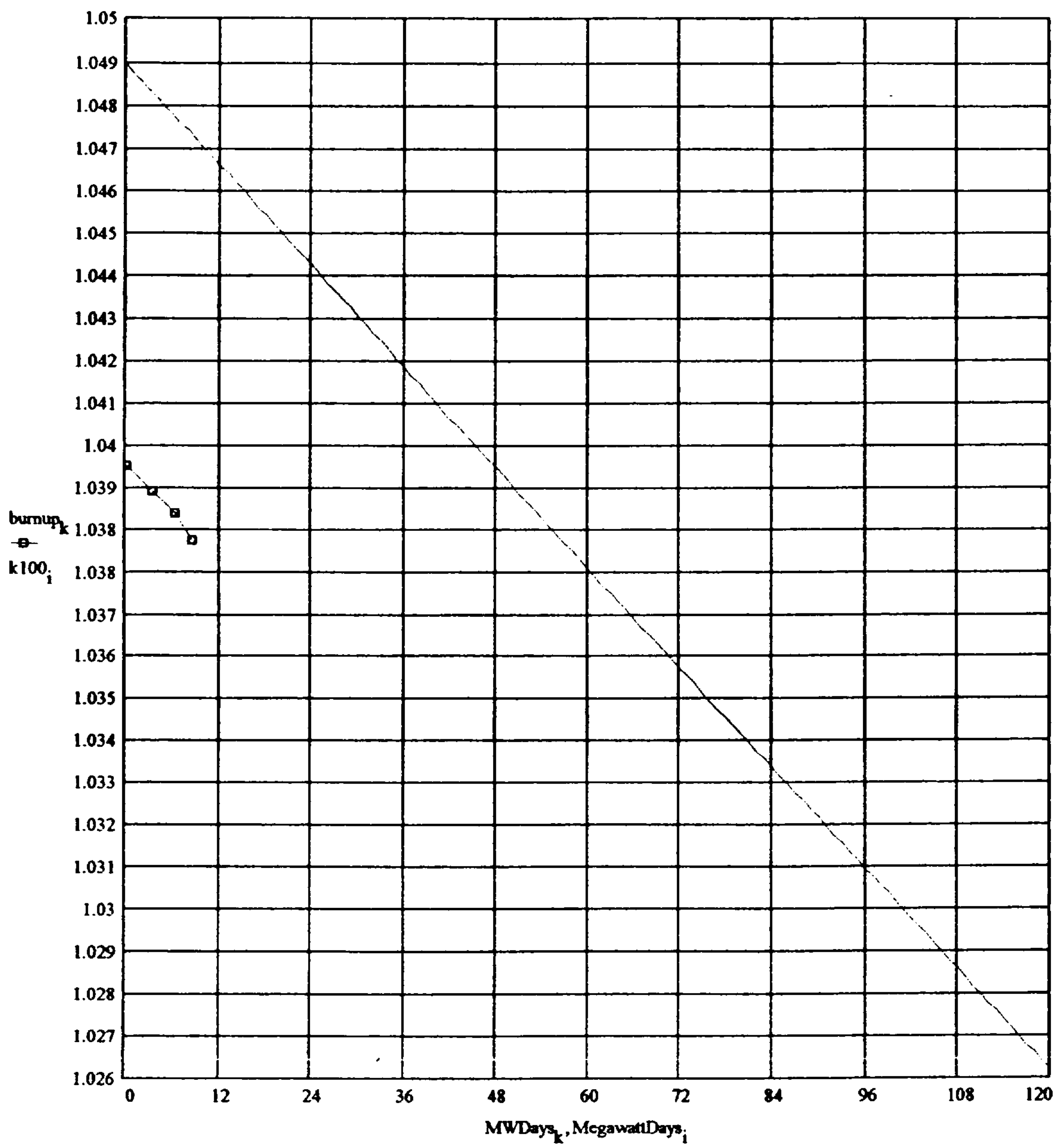
$k := 0..3$

$year_k :=$	$burnup_k :=$	$MWHRs_k :=$
1992	5.70	3.4
1993	5.62	87.8
1994	5.55	152.2
1995	5.46	205.3
1996	5.48	225.6

$burnup := 1 + burnup \cdot .007$

$MWDays_k := \frac{MWHRs_k}{24}$

$$year = \begin{bmatrix} 1992 \\ 1993 \\ 1994 \\ 1995 \end{bmatrix} \qquad burnup = \begin{bmatrix} 1.04 \\ 1.039 \\ 1.039 \\ 1.038 \end{bmatrix} \qquad MWDays = \begin{bmatrix} 0.142 \\ 3.658 \\ 6.342 \\ 8.554 \end{bmatrix}$$



CONTROL ROD CALIBRATION

Positive period data - Data entry tables   Transient Rod: 4-21-98

Reviewed
MM

(With no pneumatic transfer tube installed in G34) Standard Core

Define the neutron kinetics data

Delayed neutron fraction: $\beta := .007$

Prompt neutron lifetime: $l := 42 \cdot 10^{-6}$

Define the reactivity equation:

$$\rho(t) := \frac{100}{1+\frac{l}{t}} \cdot \left[\left(\frac{l}{\beta \cdot t} \right) + \sum_j \frac{f_j}{1+\lambda_j \cdot t} \right]$$

Six group neutron precursor data: $j := 0, 1 .. 5$

$$\lambda_j := \frac{\ln(2)}{thalf_j}$$

$f_j :=$	$thalf_j :=$
.038	54.5
.213	21.8
.188	6.00
.407	2.23
.128	.496
.026	.179

Enter the rod calibration data

Define the number of data points: $i := 0, 1 .. 15$

Enter control rod position data, initial time points and ending time period data:

$start_i :=$	$stop_i :=$	$t60sec_i :=$	$t90sec_i :=$	$t600min_i :=$	$t600sec_i :=$	$t900min_i :=$	$t900sec_i :=$
7.5	109.5	0	19.72	1	50.53	2	15.37
109.5	195	0	11.35	1	0.94	1	14.37
195	244	0	15.14	1	28.60	1	48.66
244	294	0	11.10	1	9.34	1	24.75
294	338.5	0	11.92	1	7.62	1	22.47
338.5	381.5	0	11.70	1	7.60	1	23.22
381.5	419.5	0	13.54	1	18.63	1	36.87
419.5	465	0	10.40	0	59.65	1	14.20
465	507	0	12.74	1	11.62	1	26.58
507	554.5	0	11.71	1	2.1	1	16.70
554.5	595.5	0	15.76	1	27.39	1	46.56
595.5	653	0	10.40	0	58.07	1	11.08
653	703	0	15.96	1	26.98	1	45.08
703	758	0	17.75	1	41.07	2	1.91
758	825	0	18.24	1	45.66	2	9.15
825	961	0	19.64	1	45.89	2	6.10

$dr_i := stop_i - start_i$

$\Delta t1 := (t600min \cdot 60 + t600sec - t60sec) \cdot \ln\left(\frac{600}{60}\right)^{-1}$

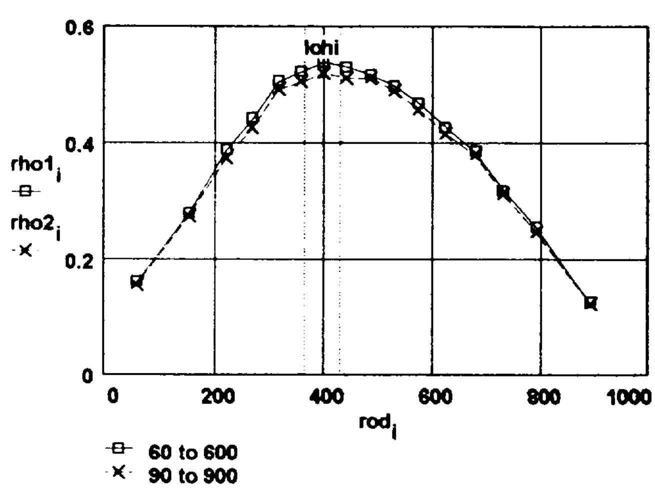
$rho1_i := \frac{\rho(\Delta t1_i)}{dr_i}$

$rod_i := 0.5 \cdot (start_i + stop_i)$

$\Delta t2 := (t900min \cdot 60 + t900sec - t90sec) \cdot \ln\left(\frac{600}{60}\right)^{-1}$

$rho2_i := \frac{\rho(\Delta t2_i)}{dr_i}$

The suspected effect of 1" Al plug in Transient rod: $\Delta x := 64$ $lo := 365$ $hi := lo + \Delta x$



$rho := rho2$

$N := \text{length}(rod)$

$N = 16$

$\sum_i \rho(\Delta t1_i) = 330.25$

$\sum_i \rho(\Delta t2_i) = 321.95$

rod_i	rho_i
58.5	0.16
152.25	0.27
219.5	0.37
269	0.43
316.25	0.49
360	0.5
400.5	0.52
442.25	0.51
486	0.51
530.75	0.49
575	0.46
624.25	0.42
678	0.38
730.5	0.31
791.5	0.24
893	0.12

WRITEPRN(rod) := rod

WRITEPRN(rho) := rho

7/24/98 [illegible]

CONTROL ROD CALIBRATION

NON-LINEAR Least squares curve fit 1998 TR no PNT

Define the data arrays x and y:

$x := READPRN(rod)$ $\quad N := length(x) \quad N = 16$

$y := READPRN(rho)$ $\quad i := 0, 1 .. length(x)$

x =	y =
58.5	0.16
152.3	0.27
219.5	0.37
269	0.43
316.3	0.49
360	0.5
400.5	0.52
442.3	0.51
486	0.51
530.8	0.49
575	0.46
624.3	0.42
678	0.38
730.5	0.31
791.5	0.24
893	0.12

Define the differential function:

$$F(x,\alpha,\beta,H) := \alpha\cdot\left[\sin\left[\pi\cdot\left(\frac{x-\beta}{H}\right)\right]\right]^2$$

Define the integral curve function:

$$Fi(\alpha,\beta,H,a,b) := \int_a^b F(q,\alpha,\beta,H)\,dq$$

Define the fitting constraint, sum of squares to be minimum:

$$i := 0 .. N-1 \qquad SSE(\alpha,\beta,H) := \sum_i \left(y_i - F(x_i,\alpha,\beta,H)\right)^2$$

Initial guess: $\alpha := .55 \qquad \beta := -150 \qquad H := 1200$

Given $\quad SSE(\alpha,\beta,H) = 0 \qquad 1 = 1 \qquad 2 = 2$

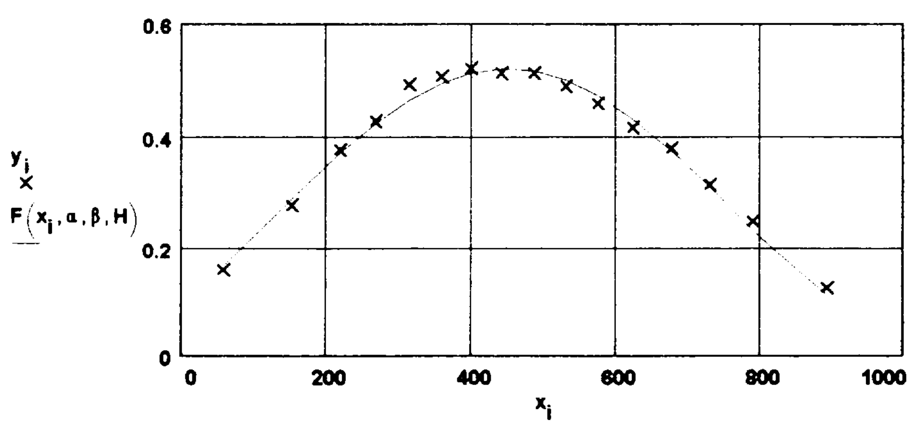
$$\begin{pmatrix}\alpha\\ \beta\\ H\end{pmatrix} := Minerr(\alpha,\beta,H) \qquad WRITEPRN(fit) := \begin{pmatrix}\alpha\\ \beta\\ H\end{pmatrix} \qquad err_i := y_i - F(x_i,\alpha,\beta,H)$$

Control Rod Parameters:

$\alpha = 0.521$

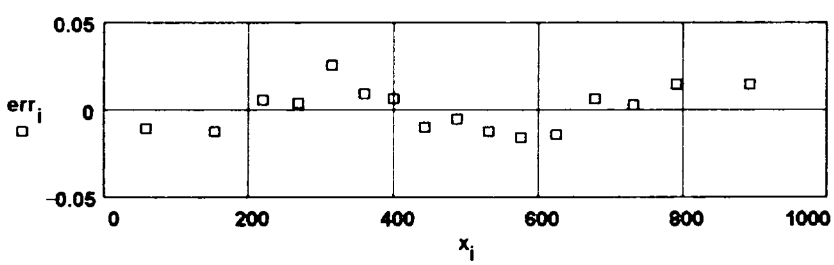
$\beta = -185.37$

$H = 1271.22$



Rod curve fit errors:

$$\frac{SSE(\alpha,\beta,H)}{(N-3)} = 1.81\cdot 10^{-4}$$



Calculate the rod total integral worth and end section worths:

$Fi(\alpha,\beta,H,\beta,H) = 337.33$

$Fi(\alpha,\beta,H,0,\beta) = -6.47$

$Fi(\alpha,\beta,H,960,H) = 8.54$

$$\int_\beta^H \alpha\cdot\left[\sin\left[\pi\cdot\left(\frac{x-\beta}{H}\right)\right]\right]^2 dx = 337.33$$

7/24/98 [illegible]

Transient rod 1998 no PNT

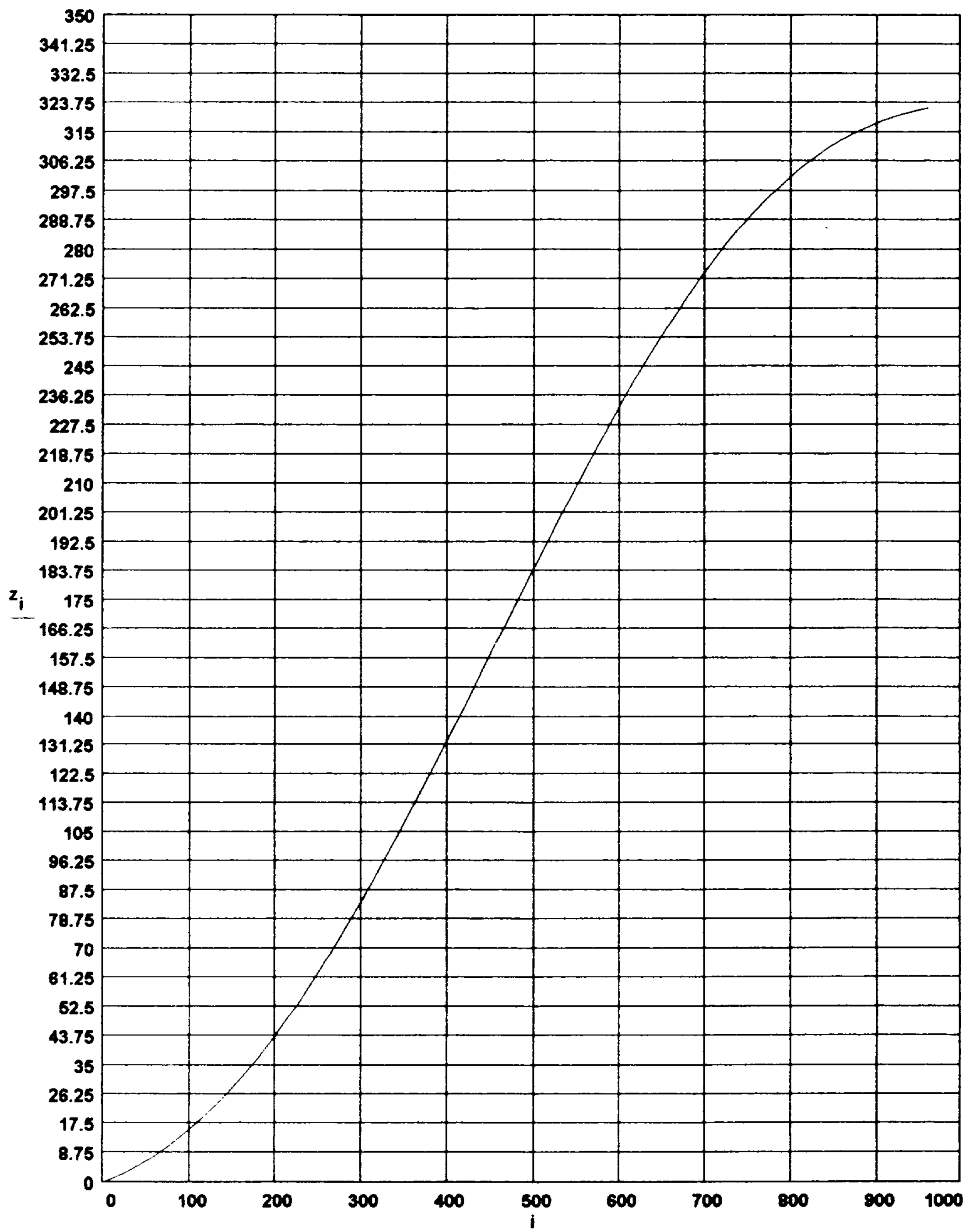
$i := 0, 6 .. 960$ $\qquad z_i := Fi(\alpha, \beta, H, 0, i)$

$j := 0, 2 .. 99$ $\qquad z000_j := Fi(\alpha, \beta, H, 0, j)$

$z100_j := Fi(\alpha, \beta, H, 0, 100 + j)$ $\qquad z200_j := Fi(\alpha, \beta, H, 0, 200 + j)$ $\qquad z300_j := Fi(\alpha, \beta, H, 0, 300 + j)$

$z400_j := Fi(\alpha, \beta, H, 0, 400 + j)$ $\qquad z500_j := Fi(\alpha, \beta, H, 0, 500 + j)$ $\qquad z600_j := Fi(\alpha, \beta, H, 0, 600 + j)$

$z700_j := Fi(\alpha, \beta, H, 0, 700 + j)$ $\qquad z800_j := Fi(\alpha, \beta, H, 0, 800 + j)$ $\qquad z900_j := Fi(\alpha, \beta, H, 0, 900 + j)$



Revised 7/24/98
M. [illegible]

Transient rod 1998 no PNT

Calculate the rod integral worth:

$$\int_{0}^{960} \alpha \cdot \left[\sin\left[\pi \cdot \left(\frac{x - \beta}{H} \right) \right] \right]^2 dx = 322.31$$

$Fi(\alpha, \beta, H, 0, 960) = 322.31$

j	$z000_j$	$z100_j$	$z200_j$	$z300_j$	$z400_j$	$z500_j$	$z600_j$	$z700_j$	$z800_j$	$z900_j$
0	0	15.82	44.08	84.25	132.94	184.73	233.45	273.67	301.99	317.86
2	0.21	16.26	44.77	85.15	133.96	185.75	234.35	274.36	302.43	318.07
4	0.42	16.7	45.47	86.06	134.99	186.77	235.25	275.04	302.86	318.26
6	0.63	17.15	46.17	86.97	136.02	187.8	236.15	275.72	303.28	318.46
8	0.85	17.61	46.88	87.89	137.05	188.82	237.04	276.4	303.7	318.65
10	1.07	18.07	47.59	88.81	138.08	189.84	237.93	277.07	304.12	318.84
12	1.3	18.53	48.31	89.73	139.11	190.85	238.81	277.73	304.53	319.02
14	1.53	19	49.03	90.66	140.14	191.87	239.69	278.39	304.94	319.2
16	1.76	19.48	49.76	91.59	141.17	192.89	240.57	279.05	305.34	319.37
18	2	19.96	50.49	92.52	142.21	193.9	241.45	279.7	305.73	319.54
20	2.25	20.45	51.23	93.45	143.24	194.91	242.32	280.34	306.12	319.71
22	2.49	20.94	51.97	94.39	144.28	195.92	243.18	280.99	306.51	319.87
24	2.75	21.43	52.72	95.33	145.32	196.93	244.05	281.62	306.89	320.03
26	3	21.94	53.47	96.28	146.35	197.93	244.9	282.25	307.27	320.19
28	3.26	22.44	54.22	97.23	147.39	198.94	245.76	282.88	307.64	320.34
30	3.53	22.95	54.98	98.18	148.43	199.94	246.61	283.5	308	320.49
32	3.8	23.47	55.75	99.13	149.47	200.94	247.46	284.11	308.36	320.64
34	4.08	23.99	56.51	100.09	150.51	201.94	248.3	284.72	308.72	320.78
36	4.35	24.52	57.29	101.04	151.55	202.93	249.14	285.33	309.07	320.92
38	4.64	25.05	58.07	102.01	152.59	203.93	249.97	285.93	309.42	321.05
40	4.93	25.59	58.85	102.97	153.63	204.92	250.8	286.52	309.76	321.18
42	5.22	26.13	59.63	103.94	154.67	205.91	251.63	287.11	310.1	321.31
44	5.52	26.68	60.43	104.91	155.71	206.9	252.45	287.7	310.43	321.43
46	5.82	27.23	61.22	105.88	156.75	207.88	253.27	288.28	310.76	321.55
48	6.13	27.79	62.02	106.85	157.79	208.86	254.08	288.85	311.08	321.67
50	6.44	28.35	62.82	107.83	158.83	209.84	254.89	289.42	311.4	321.79
52	6.76	28.92	63.63	108.81	159.87	210.82	255.7	289.99	311.71	321.9
54	7.08	29.5	64.45	109.79	160.91	211.8	256.5	290.55	312.02	322.01
56	7.4	30.07	65.26	110.78	161.95	212.77	257.29	291.1	312.32	322.11
58	7.73	30.66	66.08	111.76	162.99	213.74	258.09	291.65	312.62	322.21
60	8.07	31.25	66.91	112.75	164.03	214.71	258.87	292.19	312.91	322.31
62	8.41	31.84	67.74	113.74	165.07	215.67	259.66	292.73	313.2	322.41
64	8.76	32.44	68.57	114.74	166.11	216.64	260.44	293.26	313.49	322.5
66	9.11	33.04	69.41	115.73	167.15	217.59	261.21	293.79	313.77	322.59
68	9.46	33.65	70.25	116.73	168.19	218.55	261.98	294.32	314.05	322.68
70	9.82	34.27	71.1	117.73	169.23	219.51	262.75	294.83	314.32	322.76
72	10.19	34.88	71.95	118.73	170.27	220.46	263.51	295.35	314.58	322.85
74	10.56	35.51	72.8	119.74	171.31	221.4	264.26	295.85	314.85	322.92
76	10.93	36.14	73.66	120.74	172.35	222.35	265.01	296.36	315.11	323
78	11.31	36.77	74.52	121.75	173.38	223.29	265.76	296.85	315.36	323.07
80	11.7	37.41	75.39	122.76	174.42	224.23	266.5	297.35	315.61	323.15
82	12.09	38.06	76.26	123.77	175.45	225.17	267.24	297.83	315.85	323.21
84	12.48	38.71	77.13	124.78	176.49	226.1	267.97	298.32	316.09	323.28
86	12.88	39.36	78.01	125.8	177.52	227.03	268.7	298.79	316.33	323.34
88	13.28	40.02	78.89	126.81	178.55	227.96	269.43	299.27	316.56	323.4
90	13.69	40.68	79.77	127.83	179.58	228.88	270.15	299.73	316.79	323.46
92	14.11	41.35	80.66	128.85	180.62	229.8	270.86	300.19	317.01	323.52
94	14.53	42.03	81.55	129.87	181.64	230.72	271.57	300.65	317.23	323.57
96	14.95	42.71	82.44	130.89	182.67	231.63	272.27	301.1	317.45	323.62
98	15.38	43.39	83.34	131.91	183.7	232.54	272.97	301.55	317.66	323.67

UT-TRIGA Pulse Experiment with TCNS Wave Guide
Four Pulses at 50 watts with ~$1.92 of reactivity
Reactivity measurement from rod drive curves
Data taken 7/7/1998

Pulse No.	1	2	3	4	average	stdev	percent
Peak Temperature	267	265	264	265	265.25	1.26	0.47
Peak Power	305	303	303	304	303.75	0.96	0.32
Total Energy	9.083	9.11	9.16	9.25	9.15	0.07	0.80
Width at Half Power	26.5	26.7	26.7	26.799	26.67	0.13	0.47
Minimum Period	7.794	7.852	7.852	7.882	7.85	0.04	0.47
Reactivity	1.814	1.808	1.808	1.805	1.81	0.00	0.21

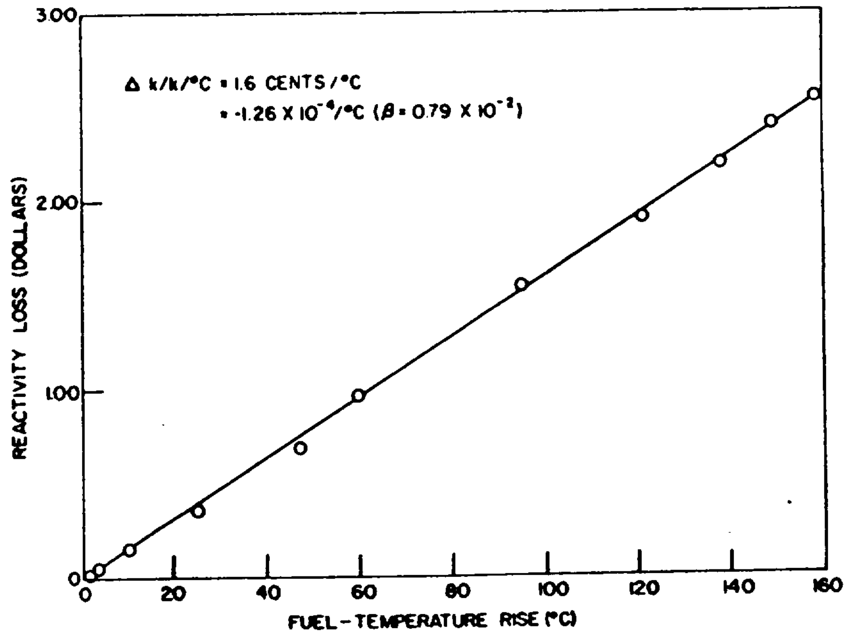


Fig. 4--Reactivity loss as a function of fuel temperature rise. Average fuel temperature is measured with respect to average core water temperature

TRANSIENT EXPERIMENTS

Prediction of Transient Behavior

The transient behavior of this reactor system can be calculated by coupling the neutron kinetic equations with a suitable thermal model for the reactor. The coupling of the neutronic system and the thermal system is done by means of the temperature coefficients measured in the quasi-equilibrium experiments. The thermal resistances for heat transfer from the fuel to the core cooling water and from the cooling water to the bulk water are calculated from the temperature curves obtained in the quasi-equilibrium experiments and from a knowledge of the fuel and core-water heat capacities.

Given these thermal parameters and the associated reactivity coefficients, it is then possible to calculate the reactor power and the reactor component temperatures as functions of time for an arbitrary reactivity

insertion. These calculations were made on the IBM-704 computer. The results were compared with the experiments and are shown as dashed lines in Figs. 5 and 6.

It should be noted that in the original calculations a theoretically estimated value of the heat capacity of the fuel elements was used. Actually, the experimental results on the fast transients provide an excellent measure of this heat capacity. Calculations performed after the experiments were completed utilize both an experimentally determined heat capacity and prompt generation time.

Experimental Procedure

For each transient, the reactor was taken to critical and the power held constant at 1 w. The control-rod configuration was adjusted so that after the transient rod was pneumatically driven out of the core, the system reactivity would be above delayed critical by the desired amount. Each succeeding reactivity insertion was chosen to decrease the reactor period by approximately a factor of 2 or to increase the fuel temperature in increments of approximately 50°C.

In each case, the transient rod was fully withdrawn from the core in less than 0.1 sec. A high-speed 36-channel galvanometer recorder was used to measure transient fuel-temperature distributions, water temperatures and pressures, and reactor power level. The core was photographed with a high-speed motion-picture camera during each transient.

After each reactivity insertion, the reactor power level was observed to increase rapidly to a peak value and then to decrease to the quasi-equilibrium value monotonically. There was no evidence of any system instability or disturbance of the water surface.

Experimental Results

A summary of the results obtained from this series of experiments is given in Table 1.

Experimental power and maximum-fuel-temperature curves for the 2-dollar step reactivity insertion are shown in Fig. 7.

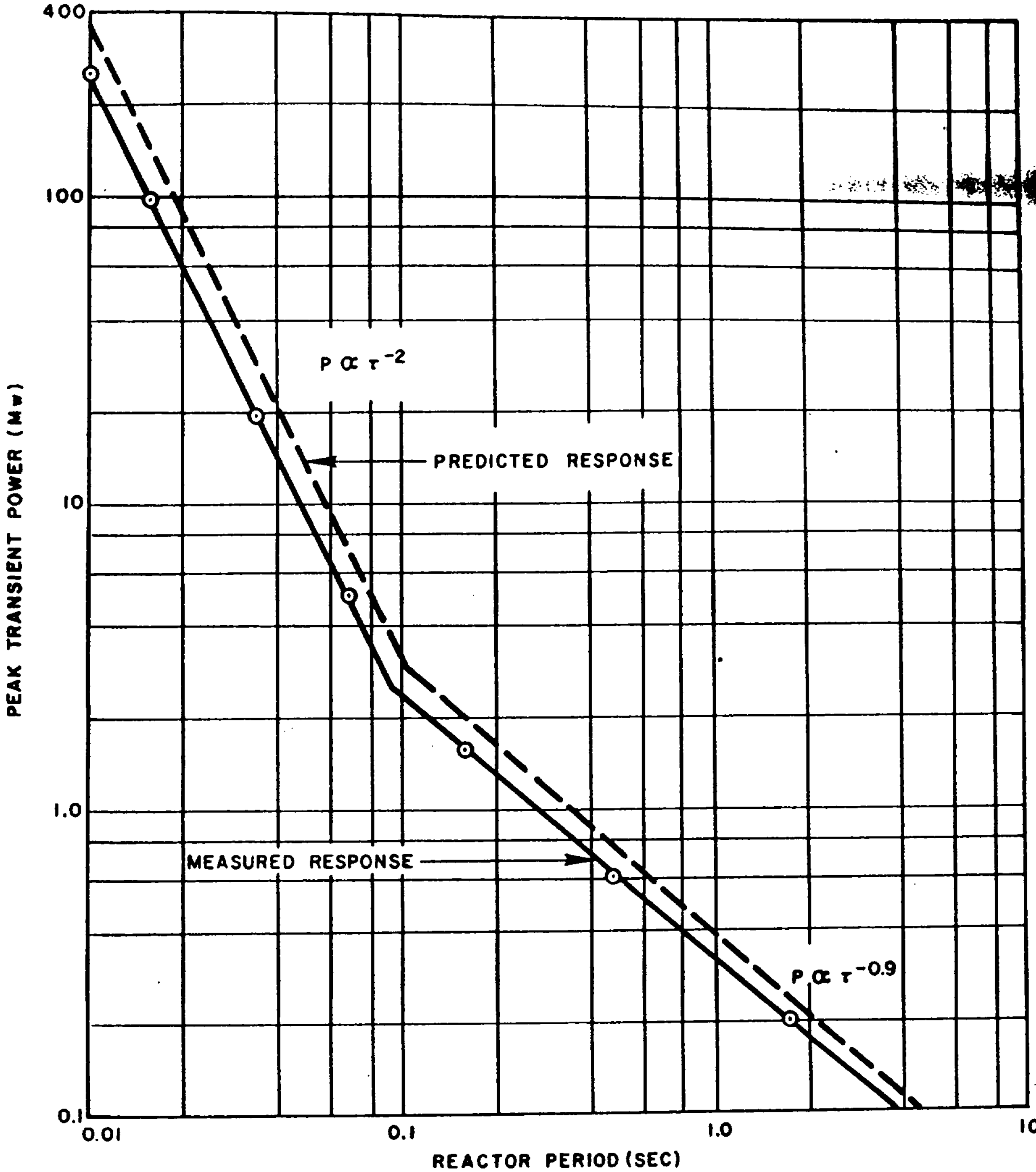


Fig. 5--TRIGA transient power response as a function of reactor period

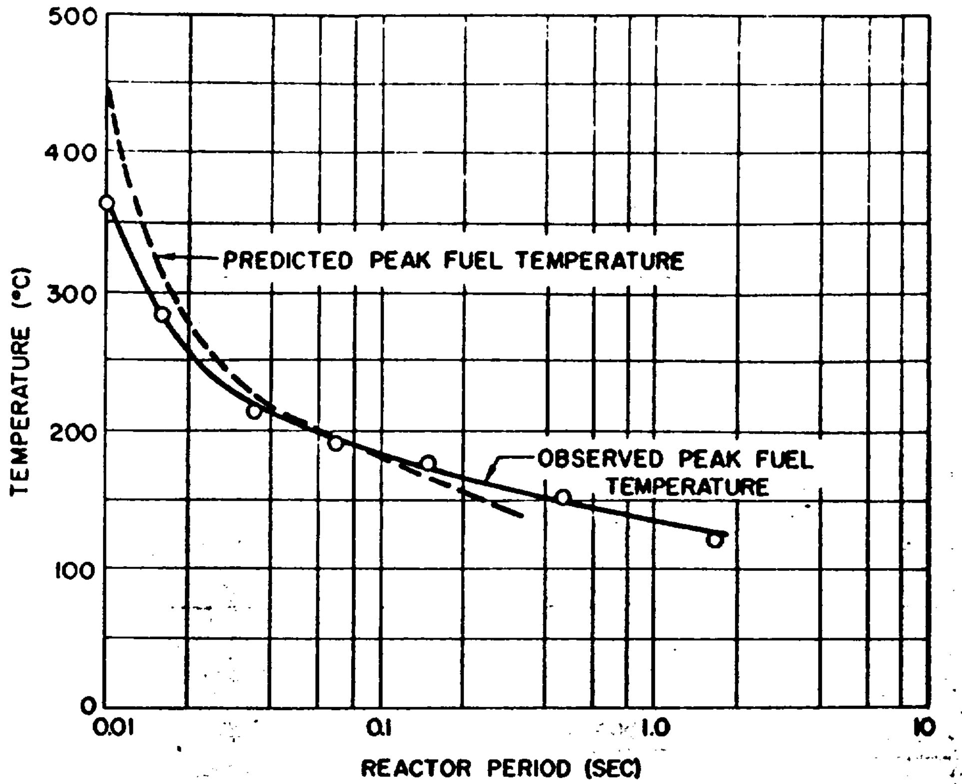


Fig. 6--TRIGA transient temperature response as a function of reactor period

Table 1

SUMMARY OF RESULTS OF TRANSIENT EXPERIMENTS

Reactivity Insertion (dollars)	Reactor Period (sec)	Peak Power (Mw)	Maximum Fuel Temperature (°C)
0.74	1.74	0.2	122
0.91	0.47	0.58	153
1.06	0.15	1.6	177
1.16	0.069	5	193
1.30	0.034	20	222
1.69	0.013	100	285
2.00	0.0098	250	360