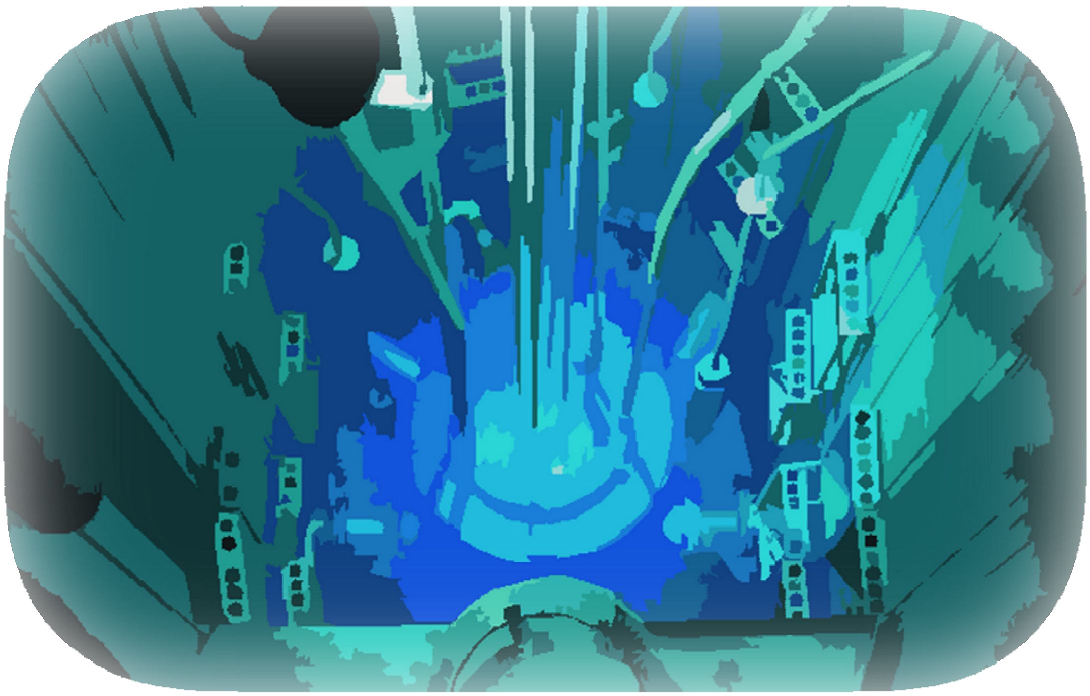
USNRC RESEARCH AND TEST REACTOR
HANDS-ON OPERATIONS TRAINING COURSE

INSTRUCTOR MANUAL (R1)



NRC-03-10-073

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USRNC REACTOR BASED TRAINING PROGRAM

TABLE OF CONTENTS

EXERCISES

REFERENCE MATERIAL

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USNRC HANDS-ON OPERATIONS TRAINING COURSE SCHEDULE & DESCRIPTION

The schedule for exercises planned in the *USNRC Hands-on Operations Training Course* over for January 24 through January 27, 2011 is provided. A brief description of the exercises follows the schedule. Exercises are designed to require approximately ½ day periods; adjustments may be possible if exercises are completed outside of the anticipated schedule.

SCHEDULE

	(Mon) Mar 14, 2011	(Tue) Mar 15, 2011	(Wed) Mar 16, 2011	(Thu) Mar 17, 2011
Morning	INTRODUCTION: Arrival, course overview, safety & security briefing EXERCISE 1: NETL Reactor and Facility Design, Walkthrough, Pulse	EXERCISE 3: Subcritical Multiplication and 1/M Approach to Critical	EXERCISE 5: Fuel Element Worth versus Position	EXERCISE 7: Reactor Pulsing and Square Wave Pulsing EXERCISE 8: Experiment Operations
Break				
Afternoon	EXERCISE 2: Reactor Startup	EXERCISE 4: Control Rod Worth and Calibration	EXERCISE 6: Power Level Instrument Calibration	EXERCISE 8: Experiment Operations CLOSEOUT: Summary, exit survey

DESCRIPTION

INTRODUCTION: Arrival, course overview, safety & security briefing

The UT Austin TRIGA Security Plan establishes a system of color coded badges to assist identification of authorized building occupants. In addition to a badge, entry into radiological areas requires personal monitoring.

EXERCISE 1: NETL Reactor and Facility Design, Walkthrough, Pulse

Based on significant RTR operating experience, significant facility systems and equipment have been identified; familiarization with material in this exercise will ensure walkthroughs are more effective in communicating concepts and issue in the context of the UT TRIGA reactor facilities. During the facility walkthrough, participants will inspect systems and equipment identified in the facility design review. Technical Specifications requirements will be emphasized. The exercise will conclude with observation of a reactor pulse.

EXERCISE 2: Reactor Startup

This training program provides experience to aid an intuitive grasp of reactor behavior. This exercise will introduce the participants to reactor controls as a basis for focusing on reactor response.

EXERCISE 3: Subcritical Multiplication and 1/M Approach to Critical

Achieving a critical condition following major changes in the reactor configuration is generally performed using a 1/M process. This exercise will introduce the participants to the use of subcritical multiplication in guiding an approach to critical.

EXERCISE 4: Control Rod Worth and Calibration

Technical Specifications requires knowledge and control of reactor reactivity parameters. Evaluating changes in reactivity parameters requires baseline and calibration of the control rods. This exercise will demonstrate two commonly used methods of control rod reactivity worth determination.

EXERCISE 5: Fuel Element Worth versus Position

Spatial distribution of the neutron flux across the reactor core determines a material's reactivity value. This exercise will demonstrate spatial dependence of the reactivity on fuel.

EXERCISE 6: Power Level Instrument Calibration

Technical Specifications requires knowledge and control of reactor thermal power. Neutron flux is the leading parameter in the process that results in thermal power from fissions. Power level instruments measure neutron flux, and this exercise will demonstrate calibration of neutron detectors as reactor power level instrumentation.

EXERCISE 7: Reactor Pulsing and Square Wave Pulsing

Temperature effects in TRIGA hydride fuel have a significant impact on reactor response. This exercise will demonstrate changes in power and temperature as a function of prompt reactivity.

EXERCISE 8: Experiment Operations

The conduct of research reactor experiments integrates Technical Specification requirements for administrative procedures and review and approval processes in planning and design with operating procedures and radiological controls. This exercise will demonstrate experiment proposal, review, insertion, and removal.

1. PURPOSE AND DISCUSSION

   The UT Austin TRIGA Security Plan establishes a system of color coded badges to assist identification of authorized building occupants. In addition to a badge, entry into radiological areas requires personal monitoring.

   Certain facility systems and equipment have been identified as significant for training purposes; familiarization with material in this exercise will ensure walkthroughs are more effective in communicating concepts and issue in the context of the UT TRIGA reactor facilities.

   During the facility walkthrough, participants will inspect systems and equipment identified in the facility design review. The exercise will conclude with observation of a reactor pulse.

2. EXERCISE OBJECTIVES

   Following completion of this exercise, participants should be able to

   2.1 Locate normal entrances, egresses and common areas

   2.2 Respond appropriately to an emergency

   2.3 Employ proper radiological precautions for entering radiological areas and/or handling radioactive materials

   2.4 Comply with security requirements related to badging

   2.5 Comply with radiological monitoring requirements

   2.6 Explain for the following systems:

   - Reactor pool and cooling systems
   - Control rod system
   - Confinement system
   - Facility radiation monitoring systems
   - Power level instrument systems
   - Process instrument systems & controls
   - Experiment facilities
   - Facility sumps & drains

   A. Purpose of the systems

   B. Safety functions associated with the system

   C. Technical Specifications related to the systems

Comment [#1]: Review map locations

Comment [#2]: Brief overview of EPan, map assembly areas, EOC, alternate EOC

Comment [#3]: Brief overview of RPP, map radiological areas, material storage areas, rad-labs

Comment [#4]: Security briefing: overview, building access, door controls

Comment [#5]: QUESTIONS:
(1) What rooms require personal monitoring devices
(2)When is contamination monitoring required

D. Approximate dimensions and flow paths of key system components

E. The purpose of system control actions, setpoints and alarms

2.7 Observe and read process instruments, and correlate instrument readings with proper operation of systems and equipment

2.8 Explain the properties of ZrH that provide inherent, passive safety

3. PREREQUISITES

3.1 Completion of safety briefing prior to entry of restricted areas

4. BACKGROUND/REFERENCES

4.1 University of Texas Safety Analysis Report

4.2 NUREG 1135, Safety Evaluation report related to the Construction Permit and Operating License for the Research reactor at the University of Texas (Docket No. 50-602)

4.3 NETL Procedures (ADMN, FUEL, HP, and OPER series)

4.4 NETL Security Plan

4.3 Attachment 1.1, NETL Reactor and Facility Design

4.4 Attachment 1.2, UT TRIGA Reactivity Summary

4.5 Attachment 1.3, Exercise 1 Review

5. INSTRUCTIONS

5.1 CHECKOUT security badge

5.2 CHECKOUT dosimetry

5.3 PARTICIPATE in safety briefing

5.4 REVIEW ***NETL Reactor and Facility Design*** (ATTACHED)

Comment [#6]: Discuss & review attachment

5.5 PARTICPATE in a tour

A. NETL laboratory area

B. Equipment and machine rooms

C. Reactor bay

D. Pool cooling room

E. Auxiliary room

F. Control room

G. HVAC rooms

5.6 OBSERVE a reactor pulse (pool side)

6. REINFORCEMENT & REVIEW

Attachment 1.3, Exercise review

6.1 Using the letter labels for components on the pool cooling and the pool cleanup schematics, label equipment in the photographs or draw lines connecting the labeled components to the equipment in the photographs

6.2 Draw lines connecting the labels to the core components in the photograph, and

6.3 Indicate which beam port supports the experiment described experiment facilities

6.4 Write the Technical Specifications value for reactivity limit described in the table

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1.0 Background

Research and test reactor significant operating experience was analyzed (Fig 1.1). The relative contribution of systems and equipment to the experiences include (ordered by contribution to occurrences):

- Reactor pool
- Control elements
- Shipping (of radioactive materials)
- Miscellaneous
- Confinement (or containment)
- Radiation Instruments & Controls
- Power Level Monitoring Instruments & Controls
- Process Systems Instruments & Controls

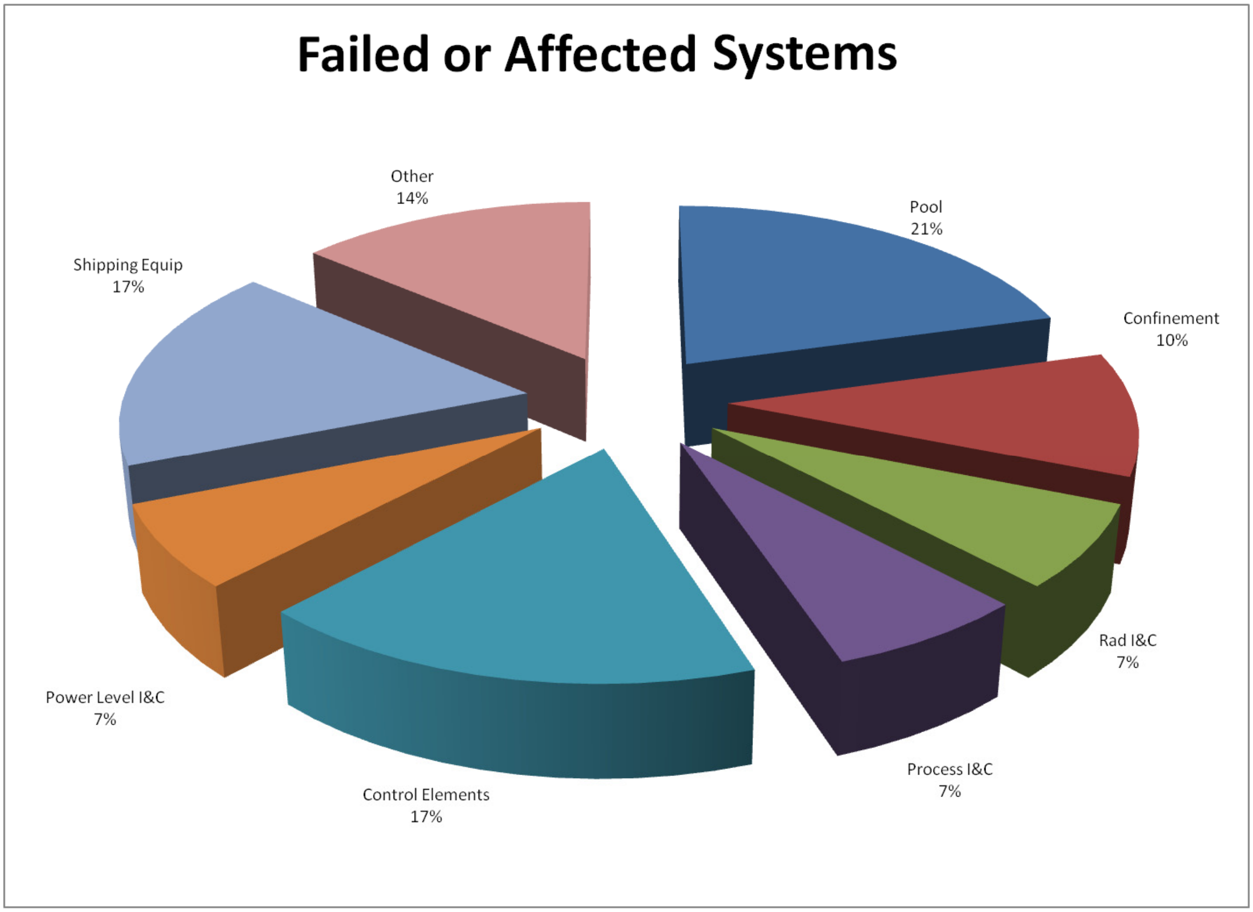


Figure 1: Contribution of Equipment or System Failures to Operating Experience

USNRC Key Topics (10/20/2010) indicate current issues relevant to the NETL facility including:

- Contaminated water releases
- Facility security
- Reactor experiments
- Radiation protection
- Safety culture

Based on this review, the following facility characteristics will be addressed and (where possible) observed in a facility walkthrough:

1.1.1 Reactor Pool and cooling

1.1.2 Control Rod System

1.1.3 Confinement

1.1.4 Facility Radiation Monitoring

1.1.5 Power level instruments

1.1.6 Process System Instruments and controls

1.1.7 Experiment Facilities

1.1.8 Facility Sumps & Drains

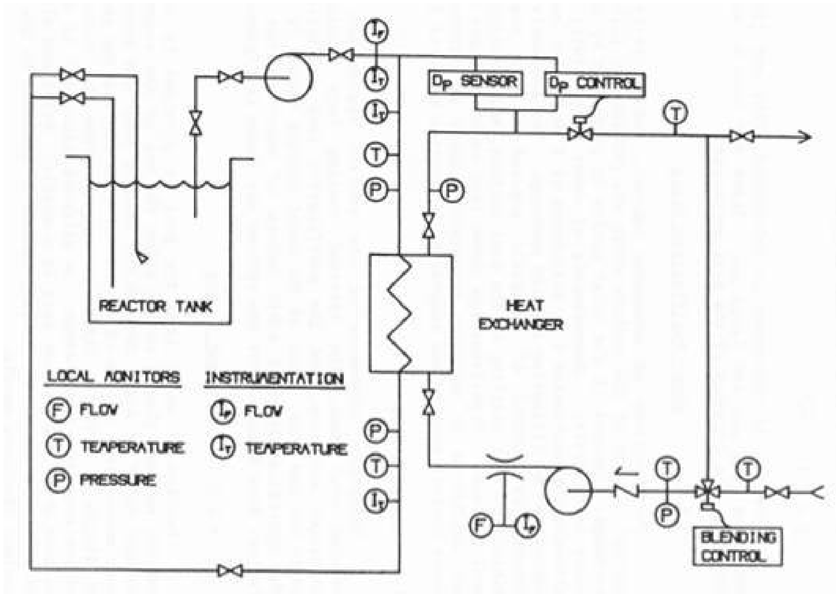
1.1 Outline

1.1.1 Reactor Pool and Cooling

Pool cooling function:

a. Reactor heat removal

b. Vertical shielding of radiation from the reactor and allow access to the reactor core



REACTOR COOLANT SYSTEM DESIGN SUMMARY

Reactor Tank			
Material	Al plate (6061)		
Thickness	1/4 in.. (0.635 cm)		
Volume (maximum)	11000 gal. (41.64 m^3)		
Coolant Lines			
Pipe	Aluminum (6061)		
Valves	Iron-Plastic liner, 316 ss. Ball and Stem		
Fittings	Aluminum (Victaulic)		
Coolant Pump			
Type	Centrifugal		
Material	Stainless steel		
Capacity	250 gpm (15.8 liter/sec)		
Heat Exchanger			
Type	Shell and tube		
Materials: shell	Carbon steel		
tubes	304 stainless steel		
Heat Duty	1000 kW		
Flowrate: tubes	250 gpm (15.8 liters/sec)		
shell	400 gpm (25.2 liters/sec)		
Typical Parameters:			
Tube inlet	100°F	42	psia
Tube outlet	69°F	27	psia
Shell inlet	48°F	55	psia
Shell outlet	67°F	48	psia

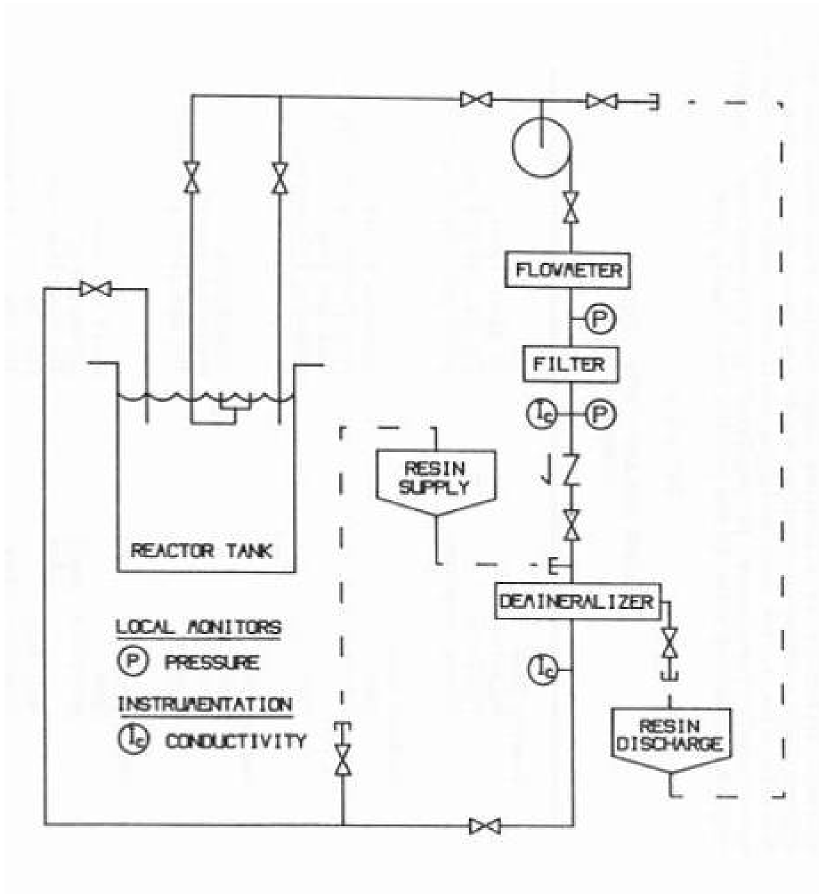
Purification:

a Reactor component (particularly fuel elements) corrosion control

b. Radioactive contamination control

c. Water (optical) clarity

d. Makeup water interface



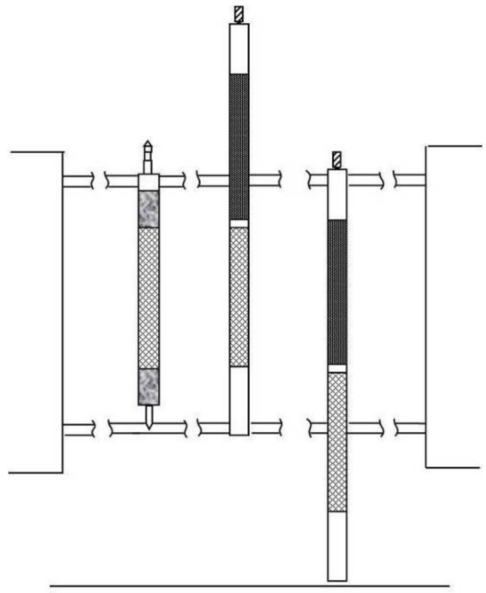
SIGNIFICANT SHIELDING & POOL LEVEL VLAUES		
Parameter of Interest	Level (meters)	Notes
CONCRETE PAD	-2.191	SAR FIG 7-1: Concrete below pool; SAR 7.2.1 indicates 1.22 meter pad; Drawing RS-6, 5 ft min
FLOOR	0	Pool floor ½” aluminum; pool wall ¼” aluminum
TOP OF SAFETY PLATE	0.197	SAR FIG 4-25, 4.4.4: ½” th. 16” <grid plate, 0.775 m below CL (7.75” above floor)
TOP OF LOWER GRID PLATE	0.642	SAR FIG 4-25, SAR 4.4.3: 1 ¼” thick, 0.332 m below CL
BOTTOM OF FUEL	0.781	SAR FIG 4-25: ½ fuel length, 7.5 in./0.191 below CL
BEAM PORT CL	**0.902**	SAR FIGS 4-25/7-1, 8.1.4.3: 6 in. diameter (0.191 m) below fuel CL
CORE CL	**0.972**	SAR FIGS 4-25/7-1
TOP OF FUEL	1.162	SAR FIG 4-25: ½ fuel length , 7.5 in. (0.191 m) above fuel CL
GRID PLATE	1.295	SAR 4.4.3: ⅝ in. thick (0.324 m) over CL

SIGNIFICANT SHIELDING & POOL LEVEL VLAUES		
Parameter of Interest	Level (meters)	Notes
MAIN LOWER SHIELDING	**4.267**	SAR FIG 7-1: 7.97 feet thick around pool
TRANSITIONAL CONCRETE STEP	**4.877**	SAR FIG 7-1: 3 feet thick around pool
SHIFT TO HIGH DENSITY CONCRETE	**5.944**	SAR FIG 7-1: 2.3 g/cc to 2.9 g/cc
MIN POOL LEVEL (TS)	**6.5**	TS 3.3, A.3.3.b: 5.25 m over the core
VACUUM BREAKERS	**6.7**	OPER-4 II.A.2.b
LOW POOL LEVEL SCARM	**7.8**	SCRAM MAIN-1 ATT 1
LOW POOL LEVEL	**8.05**	Minimum, fill required OPER-4 II.A.2.a
LOW POOL LEVEL ALARM	**8.07**	Alarm MAIN-1 ATT 1
NORMAL POOL LEVEL	**8.1**	NOMINAL OPER-4 II.A.2.a
HIGH POOL LEVEL	**8.15**	Maximum OPER-4 II.A.2.a
HIGH POOOL LEVEL ALARM	**8.17**	Alarm MAIN-1 ATT 1
TOP OF TOP DECK (NOT LEDGE)	8.534	SAR FIG 7-1

1.1.2 Reactor Core & Control Rods

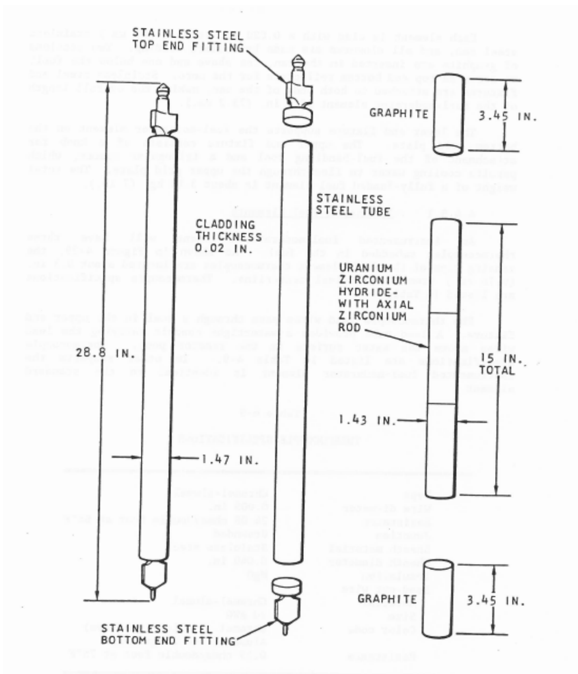
TYPICAL TRIGA CORE NUCLEAR PARAMETERS

Fuel elements	SS-clad U-$ZrH_{1.6}$
Cold clean critical loading	~64 elements
	~2.5 kg U-235
Operational loading	~90 elements
	~3.4 kg U-235
ℓ, Prompt neutron lifetime	41 μsec
β, Effective delayed neutron fraction	0.0070
α, Prompt negative temperature coefficient	~1.0 x 10^{-4} $\delta k/k$°C
T_f Average fuel temperature (1.1 MW)	265°C
T_w Average water temperature (1.1 MW)	65°C
Water coolant volume to cell volume ratio	~1/3

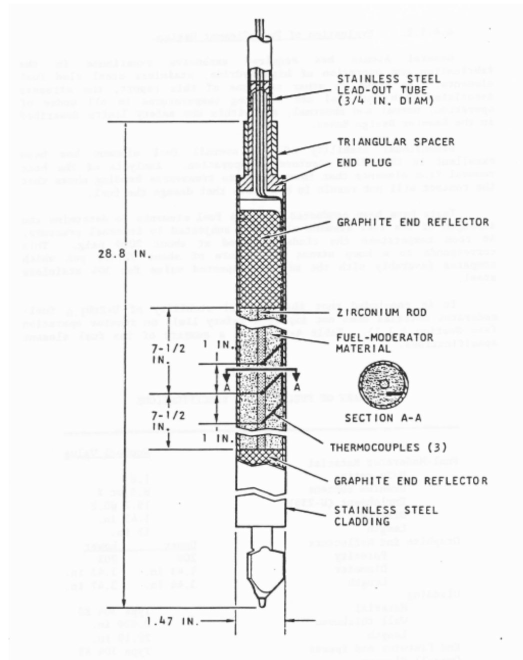


SUMMARY OF FUEL ELEMENT SPECIFICATIONS

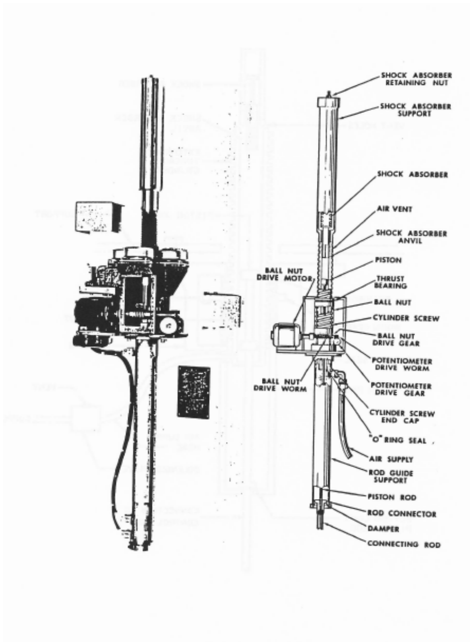
		Nominal Value
Fuel-Moderator Material		
H/Zr ratio		1.6
Uranium content		8.5 wt %
Enrichment (U-235)		19.7 #0.2
Diameter		1.43 in.
Length		15 in.
Graphite End Reflectors	Upper	Lower
Porosity	20%	20%
Diameter	1.43 in.	1.43 in.
Length	3.44 in.	3.47 in.
Cladding		
Material		Type 304 SS
Wall thickness		0.020 in.
Length		22.10 in.
End Fixtures and Spacer		Type 304 SS
Overall Element		
Outside diameter	1.47 in.	(3.73 cm)
Length	28.37 in.	(72.06 cm)
Weight	7 lb.	(3.18 kg)



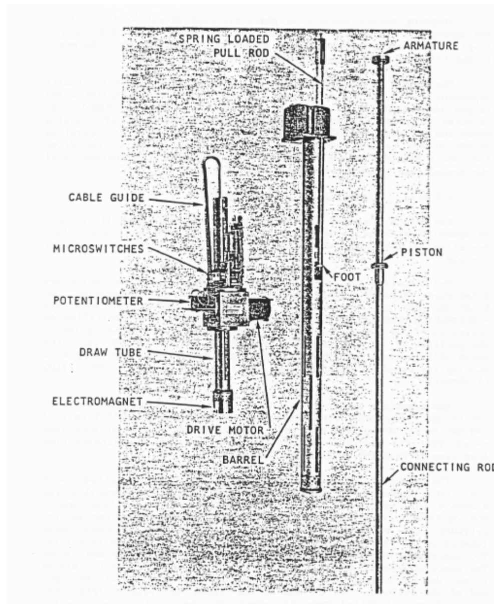
TRIGA STAINLESS STEEL CLAD FUEL ELEMENT WITH END FITTINGS



CONTROL ROD SPECIFICATIONS		
	Transient	Shim & Regulating
Cladding		
Material	Al	Type 304 SS
OD	1.25 in. (3.18 cm)	1.35 in. (3.34 cm)
Length	36.75 in. (93.35 cm)	43.13 in. (109.5 cm)
Wall Thickness	0.028 in. (0.071 cm)	0.020 in. (0.051 cm)
Absorber		
Material	Boron Carbide (solid form)	Boron Carbide (solid form)
OD	1.19 in. (3.02 cm)	1.31 in. (3.32 cm)
Length	15 in. (38.1 cm)	14.25 in. (36.20 cm)
Follower		
Material	Air	Fuel
OD	1.25 in. (3.18 cm)	1.31 in. (3.34 cm)
Length	20.88 in. (53.02 cm)	15 in. (38.1 cm)



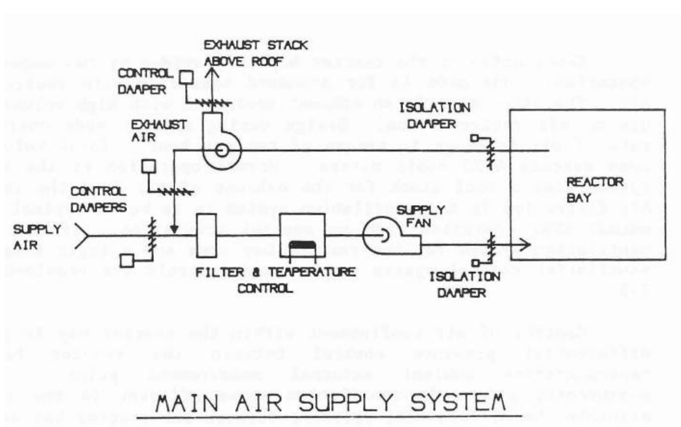
ADJUSTABLE TRANSIENT ROD

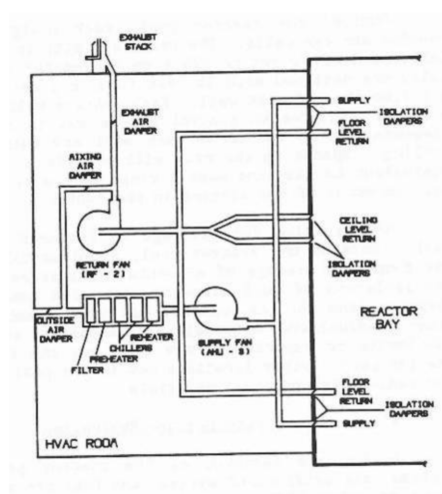


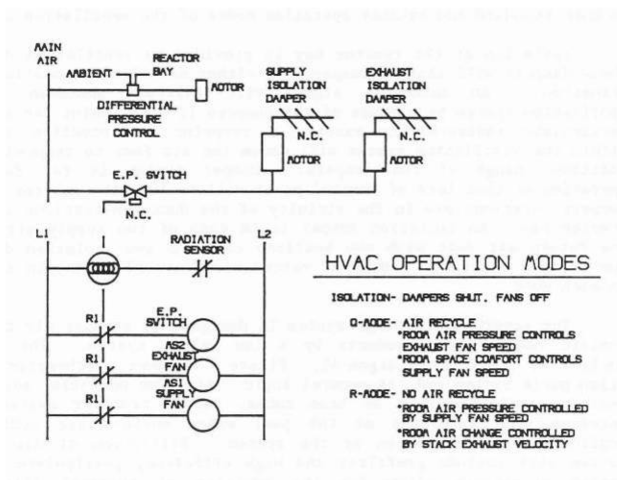
RACK AND PINION CONTROL ROD DRIVE

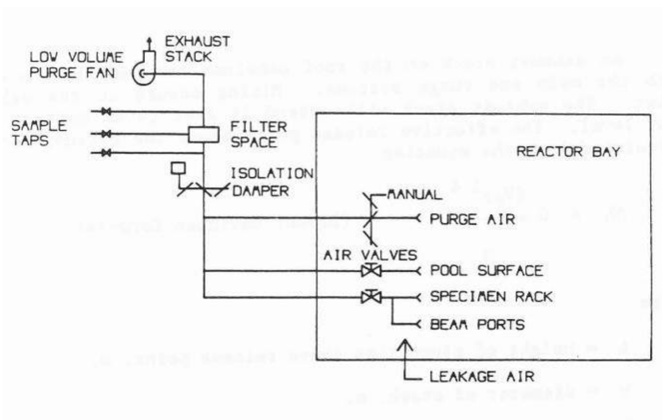
1.1.3 Confinement System

SAR 7.2.2, The ventilation system is designed to maintain a negative Pressure within the reactor bay with respect to the building exterior and other building areas. Confinement and isolation is achieved by air control dampers and leakage prevention material at doors and other room penetration points. A separate system is designed to exhaust air from several locations within the reactor that could contain airborne radionuclides such as argon-41. Manual operation of start/stop controls of both main and purge air systems will be available in the reactor control room.

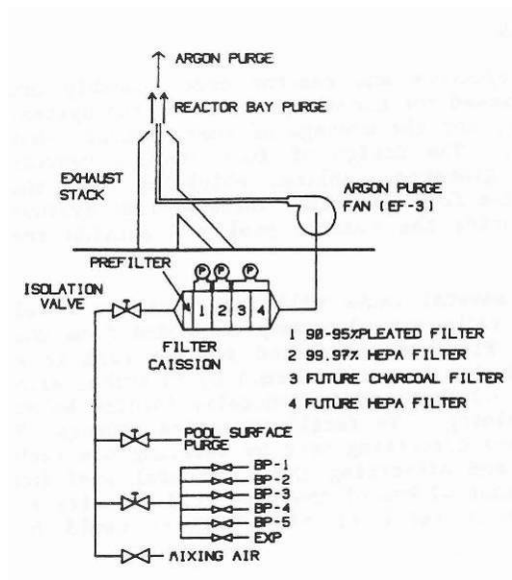


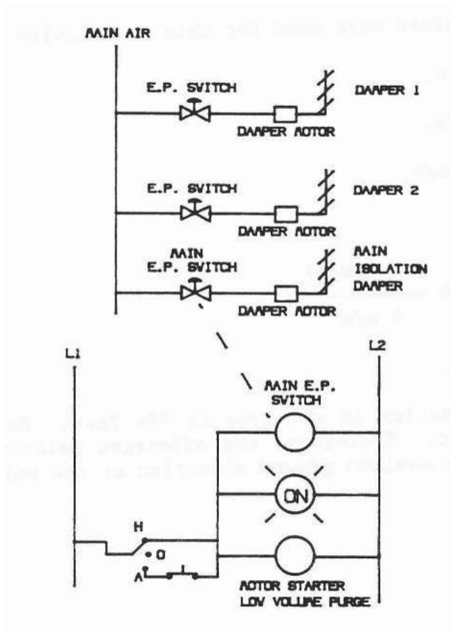






AUXILIARY AIR EXHAUST SYSTEM





Velocity	
3900 fpm	Argon purge flow velocity
1800 fpm	HVAC flow velocity
Flow Rate	
7200 cfm	HVAC flow
1100 cfm	Argon purge flow

1.1.4 Facility Radiation Monitoring

Safety Analysis Report, 9.2. RADIATION MONITORING:

Radiation monitoring shall consist of fixed, portable, or sampling type systems. Monitoring systems will be applied to measurement of radiation areas and high radiation areas around the reactor facility, significant contamination within and adjacent to the reactor facility, and radioactive materials and their concentrations in effluents.

Continuous monitoring or control of radiation fields in the restricted area around the reactor shall occur whenever levels greater than 100 mrem/hr are produced in accessible areas.

Airborne radioactive monitoring shall consist of continuous sampling of air particulate activity in the reactor area. Warning levels and action levels will be determined relative to allowable maximum permissible concentrations. Measurements should be sensitive to one maximum permissible concentration change in one hour.

Effluent monitoring shall be provided for the discharge of the radioactive noble gas argon-41. Monitoring will consist of either the use of integrating dosimeters at a location of interest or sampling of a point in the release path. Measurements shall determine that the dose at a location of interest is either less than ten mrem per year above natural background or two percent of the allowable maximum permissible concentration for the year.

Fixed area gamma monitors shall have remote readouts with audible and visual alarms at the reactor control console. Local readouts should be provided in areas with significant radiation levels and routine personnel access. A multiple channel area monitoring system with GM type detector probes will be installed. The system should have at least four channels functional. Measurement should include the dose range of 1 millirem/hour to 1 rem/hour.

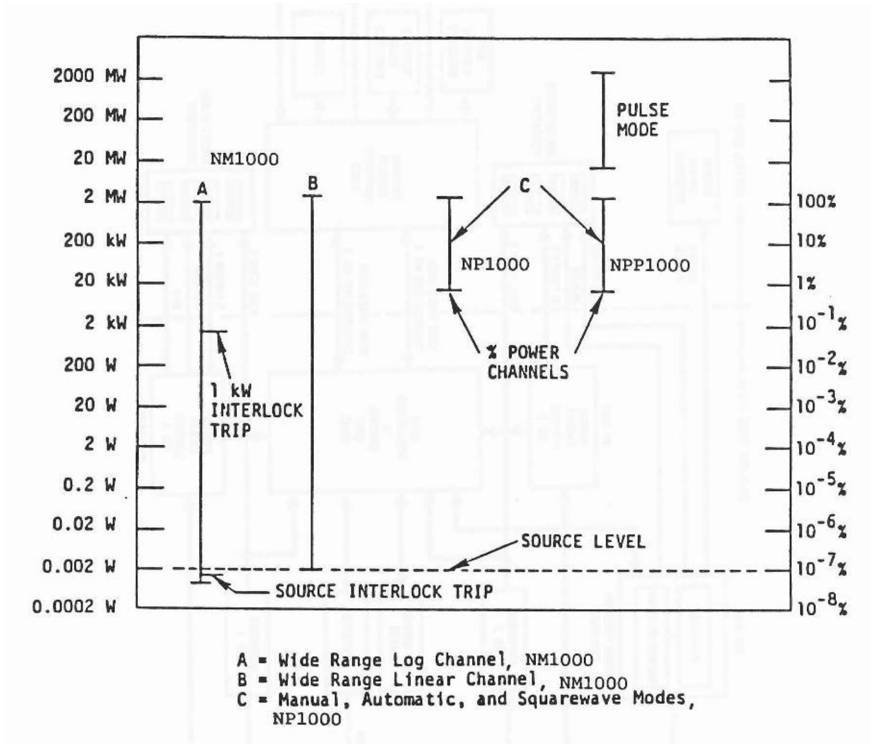
A continuous air particulate monitor with audible and visual alarms shall be functional in the reactor vicinity during reactor operations. A fixed filter beta particulate monitor with 70 1pm flow rate capacity or equivalent system will provide air particulate monitoring. Detectors such as thin window GM detectors will monitor activity and provide alert and alarm conditions with visual and audible annunciators. Count rate of the instrument should include the range of 50 to 50,000 counts/minute.

A gas monitor system for the noble gas effluent, argon-41, shall also be operable during operation, or sufficient data available to demonstrate a calculated release quantity. Design goal for the argon-41 monitor is a sensitivity of 50% of 1 mpc (maximum permissible concentration) for unrestricted areas. Test measurements indicate sensitivity for a ten minute count of 2×10^{-8} μCi/cc.

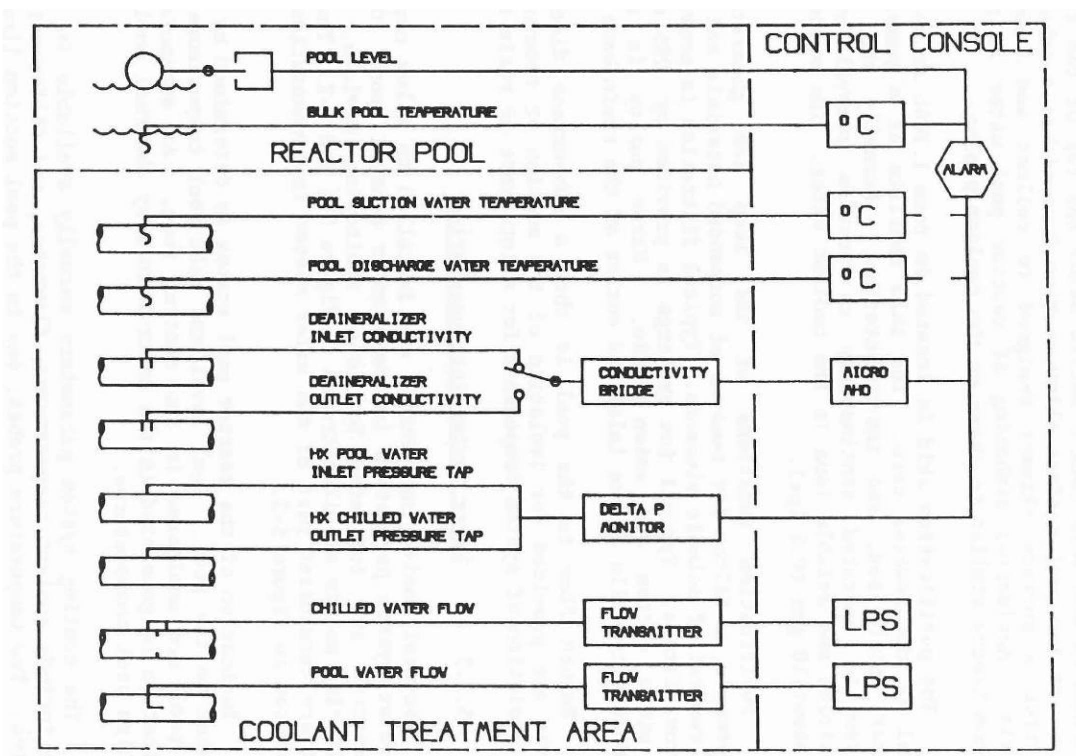
1.1.5 Power Level Instrumentation

Safety Analysis Report, Chapter 6: The Instrumentation and Control System for the TRIGA reactor is a computer-based design incorporating the use of one multifunction, NM-1000 microprocessor neutron flux monitoring channel and two companion NP-1000 current mode neutron monitoring safety channels. The combination of these two systems provides an independent operating channel and the redundant safety function of percent power with scram.

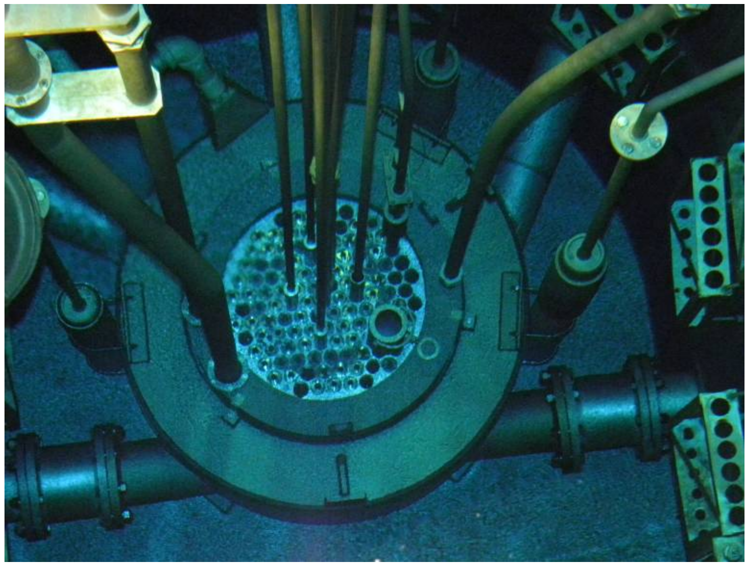
The NM-1000 provides wide range log power and multi-range linear power from source level to full power. The control system logic is contained in a separate control system computer (CSC) with graphics and text displays which are the interface between the operator and the reactor. Another system for data acquisition and control (DAC) functions as the interface point for interface circuitry, process signals and communications.

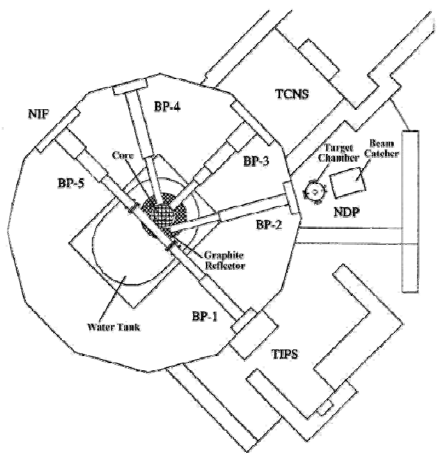


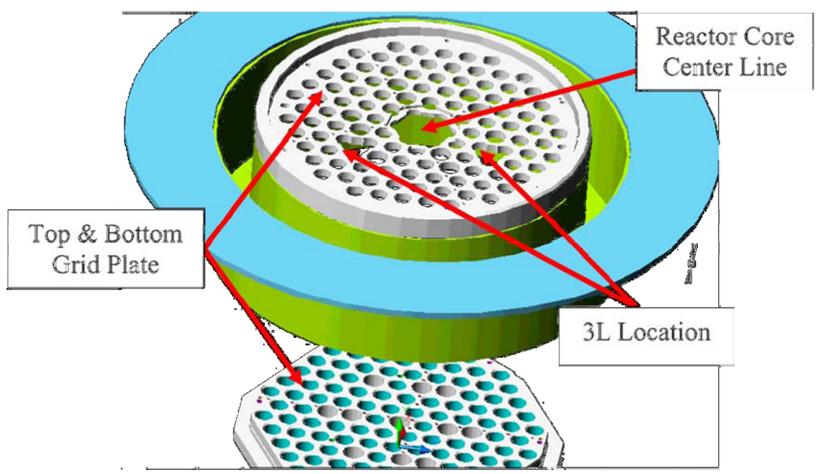
1.1.6 Process System Instruments and controls

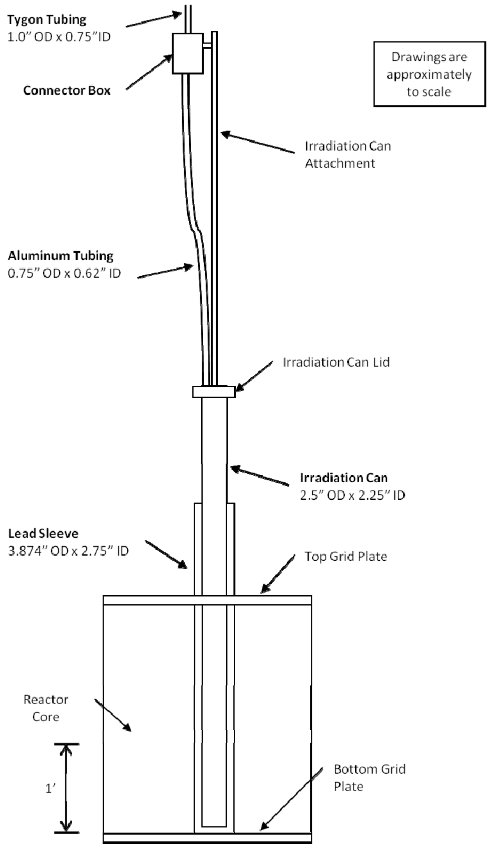


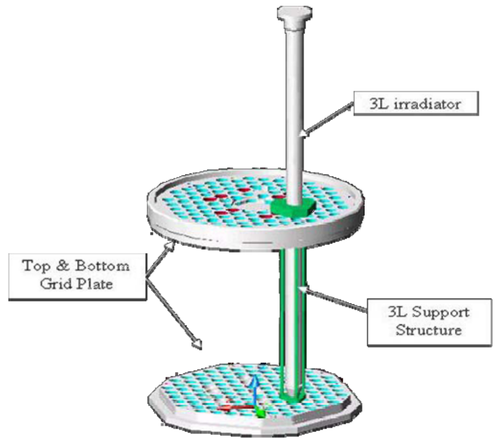
1.1.7 Experiment Facilities & Operations











1.1.8 Facility Sumps & Drains

Pedestal foundation drain; automatically discharges to sanitary sewerage

Liquid radiological waste holding tank; services radiological labs, safety shower, decontamination shower, with batch drain from waste holding tanks

Reactor building foundation drains sump in the truck access ramp; automatically discharges to concrete diversion structure between NETL and Burnet Rd

1.2 References

1.2.1 Reactor Pool and Cooling

Safety Analysis report The University of Texas TRIGA, Chapter 5 (Reactor Coolant System)

Pool and Cooling

Heat dissipation is satisfied by natural convective flow of pool water through the reactor core and forced circulation of the pool water through an external heat exchanger. The pool coolant volume is composed of approximately 41.0 cubic meters in a two by three meter oval pool with a vertical depth of 8.1 meters. A vertical shield is provided by about 6.8 meters of water above the reactor core.

Supplemental cooling of the reactor pool is required for continuous operation at the rated power level. A heat rate of 20.7°C/hour is expected with the reactor operated at 1000 kW. Heat removal from the pool is provided by heat exchange with a chilled water supply. The chilled water supply is operated by the University for cooling of Research Center buildings and equipment. Chilling capacity is provided by multiple 1200 ton (4220 kW) units. At reactor rated power the heat removal capacity required is represented by about 25% of the chilling system capacity of one unit. A tube and shell heat exchanger is installed for heat removal from the reactor pool to the available chilled water system.

Maximum design temperature of the coolant system, coolant inlet temperature, is 120°F (48.9°C). The maximum allowable peak heat flux at this temperature is 325 kBtu/hr-ft^2 (103 watts/cm^2) corresponding to a power level of 1900 kW for an 85 element core. Since the maximum licensed power level is 1100 kW, the resulting maximum heat flux will be 188 kBtu/hr-ft^2 (59.4 watts/cm^2) which is well below the value at which clad integrity may be questioned.

Suction of water from the pool is provided by an inlet which extends no more than 2 meters below the top of the reactor tank. The coolant water is drawn through the coolant pump and forced through the heat exchanger. Return of cooled water to the reactor pool is provided by single or multiple discharge outlets above the reactor core or an outlet near the tank bottom. A diffused water jet is created at the outlets above the reactor core by a nozzle. Delay and diffusion of the reactor core convective coolant column is enhanced by the action of the coolant discharge nozzle.

Accidental siphoning of reactor pool water is prevented by the presence of suction breaks on both suction and discharge lines of the coolant system. Siphon breaks are created by holes located in the lines approximately half a meter below the normal water level.

Indication of the reactor pool status is determined by two sensors located in the pool. Pool level and bulk pool temperature sensors in the pool are monitored in the control room. An annunciator alarm indication is generated in the control room by abnormal pool levels and by high pool temperatures.

Heat removal capacity and thus pool heat rate is specified by analysis of a tube and shell heat exchanger. At a flow rate of 400 gal/min (25.2 liters/sec) of chilled water at 48°F (8.89°C) a heat removal rate of 1140 kW is expected.

The heat exchanger and pump, the major components of the cooling system, are located in room 1.104b at about the same vertical level as the reactor core. Valves are provided in the coolant loop for control and isolation of the cooling system function. Specifications of cooling system components are listed in Table 5-1. A positive pressure difference of 1 psi (7 kilopascals) between the shell side outlet and tube side inlet of the heat exchanger is designed to prevent leakage of primary pool coolant into the secondary chilled water system. The pressure difference is maintained under varying flow conditions by a differential pressure controller which regulates the position of a throttle valve in the heat exchanger shell side outlet pipe. Coolant water supply temperature is regulated by a temperature controller coupled to a mixing valve in the chilled water supply line.

The cooling system parameters normally available in the control room include coolant temperatures, flowrates, and differential pressure status. Two temperature probes, one in the pool suction line and one in the pool discharge line, allow monitoring of heat exchanger cooling function. Typical temperature probes used are resistance temperature detectors (RTD's). Two flow meters, one in the chilled water line and one in the pool water line provide information on system flow rates. A differential pressure monitor provides an alarm if the pressure at the high pressure point on the heat exchanger tube side is not less than the low pressure point on the shell side. The differential pressure is designed for a difference substantially greater than 7 kilopascals (1 lb/sq. in.).

Suction of water from the pool is provided by two inlets, neither of which extend more than 2 meters below the top of the reactor tank. Valves at the pool surface allow suction from either a subsurface inlet or from a surface skimmer designed to collect and remove floating debris. Accidental siphoning of reactor pool water is prevented by siphon breaks similar to those on the coolant piping.

Purification System

The purification skid is located in room 1.104b at about the same level as the reactor core. The skid consists of a pump, flowmeter, filter, resin bed, and instrumentation. Normally the purification system is operated continuously to provide removal of suspended particles and soluble ions in the coolant water. The system flow rate is about 10 gpm (0.6 1ps).

Purification functions of the loop are generated by two components, a filter for removal of suspended materials and a resin bed for removal of soluble elements. Typical filtration is provided with 25 micron filters. Typical ion exchange is provided by .085 cubic meters of mixed cation and anion resin. Water purity is measured by conductivity cells at the inlet and outlet of the resin beds.

Return flow to the pool is thru a subsurface discharge pipe. Valves are provided for isolation of the suction or return lines, and for isolation of system components for maintenance or resin replacement.

Water quality is measured by two conductivity cells in the purification loop. The cells are located on inlet and outlet lines of the demineralizer that readout locally in the control room. Typical conductivity cells are composed of two parts, titanium electrodes shielded by ryton for conductivity measurement, and a thermister for temperature compensation. A Wheatstone bridge circuit on the purification skid is connected to the cells. A switch allows selection of either inlet or outlet conductivity.

A connection from the purification system to the domestic water system provides makeup water to replenish pool evaporation losses. The makeup water connection includes features that isolate the two water systems. Two of these features are valves for flow control and a quick release connection. The valves include a check valve for limiting flow direction and one or more block valves for stopping water flow. Use of the quick release connection allows physical separation of the two systems except during periods in which the makeup process is operating.

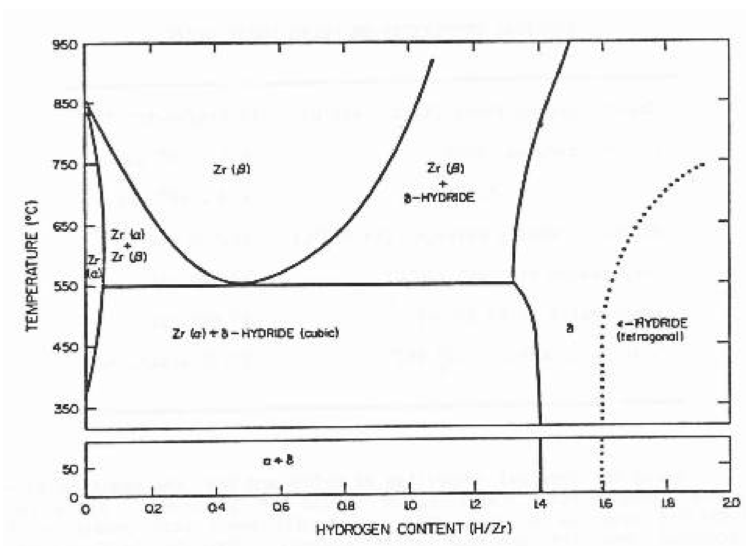
An indication of radioactive release as alternatives to CAM and Ar^{41} monitor is provided if a water activity monitor is installed or by a GM detector area monitor.

1.2.2 Reactor Core & Control Rods

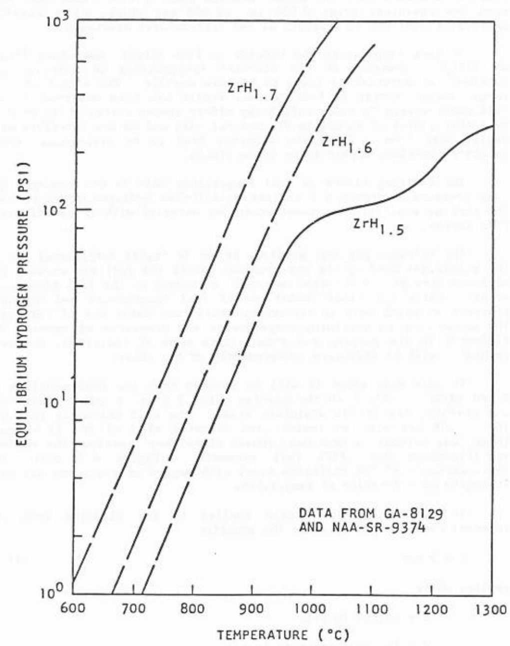
The TRIGA reactor system has three major areas which are used to define the reactor design bases:

a. Fuel temperature,
b. Prompt negative temperature coefficient,
c. Reactor power.

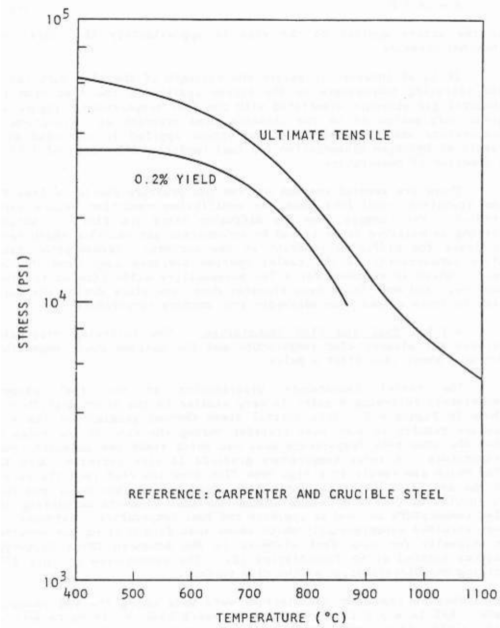
These areas are interrelated, and fuel temperature is selected as the safety parameter most directly applicable to fuel cladding integrity. A ratio of 1:1.6 was selected to ensure stable structure and volume.



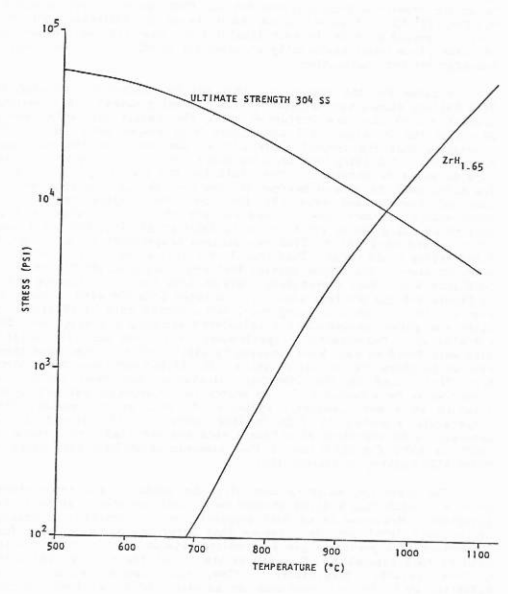
Zr H Phase Diagram



EQUILIBRIUM HYDROGEN PRESSURE
VERSUS TEMPERATURE FOR ZIRCONIUM-HYDROGEN

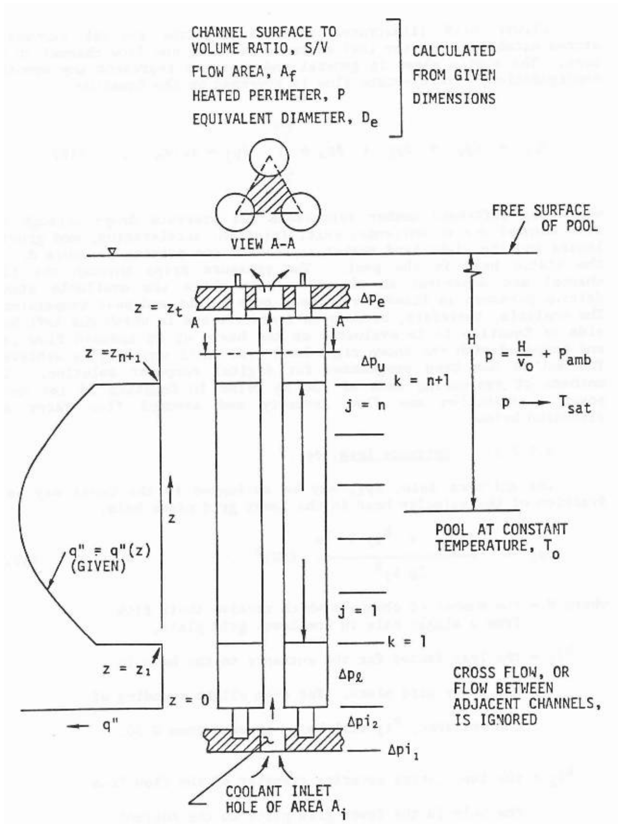


STRENGTH OF TYPE 304 STAINLESS STEEL
AS A FUNCTION OF TEMPERATURE

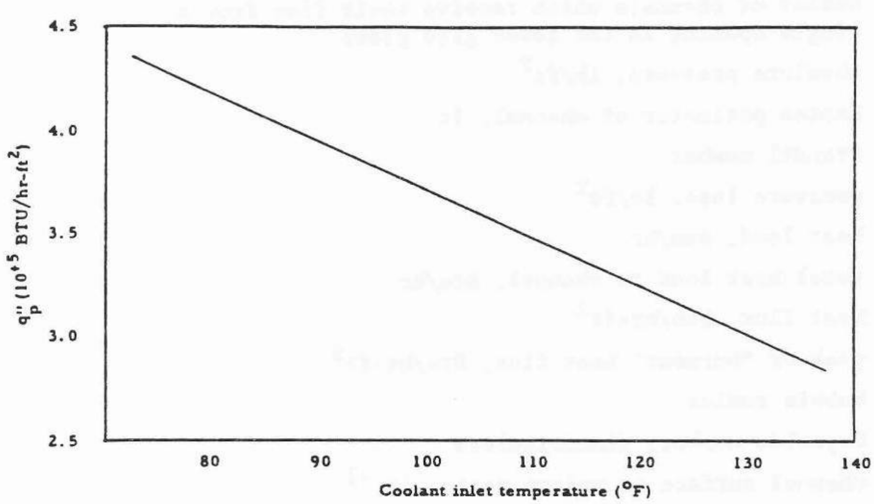


STRENGTH AND APPLIED STRESS AS A FUNCTION OF
TEMPERATURE, EQUILIBRIUM HYDROGEN DISSOCIATION PRESSURE

Thermal hydraulic analysis was performed using radial power peaking factors, TRIGA fuel geometry, and cooling channel geometry defined by the core grid plate and fuel elements.



Fuel temperatures were found to be acceptable up to departure form nucleate boiling. The analysis demonstrates that TRIGA fuel cooled by natural convection is capable of about 1.9 MW power without experiencing departure from nucleate boiling.



PLOT FOR WHICH DNB RATIO IS 1.0 OF
MAXIMUM HEAT FLUX VERSUS COOLANT TEMPERATURE

The UT TRIGA maximum power level limit was established at 1.1 MW to provide a wide margin to departure from nucleate boiling under all conditions. TRIGA fuel has an extremely large negative feedback coefficient associated with the fundamental physics of ZrH.

Relative magnitude of Contributing Components to the Prompt Negative Temperature Coefficient of TRIGA Reactors ($ZrH_{1.6}$ – SS cladding)	
Contributing Component	(%)
Zr Hydride Effect	55
Inhomogeneities (control rod locations)	5
Doppler broadening of ^{238}U resonances	20
Leakage from Core	20

Therefore enough uranium has to be added to compensate for (1) operation at elevated temperatures, (2) xenon generation, and (3) fuel burnup.

ESTIMATED FUEL ELEMENT REACTIVITY WORTH COMPARED WITH WATER AS A FUNCTION OF POSITION IN CORE

Core Position	Worth (% $\delta k/k$) SS Clad $U\text{-}ZrH_{1.6}$	Number of Fuel Positions
B ring	1.07	6
C ring	.85	12
D ring	0.54	18
E ring	0.36	24
F ring	0.25	30
G ring	0.19	36

Control rod design is required to assure the reactor remains subcritical with the most reactive rod failed in the fully withdrawn position.

ESTIMATED CONTROL ROD NET WORTH

Control Rod	diameter in. (cm)	$\delta k/k$ %
C ring - transient	1.25 (3.18)	2.1
C ring - regulating	1.35 (3.43)	2.6
D ring - shim 1	1.35 (3.43)	2.0
D ring - shim 2	1.35 (3.43)	2.0

1.2.3 Confinement System

Safety Analysis Report, Chapter 6: Ventilation of the reactor bay is provided by two modes of system operation. One mode is for standard operation with recirculation of air. The other mode is an exhaust operation with high volume flow that has no air recirculation. Design during exhaust mode operation is a rate of air exchange in excess of two per hour. Total volume for the room exceeds 4120 cubic meters. Normal operation of the ventilation system uses a roof stack for the exhaust of air from the reactor bay.

Isolation of the reactor bay is provided by ventilation dampers. These dampers will shut in response to either manual or automatic signal actuation. An automatic signal will initiate shutdown of the ventilation system by closure of the dampers if a set point for airborne particulate radioactivity exceeds a set point, Protective switches within the ventilation system will cause the air fans to respond to the position change of the dampers. Damper design is for fail-safe operation so that loss of control power will isolate the reactor bay. The separate air purge system is designed to exhaust air that may contain radionuclide products by a low volume system.

The exhaust stack will extend 14 feet (4.24 meters). Ground elevation in the area is 794 feet. Roof elevation at the stack is 843 feet. Therefore, the effective release point is at least 50 feet above the maximum ground elevation at the building.

Relative to ambient atmospheric pressure, the design goals for the reactor bay, adjacent zones and academic wing of the building is a negative pressure difference. The differential pressures are 0.06: 0.04: 0.03 inches (0.15: 0.10: 0.80 cm.) of water. Air balance values relate typical or standard values except in the case of the reactor bay.

Argon-4l is produced in the reactor pool as a result of the (n,γ) reaction with argon-40 dissolved in the pool water. Most of the argon-4l remains in solution but some of it is transferred to the reactor room air at the pool surface. Calculations of the argon released from the pool surface estimate a concentration in the room of 1.6×10^{-6} pCi/cc with the reactor operating at 1000 kW.

1.2.4 Facility Radiation Monitoring

SAR 9.5. EVALUATION OF MONITORING SYSTEMS

The radiation monitors provide information to operating personnel of impending or existing hazards from radiation so that there will be sufficient time to take the necessary steps to control the exposure of personnel and the release of radioactivity or to evacuate the facility.

Three types of radiation monitors are used: a continuous air particulate monitor for determining radiation levels due to particulate radioisotopes suspended in the reactor room air, a continuous air gaseous monitor for determining radiation levels due to argon-

41 in the room air, and area radiation monitors for determining the gamma field at several locations in the facility.

Each type of radiation monitor has a specific radiological purpose. The particulate air monitor is used to detect radioisotopes released due to fuel element failure (a design basis accident). The gaseous air monitor is used to determine the effluent radiation release of argon-41. Argon, a component of air (.04% by volume) may be activated to produce argon-41 in potentially significant quantities. Finally, the area radiation monitors are used to minimize personnel radiation exposures. The facility radiation monitors discussed in SAR sections 9.5.1., 9.5.2, and 9.5.3 are typical instruments at the time of original installation. Replacements may have slightly different characteristics.

Set points for the particulate continuous air monitor warn of the presence of particulate fission product nuclides. Since gaseous and volatile elements such as krypton, xenon, bromine, and iodine have particulate decay products, the presence of some of their radioisotopes should also be detected. An alarm set point at 2000 picoCuries/milliliter detects particulate activity concentrations at the occupational values of 10 CFR 20 for 70% of the relevant isotopes in the ranges 84-105 and 129-149. These ranges of isotopes represent the one percent yield for fission products of uranium-235.

The air monitor in use is a Ludlum Model 333-2 beta air monitor, configured for continuous sampling of airborne beta-emitters. It uses two standard pancake G-M tubes, each having a 1.75 inch effective diameter. The two detectors are arranged in line so that gamma background subtraction is performed. This increases the accuracy of the beta count. The 333-2 will accept air flow rates ranging from 10-100 liters per minute. Particle accumulation on a fixed filter continues at a constant rate. Activity on the filter, however, is a function of the air flow rate, filter collection efficiency, and the decay rates for nuclides that are present on the filter.

A high radiation area, defined in 10 CFR 20 as having a radiation level >100 mrem/hr, may exist above the pool access area during some operations. The area radiation monitor located above the pool access area will have an alarm set point of 100 memr/hr. Although the radiation levels within the pool protection railings may exceed doses of 100 mrem/hr, the dose exists only in the immediate area of the pool surface. At other locations of the pool shield platform level, the dose rates are significantly less than 100 mrem/hr, but may exceed the 2 to 5 mrem/hr range. While the reactor is operating, one area radiation monitor will operate above the pool access area, as well as at least two additional area radiation monitors located at other positions around the reactor shield and at the beam port facilities. The radiation monitor system consists of six units with GM tubes that detect dose rates from 0.1 mR/hr to 10 mR/hr.

1.2.5 Power Level Instrumentation

Safety Analysis Report, 5.1.1. NM-1000 Neutron Channel The NM-1000 nuclear channel has multifunction capability to provide neutron monitoring over a wide power range from a single detector. The selectable functions are any or all of the following:

a. Percent power.
b. Wide-range log power
c. Power rate of change.
d. Multi - range linear power.

For the TRIGA ICS, one NM-1000 system is designated to provide the wide-range log power function and the multirange linear power function. The wide-range log power function is a digital version of the patented GA 10-decade log power system to cover the reactor power range from below source level to l50% power and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the Campbelling (root mean square, RMS, signal processing) range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the full range of power. The microprocessor automatically tests the system to ensure that the upper decades are operable while the reactor is operating in the lower decades and vice versa when the reactor is at high power.

For the multirange function, the NM-1000 uses the same signal source as for the 1og function. However, instead of the microprocessor converting the signal into a log function, it converts it into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power. The same self-checking features are included for the log function. A linear power level signal is available for the percent power safety function.

The NM-1000 system is contained in two National Electrical Manufacturers Association (NEMA) enclosures, one for the amplifier and one for the processor assemblies. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low voltage power supplies, detector high-voltage power supply, and digital diagnostics and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics (between amplifier and processor), the microprocessor, a control/display module, low-voltage power supplies, isolated 4 to 20 mA outputs, and isolated alarm output. Outputs are Class lE as specified by IEEE. Communication between the amplifier and processor assemblies is via two twisted-pair shielded cables.

The system is automatically calibrated and checked (including the testing of trip levels) prior to operation. The checkout data is recorded for future use, and operation cannot proceed without a satisfactorily completed checkout. The neutron detector uses the standard 0.2 counts per nv fission chamber that has provided reliable service in the past. It has, however, been improved by additional shielding to provide a greater signal-to-noise ratio. The low noise construction of the chamber assembly allows the system to respond to a low reactor shutdown level which is subject to being masked by noise.

The NP-1000 Power Safety Channel is a complete linear percent power monitoring system mounted within one compact enclosure which contains current to voltage conversion signal conditioning, power supplies, trip circuits, isolation devices, and computer interface circuitry. The power 1evel trip circuit is normally hardwired into the scram system and the isolated analog outputs are monitored by the CSC as well as being hardwired to a bar graph indicator. A special version of the safety channel, the NPP-1000, provides measurement functions for peak pulse power, total pulse energy, automatic gain change and related trip points. The control system automatically selects proper gain setting for steady-state or pulse mode when the operator determines the reactor operating mode. Peak pulse power and total pulse energy are also set by the pulse operation mode. Both safety channels, the NP-1000 and the NPP-1000, are identical except for the peak and energy circuits. The detector for each safety channel is either an ionization chamber or self-powered in-core detector.

1.2.6 Process System Instruments and controls

Several monitoring sensors are installed to allow remote readout of water system parameters in the reactor control room. Other system parameters are indicated by local monitoring devices.

Indication of the reactor pool status is determined by two sensors located in the pool. Pool level and bulk pool temperature sensors in the pool are monitored in the control room. An annunciator alarm indication is generated in the control room by abnormal pool levels and by high pool temperatures. The cooling system parameters normally available in the control room include coolant temperatures, flowrates, and differential pressure status. Two temperature probes, one in the pool suction line and one in the pool discharge line, allow monitoring of heat exchanger cooling function. Typical temperature probes used are resistance temperature detectors (RTD's). Two flow meters, one in the chilled water line and one in the pool water line provide information on system flow rates. A differential pressure monitor provides an alarm if the pressure at the high pressure point on the heat exchanger tube side is not less than the low pressure point on the shell side. The differential pressure is designed for a difference substantially greater than 7 kilopascals (l lb/sq. in.).

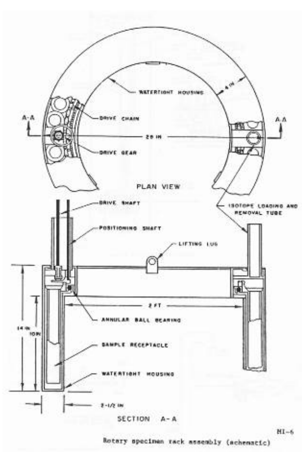
Numerous water system parameters are measured by local pressure or temperature sensors in the system lines. Both temperature and pressure probe are located on the inlet and outlet lines of the pool water side and chilled water side of the heat exchanger. A local indication of flow in chilled water side of the coolant loop is provided by the pressure drop across a venturi in the flow path. Purification loop flow is measured by an in line flow meter. Water pressure before and after the filter in the purification loop is measured for indication of filter condition. Water quality is measured by two conductivity cells in the purification loop. The cells are located on inlet and outlet lines of the demineralizer that readout locally in the control room. Typical conductivity cells are composed of two parts, titanium electrodes shielded by ryton for conductivity measurement, and a thermister for temperature compensation. A Wheatstone bridge circuit on the purification skid is connected to the cells. A switch allows selection of either inlet or outlet conductivity.

A connection from the purification system to the domestic water system provides makeup water to replenish pool evaporation losses. The makeup water connection includes features that purify the domestic water before introduction into the pool water, and valves that isolate the two water systems. Two of these features are valves for flow control and a quick release connection. The valves include a check valve for limiting flow direction and one or more block valves for stopping water flow. Use of the quick release connection allows physical separation of the two systems except during periods in which the makeup process is operating.

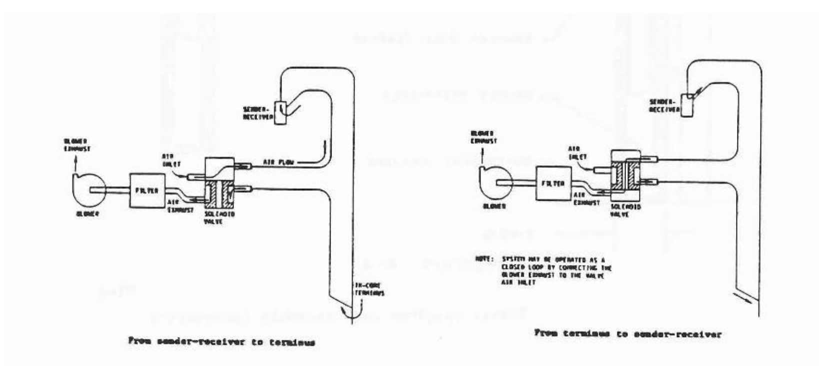
1.2.7 Experiment Facilities & Operations

Central thimble: aluminum tube that fits through the center hole of the top and bottom grid plates 1.5 in. o.d. (3.81 cm.) and 1.33 in. i.d. (3.38 cm.). Holes in the tube above the lower grid plate assure that it is normally filled with water in the entire core region.

Rotary Specimen Rack: multiple-position (40) specimen rack located 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 multiple samples. All positions in this rack are exposed to neutron fluxes of comparable intensity. Specimen positions at 1.23 in. (3.18 cm.) in diameter by 10.80 in. (27.4 cm.) in depth. Samples are loaded from the top of the reactor through a water-tight tube. The facility rotates either manually or via a motor which shares the power supply for the pool lights

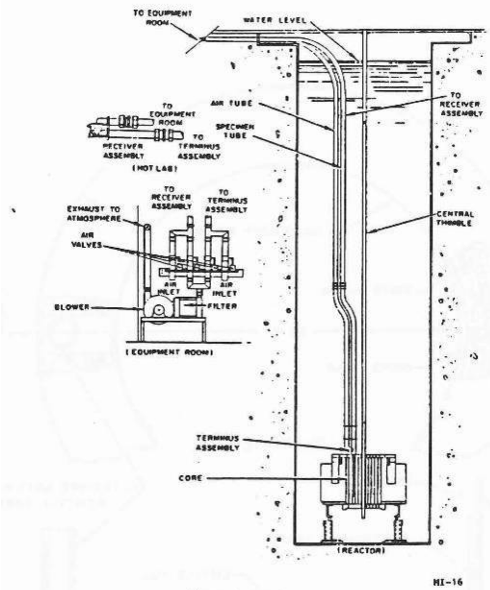


Rotary specimen rack assembly (schematic)



GA pneumatic specimen tube: sample capsule (rabbit) is conveyed to a receiver-sender station via 1.25 in. o.d. (3.18 cm.) aluminum tubing. Effective space in the specimen transfer capsules is 0.68 in. (1.7 cm.) diameter by 4.5 in. (f1.4 cm.) height. An optional transfer box may be employed to permit the sample to be sent and received from up to three different receiver-sender stations.

UT-design pneumatic tube system: the sample capsule is conveyed through a transit section of polyethylene tubing ¾" i.d. to the in-pool/core terminus slightly less than 5/8" i.d. Two in-pool/core assemblies provide cadmium covered section or bare.

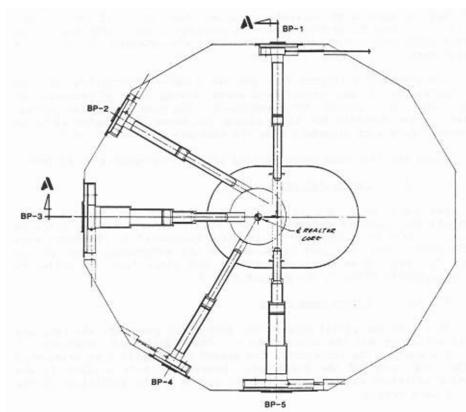


Pneumatic transfer system schematic

There are five beam ports divided into two categories as follows:

Tangential beam ports: Two beam ports are oriented tangential to the reactor core, penetrate the graphite reflector, the coolant water, and the concrete shield. A hole is drilled in the graphite tangential to the outer edge of the core. One beam port terminates at the tangential point to the core. The other beam tubes extend both directions from the reflector and out opposite sides of the reactor shield.

Radial beam ports: There are two radial beam ports, each which penetrate the concrete shield structure and the coolant water. One radial port terminates at the inner edge of the reflector. The second radial port also terminates at the outer edge of the reflector, however a hole drilled in the graphite reflector extends the effective source of radiation to the core fuel region.



Three element feature: the 3-El is a 2.062 in. (5.24 cm) diameter hole. Location of the hole center coincides with the center of a three element sub array. Total area of the hole is 3.34 in.^{2} (21.5 cm^2), but also contains additional area for each of the three elements of the original sub array. The area available under the cutout for experiments is limited by clearance for adjacent elements to less than the area of the central and three adjacent fuel element positions. Three single element holes have a total area of 5.34 in.^{2} (34.4 cm^2). One three element sub array cutout in the sub array is one D ring and two E ring locations. The other three element sub array is one E ring and two D ring locations.

Six element feature: A 6-El removable grid space is located at the center of the core grid structure controls the position of the center hole and adjacent B ring elements. Removal of the A and B ring element array will create a 4.005 in. (10.17 cm) diameter hole. This hole has an area of 12.6 in.^{2} (81.3 cm^2), centered at reactor core center. Additional space is available in the partial element holes that remain in the B ring. This additional space is the result of the experiment hole diameter, which exceeds the diameter of the center hole, but is less than a diameter that would include all the B ring holes. A limit shall exist on the use of this space such that the total area available to the experiment remains less than 15.8 in.^{2} (30 cm^2).

Seven Element Feature: during reflector change out, a facility similar to the 6-El was fabricated, but displaced into the outer rings.

Experiment evaluation requirements:

Double encapsulation is required for experiments containing materials:

- Corrosive to reactor components, compounds highly reactive which water, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated.

- NFPA reactivity ratings greater than two or fire ratings equal to four should be doubly encapsulated. The encapsulation is to protect against the energy release and corrosion properties.

- Materials having a reactivity rating of 1 or a flammability rating greater than 1 should be evaluated individually to determine if double encapsulation is warranted.

- More than 100 ppm of thorium, uranium, or plutonium.

No more than 25 mg TNT equivalent explosive material is permitted

In fueled experiments, total radioactive iodine inventory (131-135) shall be no greater than 750 mCi, with maximum strontium inventory no greater than 2.5 mCi

Experiment materials, except fuel materials which could off -gas, sublime, volatize, or produce aerosols under one of the following conditions shall be limited to occupational levels for airborne radioactivity concentration, as specified in 10CFR20, where averaging over a year for:

- Normal operating conditions of the experiment or reactor
- Credible accident conditions in the reactor
- Possible accident conditions in the experiment.

1.2.8 Facility Sumps & Drains

1.3 Technical Specifications

1.3.1 Reactor Pool and Cooling

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
3.3.1.a	Pool cooling	48°C	4.3.1.a	Annual	
			4.3.1.a	Monthly	
			4.3.1.a	During operation	
3.3.1.b	Pool cooling	6.5 m	4.3.1.b	Annual	
			4.3.1.b	Monthly	
			4.3.1.b	During operation	
3.3.1.c	Pool cooling	5.0 µmho <mo>	4.3.1.c	Annual	
			4.3.1.c	Monthly	SURV-4 Reactor Water Systems Surveillance
			4.3.1.c	During operation	
3.3.1.d	Pool cooling	<1 psig HX dp	4.3.1.d	Daily & retest	
			4.3.1.d	During operation	
3.3.1.e	Pool cooling	Periodic measurement ph	4.3.1.e	Quarterly	SURV-4 Reactor Water Systems Surveillance
3.3.1.e	Pool cooling	Periodic measurement α-β	4.3.1.e	Quarterly	SURV-4 Reactor Water Systems Surveillance
3.3.1.e	Pool cooling	Periodic measurement γ spec	4.3.1.e	Annually	
5.2.1	Reactor coolant system	Natural convection			
5.2.2	Reactor coolant system	Siphon breaks			
5.5.a	Reactor pool irradiator	≤ 10KCi Co-60			
5.5.b	Reactor pool irradiator	≥4.5 m under water			
5.5.b	Reactor pool irradiator	≥ 0.5 m form wall			
5.5.c	Reactor pool irradiator	Pool water 2.5X10x test gamma irradiator			

1.3.2 Reactor Core & Control Rods

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
2.1	Safety limit Fuel Temp	Clad <500°C, <1150°C			
2.1	Safety limit Fuel Temp	Clad >500°C, <950°C			
2.2.1	Fuel temp LSSS	< 550°C			
2.2.1	Fuel temp LSSS	IFE in B or C ring			SURV-1 Fuel temperature calibration
2.2.2	Power level	1100 kW			
2.2.3	Reactivity insertion	2.2% Δk/k (pulse)			
3.1.1	Excess reactivity	4.9% Δk/k	4.1.1	Annually or post core/control rod changes	SURV-3 Excess Reactivity and Shutdown Margin
3.1.2	Shutdown margin	0.2% Δk/k (ref core, 1 rod out, moveable experiments most reactive)	4.1.2	Annually or post core/control rod changes	SURV-3 Excess Reactivity and Shutdown Margin
3.1.3	Transient rod	2.8% Δk/k worth			SURV-6 Control Rod Calibration
			4.1.3	Annual	SURV-7 Pulse Characteristic Comparison
3.1.3	Transient rod	< 15 s withdrawal time	4.1.3	Annually or post core/control rod changes	SURV-6 Control Rod Calibration
3.1.4.a	Fuel elements	< 2.54 mm elongation	4.1.4		
3.1.4.b	Fuel elements	< 1.5875 mm bend	4.1.4		
3.1.4.c	Fuel elements	No clad defect (fission product release)	4.1.4	Biennial	
3.2.1.a	Control rods	No apparent damage	4.2.1.a	Biennial	
3.2.1.b	Control rods	< 1 s scram	4.2.1.b	Annually or retest	SURV-6 Control Rod Calibration
3.2.1.c	Control rods	< 0.2% Δk/k s^{-1}	4.2.1.c	Annually or retest	SURV-6 Control Rod Calibration
3.2.2.b	Control rods	multiple simultaneous manual rod withdrawal prevented	4.2.2	Semiannually or retest	
3.2.2..c	Transient Rod	Interlock prevents pulsing unless down	4.2.2	Semiannually or retest	
3.2.2.d	Standard Control rods	Withdrawal in pulse mode prevented	4.2.2	Semiannually or retest	
3.2.2..e	Transient Rod	Pulse prevented if > 1 kW	4.2.2	Semiannually or retest	
3.2.3.a	Safety System	550°C (2)	4.2.3	Annually	SURV-1 Fuel temperature calibration
			4.2.3	Daily channel check	
3.2.3.b	Safety System	<1.1 MW SS	4.2.3	Annually or retest	SURV-2 reactor power calibration

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
			4.2.3	Daily	
3.2.3.b	Safety System	<2000 MW P	4.2.3	Annually or retest	
3.2.3.e	Manual SCRAM	Operable	4.2.3	Semiannually or retest	

1.3.3 Reactor Pool & Pool Cooling

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
3.3.1.a	Pool cooling	48°C	4.3.1.a	Annual	
			4.3.1.a	Monthly	
			4.3.1.a	During operation	
3.3.1.b	Pool cooling	6.5 m	4.3.1.b	Annual	
			4.3.1.b	Monthly	
			4.3.1.b	During operation	
3.3.1.c	Pool cooling	5.0 µmho <mo>	4.3.1.c	Annual	
			4.3.1.c	Monthly	SURV-4 Reactor Water Systems Surveillance
			4.3.1.c	During operation	
3.3.1.d	Pool cooling	>1 psig HX dp	4.3.1.d	Daily & retest	
			4.3.1.d	During operation	
3.3.1.e	Pool cooling	Periodic measurement ph	4.3.1.e	Quarterly	SURV-4 Reactor Water Systems Surveillance
3.3.1.e	Pool cooling	Periodic measurement α-β	4.3.1.e	Quarterly	SURV-4 Reactor Water Systems Surveillance
3.3.1.e	Pool cooling	Periodic measurement γ spec	4.3.1.e	Annually	
5.2.1	Reactor coolant system	Natural convection			
5.2.2	Reactor coolant system	Siphon breaks			
5.5.a	Reactor pool irradiator	≤ 10KCi Co-60			
5.5.b	Reactor pool irradiator	≥ 4.5 m under water			
5.5.b	Reactor pool irradiator	≥ 0.5 m from wall			
5.5.c	Reactor pool irradiator	Pool water $2.5X10^{-5}$, test gamma irradiator			

1.3.4 Confinement System

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
3.3.2.a	Confinement isolation	Supply & Exhaust damper & Fan trip on radiation level	4.3.2.b	Monthly	SURV-5 Air Confinement System Surveillance

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
3.3.2.c	Confinement isolation	Air purge 2 exchanges per hour	4.3.2.c	Monthly	SURV-5 Air Confinement System Surveillance

1.3.5 Facility Radiation Monitoring

SECT	DESCRIPTION	SPECIFICATION	SR	FREQ	Procedure/record
3.3.3.a	Rad Monitoring Systems – CAM (particulate) with readout and alarm	Operable	4.3.3.a	Semiannually	
			4.3.3.a	Weekly	
3.3.3.a	Rad Monitoring Systems – CAM (particulate)	<5 meters from pool		Na	
3.3.3.a	Rad Monitoring Systems – CAM (particulate)	$2x10^{-9}$ µCi/cm^3 W 2 h particulate accumulation		Na	
3.3.3.a	Rad Monitoring Systems – CAM (particulate)	OOS for 1 week if filter evaluated daily & Ar-41 monitor available		Na	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41) with readout and alarm	Operable	4.3.3.b	Monthly	
			4.3.3.b	Biennially	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41)	Sample purge exhaust when operating		Na	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41) alarm setpoint	<$2x10^{-5}$µCi/cm^3 Daily release		Na	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41) releases	<$2x10^{-6}$µCi/cm^3 Annual release		Na	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41)	If OOS, air purge <10 days		Na	
3.3.3.c	Area radiation monitors with readout and alarm	Operable	4.3.3.c	Semiannually	
			4.3.3.c	Weekly	
3.3.3.c	Area radiation monitors	Setpoint ≤ 100 mR/hr		Na	
3.3.3.c	Area radiation monitors	Pool level		Na	
3.3.3.c	Area radiation monitors	2 other areas		Na	

1.3.6 Power Level Instrumentation

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
2.2.2	Power level	1100 kW			
2.2.3	Reactivity insertion	2.2% $\Delta k/k$ (pulse)			
3.2.2.a	Interlock	<2 cps S/U	4.2.2	Semiannual or retest	
3.2.2.e	Interlock	1 kW pulse	4.2.2	Semiannual or retest	
3.2.3.b	Safety System	<1.1 MW SS	4.2.3	Annual or retest	SURV-2 reactor power calibration
			4.2.3	Daily	
3.2.3.b	Safety System	<2000 MW P	4.2.3	Annual or retest	
			4.2.3		
3.2.3.f	Safety System	Watchdog (scan rate)	4.2.3	Annual or retest	
3.2.4.b	Power level	2 channels			
3.2.4.c	Pulse power	1 channel			
3.2.4.d	Pulse energy	1 channel			

1.3.7 Experiment Facilities & Operations

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
3.4.1.a	Experiment reactivity	Moveable, <$1.00	4.4.1		
3.4.1.b	Experiment reactivity	Secured <$2.5	4.4.1		
3.4.1.c	Experiment reactivity	Total possible < $3.00	4.4.1		
3.4.2.a	Experiment Materials	Corrosive, reactive, explosive double encapsulated	4.2.2		
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, remove			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, inspect			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, corrective action			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, Director review prior to operation			
3.4.c	Experiment Materials	<25 mg explosives			
3.4.c	Experiment Materials	Pressure calculation (or exp) within capsule design			
3.4.d	Experiment Materials	<750 mCi I131-I135, <2.5 mCi Sr (fueled)			
3.4.e	Experiment Materials	<MPC for 100% release if volatile			
3.4.f	Experiment Materials	3.4.e calc assume 10% release			

1.3.8 Sumps

NA

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$	Parameter	TS LIMIT	CURRENT VALUE	NOMINAL VALUE
-$1.16	3-element cadmium lined canister, water filled[2] (D & E)			■
-$0.92	3-element cadmium lined canister[2] (D & E)			■
-$0.90	PNT (position A1) worth[1]			■
-$0.50	CT (position A1) void worth[1]			■
-$0.45	through tube void worth[1]			■
-$0.40	RSR poison 40 palaces worth[1]			■
-$0.35	piercing tube void worth[1]			■
-$0.33	Cd pneumatic tube[1]			■
-$0.16	PNT (position G1) worth[1]			■
$0.19	G-Ring fuel element worth[1]			■
$0.20	Maximum dummy worth[1]			■
$0.25	F-Ring fuel element worth[1]			■
$0.286	**Min SDM TS 3.1.2 (1 rod out, 0.2% Δk/k)**	■		
$0.3	**Max xenon worth "Ref, Core Condition" TS 1.20**	■		
$0.36	E-Ring fuel element worth[1,3]			■
$0.50	Minimum dummy worth[1]			■
$0.54	D-Ring fuel element worth[1,2]			■
$0.85	C-Ring fuel element worth[1]			■
$1.00	**"Reactor Shutdown" using "Ref. Core Cond." TS 1.20**	■		
	Max single moveable experiment TS 3.4.1.a	■		
	Expt. movement requires SRO TS 6.1.3	■		
$1.07	B-Ring fuel element worth[1]			■
$1.25	Fuel in 3-El. facility {[E22/23, D17] or [F22/23, D17]}			■
$2.50	**Max single fixed experiment worth TS 3.4.1.b**	■		
$2.54	SHIM 1 current worth [D14]		■	
$2.90	REGULATING ROD current worth [C1]		■	
$3.00	max sum of all experiments reactivity TS 3.4.1.c	■		
$3.10	SHIM 2 current worth [D6]		■	
$3.14	**Maximum pulsed reactivity TS 2.2.3 (2.2% Δk/k)**	■		
	TRANSIENT ROD current worth [C7]		■	
$4.00	A-Ring fuel element worth			■
$4.00	**Transient rod worth TS 3.1.3 (2.8% Δk/k)**	■		
	Max reactivity insertion analysis (substantial margin)			■
$5.79	Current core excess reactivity		■	
$5.89	Current critical rod worth		■	
$6.42	Fuel in 6-El. facility { B1/2/3/4/5/6, CT }[1]			■
$7.00	**Max excess reactivity TS 3.1.1 (4.9% Δk/k)**	■		
$11.68	Total control rod worth		■	

[1] Reactor Reference Data Notebook, Safety Analysis report Table 4-5; SAR Table 4-6 indicates CT Fuel $0.90, CT Void -$0.15, PNT Void -$0.10, RSR void -0.20

[2] Significant deviation from values in 3-Element Experiment Authorization (cf. E-Ring ~$0.50 & D-Ring $0.95)

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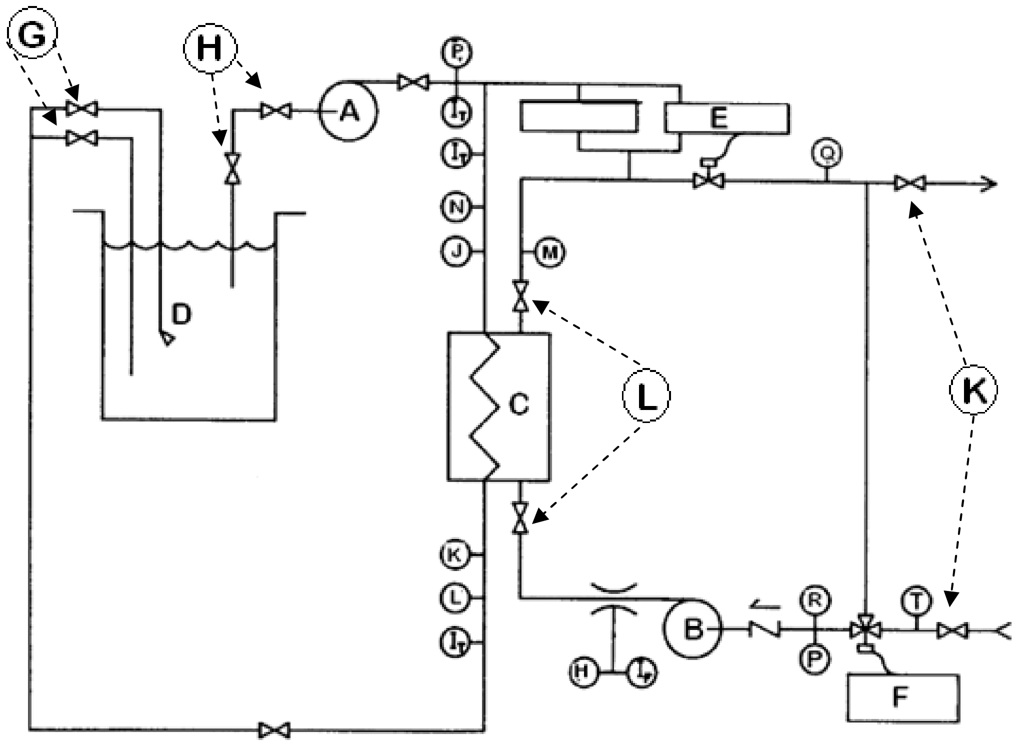

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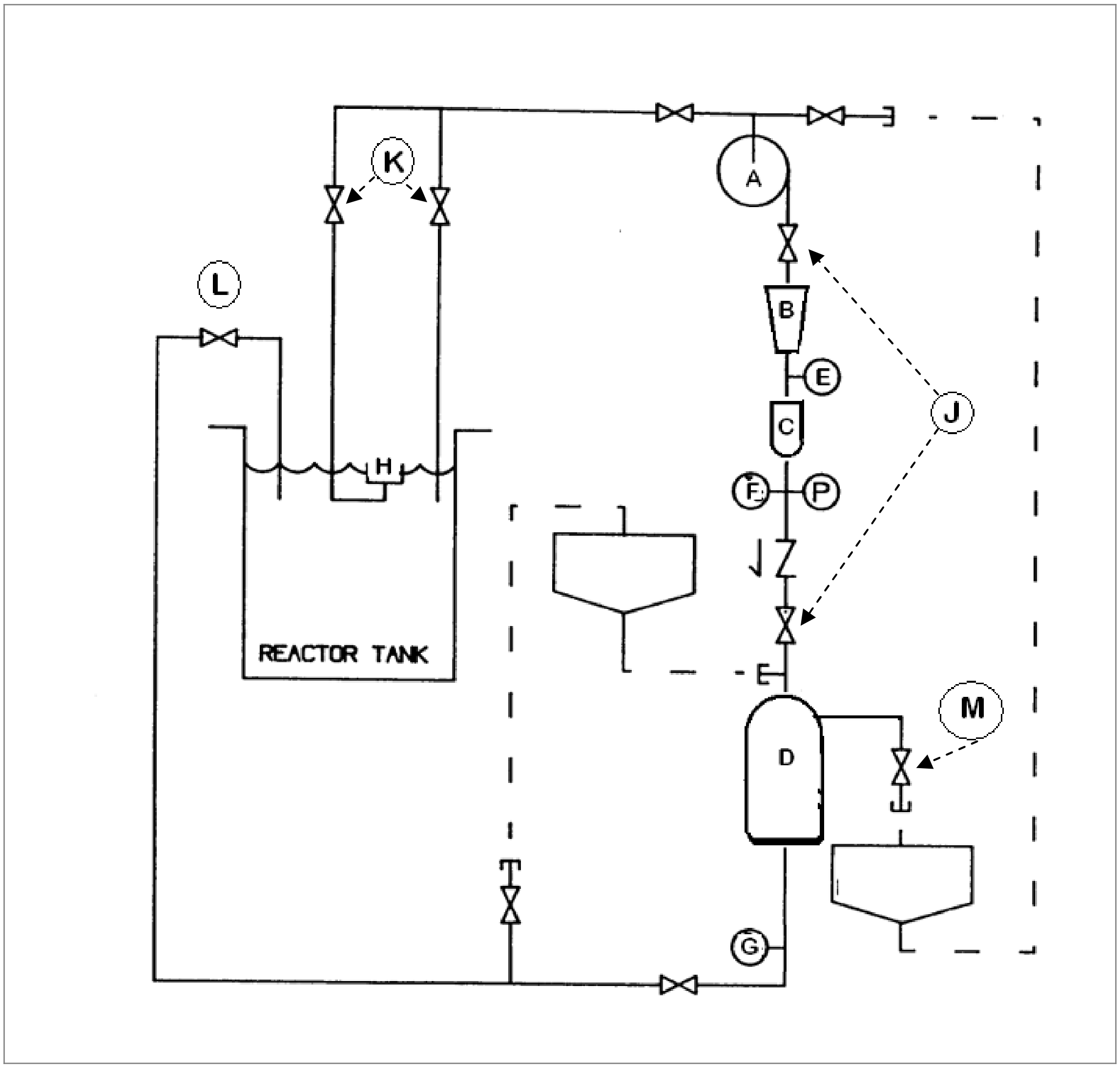
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CAUTION



Pneumatic Tube

RSR Loading Port

RSR Operating Shaft

7-Element Facility

Pneumatic tube

Fission Chamber

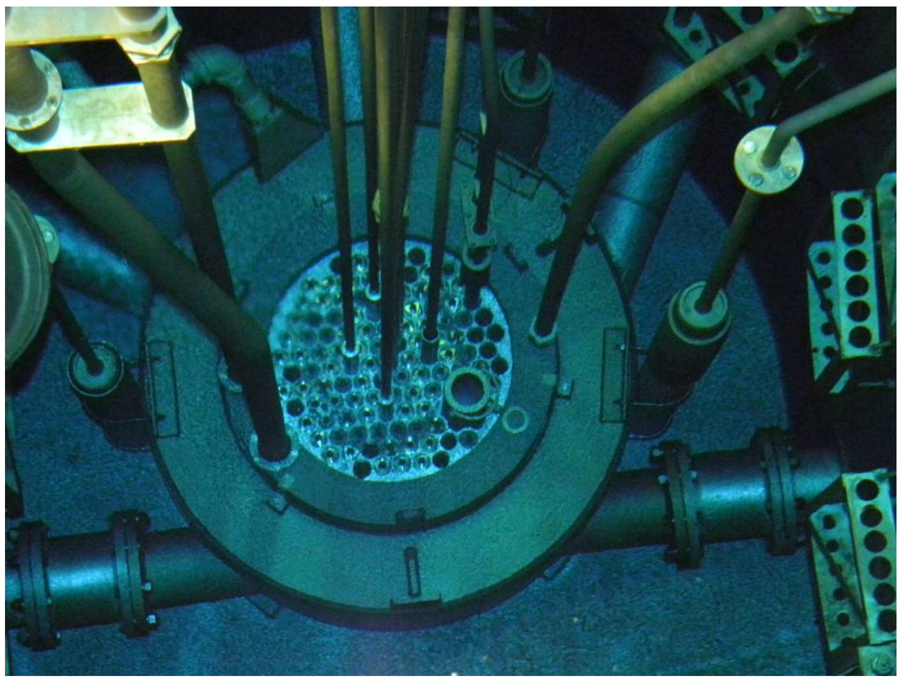
Safety Channel 1/2

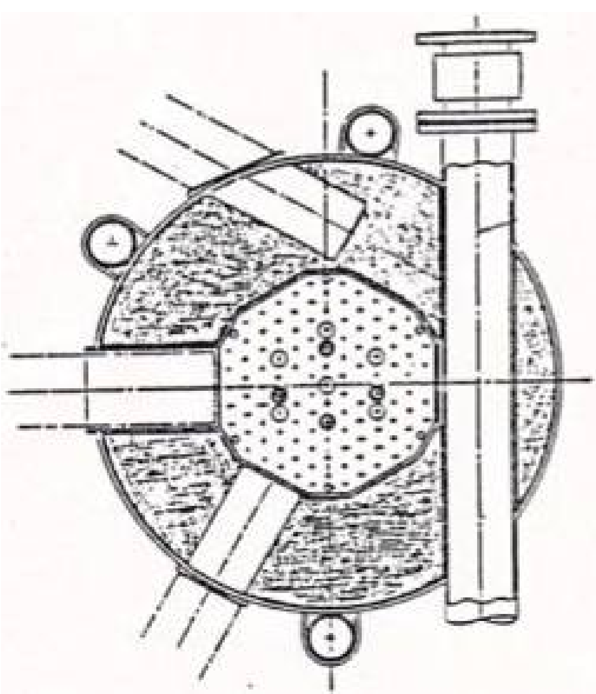
N16 Diffuser

Regulating Rod

Transient Rod

Shim rod 1





Positron Facility

Neutron Radiography Facility

Texas Cold Neutron Source

Neutron Depth profiling Facility

Parameter	**$ or** % Δk/k
Min SDM TS 3.1.2 (1 rod out)	
Max xenon worth “Ref, Core Condition” TS 1.20	
“Reactor Shutdown” using “Ref. Core Cond.” TS 1.20	
Max single moveable experiment TS 3.4.1.a	
Expt. movement requires SRO TS 6.1.3	
Max single fixed experiment worth TS 3.4.1.b	
max sum of all experiments reactivity TS 3.4.1.c	
Maximum pulsed reactivity TS 2.2.3	
Transient rod worth TS 3.1.3	
Max excess reactivity TS 3.1.1	

PROCEDURE

OPER-1

Startup - Shutdown Checks

Version
1.00

Approvals:

Facility Director, NETL — Date: 4 / 4 / 02

Chairperson, Nuclear Reactor Committee — Date: 4 / 4 / 02

Number of Pages: 6
Number of Words: 1062
Number of Characters: 5889

NUCLEAR ENGINEERING TEACHING LABORATORY
J. J. PICKLE RESEARCH CAMPUS
THE UNIVERSITY OF TEXAS AT AUSTIN

I. INTRODUCTION

A. Purpose

This procedure describes the facility and reactor system checks to be done prior to startup of the reactor and subsequent to shutdown of the reactor including a check of operation requests for valid experiments.

B. Description

Operation of the reactor requires test and verification of Instrumentation Control and Safety (ICS) System functions and other facility systems. Checklists for the test and verification of ICS System and checks of support systems assure that systems are operable prior to or during operation of the reactor.

Several facility systems must function properly for the reactor systems to operate safely. The two most important are the pool water system and air confinement system. Other equipment such as communication equipment and radiation monitoring equipment are also necessary for operation. A checklist documents the status of various systems. Both pre-start checks and post shutdown checks are for the purpose of verifying the operability or condition of important systems.

Prior to actual operation, a review of the operation requirements and check of valid experiment requests and approvals must be made. An operation request form documents the request and the valid experiment approval. All actions of this procedure require the direct supervision of a reactor operator with a valid license.

C. Schedule

Apply this procedure each day the reactor is taken through an operation cycle of startup and shutdown.

D. Contents

E. Attachments

Operation Request Instructions	1 page
Operation Request	~~2~~ 4 pages
~~Irradiation Request Sample Evaluation~~	~~1 page~~
~~Exposure Request Sample Evaluation~~	~~1 page~~
Startup Shutdown Checklist	2 pages
Supplemental Heat Exchanger Startup/Shutdown Checklist	1 page
Experiment System Checklist	1 page
CTR Load	1 page
RSR Load	1 page
PNT Load	1 page
3L Load	1 page
BP Load	1 page

F. Equipment, Materials

TRIGA research reactor
Operation Support Systems
Instrument Control and Safety System

G. References, Other Procedures

Docket 50-602 SAR
ANS 15-6, Reg. Guide 2.2

OPER-2	Reactor Startup and Shutdown
OPER-3	Reactor Operation Modes
OPER-4	Operation of Reactor Water Systems
OPER-5	Operation of Air Confinement System
MAIN-4	Area Radiation Monitoring Systems

II. PROCEDURE

Review the requirements of operation, (see section A). Prior to the day's operation perform the startup checks (refer to section B). At the conclusion of the day's operation perform the shutdown checks (refer to section C).

A. Operation Request
Review the operation request form for each experiment.

1. Review each operation and experiment to document a valid experiment approval authorization has been done. The Reactor Supervisor (SRO in his absence) shall approve the operation request.

2. Active operation request forms should be available at the reactor console during all reactor operations for that request.

3. The ~~some~~ operation requests include an evaluation of samples or materials subject to irradiation or exposure and a list of samples on a form such as HP6 Sample Logs. These forms should ~~to~~ be kept with the operation request until the irradiation or exposure is complete.

4. Place operation requests that are no longer active in the appropriate permanent files.

B. Startup Checks
Perform the following actions and record data on the Startup-Shutdown Checklist.

1. Identify experiment classification and personnel requirements.
   a. Perform visual inspection of reactor and experiment areas.
   b. Review the operation request (see section A).
   c. Designate the SRO, RO and experimenter, if any.

2. Check support facility conditions;
Several systems must be operating or operable.
   a. Room 1.104 Evacuation Alarm
   b. Communication - telephone and intercom (1 way)
   Telephone must be operating at the time of startup.
   c. Operate the following radiation monitoring systems:
      i. Air particulate activity monitor.
      ii. Argon-41 gas effluent activity monitor.
      iii. Area radiation monitors (at least 3); The pool area monitor and two additional area monitors must be operating.

3. Set reactor room ventilation conditions, as follows:

   a. Switch room HVAC fan to "Reactor ON" mode (refer to OPER-5). "Reactor ON" mode of the HVAC system should be the normal mode during operation of the reactor.
   b. Start argon purge fan and align source valves ON (refer to OPER-5). This system must be operating if the "Reactor ON" mode of the HVAC is not available and the reactor is operating.

4. Set reactor pool cooling conditions, as required.

   a. Note status of water purification loop. Pool water purification system should be operating (refer to OPER-4).
   b. Operate heat exchanger coolant system pool and chilled water loops for requested reactor power levels greater than 100 kilowatts (refer to OPER-4). Under normal conditions the cooling system should be operating prior to reactor startup. Use primary checklist for first system startup and last shutdown of the day. Use supplemental cooling system checklist for intermittent shutdown and restarts.

5. Check operability of ICS System (requires SRO approval).

   a. Verify ICS operating or initiate ICS bootstrap sequence.
   Refer to Chapter 1 & 2 of ICS Operation Manual.
   b. Verify successful ICS bootstrap sequence.
   Refer to Chapter 2 of ICS Operation Manual.
   c. Perform ICS Pre-start checks sequence.
   Refer to Chapter 2 of ICS Operation Manual.

6. Check operability of support and experiment systems.
(Checklists require SRO approval)

   a. Complete Startup Checklist.
   b. Complete any (applicable) Experiment Systems Checklist.

C. Shutdown Checks
Verify the following tasks are complete. Record on the Startup-Shutdown checklist.

1. Turn Reactor Control Console (RCC) key switch from ON to OFF.
Perform operator log OFF.

2. Remove and secure RCC key (give to SRO/place in locked storage).

3. Secure experiment areas, radiation areas, and radioactive materials.

4. Complete shutdown checklist:
   a. Secure operation of heat exchanger system (refer to OPER-4).
      i. Turn OFF power to pool water and chiller water pumps.
      ii. Close chilled water valves to heat exchanger (2) and the pool water isolation valves (3).

   b. Secure operation of room ventilation exhaust (refer to OPER-5).
      i. Turn off argon purge fan and close source valves.
      ii. Record integral Argon counts and secure Argon CAM.
      iii. Switch room HVAC fan mode from "Reactor ON" to "Reactor OFF".

   c. Perform inspection of reactor and experiment areas.

5. File previous operating records, checklists, and other data-sheets.

Operation Request Instructions

1. Fill in **Operation Request** (all sections except italicized review & approvals)

2. Determine Experiment Approval Status from table below. Only Checked experiments are approved. Other experiments require evaluation and Reactor Committee approval before performance.

3. If an in-core material irradiation, such as B3.X or B4.X is requested, complete & attach ~~an Irradiation Form and~~ an applicable **HP6 Sample Load Data** sheet(s).

4. If an ex-core material exposure, such as A3.1, A3.2, A3.3, or A4.X complete & attach an ~~Exposure Form and~~ applicable **HP6 Sample Load Data** sheet(s).

5. If desired experiment is not approved contact SRO

Experiment Listing (Only ✓ Experiments Approved)

✓	A1.0	ICS Operation	✓	A1.1	ICS Pre-start checks	✓	A1.2	ICS system calibration	✓	A1.3	ICS system changes
	A2.0	Core Reactivity Adjustments		A2.1	Critical mass experiment	✓	A2.2	Fuel element movements	✓	A2.3	Control rod elements
	A3.0	Radiation shield configurations		A3.1	Vertical beam ports	✓	A3.2	Beam ports 1, 2, 4 A. Temporary (HP) B. Routine (no HP)	✓	A3.3	Beam ports 3, 5 A. Temporary (HP) B. Routine (no HP)
	A4.0	Special Projects (Circle A, B, or C)	✓	A4.1	TCNS a. Routine usage no moderator or cooling b. Routine usage with moderator & cooling c. TCNS pulsing d. Use of Deuterated Mesitylene moderator		A4.2	Neutron radiography a. < 250 kilowatt b. > 250 kilowatt (approve as A3.2a)		A4.3	Positron production a. < 8 hour (?) b. > 8 hour (?) (no approval for b)
✓	B1.0	Routine operations -Training		B1.1	Reactivity coefficients -Voids and materials	✓	B1.2	Reactivity coefficients		B1.3	Step reactivity insertion -Positive & negative
✓	B2.0	Routine Operations -Demonstration	✓	B2.1	Power operation	✓	B2.2	Pulse operation		B2.3	Special projects
✓	B3.0	Neutron Activation (Circle A, B, or C)	✓	B3.1	RSR (long-lived) (A) Biological (B) Geological (C) Engineering	✓	B3.2.1 (large) B3.2.2 (small) B3.2.3 (sm Cd)	PNT (short-lived) (A) Biological (B) Geological (C) Engineering	✓	B3.3	Special projects (3L) (A) Biological (B) Geological (C) Engineering
	B4.0	Isotope Production	✓	B4.1	Isotope production RSR (long-lived)		B4.2	Isotope production PTS (short-lived)		B4.3	Special projects (3L)
	B5.0	Reactor core exposures		B5.1	Spare		B5.2	Spare		B5.3	Spare
✓	B6.0	Beam port exposures		B6.1	Beam ports 1, 5		B6.2	Beam ports 2, 4		B6.3	Beam port 3
	Other	(Comments / Description:)									

OPERATION REQUEST

Date: ____/____/____

Requested by: ______________________ Phone: ____________

Project Description:

*Request No.*________

Exp # ______________

*NETL Tracking No.*________

☐ ***Research***

☐ ***Service***

☐ ***Internal***

Mode of Operation: ☐ Manual ☐ Auto ☐ Square

Power Level: _______kW

Time at Power: ______hr

☐ Pulse

Reactivity $'s_______

Number:___________

Date Needed: ____/____/____

Expected Completion: ____/____/_____

☐ **Class A or ☐Class B, Experimenter**______________________________

☐ In-Core (Irradiation): ☐ CT ☐ RSR ☐ 3L (Cd) ☐ 3L (No Cd) ☐ ePNT (Cd) ☐ tPNT (No Cd) ☐ PNT-GA ☐ Other

☐ Ex-Core (Exposure): Beam Port #:___________ Other:______________

☐ New Experiment (Safety Evaluation Required)

☐ **Class C experiment, Operator:**________________ **Experimenter:**______________

☐ Non-Reactor: Room Number____________

☐ Experiment in Reactor: ☐ Pool ☐ Area

NOTES/SPECIAL INSTRUCTIONS:

Material Evaluation

☐ Exposure (ex-core)
☐ Irradiation (in-Core)
☐ No Samples

Request No. ________
Exp.No. ___________
NETL Tracking No. _______
☐ ***Research***
☐ ***Service***
☐ ***Internal***

Sample Description:
____
____

# of Samples ___________ Composition: ☐ solid AND ☐ Biological
Avg. Mass: ________g (solid) ☐ liquid ☐ Geological
________ml (liquid) ☐ gas ☐ Engineering
_________cc (gas) (Act. Calc. Req'd)

HAZARDS: ☐ Toxic ☐ Explosive ☐ Gamma Activity ☐ Other (explain)
☐ Corrosive ☐ Combustible ☐ Neutron fission
☐ Volatile ☐ Beta Activity ☐ Neutron Absorber

Elements: Major Isotopes):____
Trace (Isotopes):____

Encapsulation: (Liquids require double encapsulation. Other hazards require special consideration.)

Type	# of Samples	1539-LG	2/5 dram	2 dram poly	Heat seal	PNT rabbit	RSR rabbit	Poly bag	Aluminum	Quartz	Glass	Other	None
Solid													
Liquid													
Gas													

OPERATIONS USE ONLY:
Time of operations (hrs): ___________
Setup and breakdown time (hrs): _______
Total time (min. 1.0 hour): _____________

LABORATORY USE ONLY:
*Sample Preparation (hrs):*___________
*Irradiation/Counting (hrs):*__________
Analysis (hrs):_______________________
Lab Manager Approval_________________

ADMIN USE ONLY:
*Cost:*_________
*Account #:*__________
*Billed:*_____________
Payment Received: __________

Customer Information
____
____
____
____

Operations Approvals and Review (Reactor Supervisor Signature)
Experiment Type: ☐ *Authorization (New)* ☐ *Special* ☐ *Routine*
Approval: ________________ *Date:*_________ *Review:* ________________ *Date:*_________

1

Material Evaluation (Irradiation)

Standard Ref and/or Flux Monitor.				
	Material:			
	Ref No.			
	Mass (grams)			
Form - ✓ Applicable:	❒ solid	❒ powder	❒ liquid	❒ gas
Sample composition				
	Physical form	❒ Biological	❒ Geological	❒ Engineering (Act. Calc. Req.)
	Chemical form			
	Mass (grams)			
Form - ✓ Applicable:	❒ solid	❒ powder	❒ liquid	❒ gas
Hazards:	❒Toxic ❒Volatile ❒Corrosive ❒Explosive ❒Combustible		❒ ____________ ❒Beta activity ❒Gamma activity ❒Neutron fission ❒Neutron absorber	

Delete

Entire

Sheet

Elements				Isotopes			
Major		Minor		Major		Minor	
1	4	1	4	1	6	1	6
2	5	2	5	2	7	2	7
3	6	3	6				

2

Material Encapsulation:

Reference No's.		to		to	
Enter the number of vials in each category		*Standards*		*Samples*	
	Solid				
	Fine powder				
	Coarse powder				
	Liquid				
	Volatile				
(A) Total number of samples: (note: must match B & C)					
Encapsulation Type:		***Standards***		***Samples***	
Single and double encapsulation		1	2	1	2
MATERIAL:	Polyethylene (low density)				
	Polyethylene (hi density)				
	Aluminum				
	Quartz				
	Glass				
	Plastic bag				
	Other (i.e. boron or cadmium)				
(B) Total: (note: must match A & C)					
Seal Type:		***Standards***		***Samples***	
	Press fit				
	Thermal				
	Other				
(C) Total: (note: must match A & B)					

Operation Approvals (Reactor Supervisor Signature)			
Approval		Mm/dd/yy	

1

Material Evaluation (Exposure)

Delete Entire Sheet

Standard Ref.				
	Material:			
	Ref. No.:			
	Mass (grams)			
Form - ✓ Applicable:	❒ solid	❒ powder	❒ liquid	❒ gas
Sample composition				
	Physical form	❒Biological	❒Geological	❒ Engineering(Act. Calc. Req.)
	Chemical form			
	Mass (grams)			
Form - ✓ Applicable:	❒solid	❒powder	❒liquid	❒gas

Hazards:

❒Toxic
❒Volatile
❒Corrosive
❒Explosive
❒Combustible
❒Beta activity
❒Gamma activity
❒Neutron fission
❒Neutron absorber

Elements				Isotopes			
Major		Minor		Major		Minor	
1	3	1	3	1	3	1	3
2	4	2	4	2	4	2	4

2

Material Encapsulation:

Reference No's. / Enter the number of items in each category		to *Standards*	to *Samples*
	Solid		
	Fine powder		
	Coarse powder		
	Liquid		
	Volatile		
(A) Total number of samples: (note: must match B)			

Encapsulation Type:		*Standards* 1	*Standards* 2	*Samples* 1	*Samples* 2
(No requirements)					
MATERIAL:	Polyethylene (low density)				
	Polyethylene (hi density)				
	Aluminum				
	Quartz				
	Glass				
	Plastic bag				
	Other (i.e. boron or cadmium)				
	None				
(B) Total: (note: must match A)					

3

Comments:

4

Operation Approvals (Reactor Supervisor Signature)			
Approval		Mm/dd/yy	

5

STARTUP CHECKS

By:________________ Run #:____________ Date____/____/____

General Conditions: (✓ Conditions)

(Inspect Areas): Core _____ Pool _____ Room _____

Previous Days Experiments Removed? ☐Yes ☐No, Required To Stay

Experiment areas: (✓or circle condition)

☐Pool BP1 BP2 BP3 BP4 BP5 (S - Secure, Shielding in place)

S/E S/E S/E S/E S/E (E - Secure, Setup for Experiment)

Video Monitors ☐ON Communication System: ☐OK Rm Evacuation Alarm: ☐OK

Radiation Monitors: (✓ Units)

(✓ Unit Operable) (Requires 3 ✓) Area:	BP1	BP2&3	BP4&5	Pool	3.208	Mid
	☐1	☐2	☐3	☐4	☐5	☐6

Area:	3.102	Pipe	RAM	Resin	Hood 3.102	Hood 3.206
	☐a	☐b	☐c	☐d	☐1	☐2

Air activity:	Alert	Alarm	Current	Time ON:
Ar-41 CAM	2000	10000	✓______ cpm	____:____
PART CAM	4000	10000	✓______ cpm	

Portable Unit: __________________ Checks ☐OK

Room Ventilation System: (✓ Set Operating State)

HVAC Mode (exhaust): ☐ON ☐OFF Stack Velocity _______ fpm

Argon Purge (exhaust) ☐ON ☐OFF Stack Velocity _______ fpm

Purge Alignment ON: ☐Pool Surface Purge ☐Beam Port Purge (x On)

Manifold valves: ☐BP3 ☐BP4 ☐BP5 ☐RSR ☐BP1 ☐BP2 (x open)

Room Pressure ✓Ok Δp Level: ☐1 ☐2 ☐3 Rx Δp _______(in. H_2O)

✓Fume - Sort hoods: PNT Port Rm 3.102: ☐ON ☐OFF Glove Box Sort Hood Rm 3.206: ☐ON ☐OFF

Pool Water System: (✓ Set Operating State)

Purification: Pump: ☐ON ☐OFF Conductivity: In:____ Out:____

Pool Isolation: Valves OPEN ☐In ☐Out Align N16 _____%

Heat Exchanger: ✓ Δp Alarm ☐Ok

Hx Chill side: Valves OPEN ☐In ☐Out Pump ☐ ON

Hx Pool side: Valves OPEN ☐In ☐Out Pump ☐ ON Time ON: ____:____

Chill Water			Pool Water		
Flow:	______ gal/min		Flow:	______ %	
	(psig)	Temp.(F)		(psig)	Temp.(F)
Inlet	______	______	Inlet	______ b*	______
Outlet	______ a*	______	Outlet	______	______
Sensor Δp	______ psig		Calc. Δp (a*-b*):	______ psig	

ICS System: (✓ Operating State)

ICS program boot: ☐Yes Successful: ☐Yes ☐No
Boot type: Ctrl-C ☐ Remote Only (DAC)
Power ON ☐ Local and Remote (CSC & DAC)
☐No Previous QNX Boot date: ____/____/____

ICS Auto-Calibration: Pre-start Checks ___ Completion OK

Error condition: ______________________________

Print: Pre-start Diagnostic ☐ Fuel Temp (2) Channel Check ☐OK
Status Window ☐ Power Level (3) Channel Check ☐OK

Key ON Tests (*Initial*): ✓Manual SCRAM Ok ______ *(Use each rod @ least once)*
✓ICS Watchdog Ok DAC ______ CSC ______
✓External SCRAM OK / NA ______: 1 ☐ 2 ☐

Additional Startup Checklists Required Attached (✓ Operating State)
☐Heat Exchanger ☐Experiments ☐None

SRO Approval By:

SHUTDOWN CHECKS By:

ICS System: (Verify and Initial)
All Rods Down ____ Scram Mode ____ CSC Chart Recorder OFF ☐
Remove Key ____ # of SCRAMS ____ Print Acc. Operator Time ☐
ICS System Operator LOG-OFF ____ Video / Intercom OFF ☐
Archive File: ☐No ☐Yes Filename/Diskette #: ______________

Pool Water System: (✓ Set Operating State)

Heat Exchanger: Time OFF: _____:_____
Hx Pool Side: Pump ☐ OFF Valves OPEN: ☐In ☐Out
Hx Chill Side: Pump ☐ OFF Valves SECURE: ☐In ☐Out
Pool Isolation: Valves SECURE ☐In ☐Out ☐N16

Purification:
Pump ☐ON ☐OFF Pool Level: ________ Meters

Room Ventilation System: (✓ Operating State)

HVAC ☐Normal PART CAM ________cpm <u>Ar Time OFF</u> <u>Ar Pump/Chart</u>
Ar Purge ☐SECURE Ar41CAM ____________cnts _____:_____ ☐OFF

Experiment Systems: ☐ Shutdown Complete

Comments:

Attachment Heat Exchanger Startup/Shutdown Checklist

Date ____/____/____

STARTUP CHECKS:

Heat Exchanger Operation: (✓ operating state)

Pool Isolation: Open Valves In__ Out__ Align N16 _____%
Hx Pool Side: Open Valves In__ Out__ Pump ON__
Hx Chill Side: Open Valves In__ Out__ Pump ON__ Time: ____:_____

Check Pool Level __ By:____________________

SHUTDOWN CHECKS:

Pool Isolation: Close Valves In__ Out__ Close N16 ____
Hx Pool Side: Close Valves In__ Out__ Pump OFF__
Hx Chill Side: Close Valves In__ Out__ Pump OFF__ Time: ____:____

Check Pool Level __ By:____________________

STARTUP CHECKS:

Heat Exchanger Operation: (✓ operating state)

Pool Isolation: Open Valves In__ Out__ Align N16 _____%
Hx Pool Side: Open Valves In__ Out__ Pump ON__
Hx Chill Side: Open Valves In__ Out__ Pump ON__ Time: ____:_____

Check Pool Level __ By:____________________

SHUTDOWN CHECKS:

Pool Isolation: Close Valves In__ Out__ Close N16 ____
Hx Pool Side: Close Valves In__ Out__ Pump OFF__
Hx Chill Side: Close Valves In__ Out__ Pump OFF__ Time: ____:____
Check Pool Level __ By:____________________

STARTUP CHECKS:

Heat Exchanger Operation: (✓ operating state)

Pool Isolation: Open Valves In__ Out__ Align N16 _____%
Hx Pool Side: Open Valves In__ Out__ Pump ON__
Hx Chill Side: Open Valves In__ Out__ Pump ON__ Time: ____:_____
Check Pool Level __ By:____________________

SHUTDOWN CHECKS:

Pool Isolation: Close Valves In__ Out__ Close N16 ____
Hx Pool Side: Close Valves In__ Out__ Pump OFF__
Hx Chill Side: Close Valves In__ Out__ Pump OFF__ Time: ____:____

Check Pool Level __ By:____________________

EXPERIMENT STARTUP/SHUTDOWN CHECKS

By ____________________ Date: ____/____/____

✓☐ **Experiment Requested** Run #

☐ **CENTER TUBE EXPERIMENT (CT)**

S/U: Center Tube in A-1 ☐ Sample Assembly Loaded in Tube ☐
S/D: Status of Sample Assembly: Decay In Core ☐
Remove From Core ☐ In Pool? ☐Yes ☐No

☐ **ROTARY SPECIMEN RACK EXPERIMENT (RSR)**

S/U: RSR Ar Purge Valve ☐OFF
RSR Rotation Direction ☐CW ☐CCW
RSR Drive Motor ☐ON
S/D: RSR Drive Motor ☐OFF

☐ **PNEUMATIC TRANSFER SYSTEM EXPERIMENT (PNT)**

S/U: PNT Terminal in Core: ☐ GA Small Tube: ☐ NCd ☐ Cd
CO_2 Gas Bottle Valve: ☐ OPEN
CO_2 Pressure ________ psig
PNT Control System Key ☐ ON
S/D: PNT Control System Key ☐ OFF
CO_2 Gas Bottle Valve ☐ SECURE
CO_2 Pressure ________ psig
Status of PNT Terminal: In-core ☐ Yes ☐ No

☐ **THREE ELEMENT CUTOUT IRRADIATOR EXPERIMENT (3L)**

S/U: ☐ Verify or ☐ Reconfigure Fuel & Graphite Set For 3L Operation
☐ Review Calibration Data: ☐ Core Reactivity, ☐ Rod Worth
If ☐Required ☐N/A Reconfigure fuel & graphite for 3L Operation,
Full Power SCRAM's Reset ☐ N/A To ______% ☐ NPP ☐ NP ☐ NM
Load 3L Canister Into Core: NCd ☐ Cd ☐
Install Rotation Mechanism IN PLACE/ON ☐
S/D: Remove Rotation Mechanism SECURE/OFF ☐
3L Canister Removed From Core: ☐No ☐Yes (Do not remove from pool)
If ☐Required ☐N/A Reconfigure fuel & graphite for Normal Operation,
Full Power SCRAM's Reset ☐ N/A To **Normal**: ☐ NPP ☐ NP ☐ NM

☐ **BEAM PORT EXPERIMENT AREAS (BP)**

Check Beam Shutter, Access Gate, and Beam Stop

BEAM Exp.	PORT #	Work in Area	SHUTTER OK/GATE SECURE	BEAM STOPS IN PLACE
☐	#1	☐NO, ☐YES	At S/U ☐, At S/D ☐	At S/U ☐, At S/D ☐
☐	#2	☐NO, ☐YES	At S/U ☐, At S/D ☐	At S/U ☐, At S/D ☐
☐	#3	☐NO, ☐YES	At S/U ☐, At S/D ☐	At S/U ☐, At S/D ☐
☐	#4	☐NO, ☐YES	At S/U ☐, At S/D ☐	At S/U ☐, At S/D ☐
☐	#5	☐NO, ☐YES	At S/U ☐, At S/D ☐	At S/U ☐, At S/D ☐

☐ **OTHER** ______________________________ (____)

S/U: Complete ☐
S/D: Complete ☐

Comments:

Project Description/#________________ Operation Request #_________

CENTRAL THIMBLE FACILITY:

Install:

Install CT experiment tube assembly: ❐Yes ❐Already In
(Record on irradiation log sheet)
Apply air source to expel tube water ❐Yes ❐No
(If Yes a radiation beam will be present at the pool surface)

Remove:

Remove CT experiment tube assembly: ❐Yes ❐Leave In
(Record on irradiation log sheet)

CENTRAL THIMBLE SAMPLE HOLDER:

Prepare RWP for load and unload activities ❐Yes ❐No
Request (require) HP assistance: ❐Yes ❐No

Sample Loading:

Loading By: ____________________
Inspect Sample Holder: ❐O-Ring & Thread Lube OK
Assemble sample assembly: ❐
Record Sample Log HP6A ❐Yes
Install sample assembly: ❐ Position__________

Irradiation Times:

Time At Power _____:______ at _______ Kw
Shutdown Time _____:______

Sample Unloading:

Unload By: ____________________
Remove sample assembly: ❐
Assembly Dose rate (HP6A) __________@__________
Sample Dose rate (HP6A) __________@__________
Un-package sample assembly: ❐
Store and Shield Sample Holder ❐

RECORDS:

This list should be used for each activity and attached to the operation request.

Project ________________ Operation Request #__________________

RSR LOADING:

Use RSR RWP to load and unload samples.
For each sample batch:

Total number of samples: ____________
RSR Capsule type: Polyethylene _____
Aluminum _____
Other _____
Total number of RSR capsules (load): ____________

Load:

- Test sample loading device. ☐
- Remove loading tube cap and plug ☐
- Ventilate RSR (10 minute purge) ☐
- Load samples Total: ________________ ☐
- Record Load data on HP6A ☐
- Reinstall loading tube plug and cap ☐
- Secure RSR Purge, valve OFF ☐

Irradiation Times:

Time At Power ______:_______ at ________ Kw
Shutdown Time ______:_______

Unload:

- Switch rotation to OFF: ☐
- Ventilate RSR (10 minute purge) ☐
- Remove loading tube cap and plug ☐
- Unload samples ☐
- Record Unload data on HP6A ☐
- Secure RSR Purge, valve OFF ☐
- Reinstall loading tube plug and cap ☐

QA check:

Total number of RSR capsules: (unload) ____________

RECORDS:

These steps should be followed for each activity but a completed copy of this list is not required for each run. If a copy is completed it should be attached to the operation request.

PNT:

For each sample do the following:

Load:

Listen for sample actuation announcement
Acknowledge sample actuation announcement
Observe sample irradiation time
Observe sample entry to pool (red on)
Record PNT sample #, exposure time, and ρ in log

Unload:

Observe sample return from pool (red off)
Observe sample return to port (red green)
Observe PNT port dose rate

Records:

No copy of this list required to be filed for each run

Sample logs and dose rates for tracking are maintained in the PNT Logbook in 3.102

Project Description/#__________________ **Operation Request #**__________

3L IRRADIATOR FACILITY:

Install:

Install 3L Spider Spacer in core: ☐Yes ☐Already In
(Record on irradiation log sheet)
3L Canister Loaded into core at: _____:_____ on ___/___/___

Remove:

Remove 3L Spider Spacer from core:: ☐Yes ☐Leave In
(Record on irradiation log sheet)
3L Canister Removed from core at: _____:_____ on ___/___/___

Dose rate at top of canister (cap ON) __________mR/Hr @__________

3L IRRADIATOR SAMPLE LOADING:

Use RWP for load and unload activities ☐Yes
Request (require) HP assistance: ☐Yes ☐No
3L Canister Used ☐Cd ☐Non-Cd

Sample Loading:

Loading By: ____________________

Inspect Sample Holder, ✓OK ☐O-Ring ☐Thread Lube
Record Sample Log HP6A ☐Yes ☐Other form

Install sample assembly: ☐ Position__________
Cap Tightened: ☐OK
2 Minute CO_2 Purge: ☐OK
Purge Valve Closed: ☐OK
Push down on relief valve top: ☐OK

Irradiation Times:

Time At Power ______:______ at ________ Kw
Shutdown Time ______:______

Sample Unloading:

Unload By: ____________________
Purge Valve Open: ☐
Can Dose Rate (Top Cap OFF) __________mR/hr @__________
Sample Package Dose Rate __________mR/hr @__________
Sample Dose rate (HP6A) __________mR/hr @__________

3L Canister Storage:

Storage: ☐Loading Dock with ☐Shielding in Place
☐In Pool with ☐Lid ON, ☐Valves Closed

RECORDS:

This list should be used for each activity and attached to the operation request.

Beam Port Exposure:

SAMPLE LOADING:
- Follow Requirements of applicable Beam Port RWP
- Record Sample Data on HP6A or approved RWP Sample Log Sheet

SAMPLE UNLOADING:
- Follow Requirements of applicable Beam Port RWP
- Record Sample Data on HP6A or approved RWP Sample Log Sheet
- Follow RWP requirements for removal of sample from BP area

RECORDS:
- All records shall be recorded in the RWP as required by the RWP

PROCEDURE

OPER-2

Reactor Startup and Shutdown

Version
1.00

Approvals:

Randall Charbeneau 4 / 4 / 02
Facility Director, NETL Date

Larry S Baker 4 / 4 / 02
Chairperson, Nuclear Reactor Committee Date

Number of Pages: 9
Number of Words: 1924
Number of Characters: 9780

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J. J. PICKLE RESEARCH CAMPUS
THE UNIVERSITY OF TEXAS AT AUSTIN

I. INTRODUCTION

A. Purpose

This procedure specifies actions to be done for startup and shutdown of the reactor.

B. Description

Actions for reactor startup and shutdown require certain specific conditions. Prior to startup the correct operating conditions are checked by the performance of a Startup Checklist. A Pre-start Check tests ICS functions. Subsequent to reactor shutdown a Shutdown Checklist documents the condition of important systems.

Guidance for startup of the reactor is available in the console operator's manual. Features of this procedure provide requirements and guidance. Some deviations will occur depending on the experience of the operator and the operation requests. An example of the guidance is the presentation of a typical startup sequence.

All actions of this procedure require the direct supervision of a reactor operator with a valid license. Abnormal shutdown or SCRAMS require a SRO approval prior to restart of the reactor to a critical condition.

C. Schedule

Apply this procedure each day the reactor is taken through an operation cycle of startup and shutdown.

D. Contents

E. Attachments

Console Operation Log	2 pages
SCRAM Log Record	2 pages
Reactivity Configurations	1 page

F. Equipment, Materials

Instrumentation Control and Safety System (ICS)

G. References, Other Procedures

Control Console Operators Manual

OPER-1	Startup – Shutdown Checks
OPER-3	Reactor Operation Modes
OPER-4	Operation of Reactor Water Systems
OPER-5	Operation of Air Confinement System
OPER-6	Reactor Bay Systems
SURV-7	Pulse Characteristic Comparison

II. PROCEDURE

A. Reactor Startup

1. Review Operation Procedures for the mode of operation. Refer to ICS Operation Manual for specifics of the ICS System operation. Console log sheets for each operation will record operator comments regarding important system conditions. These log sheets in the console logbook will supplement computer printouts from the ICS system. Other datasheets such as the operation request, startup-shutdown checklist, and sample irradiation and exposure log complete the documentation for a typical reactor run.

2. Review completion of the startup checklist and note the recorded conditions. If a problem occurs at the time of reactor startup or shutdown use the comment section of the startup-shutdown checklist to document unusual conditions. Record any abnormal CSC or NM1000 messages on the appropriate OPER-6 attachments. If any question exists regarding acceptability of the condition consult the supervisory senior reactor operator.

3. Perform operator log-on function. Turn MAGNET POWER key switch from OFF to RSET to ON. Set mode to steady-state Manual. Verify the following scrams are functional use each rod at least once):

   a. Check Manual SCRAM, if this is the first startup.
   b. Check DAC Watchdog SCRAM if this is the first startup;
      Use SCRAM Test button for test of DAC WD relay.
   c. Check CSC Watchdog SCRAM if this is the first startup;
      Use SCRAM Test button for test of CSC WD relay.
   d. Check External SCRAM, if an experiment shutdown is applicable.
   e. Verify the cause of any previous SCRAM condition.

4. Verify minimum staffing requirements are met:

   a. SRO, with second person in building, or
      RO (SRO present or available in the building), or
      RO (SRO on call within 30 minutes), second person in building.
   b. Verify SRO present for initial startup of the day.
   c. Verify SRO present for any significant power changes (> 100 kw).
   d. Verify SRO approval of startup for an experiment.
   e. Verify SRO approval of restart from a SCRAM condition.

5. Determine mode of operation (refer to OPER3).
   Review typical startup sequence (refer to Section II.B)

6. Refer to Section II.C of procedure for termination of operation.

B. Typical Startup Sequences
Rod positions range from 0 to 960 (0 to 15 inches).
The percent rod position is about one tenth of the position indicator value.
To enter desired operation mode Refer to OPER-3.

1. Manual Mode:

   a. Withdraw transient rod to approximately 50% withdrawn position (similar to critical in recent previous startup with similar experiments and burn-up).
   b. Withdraw regulating rod to approximately 50% withdrawn position (similar to critical in recent previous startup with similar experiments and burn-up).
   c. Withdraw each shim rod in steps of 50 units or less.
   d. Check for positive period of about 20 seconds.
   e. Do not exceed a 10-second period.
   f. Adjust shim rods to maintain period.
   g. Move shim rods to stabilize power level.
   h. Move all rods to a bank configuration for "best" operation.

2. Auto Mode:

   a. Set demand power switches to desired power level.
   b. Withdraw transient rod to approximately 50% withdrawn position (similar to critical in recent previous startup with similar experiments and burn-up).
   c. Withdraw each shim rod to approximately 50% withdrawn position (similar to critical in recent previous startup with similar experiments and burn-up).
   d. Press Auto Mode switch.
   e. Minimize transient, shim1 or shim2 rod UP motion, if Reg motion is:
      i. UP and power is not within 10% of demand value,or
      ii. DN and power is not within 10% of demand value.
   f. Move rods to a bank configuration for "best" operation.

3. Square Wave Mode:

   a. Sequences for square wave mode operation use transient rod (TR) positions that create a positive reactivity insertion as the rod moves from a set position x to y% withdrawn. The motion of the TR may be chosen to end in the 100% withdrawn position or a position typical for a banked rod condition at the desired steady state power, therefore the transient rod will move from 0% to the anticipated banked position or a maximum of 100%. At the time of actuation of the fire button, the core will be sub-critical by an amount equal to the anticipated banked TR worth minus the expected value of the square wave. Calculate TR position for reactivity insertion:

      i. Choose desired final position of TR, banked position or full out
      ii. From TR rod worth curve calculate total reactivity ρ_f in the TR at desired position. ρ_f = ____________ cents at __________ rod position
      iii. Estimate required square wave insertion amount required for desired final power level. Demand power must be reached within 10 seconds for a successful Square Wave. Use $P = P_o * e^{-t/T}$ (P = final power, Po=initial power, and t$\leq$ 10 seconds) to determine required period to reach the desired power. Then look up the reactivity insertion ρ_{sq} from an in-hour table (the corresponding reactivity must be less than \$1.00): ρ_{sq} = ____________ cents
      iv. Subtract the reactivity obtained in step 3.a.iii from that in step 3.a.ii and look up the resulting reactivity ρ_{50} and TR position in the TR worth table. This is the pre-square wave position the TR must be placed in for the initial 50 watt critical condition.
      $\rho_{50} = \rho_f - \rho_{sq}$ =______________ at ________rod position.
      v. Compare estimated values with previous insertions if possible.

   b. Obtain steady state (SS) power, 50 watts is typical (< 1 kW):

      i. Withdraw transient rod to the position calculated in step 3.a.iv above.
      ii. Withdraw reg rod to approximate 50% withdrawn position.
      iii. Withdraw each shim rod in steps of 50 units or less.
      iv. Check for positive period of about 20 seconds.
      v. Do not exceed 10-second period.
      vi. Adjust shim rods to maintain period.
      viii. Move shim and reg rods to stabilize power in banked position.

c. Initiate Square Wave:

i. Insert transient rod to 0% position.
ii. Check rod at low limit, turn air pressure off.
iii. Withdraw drive cylinder to position determined in step 3.a.ii above.
iv. Check power < 1kw, DPM < + 1.
v. Set Demand Power thumbwheels to desired SS power
vi. Press Square Wave switch.
vii. Press Fire switch.
viii. System will switch to AUTO mode if demand power reached in 10 seconds. Adjust rods to banked position for "best" performance.
ix. System will switch to MANUAL mode if demand power is not reached in 10 seconds.

4. Pulse Mode:

a. Sequences for pulse mode operation use TR positions that create a positive reactivity insertion as the rod moves from a set position x to 100% withdrawn. The motion of the control rod, however, will move the full range from 0% to 100%. At the time of actuation of the fire button, the core will be sub-critical by an amount equal to the total control rod worth minus the expected value of the pulse. Calculate TR position for reactivity insertion:

i. From TR rod worth curve calculate total reactivity ρ_f in the TR at 100 % withdrawn position.
ρ_f = ____________ cents at __________ rod position
ii. Determine the required pulse insertion amount (≤ \$3.00)
ρ_p = ____________ cents
iii. Subtract the reactivity obtained in step 4.a.ii from that in step 4.a.i and look up the resulting reactivity ρ and TR position in the TR worth table. This is the pre-pulse position the TR must be placed in for the initial 50 watt critical condition.
$\rho = \rho_f - \rho_p$= ______________ at ________rod position.
iv. Compare the estimated values with previous insertions, refer to SURV-7. Due to the infrequent operation in pulse mode, verify the SURV-7 annual comparison pulse is current.

b. Obtain steady state power; 50 watts is typical (< 1 kW):

i. Withdraw transient rod to the position calculated in step 4.a.iii above.
ii. Withdraw reg rod to approximate 50% withdrawn position
iii. Withdraw each shim rod in steps of 50 units or less.
iv. Check for positive period of about 20 seconds.
v. Do not exceed 10-second period.
vi. Adjust shim rods to maintain period.
vii. Move shim rods to stabilize power.
viii. Move reg and shim rods to banked position to maintain power level

c. Initiate Pulse Mode:

i. Insert TR to 0% position.
ii. Check rod at low limit, turn air pressure off.
iii. Withdraw TR drive cylinder to 100% position.
iv. Check power < 1kw, DPM < + 1.
v. Press Pulse Mode switch.
vi. Enter record information for pulse data.
vi. Verify mode is Pulse-Ready
vii. Press Fire switch.
viii. System will switch to SCRAM mode at pulse conclusion to display the pulse.
ix. Refer to OPER-3 to resume operation following pulse display.

C. Reactor Shutdown

1. Normal shutdown:

a. Switch to Manual mode and insert Reg rod to the 0% withdrawn position. (One may insert all four rods simultaneously.)
b. Insert <u>each</u> Shim rod to the 0% withdrawn position.
c. Insert Transient rod to the 0% withdrawn position.
d. Assure that all rod drives and control rods are in the down position.
e. Turn MAGNET POWER key switch from ON to OFF.
Perform operator log-off function to set mode from steady-state to scram.

2. Abnormal shutdown:

   a. Press SCRAM button for exit from any operation mode. The SCRAM is an immediate shutdown for response to abnormal conditions or a severe emergency.
   b. Record all abnormal shutdowns in the SCRAM log. These do not include manual scrams by the operator that are for non-emergency conditions.
   c. Abnormal shutdowns are of the following types:
      i. *Manual SCRAM (MS)*
         Operator activation of SCRAM button,
         Operator activation of magnet key switch.
      ii. *Limiting Safety System Setting (LSSS)*
         Fuel temperature (#1, #2),
         Percent power (#1, #2),
         Linear power (NM1000).
      iii. *ICS Operable (ICSO)*
         HV (#1, #2, NM1000),
         Pool Level (1 of 2),
         External (1 of 2 if in use),
         WD (CSC, DAC),
         Other (program conditions).

1

UT-TRIGA ICS Console Operation Log		
UT-TRIGA The University of Texas *NETL Pickle Research Campus*		/ /
		Run No. ____

2

Startup Checks			
Complete checks	☐ S/U OK		
Operation:	☐ Non pulse		☐ Pulse
✓ All that apply	☐ Manual	☐ Auto	☐ Square wave

	Req. Nos.:		Experiment Nos.:		Exp. facilities	
	1	4				
	2	5				
	3	6				
	Samples:		In-core:		In-beam:	
			Yes ☐	No ☐	Yes ☐	No ☐

3

Reactivity:				
	List Exp. facilities			Limit
Static		Subtotal		< $2.50
Moveable		Subtotal		< $1.00
		Total ($'s)		< $3.00
SCRAM Recovery Approval		Type MS, LSSS, ICSO (refer to OPER3)		
(SRO sign in written log at event)		# _____ recorded in Scram Log		

4

Shutdown Checks							
	Operation Time		Energy (burnup)		Number of pulses done:	S/D Checks Complete:
	______ Hours		______ KWHRs		______	☐
Samples:	In core:		In beam:		Location:	
	Yes ☐	No ☐	Yes ☐	No ☐	____________	
Comments:						

5

	Operator	SSRO✓	Time In/Out			Operator	SSRO✓	Time In/Out	
1		☐			09		☐		
2		☐			10		☐		
3		☐			11		☐		
4		☐			12		☐		
5		☐				Second Person (After Hrs)		Time In/Out	
6		☐			1				
7		☐			2				
8		☐			3				

6

ICS Console Operation Log Sheet (Continuation Page)	Run No. ____
Time	**Comments**
	Page ___ of ___

1

Scram Log
2 **for General System**

A. SCRAM MANUAL	A. SCRAM EXTRN 1
B. SCRAM POOL LO	B. SCRAM EXTRN 2
A. SCRAM FT 1	A. SCRAM NO OPERATOR
B. SCRAM FT 2	B. SCRAM KEY OFF
C. SCRAM %PWR1(NPP)	C. SCRAM NPP1000 HV
D. SCRAM %PWR2 (NP)	D. SCRAM NP 1000 HV
E. SCRAM NM% or HV	E. SCRAM NM 1000 HV
F. SCRAM PPWR HI	F.

Date	Time					Comments:
Totals:						

3 Record each scram with the type designation. Note the cause or other information as a comment.
4

Scram Log
for Digital Systems

A. SCRAM NM COMM
B. SCRAM NM DATA

A. CSC WD FLT
B. CSC WD TO
C. DAC WD FLT
D. DAC WD TO

A. SCRAM IC NET
B. SCRAM CSC DIS064
C. SCRAM DAC DIS064S
D. SCRAM DBASE

A. SCRAM DOM32
B. SCRAM AIO16 1
C. SCRAM AIO16 2

Date	Time					Comments:
Totals:						

Record each scram with the type designation. Note the cause or other information as a comment.

1

Startup Reactivity Calculation - Core Reactivity Record

2

Reactor Core k_{excess} condition:

Today mm / dd / yy				Burnup Mw-days				Reference mm / dd / yy				Reactivity Cents			
			Today			.					Reference				
			Reference –			.					Today	–			
						.					(A)				

3

Burnup from reference core	x 2.5	cents/Mw-day		–			
			Subtotal less BU:				

4

Presence of fission products (Xe):

Previous Reference	mm / dd / yy			Cold-clean critical core (no Xe):	–				
				Measurement: (a)					
Current Estimate				Calculation: (b)					
				Xe (a or b):	–				
				Subtotal less Xe:					

5

Experiment facilities:

Reactor In-Core Facilities					Beam Ports & Ex-Core Facilities			
	Fixed		Moveable (samples)	Comment		Fixed	Moveable (samples)	Comment
RSR			<20		BP-1	< ± 5	0	TIPS
CT	<±5		?	A-1	BP-2	< ± 5	0	NDP
GA-PNT	<±5		<±5	G-34	BP-3	< ± 5	< ± 5	TCNS Hot-Cool
PNT NC Cd	<±5	-30	<±5	G-34	BP-4	–		
3L – NC Cd	?	≈ -95		D17,E22,3	BP-5	< ± 5	0	NRad
Other					Other			
Subtotal change:								

6

Summary:	(All measurement in cents)	Total core reactivity:	
ECP	50 watts	Initial:	Date:

7

8

Reactivity Configuration | Stamp(Original-Red, Copy-Blue)

Date of Change | | | | |

NETL Director Approval | | | | |

COPY

PROCEDURE

OPER-3

Reactor Operation Modes

Version
1.00

Approvals:

Randall Charbeneau 4 / 4 / 02
Facility Director, NETL Date

Kenneth S Ball 4 / 4 / 02
Chairperson, Nuclear Reactor Committee Date

Number of Pages: 5
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NUCLEAR ENGINEERING TEACHING LABORATORY
J. J. PICKLE RESEARCH CAMPUS
THE UNIVERSITY OF TEXAS AT AUSTIN

I. INTRODUCTION

A. Purpose

This procedure describes the different operation modes of the reactor.

B. Description

The ICS system operation modes control the program logic interlock requirements and set the operation conditions for the reactor control rod drives. There are four operation mode conditions set by the control panel switches. Mode descriptions in the display annunciation box may differ from the switch labels to qualify conditions within a mode. The SCRAM mode is a non-operation mode (no mode light on).

All actions of this procedure require the direct supervision of a reactor operator with a valid license.

C. Schedule

Apply this procedure each day the reactor is taken through an operation cycle of startup and shutdown.

D. Contents

E. Attachments

None

F. Equipment, Materials

Instrumentation Control and Safety System (ICS)

G. References, Other Procedures

Control Console Operator's Manual

OPER-1	Startup – Shutdown Checks
OPER-2	Reactor Startup and Shutdown
OPER-4	Operation of Reactor Water Systems
OPER-5	Operation of Air Confinement System

II. PROCEDURE

A. Manual Mode

1. Review Manual Mode operation.
   Refer to Chapter 4 of ICS Control Console Operator's Manual.

2. Determine desired reactor power level.
   Estimate control positions at steady-state power. Refer to:
   a. Typical rod positions (adjusted for any experiments and burn-up), or
   b. Previous operation history (similar core conditions), or
   c. Rod worth curves (relative to shut down margin).

3. Move control rods to achieve power level or go to step 5 to enter another mode.
   Typical startup sequences are in OPER-2.
   a. A printout of the *Status Window* should be done:
      i. After each power change,
      ii. At ≈ 30 minutes intervals while at a steady-state power level. (Recommendation: Do a status printout on each ½ hour.)
   b. No Status Window print is necessary for some special cases. For example an operation such as control rod calibrations that require "continual" power changes do not require a printout for each power change.
   c. Initiate a linear chart recording (normally done but not required):
      i. Record Start of Run (use "SOR") and time on chart;
         Record the Date and Run # on the chart;
         Switch recorder power ON.
         Verify recorder chart speed set to slow (100 sec/cm).
      ii. Record End of Run (use "EOR") and time on chart.
         Switch the recorder power OFF and cover pen tip.

4. Monitor operation of system.
   Monitor power level, control rod positions, and other data.
   Print data logs at recommended intervals.

5. Refer to Operation Procedure, "MODE", for exit to alternate operation mode.

a.	Auto Mode:	Procedure; Section B	ICS Manual; Chapter 5
b.	Square Wave:	Procedure; Section C	ICS Manual; Chapter 5
c.	Pulse (Ready):	Procedure; Section D	ICS Manual; Chapter 6

6. Press SCRAM button for exit from Manual Mode to Scram Mode.

B. Auto Mode

1. Review Auto Mode operation and typical Auto mode in OPER-2.
   Refer to Chapter 5 of ICS Control Console Operator's Manual.

2. Set Demand Power switches to desired power level.

3. Press AUTO mode switch
   Verify AUTO light illuminates.
   Verify console system mode is <u>Auto</u>.

4. Monitor system operation.
   Monitor power level, and control rod positions at periodic intervals.
   Record data logs at recommended intervals for Manual Mode.

5. Press MANUAL switch for exit from Auto mode to Manual Mode.

6. Press SCRAM button for exit from Auto mode.

C. Square Wave Mode

1. Review Square Wave Mode operation and typical square wave in OPER-2.
   Refer to Chapter 5 of ICS Control Console Operator's Manual.
   Determine Transient reactivity and final Transient Rod (TR) position.

2. Start up to the desired Steady State pre-square wave power (<1Kw).

3. Press SQUARE WAVE mode switch.
   Verify SQUARE WAVE light illuminates.
   Verify console system mode is <u>Square Wave Ready</u>.

4. Set Demand Power switches to desired power level.
   Adjust TR cylinder to Square Wave position.

5. Press FIRE button.
   Verify system mode is <u>Square Wave Ramp-up</u>.
   Exit Ramp-up mode is to Auto mode.
   If demand power is not reached in 10 seconds,
   control system automatically exits to Manual mode.

D. Pulse Mode

1. Review Pulse Mode operation and typical Pulse operation in OPER2.
Refer to Chapter 6 of ICS Control Console Operator's Manual.

2. Determine Transient reactivity and pre-pulse TR position.
Start up to desired Steady State pre-pulse power (<1Kw)
Adjust Transient rod drive cylinder to pre-pulse position.
Verify power level is less than 1 kilowatt.

3. Drop Transient rod by actuation of AIR button.
Reposition TR cylinder to 100% withdrawn position.

4. Press PULSE mode switch.
Verify PULSE light illuminates.
Verify system mode is <u>Pulse Ready</u>.

5. Determine alphanumeric pulse description and enter string.

Recommendation:	Routine:	MMYY-###-cents
	Comparison:	CMPR-cents-MM/YY

6. Press FIRE button.
Verify system mode is <u>Pulse</u>.
Pulse mode exit is to Scram Mode for Pulse Display.

7. Refer to Operation mode, Manual procedure.
Do the following SCRAM circuit tests to establish original circuit conditions (risk of failure).
Verify pulse bypass functions are reset to non-pulse conditions:

	<u>Check (apply test signal)</u>	<u>Reason for test</u>
a.	NP1000 for Hi Power scram	Test reset of bypass relay.
b.	NPP1000 for Hi Power scram	Test reset of the gain relay.
c.	NM1000 power signal output	Observe decay of pulse power.

A Mode 7 at the NM1000 will cause a Mode 4 (CMB low) for 10 seconds, then a Mode 0 (Normal) for 10 seconds. Observe signal response to verify system returns from Mode 4 to Mode 0.

Note: Reset of the NM1000 bypass relay does not apply at this time; the relay is not currently in the scram loop. In the square wave and pulse mode the NM1000 will momentarily bypass the Hi-power SCRAM (Mode 6 or 7) function.

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USNRC TRAINING PROGRAM – 01/2011	
EXERCISE 2: Reactor Startup	Page 1 of 3

1. PURPOSE AND DISCUSSION

The overall purpose of the USNRC training program is to provide experience to develop an intuitive grasp of reactor behavior. This exercise will introduce the participants to reactor controls as a basis for focusing on reactor response.

2. EXERCISE OBJECTIVES

Following completion of this exercise, participants should be able to:

2.1 Given a control rod position change,

A. Recognize the prompt jump/drop

B. Determine whether the reactor is subcritical, critical, or supercritical

2.2 Given a desired control rod position from one subcritical position at an observed power level to the next, calculate the expected final steady state power level

2.3 Given two graphs of reactor response to arbitrary positive changes in reactivity while subcritical, determine which reactivity change was greater

2.3 Given a control rod position increase while critical,

A. Measure the resulting positive period

B. Determine the reactivity associated with the change

2.4 Recognize the effect of temperature changes on reactivity

3. PREREQUISITES, LIMITS & PRECAUTIONS

3.1 Preoperational Checks complete

3.2 See Attachment 1, Relevant Technical Specifications

4. BACKGROUND/REFERENCES

4.1 Basic Nuclear Principles 6 *Reactor Modes*

4.2 Basic Nuclear Principles 7 *Four Factor Formula*

4.3 Basic Nuclear Principles 8 *Neutron Non-Leakage*

4.4 Basic Nuclear Principles 9 *Reactor Kinetics*

4.5 UT Procedure OPER 1, Startup and Shutdown Checks

Comment [#1]: Review BNP6

Comment [#2]: Review BNP7

Comment [#3]: Review BNP8

Comment [#4]: Review BNP9

Comment [#5]: Review OPER1

4.6 UT Procedure OPER 2, Reactor Startup and shutdown

Comment [#6]: Review OPER2

4.7 UT Procedure OPER 3, Reactor operation Modes

Comment [#7]: Review OPER3

4.8 Attachment 2.1, Relevant Technical Specifications

Comment [#8]: Review ATT2.1

4.9 Attachment 2.2, Exercise 2 Review

5. INSTRUCTIONS

5.1 OBSERVE and RECORD

Comment [#9]: Discuss how observed items may have changed since previous operation

A. Control rod position from most recent startup

B. Reactivity addition required to establish criticality from the most recent startup

C. Current reactor power level with all rods fully inserted

5.2 CALCULATE control rod position to add ½ of the reactivity required for criticality

5.3 ESTABLISH control rod configuration to add ½ of the reactivity required for criticality

AND

OBSERVE reactor power level chart recorder trace as each control rod is moved from full in to the ½ critical position, including

Comment [#10]: Identify traces, emphasizing which is linear and which is log. Discuss how exponential behavior is recorded on each trace.

- Amount of reactivity added
- Relative magnitude of response
- Time required to establish new equilibrium

5.4 POSITION control rods to establish critical 50 Watt condition

NOTE: Automatic mode may be used to establish equilibrium.

5.5 POSITION (in manual control) regulating rod 30 units greater than the critical configuration and OBSERVE chart recorder trace

Comment [#11]: Ask students to explain trace; discuss relationship between doubling time and period, period and reacitivty

5.6 IDENTIFY prompt jump on the chart recorder

5.7 MEASURE the time difference required to move from an arbitrary power level and twice the power level

5.8 CALCULATE the reactivity required to achieve the period

5.9 COMPARE the calculated response to the reactivity addition as originally determined from control rod position

5.10 REPEAT Steps 5.7 and 5.8 UNTIL period begins to change

5.11 OBSERVE fuel temperature and period

Comment [#12]: Direct students attention to digital temperature readings, period meter, and power level

5.12 WHEN reactor power is stable,

RECORD fuel temperature

CALCULATE fuel temperature coefficient of reactivity

5.13 SHUTDOWN reactor

6. REINFORCEMENT & REVIEW

6.1 Attachment 2.2, Exercise 2 Review

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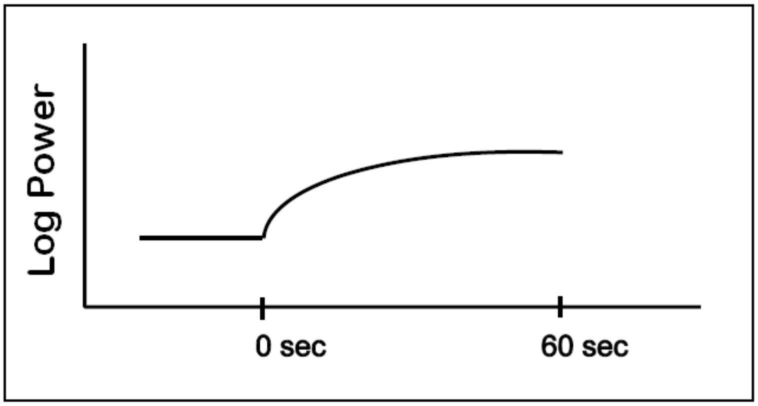
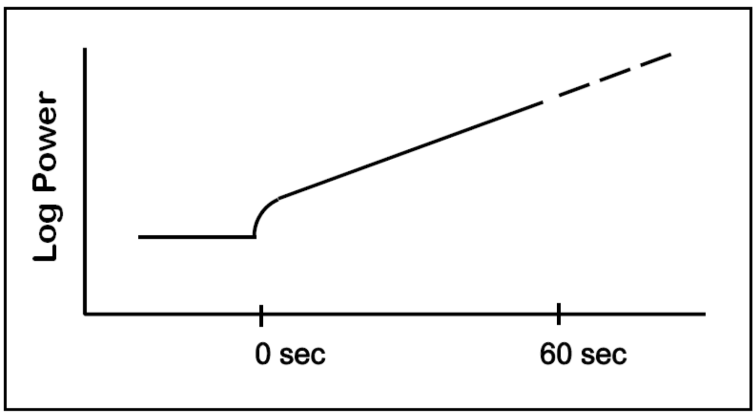
ATTACHMENT 2.1: Relevant Technical Specifications

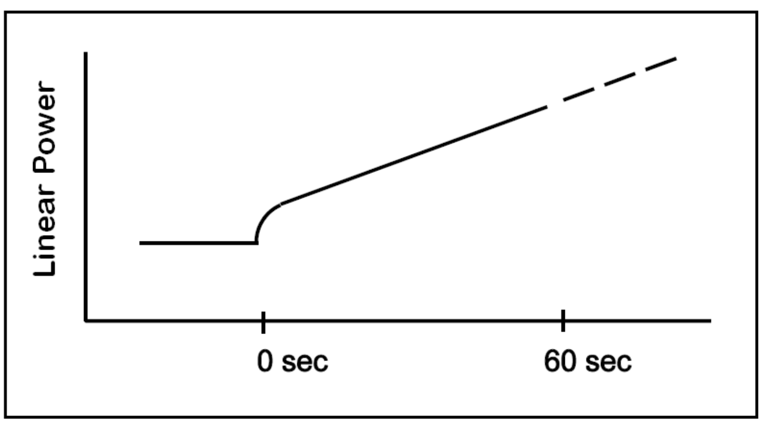
SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	TEST	Procedure/record
2.2.1	Fuel temp LSSS	<550°C				
2.2.2	Power level	1100 kW				
3.1.1	Excess reactivity	4.9% Δk/k	4.1.1	Annually or post core/control rod changes		SURV-3 Excess Reactivity and Shutdown Margin
3.2.2.a	Interlock	<2 cps S/U	4.2.2	Semiannual or retest	Test	
3.2.2.b	Interlock	Multiple rods	4.2.2	Semiannual or retest	Test	
3.2.2.c	Interlock	Pulse rod up/air	4.2.2	Semiannual or retest	Test	
3.2.2.d	Interlock	STD rod, pulse	4.2.2	Semiannual or retest	Test	
3.2.2.e	Interlock	1 kW pulse	4.2.2	Semiannual or retest	Test	
3.2.3.a	Safety System	550°C (2)	4.2.3	Daily channel check	& test	
3.2.3.b	Safety System	<1.1 MW SS	4.2.3	Daily	channel check & test	
3.2.3.b	Safety System	<2000 MW P	4.2.3		Daily channel check & test	
3.2.3.c	Safety System	HVPS	4.2.3		Daily channel check & test	
3.2.3.d	Safety System	Magnet current	4.2.3		Daily channel check & test	
3.2.3.e	Safety System	Manual	4.2.3		Daily channel check & test	
3.2.3.f	Safety System	Watchdog (scan rate)	4.2.3	Annual or retest	calibration	
3.2.4.a	Temp	2 channels	4.2.3	Daily	channel check & test	
3.2.4.b	Power level	2 channels				
3.2.4.c	Pulse power	1 channel				
3.2.4.d	Pulse energy	1 channel				
3.3.1.a	Pool cooling	48°C	4.3.1.a	During operation	operation	
3.3.1.b	Pool cooling	6.5 m	4.3.1.b	During operation	operation	
3.3.1.c	Pool cooling	5.0 µmho <mo>	4.3.1.c	During operation	operation	
3.3.1.d	Pool cooling	<1 psig HX dp	4.3.1.d	Daily & retest	channel test	
			4.3.1.d	During operation	Operation	
3.3.2.a	Confinement isolation		4.3.3.d	Daily	check ventilation system alignment	
3.3.3.a	Rad Monitoring Systems – CAM (particulate)	readout	4.3.3.a	Daily	check operability of particulate CAM	
3.3.3.a	Rad Monitoring Systems – CAM (particulate)	alarm	4.3.3.a	Daily	check operability of particulate CAM	
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41)	Sample purge exhaust when operating				
3.3.3.b	Rad Monitoring Systems – CAM (Ar-41)	$<2x10^{-6}$ µCi/cm^3 Annual average				
3.3.3.c	Area radiation monitors	SP ≤ 100 mr/hr				
3.3.3.c	Area radiation monitors	Pool level				
3.3.3.c	Area radiation monitors	2 other areas				

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	TEST	Procedure/record
3.4.1.a	Experiment reactivity	Moveable, <$1.00	4.4.1		Δk checked before an experiment is functional	
3.4.1.b	Experiment reactivity	Secured <$2.5	4.4.1		Δk checked before an experiment is functional	
3.4.1.c	Experiment reactivity	Total possible < $3.00	4.4.1		Δk checked before an experiment is functional	
3.4.2.a	Experiment Materials	Corrosive, reactive, explosive double encapsulated	4.2.2		Any surveillance conditions or special requirements specified in approval	

1. Identify prompt jump
2. Identify whether the graph is subcritical, critical, or supercritical.







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1. PURPOSE AND DISCUSSION

   This exercise will demonstrate two commonly used methods of control rod reactivity worth determination.

2. EXERCISE OBJECTIVES

   Following completion of this exercise, participants should be able to:

   2.1 Perform control rod manipulations to support control rod worth determinations

   2.2 Analyze reactor transients to support control rod worth determinations

   2.3 Compensate for the contribution of delayed neutrons in stabilizing reactor power

3. PREREQUISITES

   3.1 Preoperational Checks complete

4. BACKGROUND/REFERENCES

   4.1 Basic Nuclear Principles 9 *Reactor Kinetics*

   4.2 UT Procedure OPER 1, Startup and Shutdown Checks

   4.2 UT Procedure OPER 2, Reactor Startup and shutdown

   4.2 UT Procedure OPER 3, Reactor operation Modes

   4.3 UT Procedure SURV 6 Version 100, Control Rod Calibration

   4.4 ATTACHMENT 4.1: Relevant Technical Specifications

Comment [#1]: Review BNP9

Comment [#2]: Provide an overview of the procedure, discuss how positive period measurements and rod drops are used to determine reactivitity

Comment [#3]: Review TS requirements for control rods and control rod worth; indicate which TS requirements use control rod worth

5. INSTRUCTIONS

   5.1 PERFORM SURV 6

   5.2 CREATE a differential control rod worth curve

   5.3 CALCULATE integral rod worth

   5.4 PERFORM control rod drop calibration to determine the worth of a control rod from a designated position

6. REINFORCEMENT & REVIEW

6.1 Compare control rod worth as determined from the two methods

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
2.2.3	Reactivity insertion	2.2% Δk/k (pulse)			
3.1.1	Excess reactivity	4.9% Δk/k	4.1.1	Annually or post core/control rod changes	SURV-3 Excess Reactivity and Shutdown Margin
3.1.2	Shutdown margin	0.2% Δk/k (ref core, 1 rod out, moveable experiments most reactive)	4.1.2	Annually or post core/control rod changes	SURV-3 Excess Reactivity and Shutdown Margin
3.1.3	Transient rod	2.8% Δk/k worth			SURV-6 Control Rod Calibration
			4.1.3	Annual	SURV-7 Pulse Characteristic Comparison
3.1.3	Transient rod	<15 s withdrawal time	4.1.3	Annually or post core/control rod changes	SURV-6 Control Rod Calibration
3.2.1.a	Control rods	No apparent damage	4.2.1.a	Biennial	
3.2.1.b	Control rods	<1 s scram	4.2.1.b	annually or retest	SURV-6 Control Rod Calibration
3.2.1.c	Control rods	<0.2% Δk/k s^{-1}	4.2.1.c		SURV-6 Control Rod Calibration
3.2.2.a	Interlock	<2 cps S/U	4.2.2	Semiannual or retest	
3.2.2.b	Interlock	Multiple rods	4.2.2	Semiannual or retest	
3.2.2.c	Interlock	Pulse rod up/air	4.2.2	Semiannual or retest	
3.2.2.d	Interlock	STD rod, pulse	4.2.2	Semiannual or retest	
3.2.2.e	Interlock	1 kW pulse	4.2.2	Semiannual or retest	
3.4.1.a	Experiment reactivity	Moveable, <$1.00	4.4.1		
3.4.1.b	Experiment reactivity	Secured <$2.5	4.4.1		
3.4.1.c	Experiment reactivity	Total possible < $3.00	4.4.1		
5.3.1.a	Reactor core & fuel	8.5 w% 19.7 enriched			
5.3.1.b	Reactor core & fuel	1.6 ZrHx			
5.3.1.c	Reactor core & fuel	304 SS cladding 0.02 in.			
5.3.2.a	Control Rods	SS or Al clad, air, al or fuel (excepting transient) follower			
5.3.2.b	Control rods	Borated graphite, B4C powder, or boron and compounds solid			
5.3.2.c	Control rods	Transient rod, mechanical limit			
5.3.2.d	Control rods	2 shim, 1 reg, transient			

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PROCEDURE

SURV-3

Excess Reactivity and Shutdown Margin

Version
2.00

Approvals:

Randall Charbeneau — 4 / 4 / 02
Facility Director, NETL — Date

Kenneth S Ball — 4 / 4 / 02
Chairperson, Nuclear Reactor Committee — Date

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NUCLEAR ENGINEERING TEACHING LABORATORY
J. J. PICKLE RESEARCH CAMPUS
THE UNIVERSITY OF TEXAS AT AUSTIN

I. INTRODUCTION

A. Purpose
The purpose is to determine the reactor core reactivity conditions. These two conditions are safety considerations that directly effect the possible accident consequences.

B. Description
Evaluation of the TRIGA safety analysis demonstrates the limiting safety system settings (LSSS's) and limiting conditions for operation (LCO's). Excess reactivity and shutdown margin are directly related to reactor safety by defining the available control capability of the reactor. Operation of the reactor core within these limits is a necessity to maintain the proper control functions for all credible conditions.
Excess reactivity and shutdown margin are to be done annually or after significant changes to the core configuration. Normal practice, however, should check the excess reactivity and shutdown margin after any suspect core changes or if a high amount of burnup has occurred since the previous check.

C. Schedule
A K-excess and K-shutdown measurement of the reactor configuration must be done each year. Measurements should be done in but shall not exceed longer than 15 months from preceding measurement.

D. Contents

E. Attachments
K-excess K-shutdown Data 1 page

F. Equipment, Materials
Reactor Core System
Reactor Pool System
Instrument Control and Safety System

G. References, Other Procedures
Docket 50-602 Technical Specifications
TRIGA Control Rod Calibration Curves
Reactor Core load configuration

II. PROCEDURE

A. Requirements:

1. Verify that the reactor core is in a cold clean critical condition.
   a. Check logbook for previous operation history.
   b. No power operation above 10 kilowatts in past 3 days.
   c. If previous power operation then do not continue.

2. Perform a routine pre-start check.

3. Remove experiments from core if possible.
   Determine reactivity worth of any experiments that remain in the core.

B. Measurement of excess reactivity in normal operating configuration. (reference for experiment $\Delta k/k$):

1. Proceed with a routine startup to 50 watts.
   Bank all rods to approximately equal positions.

2. Calculate the amount of excess reactivity.
   Determine reactivity from calibration curves for each rod remaining in the core.

3. Determine excess reactivity as follows:

	Transient rod excess worth
+	shim 1 rod excess worth
+	shim 2 rod excess worth
+	reg. rod excess worth
=	total core excess worth
+/-	adjustment for experiments
=	\$ total core available excess worth
x	0.7 % $\Delta k/k$ per \$
=	% $\Delta k/k$ core excess (rods banked)

C. Measurement of reactivity limits:
Apply the following control rod configuration for measurement of the limiting excess and shutdown core reactivity. This measurement determines the maximum excess reactivity and minimum shutdown margin of the core.

1. Maintain transient rod in the fully inserted position, DN lamp illuminated.

2. Withdraw the regulating rod to the fully removed position, UP lamp illuminated.

3. Startup the reactor to 50 watts:
   i. Remove the highest worth shim rod to the full up position.
   ii. Adjust the other shim for criticality at a power of about 50 watts. (At 50 watts the neutron source should be less than 5 cents of reactivity.)
   iii. Verify power changes less than ±1watt per 200 seconds.
   iv. Print Status Window data.

4. Shutdown.

D. Calculate limits:

1. Shutdown margin (minimum >0.2% Δk/k):
   Determine minimum shutdown margin (excl. most reactive rod) as follows:

	reg. rod worth withdrawn
+	shim 1 rod worth withdrawn
+	shim 2 rod worth withdrawn
=	shutdown margin (all rods down)
+/-	adjustment for experiments
=	shutdown margin
-	most rx rod worth (reg)
=	$ min shutdown margin
x	0.7% k/k per $
=	% Δk/k min shutdown margin

2. Excess reactivity (maximum <4.9% Δk/k):
   Determine excess reactivity (total core reactivity) as follows:

	Transient rod total worth
+	shim 1 rod total worth
+	shim 2 rod total worth
+	reg. rod total worth
=	total core rod worth
+/-	adjustment for experiments
=	total core rod worth
-	s/d margin (all rods down)
=	$ k excess
x	0.7% Δk/k per $
=	% Δk/k max excess reactivity.

Core Excess and Shutdown Reactivity

Measurement of excess reactivity (Bank)

Transient rod excess worth	______	
Shim 1 rod excess worth	______	
Shim 2 rod excess worth	______	
Reg. rod excess worth	+ ______	
Total core excess worth	______	
Adjustment for experiments	+/- ______	
Total core available excess	______	
$\Delta k/k$ per \$ (β)	x 0.7 %	
$\Delta k/k$ core excess		______ %

Calculate limits Shutdown margin:

Shim 1 rod worth withdrawn	______	
Shim 2 rod worth withdrawn	______	
Reg. rod worth withdrawn	+ ______	
Shutdown margin(all rods down)	______	
Adjustment for experiments	+/- ______	
Shutdown margin	______	
Most rx rod worth (reg.)	- ______	
Min shutdown margin	______	
$\Delta k/k$ per \$ (β)	x 0.7 %	
$\Delta k/k$ min. shutdown margin		______ %
		(minimum >0.2% $\Delta k/k$)

Calculate limits Excess reactivity:

Transient total worth	______	
Shim 1 total worth	______	
Shim 2 total worth	______	
Reg. total worth	+ ______	
Total core rod worth	______	
Adjustment for experiments	+/- ______	
Total core rod worth	______	
S/D margin (all rods down)	- ______	
$\Delta k/k$ excess	______	
$\Delta k/k$ per \$ (β)	x 0.7%	
$\Delta k/k$ max excess reactivity		______ %.
		(maximum <4.9% $\Delta k/k$)

Date ___/___/___ **SRO Approval:** ______________________

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PROCEDURE

SURV-6

Control Rod Calibration

Version
1.00

Approvals:

Facility Director, NETL — Date: 3 / 2 / 09

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NUCLEAR ENGINEERING TEACHING LABORATORY
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THE UNIVERSITY OF TEXAS AT AUSTIN

I. *INTRODUCTION*

A. Purpose

The Control Rod Calibration Procedure benchmarks the primary system for reactor control and safety.

B. Description

Knowledge of control rod worth is necessary to assure the appropriate performance of the reactor control system and demonstrate compliance with Technical Specification limits. Both routine operating conditions and the safety functions of the control rods depend on accurate calibration data. Two separate methods of measurement are available to provide calibration data. The Rod Drop Experiment determines the approximate integral control rod worth by observation of the change in reactor power level as a function of time after the rod drop. This experiment provides the initial estimate of a rod's worth and may be used after major core rearrangements to predict approximate rod worth. The experiment may also verify the total rod worth after minor core changes. The second method of rod calibration is the Positive Period Experiment. This method provides the most accurate measurement of the differential rod worth. This experiment determines both the total control rod worth and the shape of the control rod position versus control rod worth curve. The Positive Period method should be used for normal control rod calibration

Measurement of the rod drop times verify the performance of the system safety function per Technical Specification requirement. The SCRAM switch or relay in the safety circuit initiates the safety circuit action dropping all control rods. Individual rod switches initiate the drop of each individual rod. Rod position switches sense when the rods reach the full down position. Proper performance of the safety system is indicated if all rods reach the full down position in the specified time limit.

Measurement of the control system rod removal rate coupled with the control rod peak differential worth establishes the maximum reactivity insertion rate of each control rod. This rate is limited as specified in the Technical Specifications to allow the safe control of the reactor in manual or auto mode.

C. Schedule

Control rod calibrations are to be done at least once each year and after any significant change to the reactor core configuration. Annual calibration measurements should be done in January or July but shall not exceed longer than 15 months from preceding measurement.

Measurement of control rod drop time and reactivity insertion rate should be done annually, not to exceed 15 months from preceding measurement, and/or after control rod or drive maintenance, reactor core reconfiguration, or movement of fuel adjacent to the standard rod drives.

D. Contents

E. Attachments

1. Reactivity vs. Power Ratio Plot 1 Page
2. Positive Period Data Sheet 1 Page
3. Stable Period Wait Time 1 Page
4. Inhour curve - Reactivity vs. Period Plot 1 Page
5. Rod Drop Time / Withdrawal Rate Data Sheet 1 Page

F. Equipment, Materials

TRIGA ICS System with control rod drives
Data Analysis Software such as "MathCAD"
Digital Stopwatch
Digital Storage Oscilloscope

G. References, Other Procedures

MAIN-6, Rod & Drive Maintenance, Inspection
Attachments 1 & 3 :
A. Edward Profio, "Experimental Reactor Physics", John Wiley and Sons Inc., 1976, pp 712, 716
Attachment 4 :
General Atomics Data Sheet.

II. PROCEDURE

A. Control Rod Worth Estimate by Rod Drop Method:

Use to estimate initial control rod worth following new core start-up.
May be useful following substantial core reconfigurations.

1. The reactor core condition should be cold and clean prior to measurement of rod worth. Perform ICS system pre-start checks. The reactor coolant system pumps should be off during control rod calibration.

2. Commence Startup of the reactor:

   a. Position the control rod being evaluated at the desired position – full up if the entire rod worth is desired to be estimated in one step, or partially withdrawn at selected increasing withdrawn locations if several drops are to made.

   b. Position the two rods closest to the rod being evaluated at a banked elevation, position the control rod farthest from the rod being calibrated at about 900 units to allow fine control of its reactivity for achieving criticality.

   c. Adjust control rods for criticality at a low power level such as 50 to 500 watts. The power should not be so high as to see a fuel temperature increase above ambient i.e. less than 1 Kilowatt.

   d. Remove the neutron source and readjust for criticality. The delayed neutrons should be allowed to come into equilibrium as evidenced by the indicated power remaining constant to within +/- 2% for a minimum of 3 to 5 minutes without further rod movement.

3. Setup data recording system to record reactor linear power as a function of time or use stopwatch and indication on linear power display to tabulate initial power and the indicated power after the control rod is dropped.

4. Drop control rod being evaluated by actuation of magnet button (standard rod drives) or air button (transient rod drive) and document the power vs. time data. Select times to record data based on time data plotted on the graph in Attachment 1 - Ratio of neutron density after a rod drop to the initial density (at critical), as a function of subcritical reactivity.

5. Using the data in Attachment 1, determine the reactivity associated with the rod drop based on the measured neutron density ratio (power ratio) at the specified time after the rod drop.

B. Control Rod Worth Measurement by Positive Period Method:

Use for the annual rod worth calibration.
Use as the primary rod calibration method.

1. The reactor core condition should be cold and clean prior to measurement of rod worth. Perform ICS system pre-start checks. The reactor coolant system pumps should be off during control rod calibration.

2. Commence Startup of the reactor:

   a. Position the control rod being evaluated at the desired position – full down if the entire rod worth is to be evaluated, or at predetermined locations if the shape of the differential rod worth curve has already been established.

      i. Initial control rod calibrations or calibrations after major core reconfigurations should evaluate the entire rod worth by stepwise pulling the rod in increments correlating to reactivity steps of 15 to 20 cents over its entire travel. This will require taking 10 to 20 measurements per control rod depending on its total worth.

      ii. Once the initial control rod calibration curve shape has been established, subsequent routine control rod calibrations may be made by using only 5 or 6 appropriately selected insertions of the same reactivity magnitude as above. One or two points should be selected near the rod height correlating to the peak differential rod worth. Four additional points should be selected, two in the lower and two in the upper parts of the rod travel correlating to areas spaced roughly equally on the slope portions of the differential worth curve. The data from these measurements can then be curve fit to the shape of the differential control rod worth curve to determine the actual rod worth.

   b. Position the two rods closest to the rod being evaluated at a banked elevation, position the control rod farthest from the rod being calibrated at about 900 units to allow fine control of its reactivity for achieving criticality.

   c. Adjust control rods for criticality at a low power level of 1 to 3 Watts.

3. Remove the neutron source and readjust for criticality. The delayed neutrons should be allowed to come into equilibrium as evidenced by the indicated power remaining constant to within +/- 4% for a minimum of 3.5 to 5 minutes without further rod movement. This constraint will limit measurement errors of criticality to +/- ~0.25¢ per measurement.

4. Record the Control rod positions on the Control Rod Calibration Data Sheet in Attachment 2.

5. Pull the control rod being calibrated in one smooth movement a distance correlating to an estimated reactivity worth of 15 to 20 cents which correlates to a stable period between 58 and 37 seconds. Record the rod position stop point on data sheet. (To minimize rod position hysteresis, if you inadvertently pull the rod too far, quickly move the rod back down slightly below the target point, then raise the rod to the target point.) Refer to previous calibration data to estimate the number of units to move the rod. Typical movements are 90 to 100 units for the initial and final pull at the full down or full up endpoints, decreasing rapidly to 20 to 40 units per pull in the mid range of rod travel. The reactivity per pull is limited to allow the reactor to attain a stable period prior to taking the power vs. time data thus reducing measurement errors. The time to reach a stable period is called the wait time. The wait times for 5% error are 20 to 35 seconds, for a 1% error they increase to 50 to 65 seconds respectively for 37 to 58 second stable periods. A table showing measurement errors as a function of the wait time required to attain a stable period is shown in Attachment 3.

6. Observe the power increase as indicated on the digital readout of the auto ranging linear power channel on the Animation Window. Use a stopwatch set to measure time intervals with respect to the start time. Start the primary stopwatch when the power passes the 60 watt point. Record the time when the power passes the 90 watt level, the 600 watt level, and the 900 watt level (time points should be marked at the first instant the power reaches the target value on the digital display). Time data at powers above the 1 Kilowatt level shall not be used as temperature feedback will create errors above this level.

7. Drive control rods other that the rod being calibrated down to decrease the reactor power. Leave the control rod being calibrated at the point to which it was withdrawn in step 5 if the entire rod is being stepwise calibrated. If the curve fit method is being used, reposition the rod being calibrated to its next starting point.

8. Repeat steps 2 through 7 until the remainder of the rod is completed or sufficient data points for curve fitting are obtained.

Notes: As long as the power level is not allowed to fall below the source interlock the source may be continuously left out of the core until all the data points desired are obtained.

Each sequence through this process will take approximately 15 minutes if everything is done with attention to detail so plan accordingly.

9. Analyze data either manually or via software program. Using the time data recorded, calculate the stable period resulting from each rod pull. Then use the reactivity equation or inhour curve in Attachment 4 to determine the reactivity associated with each rod pull.

10. A senior operator should review and approve the rod calibration data. If significant changes in rod worth are indicated, a review of the implications on excess reactivity and shutdown margins should also be initiated.

C. Control Rod Withdrawal, insertion, and drop time measurements.

1. Perform ICS system pre-start checks if not already completed.

2. Setup drop time measurement system. The magnet power supply voltage level controlled by the console scram switch should be used to start the timing. A signal from the control rod down limit switch should be monitored to indicate when the rod has reached the full down position
   a. Measurement equipment should be a storage oscilloscope or an electronic timer with signal start-stop features. Use of a stopwatch to measure rod drop time, manually started at the time the scram button is depressed and stopped at the time the rod visually hits bottom is also acceptable but not the preferred method of measurement.
   b. Measurement resolution for oscilloscope sweep should be set to 100 ms/div, vertical gain should be set to 5 V/div. Vertical signal probe should be set to X 10 for the transient rod, and X 1 for all other rods. Scope should be set to Auto trigger mode while setting up, and changed to single trigger or normal mode when taking the data.
   c. Connect start signal (scope trigger) to the Regulating rod positive magnet power (see table below for connection location). Set the scope trigger to DC coupling on a negative slope at a level of about 10 volts. The nominal magnet power high side is +13 volts and the low side is –6volts.
   d. Connect the signal (Channel 1) to the rod drive down limit switch (see table below for connection location) of the drive being evaluated.

Scope Input Channel	*DAC Tie Bar*	*Description*
Trigger	TB 5-3	Reg Magnet Pwr (+13V)
CH 1	TB 8-8	TR rod down limit
CH 1	TB 8-16	Shim 1 rod down limit
CH 1	TB 8-24	Shim 2 rod down limit
CH 1	TB 9-32	Reg rod down limit

3. Withdraw control rod being measured about 60 units and test drop the rod to verify the scope setup.

4. Fully withdraw the rod being evaluated, measuring the time it takes to move from full down to full up using a stopwatch. Record data on Attachment 5.

5. Drop the control rod to trigger and record a trace by initiation of the scram button. Drop time is measured from the time the scope triggered until the rod reaches full down, as evidenced by the transition of the signal on the rod down switch. Some rods may show a bounce after the initial bottom transition, typical drop time recorded is the time measured to when the rod remains full down as indicated on the trace. Record data on Attachment 5.

6. Repeat steps 2d through 5 for each remaining rod.

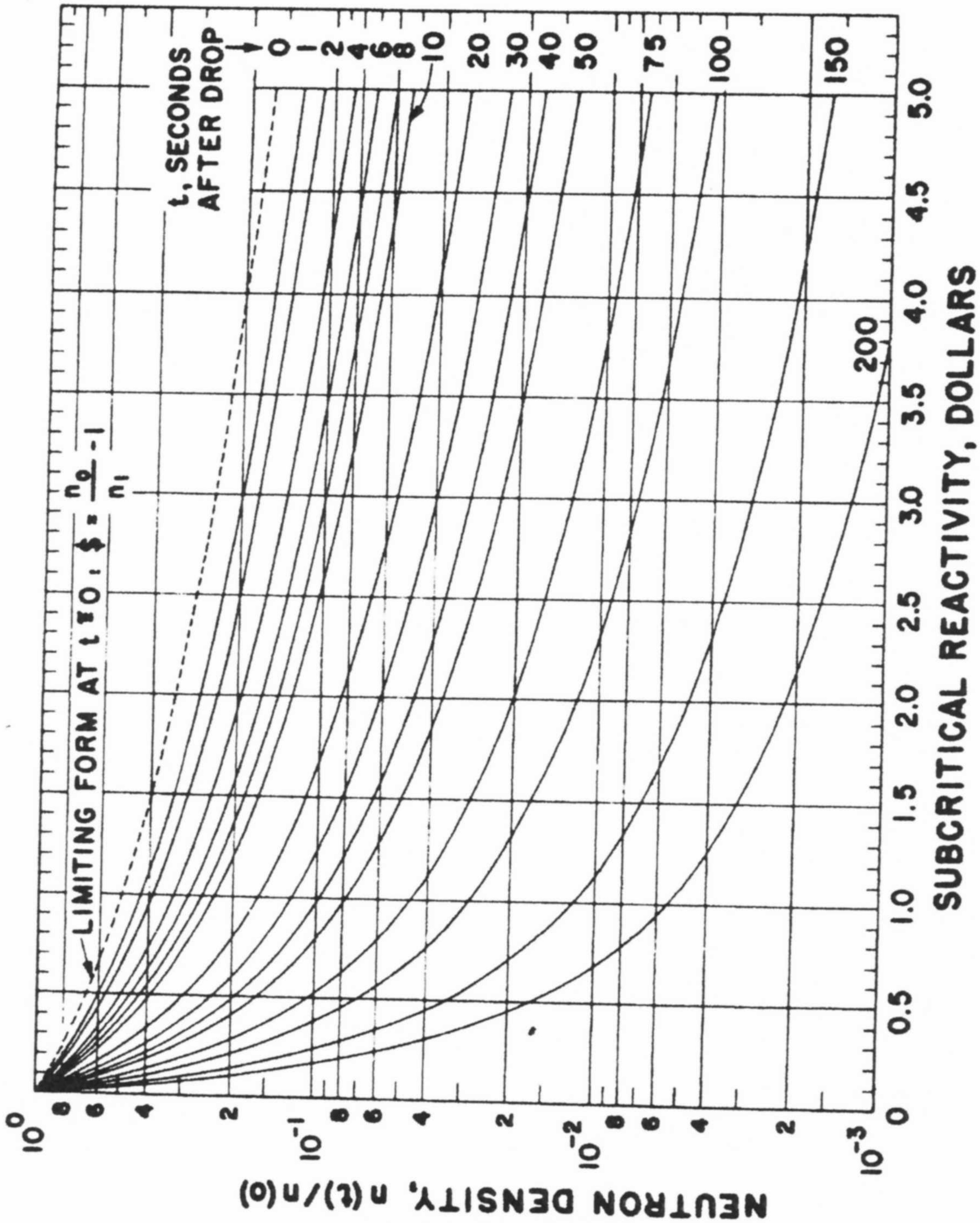
7. Calculate measured reactivity insertion rate and record data on Attachment 5:

   a.. Obtain peak differential rod worth near rod midpoint for each rod from the control rod calibration data.
   b. Calculate insertion rate (< 0.2 % Δk/k/sec) as follows:

   rate (% Δk/k/sec)=rate (units/sec)* worth (¢/unit)*(0.7% Δk/k/100¢)

8. Document any relevant notes, comments, or observations on Attachment 5 data sheet.

PAFormat.doc Attachment d:\Procedures\Surv\surv6-a1.doc	Number – Rev.: Procedure Title:	SURV-6 1.00	Date: 3/2/09 Control Rod Calibration



Reactivity vs. Power Ratio					Stamp(Original-Red, Copy-Blue)
Date of Change					COPY
NETL Director Approval					Page 1 of 1

Rod: ____________

Date: ____________

$P(\tau)=P_0e^{\tau/T}$

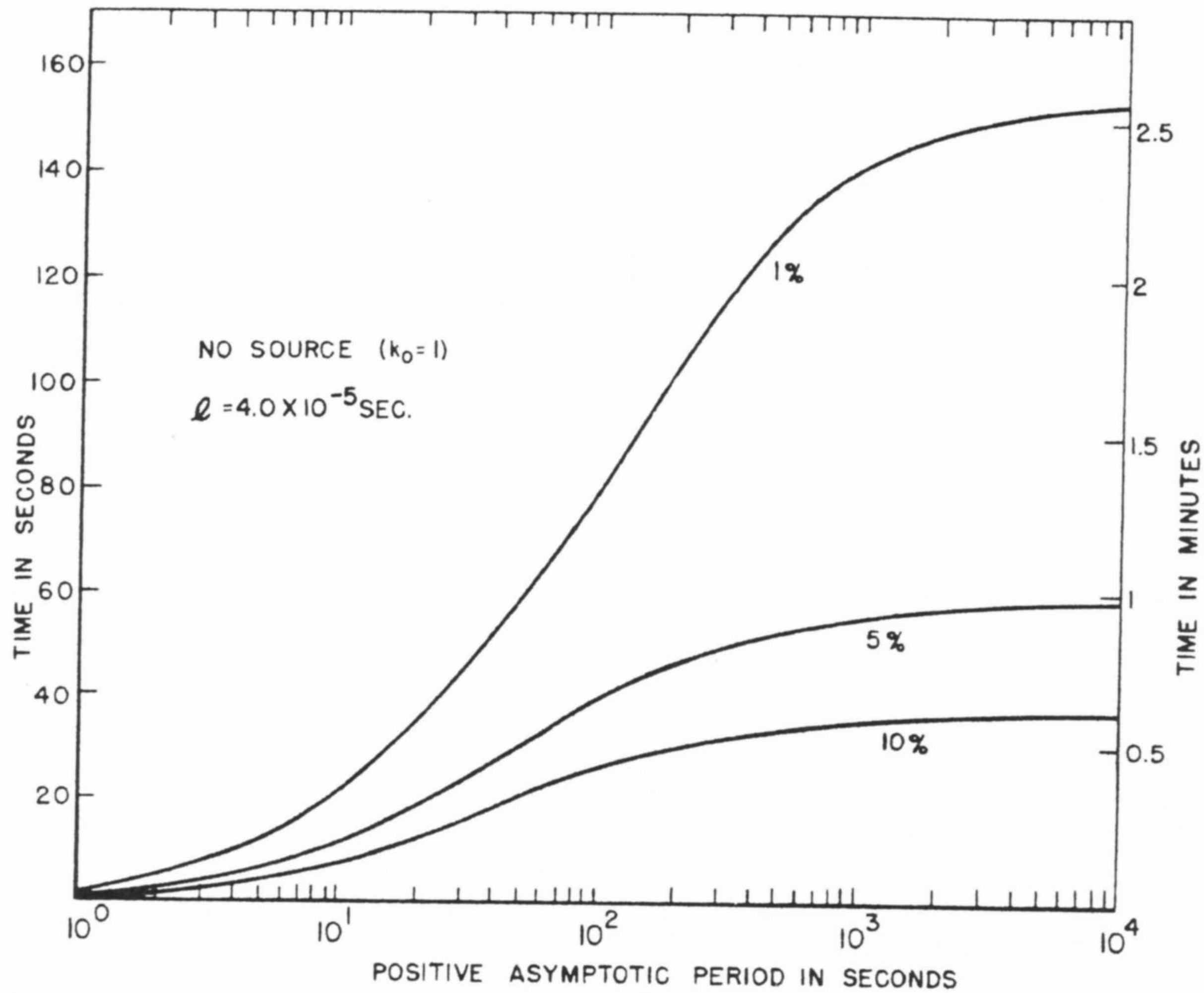
$T=\Delta\tau/\ln(P(\tau)/P_o)$

$=\Delta\tau/\ln(10)$

$= .434\ \Delta\tau$

	Steady-state: ~3 watts TR S1 S2 Reg	Rod positions start stop	Times at power 60 watts	 90 watts	 600 watts	 900 watts
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						

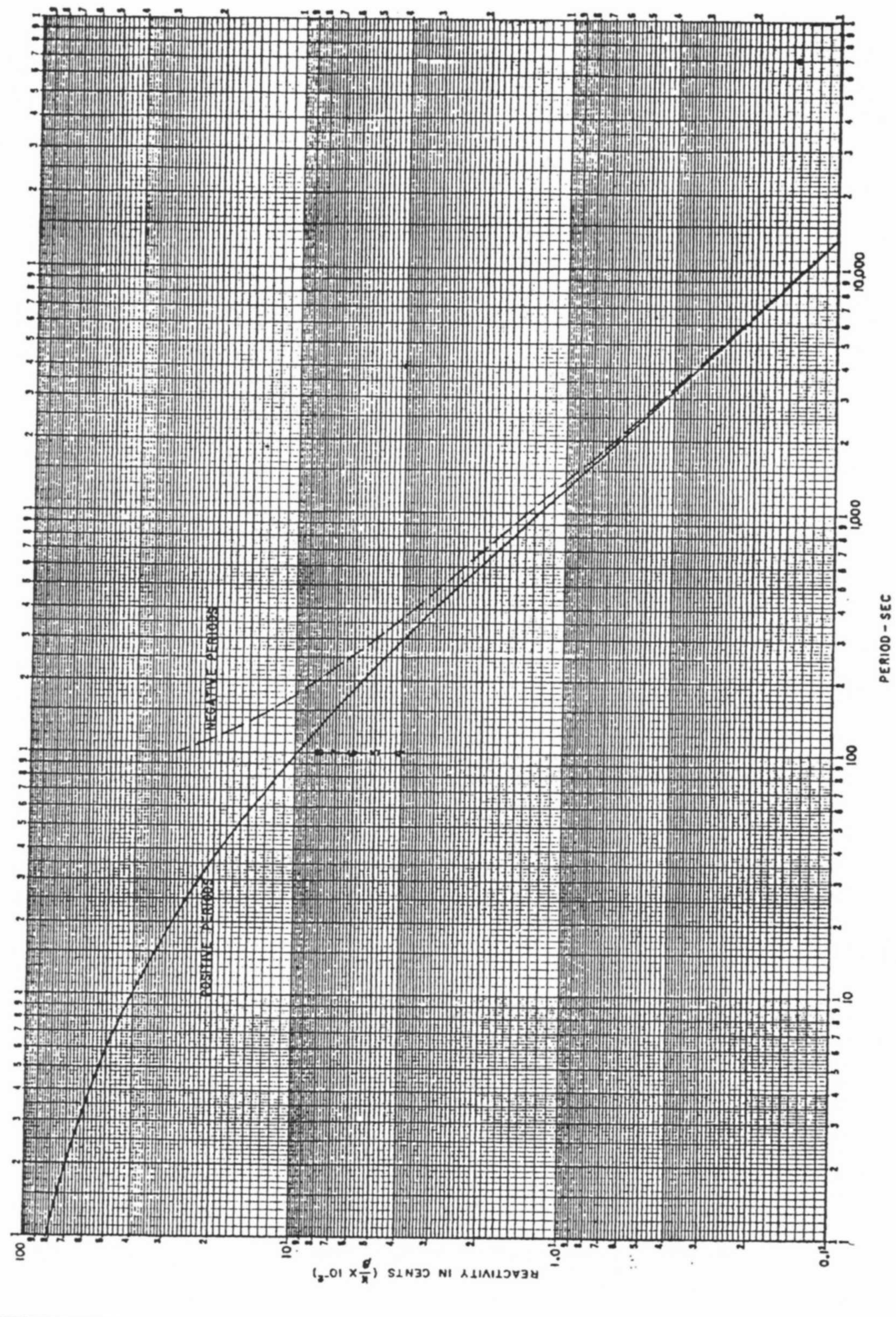
1



NO SOURCE (k_o = 1)
ℓ = 4.0 X 10^{-5} SEC.
1%
5%
10%
TIME IN SECONDS
TIME IN MINUTES
POSITIVE ASYMPTOTIC PERIOD IN SECONDS

2
3

1


NEGATIVE PERIODS
POSITIVE PERIODS
PERIOD - SEC
10,000
1,000
100
10
REACTIVITY IN CENTS
100
10
1.0
0.1

2

Rod Drop/Withdrawal Data

Rod Drop Time:
(limit: less than 1 second)

Rod	Time (Sec)	Verified OK (initial)
Transient		
Shim 1		
Shim 2		
Regulating		

Maximum Reactivity Insertion Rate:
(limit: less than 0.2 % ΔK/K/sec)

Rod	Withdrawal Time (sec)	Peak Differential Worth (¢/unit)	Insertion Rate (% ΔK/K/sec)
Transient			
Shim 1			
Shim 2			
Regulating			

Comments:

SRO Approval:______________________________ **Date:**___/___/___

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1. PURPOSE AND DISCUSSION

This exercise will demonstrate spatial dependence of the reactivity of core materials.

Comment [#1]: Discuss how flux varies across the core and how this affects reactivity; discuss limits of point kinetics in discrete systems

2. EXERCISE OBJECTIVES

Following completion of this exercise, participants should be able to:

2.1 Use the control rod worth calibration to evaluate reactivity of core materials

2.2 Evaluate how position influences reactivity worth

2.3 Safely handle TRIGA fuel

3. PREREQUISITES

3.1 Preoperational Checks complete

4. BACKGROUND/REFERENCES

4.1 UT Procedure OPER 1, Startup and Shutdown Checks

4.2 UT Procedure OPER 2, Reactor Startup and shutdown

4.2 UT Procedure OPER 3, Reactor operation Modes

4.3 UT Procedure SURV 6 (100), Control Rod Calibration

4.4 UT Procedure Fuel 1 (101), Fuel Movement

Comment [#2]: Discuss TS requirements for fuel handling, fuel handling log, communications protocol

5. INSTRUCTIONS

5.1 OBTAIN baseline data

5.1.1 In accordance with OPER 2, PERFORM reactor startup to 50 Watts

5.1.2 RECORD critical rod heights

Transient Rod Position	Shim 1 Position	Shim 2 Position	Regulating Rod Position

5.1.3 In accordance with OPER 2, PERFORM a reactor shutdown

Applies to Step 5.2, 5.4 and 5.7

NOTE

(1) The reactor is not secured during in-core fuel handling
(2) TS 6.1.3 requires direct supervision a Senior Reactor Operator for fuel element relocation within the core
(3) Fuel relocation is documented as per FUEL 1 on a fuel handling log

5.2 In accordance with FUEL 1, REMOVE an installed element from the E, F, or G ring AND PLACE the fuel element in a designated fuel storage location

5.3 OBTAIN comparison data

5.3.1 In accordance with OPER 2, PERFORM reactor startup to 50 Watts

5.3.2 RECORD critical rod heights

Transient Rod Position	Shim 1 Position	Shim 2 Position	Regulating Rod Position

5.3.3 In accordance with OPER 2, PERFORM a reactor shutdown

5.4 In accordance with FUEL 1, REMOVE the fuel element from the designated fuel storage location and AND PLACE the element in the position cleared in Step 5.2

5.5 In accordance with FUEL 1, REMOVE an installed element from the B, C, or D ring AND PLACE the fuel element in a designated fuel storage location

5.6 OBTAIN comparison data

5.6.1 In accordance with OPER 2, PERFORM reactor startup to 50 Watts

5.6.2 RECORD critical rod heights

Transient Rod Position	Shim 1 Position	Shim 2 Position	Regulating Rod Position

5.6.3 In accordance with OPER 2, PERFORM a reactor shutdown

5.7 In accordance with FUEL 1, REMOVE the fuel element from the designated fuel storage location and AND PLACE the element in the position cleared in Step 5.5

6. REINFORCEMENT & REVIEW

6.1 Graphically evaluate the worth of the fuel element in both positions

	Baseline Condition		Core Position ____		Core Position ____	
	Position	Reactivity Worth	Position	Reactivity Worth	Position	Reactivity Worth
Transient Rod						
Shim 1						
Shim 2						
Reg Rod						
SUM	NA	NA	NA		NA	
DIFFERENCE	NA	NA	NA		NA	

6.2 Compare measured fuel element worth values to those in the SAR

Core Position	Worth (% Δk/k)	Worth (Cents)	Fuel Position
B Ring	1.07	153	6
C Ring	0.85	121	12
D Ring	0.54	77	18
E Ring	0.36	51	24
F Ring	0.25	36	30
G Ring	0.19	27	36

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PROCEDURE

FUEL-1

Movement of Fuel

Version
1.00

Approvals:

Sheldon Landsberger 2 / 17 / 05
Facility Director, NETL Date

Howard M. Liljestrand 2 / 17 / 05
Chairperson, Nuclear Reactor Committee Date

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I. INTRODUCTION

A. PURPOSE

The instructions of this procedure are to control the movement of reactor fuel components within the reactor core grid structure and to or from storage locations.

B. DESCRIPTION

Reactivity changes occur as fuel is added to the reactor core or storage locations. To assure adequate safety margins for the proper performance of the control rod system, procedural controls define the requirements and limitations for fuel movement within the reactor core and storage areas. These rules include changing the arrangement of fuel components as well as fuel additions or deletions.

C. Schedule

Apply this procedure each time fuel is moved.

D. Contents

E. Attachments

Fuel Movement Log	1 Page
TRIGA Core Arrangement	1 Page
Storage Well Arrangement	1 Page
Pool Storage Rack Inventory	1 Page
MCZPR Storage Rack Inventory	1 Page

F. Equipment, Materials

Fuel Element Tool
Radiation Work Permit (RWP) - A RWP is required for this procedure only if irradiated fuel is to be moved outside of the reactor pool access area.

G. REFERENCES

Criticality Calculations

II. PROCEDURE

A. TRIGA Fuel Movement

1. A senior reactor operator shall supervise all movements of fuel, including movements to or from the reactor core grid structure and movements between storage locations. At least one person should assist with the handling of the fuel elements.

2. Restrict all fuel element arrays except the reactor core to an array limit of less than 20 elements.

    a. Store fuel elements in the fuel storage wells or in the reactor pool. Use the 19 element hexagonal array racks (these may by stacked two deep per well) or the 6 or staggered 12 element linear array racks.

    b. Elements not in storage racks or shipment casks should be in groups of three or less.

3. Plan fuel movement activities so as to minimize the number of individual moves required to achieve the desired result.

4. Move elements between the reactor core, storage racks, shipment casks or other locations with special fuel handling tool.

5. Maintain access control or restrict use of fuel handling tool by lock if fuel movements are not in progress.

6. Test fuel handling tool on non-fuel element prior to use.

7. Approve by inspection and test any device other than the fuel handling tool prior to use for movement of fuel. Handle the instrument elements with the extension tubes. Handle control followers with the extension rods.

8. Handle fuel elements carefully. Care should be taken not to bump or scrape elements. Minimize the possibility and potential consequences of an accidental drop of an element.

9. The Pool Area radiation monitor shall be functional during fuel movement in or to and from the reactor pool.

10. Verify a gamma sensitive survey instrument(with audible or alarm functions) is present in the area where the fuel movement will occur. A Radiation Work Permit (RWP) is necessary for movement of irradiated fuel beyond the immediate vicinity of the reactor pool access area.

11. The air particulate monitor or substitute monitor should be functioning.

12. Record all fuel element movements in the Fuel Element Log.

13. Acknowledge by verbal response each change of fuel handling tool opened or closed status if two persons operate the tool.

14. Acknowledge by verbal response the exchange or transfer of the fuel handling tool to another person.

15. Operate and monitor the reactor console during the movement of fuel to or from the reactor core.

    a. Prevent movement of any control rod drive by removing the neutron source from the core.
    b. Place the console in Manual Mode. Verify no control rods will withdraw.
    c. A log of any event will be automatically recorded to the control system history file.
    d. Removal of a fuel followed control rod from the core for inspection requires a minimum shutdown margin greater than 0.2% $\Delta K/K$ 2 rods out (i.e. with the rod being removed out and the remaining highest worth rod up).
    e. Movement of an instrument element requires disconnect and reconnect of instrument connections with a functional test prior to reactor operation.

16. Verify excess reactivity and shutdown margin if fuel movement is to or from the reactor core. Check by measurement or calculate by conservative estimate.

17. Compare control rod critical positions before and after movement and recalibrate if a change occurs due to movement of the fuel in the core.

18. Upon completion of fuel movement the fuel handling tool shall be surveyed for contamination, bagged, and locked securely for storage in its designated location.

B. Criticality and Inventory Control of Materials in Storage

1. Storage and handling of large quantities of special nuclear materials (SNM), such as reactor fuel, carries the risk of accidental criticality if the materials are placed into a critical geometry or moderated with hydrogenous materials. Tracking of licensed SNM is required (10CFR74) and annually reported to the NRC via the Nuclear Materials Management and Safeguards System (NMMSS). The total quantity of SNM at the NETL shall not exceed a Category III (Low Strategic Significance) quantity of material as defined in 10CFR73.2 (<10kg unirradiated). The NETL Reactor Facility License, R-129, requires all fuel elements to be stored in a geometrical array where the effective multiplication is less than 0.8 for all conditions of moderation. TRIGA elements stored in a 19 element rack and MCZPR elements stored in a linear array are sufficiently subcritical.

2. SNM, other than irradiated fuel and the subcritical assembly, in quantities greater than 1.0 gram should be stored in room 2.204B, the Auxiliary Equipment Room (AER), when not actually in use or prepared for shipment. The subcritical assembly should not be stored near other fuel in the AER, but should be stored in its 55 gal drum maintained within the confines of the reactor bay. Sealed neutron sources (e.g. PuBe) shall NEVER be stored within the AER. Neutron sources may be used within the AER for Criticality Alarm instrument checks as noted below.

3. A Criticality Accident alarm system is required by 10CFR70.24 for SNM not stored underwater when the total quantity of enriched U-235 exceeds 700 grams. Monitoring is not required for fuel stored in the storage pits if they are maintained underwater. The NETL Criticality Accident monitor is a Ludlum 375 gamma and neutron monitor or

9. The Pool Area radiation monitor shall be functional during fuel movement in or to and from the reactor pool.

10. Verify a gamma sensitive survey instrument(with audible or alarm functions) is present in the area where the fuel movement will occur. A Radiation Work Permit (RWP) is necessary for movement of irradiated fuel beyond the immediate vacinity of the reactor pool access area.

11. The air particulate monitor or substitute monitor should be functioning.

12. Record all fuel element movements in the Fuel Element Log.

13. Acknowledge by verbal response each change of fuel handling tool opened or closed status if two persons operate the tool.

14. Acknowledge by verbal response the exchange or transfer of the fuel handling tool to another person.

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    e. Movement of an instrument element requires disconnect and reconnect of instrument connections with a functional test prior to reactor operation.

16. Verify excess reactivity and shutdown margin if fuel movement is to or from the reactor core. Check by measurement or calculate by conservative estimate.

17. Compare control rod critical positions before and after movement and recalibrate if a change occurs due to movement of the fuel in the core.

18. Uppon completion of fuel movement the fuel handling tool shall be surveyed for contamination, bagged, and locked securely for storage in its designated location.

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3. A Criticality Accident alarm system is required by 10CFR70.24 for SNM not stored underwater when the total quantity of enriched U-235 exceeds 700 grams. Monitoring is not required for fuel stored in the storage pits if they are maintained underwater. The NETL Criticality Accident monitor is a Ludlum 375 gamma and neutron monitor or

equivalent meeting the requirements of 10CFR70.24. The digital monitor should be mounted outside the AER with the detectors located adjacent to the MCZPR fuel storage location. The detector system is calibrated annually and response checked quarterly using a neutron/gamma source. The audible alarm setpoint should be set between 5 and 20 mR/hr.

4. Routine entry into the AER requires the SRO issue the key. The SRO will verify the individual has been trained to properly enter and work within the AER prior to key issue.

5. Individuals entering the AER must note the readings of the Criticality Accident Monitor outside the AER and verify the digital meters are at typical background levels (expected to be less than 5.0 mR/hr and 5.0 mrem/hr) prior to entry. If necessary, movement of moderator (e.g. graphite, polyethylene, water) materials into or within the AER requires SRO supervision.

6. Fuel element (TRIGA or MCZPR) movement within the NETL requires SRO supervision. The SRO will ensure fuel elements are moved in limited quantities and maintained in a subcritical configuration during movement.

C. TRIGA FUEL REFERENCE REACTIVITY VALUES

Core Location	TRIGA fuel vs. water	% ΔK/K
Ring A		4.00
Ring B		1.07
Ring C		0.85
Ring D		0.54
Ring E		0.36
Ring F		0.25
Ring G		0.19
3 elements (1D, 2E)		1.25
6 elements (6B)		6.42

DATE	TIME	ELEMENT #	OLD LOCATION	NEW LOCATION	COMMENTS/SRO

Core Arrangement

Reference ____________
Date ____________

Configuration Number:
FUEL-GRAPHITE-WATER-OTHER
______-______-______-______

G-26 G-27 G-28 G-29 G-30
____ ____ ____ ____ ____

G-24 F-21 F-22 F-23 F-24 F-25 F-26 G-32
____ ____ ____ ____ ____ ____ ____ ____

G-23 F-20 E-17 E-18 E-19 E-20 E-21 F-27 G-33
____ ____ ____ ____ ____ ____ ____ ____ ____

G-22 F-19 E-16 D-13 D-14 D-15 D-16 E-22 F-28 G-34
____ ____ ____ ____ SHM2 ____ ____ ____ ____ ____

G-21 F-18 E-15 D-12 C-09 C-10 C-11 D-17 E-23 F-29 G-35
____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____

G-20 F-17 E-14 D-11 C-08 B-05 B-06 C-12 D-18 E-24 F-30 G-36
____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____

F-16 E-13 D-10 C-07 B-04 B-01 C-01 D-01 E-01 F-01
____ ____ ____ REG ____ ____ TRNS ____ ____ ____

G-18 F-15 E-12 D-09 C-06 B-03 B-02 C-02 D-02 E-02 F-02 G-02
____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____

G-17 F-14 E-11 D-08 C-05 C-04 C-03 D-03 E-03 F-03 G-03
____ ____ ____ ____ ____ ____ ____ ____ ____ ____ ____

G-16 F-13 E-10 D-07 D-06 D-05 D-04 E-04 F-04 G-04
____ ____ ____ ____ SHM1 ____ ____ ____ ____ ____

G-15 F-12 E-09 E-08 E-07 E-06 E-05 F-05 G-05
____ ____ ____ ____ ____ ____ ____ ____ ____

G-14 F-11 F-10 F-09 F-08 F-07 F-06 G-06
____ ____ ____ ____ ____ ____ ____ ____

G-12 G-11 G-10 G-09 G-08
____ ____ ____ ____ ____

Well # ________________ Date ____________

UPPER

C-09 C-10 C-11

_____ _____ _____

C-08 B-05 B-06 C-12

_____ _____ _____ _____

C-07 B-04 A-01 B-01 C-01

_____ _____ _____ _____ _____

C-06 B-03 B-02 C-02

_____ _____ _____ _____

C-05 C-04 C-03

_____ _____ _____

LOWER

C-09 C-10 C-11

_____ _____ _____

C-08 B-05 B-06 C-12

_____ _____ _____ _____

C-07 B-04 A-01 B-01 C-01

_____ _____ _____ _____ _____

C-06 B-03 B-02 C-02

_____ _____ _____ _____

C-05 C-04 C-03

_____ _____ _____

1

RACK #	Pool Rack Position #1	#2	#3	#4	#5	#6
R1 Lower						
R1 Upper						
R2 Lower						
R2 Upper						
R3 Lower						
R3 Upper						
R4 Lower						
R4 Upper						
R5 Lower						
R5 Upper						
R6 Lower						
R6 Upper						
R7 Lower						
R7 Upper						
R8 Lower						
R8 Upper						
R9 Lower						
R9 Upper						
R10 Lower						
R10 Upper						
T1						
T2						
T3						

2 3 By: Date:

1

RACK #	MCZPR Rack Position								
	#1	#2	#3	#4	#5	#6	#7	#8	#9
Upper Pin									
Lower Pin									

2 By: Date:

3

MCZPR Storage Rack Inventory

Date of Change

NETL Director Approval

Stamp(Original-Red, Copy-Blue)

COPY

1. PURPOSE AND DISCUSSION

   This exercise will demonstrate calibration of the reactor power level instrumentation.

2. EXERCISE OBJECTIVES

   Following completion of this exercise, participants should be able to:

   2.1 State the three methods for reactor power calibration

   2.2 Calculate thermal power of the reactor:

   A. Given time dependent pool temperature data,

   B. Given pool temperature data prior to heatup and after reactor operation,

   C. Given mass flow rates and process system temperature differences

   2.3 State the errors associated with each method

3. PREREQUISITES

   3.1 Preoperational Checks complete

4. BACKGROUND/REFERENCES

   4.1 UT Procedure OPER 1, Startup and Shutdown Checks

   4.2 UT Procedure OPER 2, Reactor Startup and shutdown

   4.2 UT Procedure OPER 3, Reactor operation Modes

   4.2 UT Procedure SURV 2 (100), Reactor Power Level Calibration

   4.3 ATTACHMENT 6.1: Relevant Technical Specifications

   4.4 GEN-39, Power Calibration for TRIGA Reactors

5. INSTRUCTIONS

   5.1 PERFORM SURV-2

   5.2 In accordance with SURV-2, CALCULATE reactor thermal power

   - Ballistic method

Comment [#1]: Discuss procedure prerequisite items accomplished prior to the exercise; discuss those (if applicable) exceptions to prerequisites taken for the purposes of this exercise

Comment [#2]: Discuss rate of rise versus ballistic methods of power level calibration

- Heat up rate method

5.3 PERFORM a reactor startup to a designated power level

Applies to Step 5.4 NOTE
It is necessary that fuel and pool temperatures be at equilibrium in order to perform a calorimetric calibration;

5.4 WHEN fuel and pool temperatures are stable, READ AND RECORD:

Comment [#3]: Identify instruments and readings

- Heat exchanger exit and inlet pool water temperature
- Heat exchanger exit and inlet chill water temperature
- Primary coolant mass flow rate
- Chill water mass flow rate
- Reactor pool bulk water temperature

Time	Pool Flow	Pool Inlet Temp	Pool Outlet Temp	CW Flow	CW Inlet Temp	CW Outlet Temp	Pool Bulk Temp
AVE							

5.5 PERFORM heat balance to determine reactor thermal power

6. REINFORCEMENT & REVIEW

Compare the thermal power calculated using each method

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
2.1	Safety limit Fuel Temp	Cladd<500°C, <1150°C			
2.1	Safety limit Fuel Temp	Cladd>500°C, <950°C			
2.2.2	Power level	1100 kW			
3.2.3.b	Safety System	<1.1 MW SS	4.2.3	Annual or retest	SURV-2 reactor power calibration
			4.2.3	Daily	
3.2.3.b	Safety System	<2000 MW P	4.2.3	Annual or retest	
			4.2.3		
3.2.3.c	Safety System	HVPS	4.2.3	Annual or retest	
			4.2.3		
3.2.4.b	Power level	2 channels			
3.2.4.c	Pulse power	1 channel			
3.2.4.d	Pulse energy	1 channel			

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PROCEDURE

SURV-2

Reactor Power Calibration

Version
1.00

Approvals:

3 / 2 / 09

Facility Director, NETL   Date

3 / 2 / 09

Chairperson, Nuclear Reactor Committee   Date

Number of Pages: 6
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NUCLEAR ENGINEERING TEACHING LABORATORY
J. J. PICKLE RESEARCH CAMPUS
THE UNIVERSITY OF TEXAS AT AUSTIN

I. *INTRODUCTION*

A. Purpose

The Reactor Thermal Power Calibration procedure determines the heat output of the TRIGA reactor by measurement of the change in the bulk pool water temperature.

B. Description

Accurate knowledge of the reactor power level depends on the total amount of water in the pool and several corrections. The corrections adjust for conditions that cause the pool-reactor system to deviate from an adiabatic condition. Power calibration depends on the pool constant that is a function of the pool water volume. A change in volume equivalent to a 10-centimeter water depth will cause a 1.2% change in the pool constant.

C. Schedule

A reactor thermal power calibration must be done once each year. Measurements should be done in July but shall not exceed longer than 15 months from preceding measurement.

D. Contents

E. Attachments

1.	Power Calibration Data	1 Page

F. Equipment, Materials

Thermocouple array (three-3 element linear arrays)
Ice bath reference junction for thermocouple array
Galvanometer or Micro-voltmeter
Computer with data acquisition interface
Data acquisition software such as Labview
Math software such as Mathcad

G. References, Other Procedures

GA Publication: "Power Calibration for TRIGA Reactors" by W.L. Whittemore, J. Razvi and J.R. Shoptaugh Jr. February 1988.

II. PROCEDURE

A. Thermal Power Calibration Experiment

Routine thermal power calibrations should be done at or near full power. Typically performing the calibration at 900 KW indicated is preferred so as to maintain some margin below the scram set points. If a major core reconfiguration was made, several calibrations at stepwise increasing powers of about 300 KW, 600 KW, and then 900 KW should be performed.

1. Install or verify installed pool water temperature measurement thermocouple array and set up ice bath for reference junction.
   Pool water temperature (Type E, 9-element array):
   - Sense array points are at approximate depths of 1, 2 and 3 meters. Locate sensor array across minor pool axis near major axis radius point.

2. Connect thermocouple signal and ground shield from ice bath junction to digital multimeter with input sensitivity of at least 1 μV (0.1 μV preferred) such as Keithley System DMM with interface to data acquisition computer. Initiate thermocouple data acquisition software such as Labview and set to acquire data at 1 minute intervals. Configuration of software to average multiple data points taken at a high sample rate may be used to reduce signal noise.

3. Record air and shield temperatures.

   a. Room air temperature (approximate measurement point):
      - At pool railing 1 meter above pool deck, south rail.
   b. Shield concrete temperature (approximate measurement points):
      - In conduit near shield surface at level 1 by BP1 ARM.
      - In conduit near shield surface at level 2 under NM

4. Adjust pool depth to 8.10 meters with bulk pool temperature at approximately 20°C.

5. Install pool stirrer mechanism into the reactor pool and secure with safety line. Initiate operation of the stirrer

6. Sub-cool pool using the coolant system to a temperature based on the selected target reactor power. Target temperature should result in the average of the pool temperature prior to start of the calibration and the final pool temperature after the conclusion of the power part of the experiment to equal the measured shield temperature.

7. Secure the operation of pool purification and coolant pumps. Close pool water isolation valves.

8. Close pool surface argon purge valve.

9. Turn off pool lights.

10. Initiate recording of pool temperature data at 1-minute intervals. Take at least 90-minutes of data prior to reactor startup to the calibration power level.

11. Complete reactor startup checks including electronic Pre-start checks.

12. Initiate logging of reactor power indications on all 3 power monitoring channels by either use of continuous electronic data capture or manual data recording at 2 minute intervals.

13. Operate reactor at 900 KW or target power for 30 minutes.
Power level is to be measured by NPP linear channel.

    a. The operation mode for startup should be <u>manual</u>.
    b. Startup rate should be equivalent to a 20-second period.
    c. Record at power time (to nearest tenth of a minute).
    d. Observe Linear Power channel #1 (NPP) to control power level.
    e. Limit control rod adjustments following the startup (first 5 minutes) to <10 units.

14. Continue recording pool temperature data during time at power at 1-minute intervals.

15. Shutdown Reactor by manual scram at end of 30 minutes and record scram time.

16. Continue record pool temperatures at 1-minute intervals for 60 minutes after shutdown.

17. Return pool conditions to pre-experiment conditions.

18. Complete shutdown checklist.

B. Thermal Power Evaluation

Use of data analysis software such as Mathcad is suggested for evaluation of power calibration results.

1. Calculate power level by the slope method based on the time rate of temperature changes during the at power portion of the run. Use a linear least squares fit of temperature data to determine the slope at constant power.

a. P_{900} = [dT/dt (°C /Hr)] / [Pool Δ (°C /MW-Hr)]
b. Where constant Pool Δ = 20.74 °C /MW-Hr.

2. Verify pool calibration by ballistic calibration method.

   a. Calculate pool temperature at startup.
      i. Use a least-squares fit of the temperatures before reactor startup.
      ii. T_i is extrapolated to the initial log time at 900 kW.

   b. Calculate pool temperature at shutdown.
      i. Use a least-squares fit of the temperatures after reactor shutdown.
      ii. T_f is extrapolated to the final log time at 900 kW.

   c. Calculate the total temperature change during constant power production.
      i. Subtract initial temperature at startup from final temperature at shutdown.
      ii. E = (T_f-T_i) (°C / (Pool Δ (°C /MWHR)

3. Evaluate ballistic method power calibration.

   a. Correct the time at constant power for startup and shutdown energy.
      i. t_i (initial time at power) for the startup, before Δt.
      ii. t_f (final time at power) for the shutdown energy, after Δt.

   b. Correct for the contribution of fission product energy.
      Fission products accumulate during and shortly following operation, ΔT.

   c. Correct for heat flow of pool inflow or pool outflow during power operation.

4. Compare results of slope method and ballistic method for agreement.
   Comparison of the two power analysis methods should be within ~2%.

5. Compare current data with data from previous calibrations.
   a. Water, air and concrete, temperatures prior to power level test,
   b. Initial pool temperature at startup,
   c. Final pool temperature at shutdown,
   d. The reactor operation time and power level indication.

6. Thermal power measurement accuracy should be ~5 %.
   Measurement errors should be less than 5% at one standard deviation.

C. Power Instrumentation Adjustments

1. Senior operator shall approve acceptance or adjustment of power channels. All power channels should read within 2% of the calculated thermal power.

2. Adjust each chamber indication to the value of the thermal power test.
   a. Instrumentation power channels shall be adjusted to within 2%.
   b. Adjust detector chamber position with reactor in manual mode only.

3. Adjustment of the NP and NPP should be by movement of detector position.

4. Adjustment of the NM is typically by change of only the digital calibration constants. If a large NM adjustment is required the chamber should be repositioned following the same procedures used for initially setting up the channel, including positioning the chamber to read the specified current followed by setting the digital constants.

5. If adjustment of the NM digital constants was required, the calibration signal potentiometers in the campbell amplifier may require adjustment to allow the prestart checks Mode 4 and 5 to pass and the shutdown crossover should be observed to verify a smooth transition. Refer to initial channel set up procedures if adjustments are required.

6. Repeat procedure if the calibration requires more than a 10% adjustment. Apply power calibration comparison to each power chamber (NM, NP, & NPP).

7. Reevaluate pool constant if there is any significant change of pool water volume. A significant change may occur if the mass of other materials in the pool changes.

Thermal Power Calibration Data

Ambient Conditions:

Room air temperature ________ °F

Reactor shield temperature: ________ °F

Test power level (NPP1000 value): _________ %

Startup:

Time at test power: ____:____:____

Shutdown:

Time at shutdown (scram): ____:____:____

ICS Configuration:

Power channel values:
(30-minute average)

NM1000: ______ %

NP1000: ______ %

NPP1000: ______ %

Comments:

Date ___/___/___ **Approval:** ______________________________

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TRIGA™ REACTORS

GEN-39

POWER CALIBRATIONS FOR TRIGA™ REACTORS

by

W.L. Whittemore, J. Razvi,
and J.R. Shoptaugh Jr.

GENERAL ATOMICS
SAN DIEGO, CALIFORNIA

February, 1988

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POWER CALIBRATIONS FOR TRIGA REACTORS

by

W.L. Whittemore, J. Razvi, and J.R. Shoptaugh, Jr.
General Atomics
San Diego, California

1.0 INTRODUCTION

Over the many years since the first TRIGA reactor was built, a number of variations have evolved for calibration of thermal reactor power. The first method (1) used by General Atomics (GA) in the startup of a number of facilities was based on use the of a calibrated electrical heater in a calorimetric procedure where the rate of rise of the bulk pit water temperature was measured using such heaters. The reactor was then operated to give the same rate of rise of water temperature. Thus the reactor power was established at the value produced by the electrical heaters. For most installations, no stirrer was used in these initial power calibrations; typically a heater with a 10-15 kW capacity was used for Mark I and Mark II reactors. The facilities with larger tanks such as that at the Armed Forces Radiological Research Institute (AFRRI) used a larger electrical heater with a 90-100 kW capacity.

Numerous problems have developed with the electrical heater technique over the past 30 years. First and foremost is the inconvenience from repeated use of the electrical heaters. Second, and almost as important, is the realization that adequate stirring of the water is necessary in order to provide greater precision in the results of the calibration. In most cases, the electrical heater power level was only a tiny fraction of the final reactor power -- typically, 10-15 kW for a 250-1000 kW Mark I or Mark II -- giving an output which was only 1.2-5% of full power. Even the 90 kW heater for the 1000 kW AFRRI TRIGA reactor gives only about 9% full power. Under these circumstances, the extrapolation from the calibration power to full reactor power involves a factor of at least 10 to as high as 20 or more. Such large scale extrapolations require careful attention to the linearity of the power monitor circuitry, especially in the case of water reflected reactors such as the Mark F.

After the first few installations of TRIGA reactors, the initial power calibration for later reactors was performed without the electrical heaters. With the reactor operating at a constant power, the rate of temperature rise was determined. With a tank constant (ΔT per hour per unit power) calculated for the applicable heat content of the system, the reactor power was then determined from the measured rate of temperature rise from operation of the reactor. Unfortunately, a stirrer was not used in many of these installations resulting in imperfect mixing during these determinations. Without a stirrer and with intermediate reactor powers (100-200 kW), the flow pattern of hot water from the core is a columnar chimney rising up about half way to the surface of the pool and then bending over in a mushroom fashion to return to the region below the reactor core. At somewhat higher power levels (500-700 kW), this columnar chimney of hot water may extend nearly all the way to the top of the reactor tank before turning over to return to the region below the core. It is easy to imagine then that the measured rate of temperature rise near the top of the pool can give quite different results depending upon where in the tank the temperature probe is located and whether the chimney reaches all the way to the top of the tank. A stirrer which will disperse this flow pattern is obviously needed to provide reproducible temperature measurements that are relatively independent of the location of the temperature probe.

During the later TRIGA installations a stirrer was used in the initial power calibration. Recognizing the problems outlined above, GA has recommended the use of an adequate stirrer for subsequent power calibrations that are based solely on the calculated heat capacity of the water in the reactor tank. A number of user facilities have adopted this procedure with varying results. The purpose of this paper is to establish a framework for the calorimetric power calibration of TRIGA reactors so that reliable results can be obtained with a precision of better than ± 5%. Careful application of the same procedures have produced power calibration results that have been reproducible to ± 1.5%. The procedures are equally applicable to the

Mark I, Mark II and Mark III reactors as well as to reactors having much larger reactor tanks (e.g., the conversion facilities with 25,000-40,000 gallons) and to TRIGA reactors capable of forced cooling up to 3 MW in some cases and 15 MW in another case. In the case of forced cooled TRIGA reactors, the calorimetric power calibration is applicable in the natural convection mode for these reactors using exactly the same procedures as are discussed below for the smaller TRIGA reactors ($<$ 2 MW).

2.0 DISCUSSION

2.1 Basic Theory.

The calorimetric procedure is essentially the same whether it involves the calorimetric determination of heat equivalent of electrical energy, the heat content of a piece of hot metal or the rate of heat generation by a research reactor core. In each case, the calorimeter contains a relatively large volume of fluid such as water and is constructed with insulated walls to reduce the flow of energy through the calorimeter walls. Extraneous heat sources (pool lights, pumps, etc.) are either accounted for or eliminated.

The basic formulation is the following:

$$\text{Power} = \frac{dE}{dt} = mC\,\frac{dT}{dt} ,$$

where: E = energy in the entire system,

$\frac{dE}{dt}$ = rate of producing energy,

m = effective mass of the system,

C = specific heat of the system, and

$\frac{dT}{dt}$ = time rate of temperature rise of mass m.

For the TRIGA system the mass m is mainly the water in the tank because of its large heat capacity. The product of the mass of the metal components times their individual heat capacity (i.e., $\Sigma\, m_i C_i$) is small compared to the heat content of the large body of water. It is important to note that the stirring produced by the motor driven impeller assures that all the water in the tank participates in the calorimetric measurement. The small rate of energy added by the pump motor is typically less than 1 kW and is negligible for power calibrations performed at 200-1000 kW.

Figure 1 shows the conceptual arrangement of a typical calorimeter. When the concept is applied to the power calibration of a TRIGA reactor core, the following items must be considered in order to establish repeatable conditions and to reduce or control the flow of extraneous heat into or out from the system.

i. Adjust tank water to the correct level to control the mass;
ii. Precool the tank water to an appropriate low temperature;
iii. Secure the cooling system and water treatment system (if this is a separate circuit);
iv. Turn off the underwater lights;
v. Minimize net heat flow through tank walls during the measurement by selecting the initial and final water temperatures to be about equally above and below the average tank wall temperature; and
vi. Install an appropriate stirrer.

2.2 Selection of Stirrer

As discussed above, the use of a stirrer is necessitated by the requirement that essentially all portions of the tank water participate in the measurement. The goal of stirring is to assure as nearly as possible a uniform temperature within the entire tank. Of course, the water column immediately above the core will always remain hotter than the bulk water.

In our experience, an adequate stirring for a Mark I or Mark II type tank is provided by a 1/3 HP motor driving a stainless steel propeller at 1725 RPM. For a larger tank, such as the GA TRIGA Mark F which contains about 24,000 gallons of water, a 2/3 HP motor (1725 RPM) with double impel-

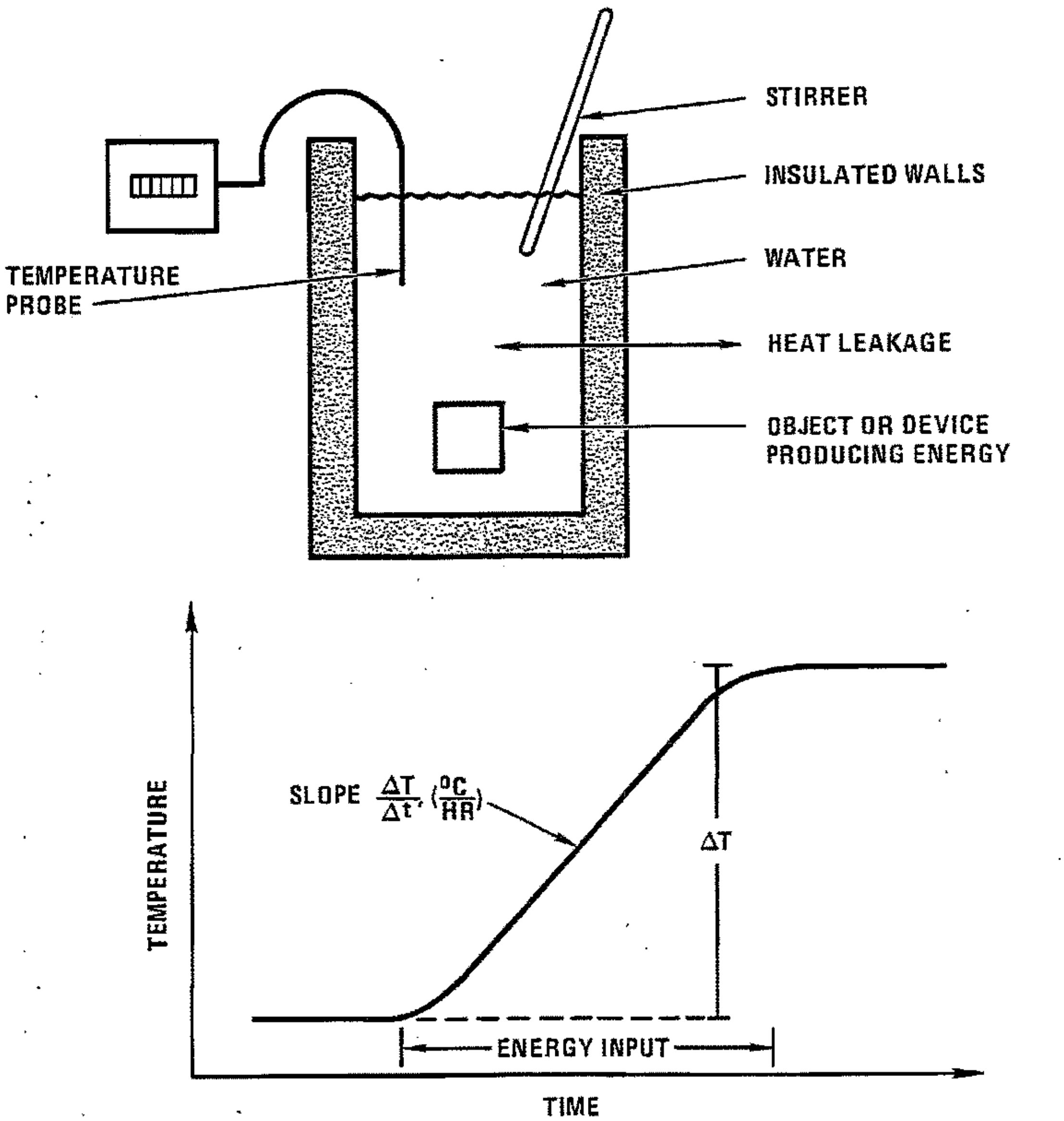

Figure 1. Conceptual Arrangement of Calorimeter and Basic Experimental Results.

lers has been demonstrated to be adequate. Both of these motors operate on 115 VAC. If there are associated canals or side pools connected to the main reactor tank, a circulating pump or additional stirrers should be used to provide good mixing of the entire water system. A 1.5 HP motor with circulating pump has been found to provide adequate mixing of the Mark F side canal (~2500 gallons) and the main tank at the GA reactor. As seen below, the electrical power equivalent of any of the above motors contributes a negligible heat input to the tank water compared to that from the reactor core.

$$1/3 \text{ HP} \times 746 \text{ watts/HP} \simeq 0.25 \text{ kW}$$
$$2/3 \text{ HP} \times 746 \text{ watts/HP} \simeq 0.50 \text{ kW}$$
$$1.5 \text{ HP} \times 746 \text{ watts/HP} \simeq 1.10 \text{ kW}$$

3. RESULTS

3.1 Calorimetric Power Calibration of the GA TRIGA Mark F Reactor.

The standard method of calibrating the 1.5 MW Mark F reactor is to establish well documented conditions and to determine by regression analysis the rate of temperature rise of the stirred pool water. From this and the determination that a 9.60°C rise is produced by 1 MW-hr in our 23,717 gallons of water, the power level is determined. To verify the validity of this procedure a related procedure (called the ballistic method) has been devised which depends upon an analysis of the heat balance for the water in the tank. The procedure in both cases is to start with the reactor pool at an initial temperature T_1 and isolated to the maximum extent possible, then operate the reactor for about one hour at power P. During this time the water temperature rises at a steady rate dT/dt until reaching the final temperature T_2. The reactor power P can be determined in one of two ways.

1. The standard method (called herein the "slope method" is to compute the power as follows:

$$P = \frac{dT/dt\ (^\circ C/HR)}{9.60^\circ C/MW\text{-}HR} .$$

2. An alternative method to determine P (called the "ballistic" method) is provided by the following considerations As a result of operating the reactor at power P for one hour, a heat input of [Px time] is applied to the 23,717 gallons of water thus raising its temperature from T_1 to T_2. Accurate values of T_1 and T_2 can be determined by extrapolation from long term temperature measurements before and after reactor operation. The slow change of the temperature of the stirred water before and after the reactor run is governed by the heat influx and heat leakage, respectively. From a direct measurement of these time rates of temperature rise or fall, the related energy leakages can be evaluated and used to correct the estimated power. In addition, corrections to [Px time] can be applied to account for: (1) the power generated during the rise to power P; and (2) the beta and gamma energy generated from the fission products after the reactor is shut down.

Figure 2 presents the experimental results for the Mark F reactor operated at about 1 MW for about one hour. The actual power as determined from the slope using a least squares fit to the steeply rising portion of the curve gives:

$$\frac{0.15327 \times 60}{9.60} = 0.958 \text{ MW or } 958 \text{ kW.}$$

As can be seen, the temperature of the tank water was monitored for about 97 minutes prior to operating the reactor and 60 minutes subsequent to the reactor scram. Little heat leakage into the tank is demonstrated by the small rate of temperature rise (0.040°C/hr) which corresponds to a heat inleakage of about 4.2 kW. At the end of the reactor run, the temperature of the water was monitored for an additional hour. The rate of temperature decrease (0.245°C/hr) corresponds to an indicated heat loss rate of about 25.6 kW. This latter estimate of heat loss after shutdown can be further adjusted to account for the energy generated by the decay of fission products. Additionally, a correction can be made to account for the heat generated by the reactor during the rise to full power.

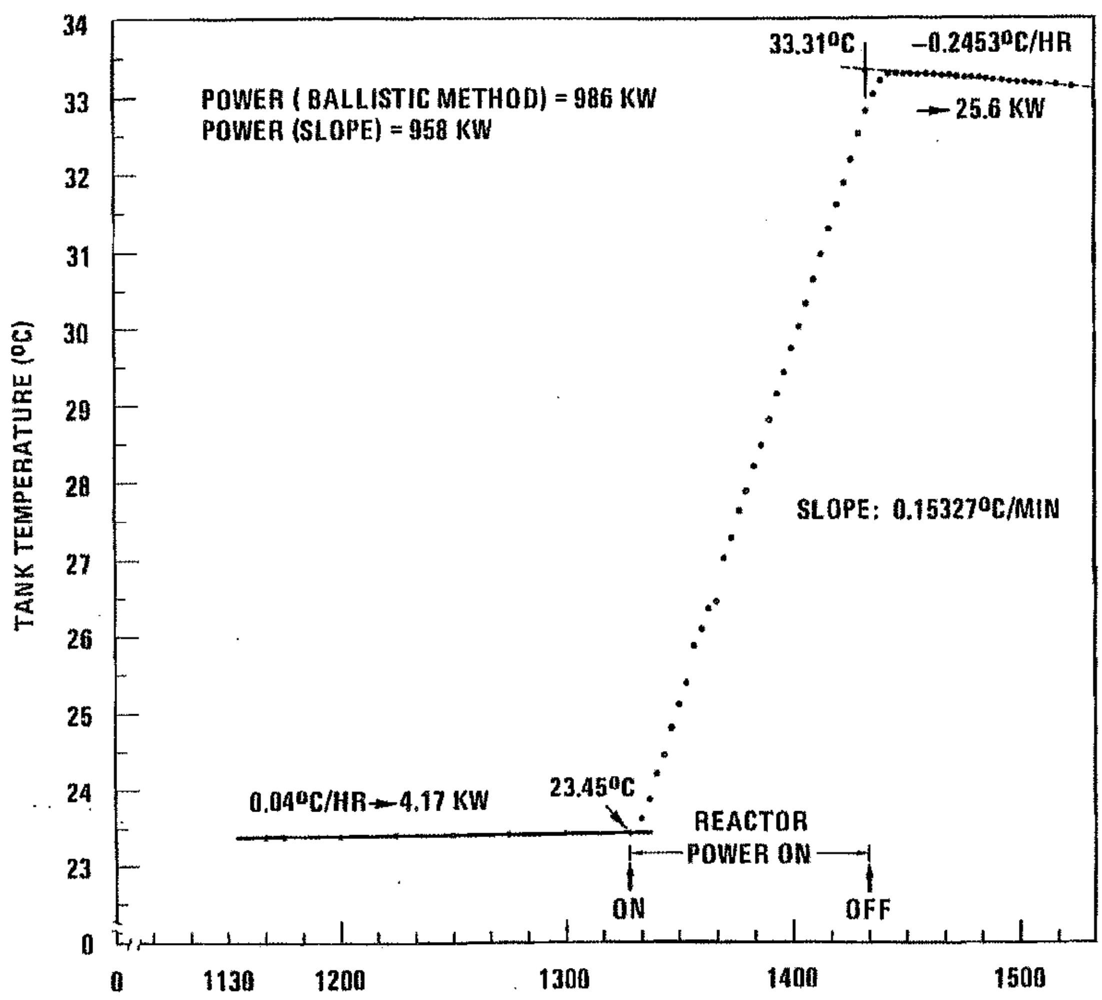

Figure 2. Calibration Curve for Mark F TRIGA Reactor (≈1000 kW).

3.1.1 Heat Generated During Rise to Power

It can easily be shown that the energy generated during the rise to power P on a reactor period τ is given simply as P x τ. To account for this additional heating, one needs only to increase the reactor on time by a time τ. A 17 second reactor period would increase the on time by 0.28 minutes. For most of our power calibrating, the reactor is taken to full power on such a relatively short period.

3.1.2 Observed Heat Input into Reactor Tank Water

The curves in Figure 2 before and after the reactor run can be used to evaluate the initial and final water temperatures using standard calorimetric procedures. Proceeding thus we find T_1 = 23.45 and T_2 = 33.31°C. With these temperatures, the corrected run time of 63.0 + 0.3 min (including the correction in Section 3.1.1), and the tank constant for 23,717 gallons of water, 9.60°C/Mw-hr, we calculate on observed heat input of

$$\frac{(33.31 - 23.45)°C}{9.60°C/Mw\text{-}hr} = 1.027 \text{ Mw-hr} = 61620 \text{ kW-min.}$$
$$= 0.973 \text{ MW} \times 63.3 \text{ min}$$

3.1.3 Heat Generated by Fission Product Decay

From Glasstone[2] and Blizzard[3] one can deduce that about 6.5 percent of reactor power is produced by the beta and gamma rays from fission product decay. Figure 3 presents data from these references for a one-hour reactor run. These data are in essential agreement with those summarized in Table 33 of Etherington [4]. After shutdown these products decay rapidly so that 3 or 4 minutes following the shutdown, the fraction of original power is less than 1.5 percent. Since the amount of fission product energy involved is small compared to that generated by the reactor operation, considerable simplification can be introduced in handling the fission product decay energy. Reference to Figure 2 shows that the temperature peaks about 5 to 6 minutes after the reactor scram. Most of this delay represents the final mixing time and includes the initial contribution from fission product decay (see Table 1). It does not correspond to any significant heat input from the reactor since the neutron power within the reactor decreases abruptly by more than a factor of 20 upon insertion of all the control rods and continues to fall promptly to essentially zero. [The neutron flux (not power) falls on the - 80 sec period.]

The energy from fission product decay can be treated as follows. During the first hour we will separate this period into the first 3.32 minutes (during which the largest rates of decay occur), and the remaining 56.68 minutes for which we will calculate an average rate of producing energy. Using the data in Figure 3, we can prepare the following useful table.

Table 1. Estimated Fission Product Energy Production after a 1-Hour Reactor Run at Estimated Power P of 970 kW.

Time Interval (min)	Energy Generated (kW-sec)		Average Power (kW)
0 - 3.32	5.376 x P = 5214.7 KJoules		26.2 kW for 3.32 min
3.32 - 10	4.368 x P = 4237		19951/3400.8 sec
10 - 30	9.0 x P = 8730	19951 KJ	= 5.87 kW
30 - 60	7.2 x P = 6984		

3.1.4 Net Heat Loss from Tank.

Using the data in Table 1 and the measured time rate of change of temperature before and after the reactor run at power (Figure 2), the net heat loss from the tank can be determined as follows. The measured rate of rise (0.04°C/hr) gives an inleakage of 4.2 kW, whereas the measured rate of decrease after the run (0.245°C/hr) corresponds to an indicated rate of heat loss of 25.6 kW. However, this rate of heat loss must be corrected for the

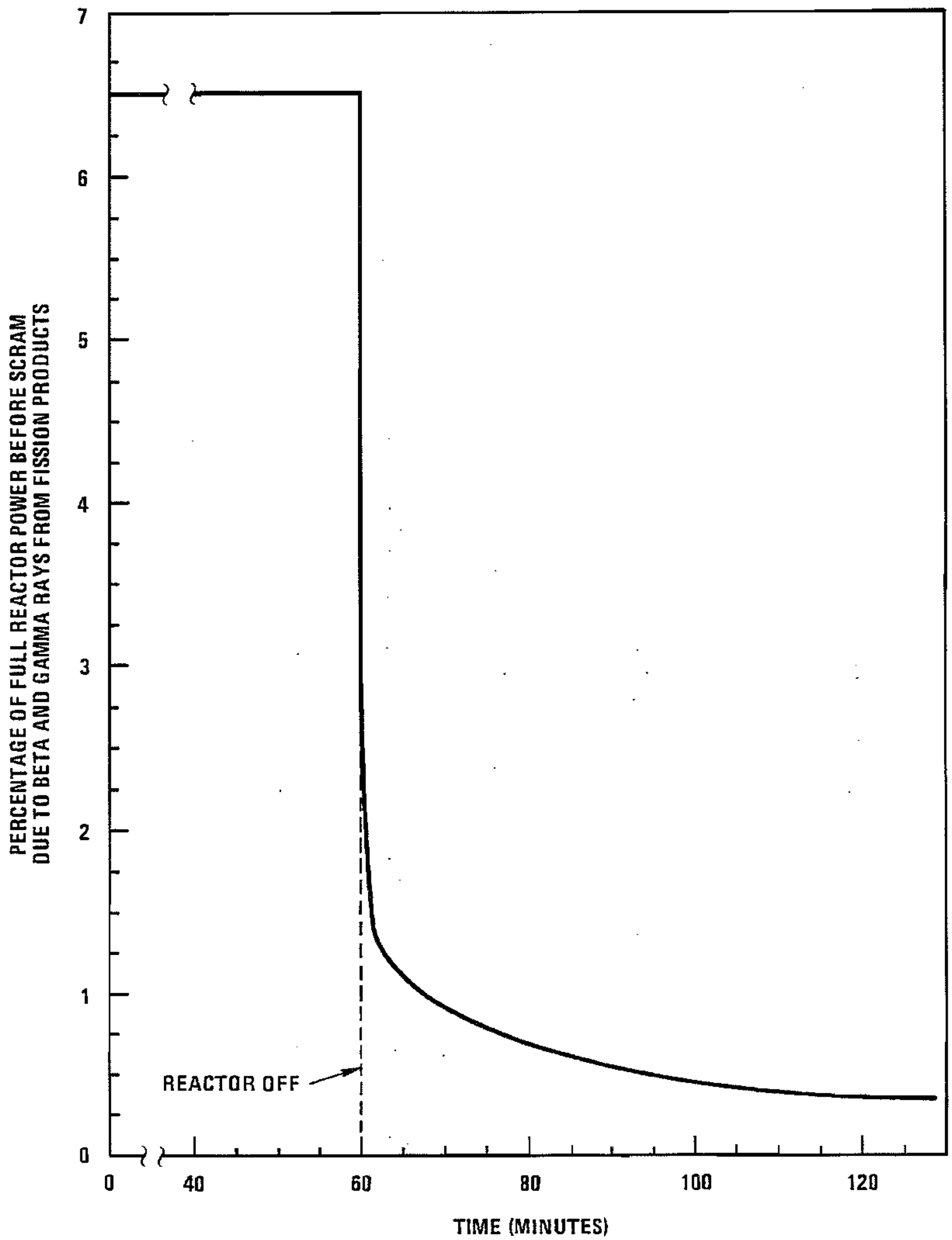

Figure 3. Percentage of Full Reactor Power from Beta and Gamma Rays from Fission Products: 1 Hour Reactor Run.

gamma heat from fission product decay after shutdown, which adds 5.87 kW of heat to the water (Table 1) in the time interval of concern. The actual net heat loss from the tank is then the sum of the measured heat loss (25.6 kW) and the added heat from fission product decay (5.87 kW), or 31.35 kW. It may be noted that we ignore the heat loss due to evaporation since this has been evaluated to be only on the order of 1 kW at most.

If a linear relationship is assumed between the heat addition and heat loss over the 63.3 minutes (exposure time corrected as in Section 3.1.1) of reactor operation, the <u>average net</u> heat loss from the tank can be calculated to be 864.0 kW-min.

3.1.5 Heat Balance for the Reactor System.

With the data generated above, we can now calculate a reasonably accurate heat balance for the power calibration run plotted in Figure 2.

$$\left[\begin{array}{c}\text{Apparent Heat Input}\\ \text{into Tank Water}\\ \text{(temp. rise)}\end{array}\right]$$

$$= \left[\begin{array}{c}\text{Heat From}\\ \text{Reactor}\\ \text{(At Power)}\end{array}\right] - \left[\begin{array}{c}\text{Net Heat}\\ \text{Loss from Tank}\\ \text{Water}\end{array}\right] + \left[\begin{array}{c}\text{Heat Input}\\ \text{From Fission}\\ \text{Products}\end{array}\right]$$

Solving for the heat generated by the reactor operation (related directly to the power level), we have:

$$\left[\begin{array}{c}\text{Heat from}\\ \text{Reactor}\\ \text{(At Power)}\end{array}\right] = \left[\begin{array}{c}\text{Apparent Heat}\\ \text{Input Into}\\ \text{Tank Water}\\ \text{(Temp. Rise)}\end{array}\right] + \left[\begin{array}{c}\text{Net Heat}\\ \text{Loss from}\\ \text{Tank Water}\end{array}\right] - \left[\begin{array}{c}\text{Heat Input}\\ \text{From Fission}\\ \text{Products}\end{array}\right]$$

Using the values computed in Section 3.1.4, we have therefore

$$\begin{bmatrix} \text{Heat from} \\ \text{Reactor} \end{bmatrix} = 61620 \text{ kW-min} + 864 \text{ kW-min} - (26.2 \times 3.32) \text{ kW-min}$$

$$= 62397 \text{ kW-min}$$

Since the reactor ran at power P for 63.3 min the average power P is thus

$$62397/63.3$$
$$= 986 \text{ kW.}$$

3.1.6 Comparison of Slope Method and Ballistic Method

The ballistic method produced an estimate of 986 kW including corrections for heat leakage through the tank walls and effects due to fissions product heat generation after reactor scram. The relative magnitude of these corrections in this case are:

Heat Leakage
Through Tank Water = $\left[864.0/62397\right] \times 100 \cong 1.4\%$
Heat from Fission Product Decay: $(87/62397) \times 100 \cong 0.14\%$

The slope method gave an estimated power of 958 kW. If only the slope data is available without long term temperature monitoring before and after the run, then no correction is possible for the heat loss from the tank water (≈1.4% in this case). On the other hand, by balancing the initial and final temperatures around the average tank wall temperature the heat loss through the walls can be reduced to a small fraction of the 1.4% estimated here.

In this case the percent difference in the estimates of power from the "ballistic" and "slope" method is

$$\frac{986 - 958}{986} \times 100 = 2.8\%$$

About 1.4% of this is due to heat loss through the walls. If this is taken into account, then the percent difference in the two results would be only 1.4%. The more accurate of the two estimates would still appear to be that from the ballistic method of calculation.

3.2 Calibration at Near Full Power

With adequate stirring the calorimetric method of power calibration can be conducted essentially at full power. This has the distinct advantage of ending the power calibration at or near the full power operating conditions. This can offer advantages for certain TRIGA reactor cores that use water as the reflector. For these reactors the temperature of the pool water can influence the power indication by several percent due to the variations in neutrons leaking to the power monitors caused by the variations in the water density as a function of pool temperatures. Although the power change may be only about 5 percent from a 10°C change in water temperature, this variation can be greatly reduced by (1) arranging to end the calorimetric power calibration at the desired pool operating temperature (such as 36°C for the Mark F reactor) and (2) by carefully controlling the water temperature during operations, for example at 36 ± 1°C as for this reactor during current operations.

Figure 4 presents data for a calorimetric calibration done at near full power which is 1.5 MW for the Mark F TRIGA reactor. This run which also includes data useful for the ballistic method was conducted with an earlier, somewhat smaller stirrer (1/3 HP motor). Using the procedures set forth in the sections under 3.1 we obtained the following:

$$P \text{ (slope method)} = 1450 \text{ kW.}$$
$$P \text{ (ballistic method)} = 1441 \text{ kW.}$$

In this case, the slope method gives a value larger by 0.8% for the power compared to that from the ballistic method with its special corrections. In this case, the percentage heat loss from the tank water leads to a correction of 2.7% as compared to a 1.4% correction at ≈ 1 MW.. If applied to the

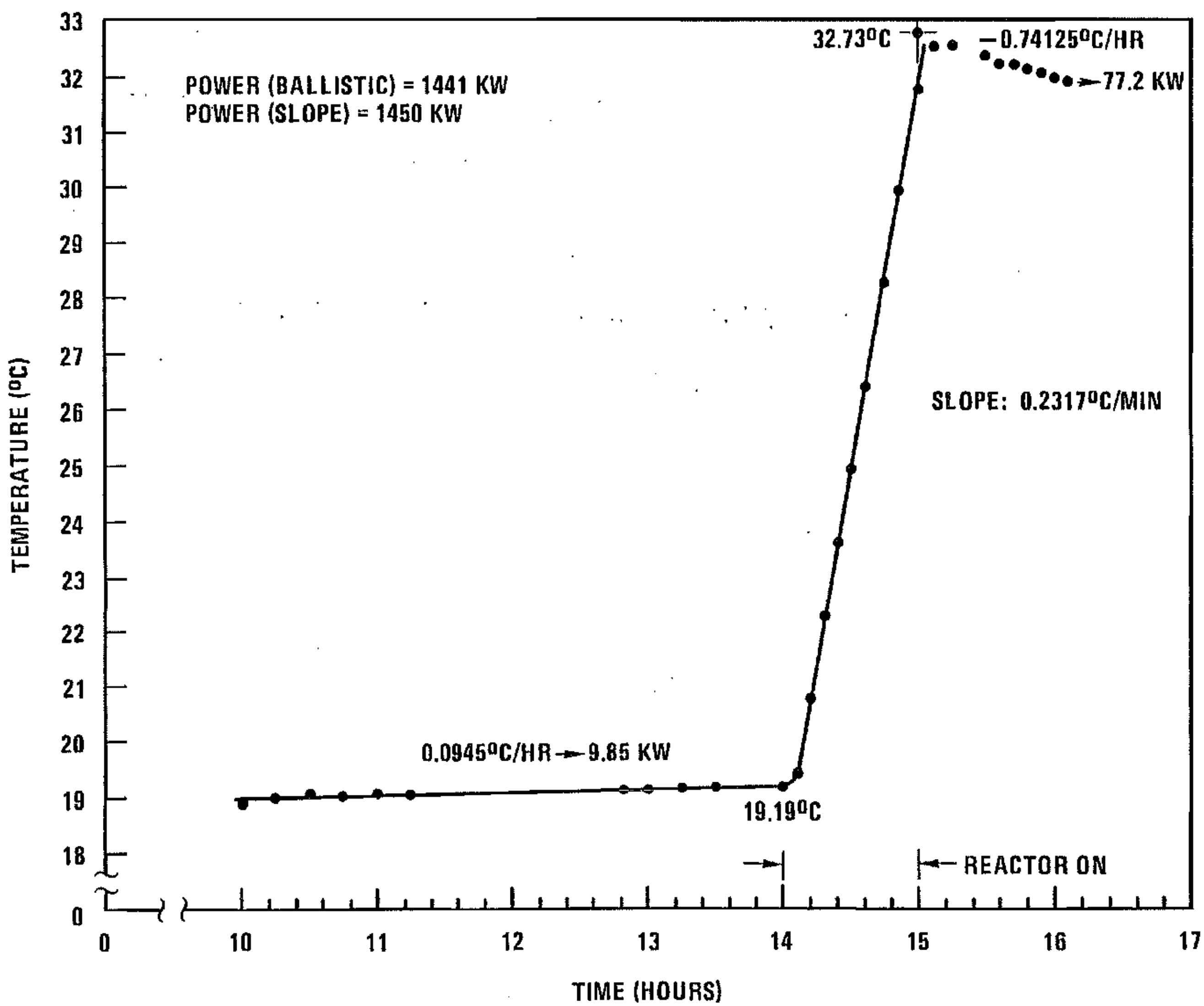

Figure 4. Calibration Curve for Mark F TRIGA Reactor (≈1450 kW).

slope method value it would drive the percentage difference from 0.6% to 3.3% whereas for the case discussed in Section 3.1 the leakage correction applied to the slope method value reduced the percentage difference from 2.8% to about 1.4%.

It can be concluded that the value of power determined by the slope method (uncorrected for heat leakage from the tank water) can differ from the best estimate (ballistic method value) approximately ± 2-4%. The results above assure that the standard slope method when applied with reasonable care will provide reliable values of the reactor power.

The results of the above demonstrations show that with adequate stirring of the tank water, the power calibrations can be conducted at the higher power levels, even at just below full power. During the past year several power calibrations of the 1.5 MW Mark F reactor have been successfully conducted in the power range 1.25 - 1.4 MW. The reproducibility under these circumstances has been excellent, to with ± 1.5%.

3.3 Effect of the Tank Water Temperature on Indicated Power

Experiments were conducted to evaluate the effect of reactor tank water temperature on <u>indicated</u> power for both the graphite reflected TRIGA core (Mark I) and the water reflected core (Mark F). Both of these reactors are operated at the GA TRIGA Reactors Facility.

3.3.1 <u>Mark F Reactor</u>. In the water reflected Mark F reactor currently instrumented for the thermionic testing program (Ref. 5), three in-core rhodium Self Power Detectors (SPD) have been installed to determine the in-core flux level (hence, power level) to within ± 1%. This instrumentation is comparable to the sensitivity to power variations exhibited by the thermionic devices and provides the reactor operator with the capability to monitor in-core power levels to 1 percent.

In earlier power calibrations, the indicated power as measured by out-of-core ion chambers was held constant during the power calibration although it was recognized that the actual core power level was continuously falling during the calibration. Figure 5 is a strip chart trace of the output from the three in-core Self Power Detectors. As indicated on the chart, the estimated in-core power decrease ranged from 4.5 to 5.2% during the power calibration conducted at constant indicated power. The tank water temperature varied from 18 to 36°C giving an 18°C variation during these measurements. Recently, the power calibration has been conducted by operating the reactor so as to hold the in-core SPD readings constant. Of course, the out-of-core monitors slowly rise during such operation. The power calibration is terminated without shutting down the reactor. The final water temperature T_2 after the calibration is selected by prior planning to be the desired long term pool temperature (36 ± 1°C in the present case). With this procedure the reproducibility of the in-core power calibration is ± 1.5% from calibration to calibration. The even more highly instrumented thermionics devices themselves confirm this precision in reproducing the power levels.

3.3.2. Mark I Reactor. The graphite reflector around the Mark I and Mark II reactors reduces the magnitude of the variation in indicated power with change in pool water temperature compared to that for a water reflected core. An experiment was conducted to determine the magnitude of this variation. Unfortunately, the Mark I reactor was not equipped with in-core SPD instrumentation. An alternative approach was used based on the fact that the power coefficient of the Mark I reactor at or near full power is about 1 kilowatt per 1 cent change in reactivity. From the change in core reactivity the actual change in reactor power was evaluated.

During a routine power calibration of the Mark I reactor, and with the indicated power held constant at about 200 kW, the control rod positions were carefully monitored as a function of the rising pool water temperature. The measurements were conducted over the temperature range from 16 to 38°C. These measurements showed a 5 cent change in core reactivity. Figure 6 shows that the change in reactivity varies linearly with water temperature from about 16°C. After the calibration was completed, a separate run was conducted to determine the change in reactor power (at about 200 kW) produced by inserting 5 cents of reactivity. The 5 cent change in reactivity

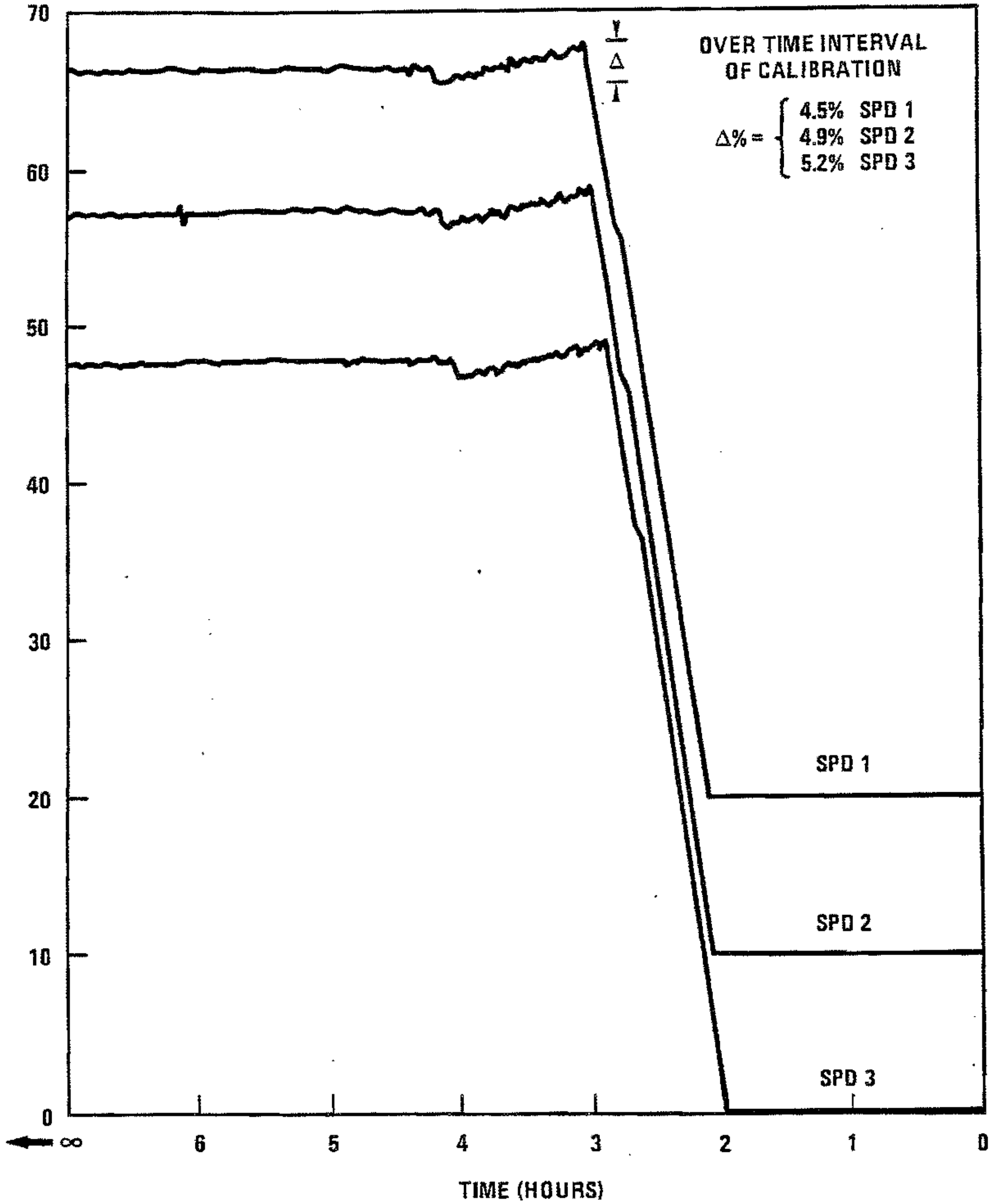

Figure 5. Output from In-Core Self Powered Detectors Showing Decrease of Reactor Power with Constant Out-of-Core Power Monitors.

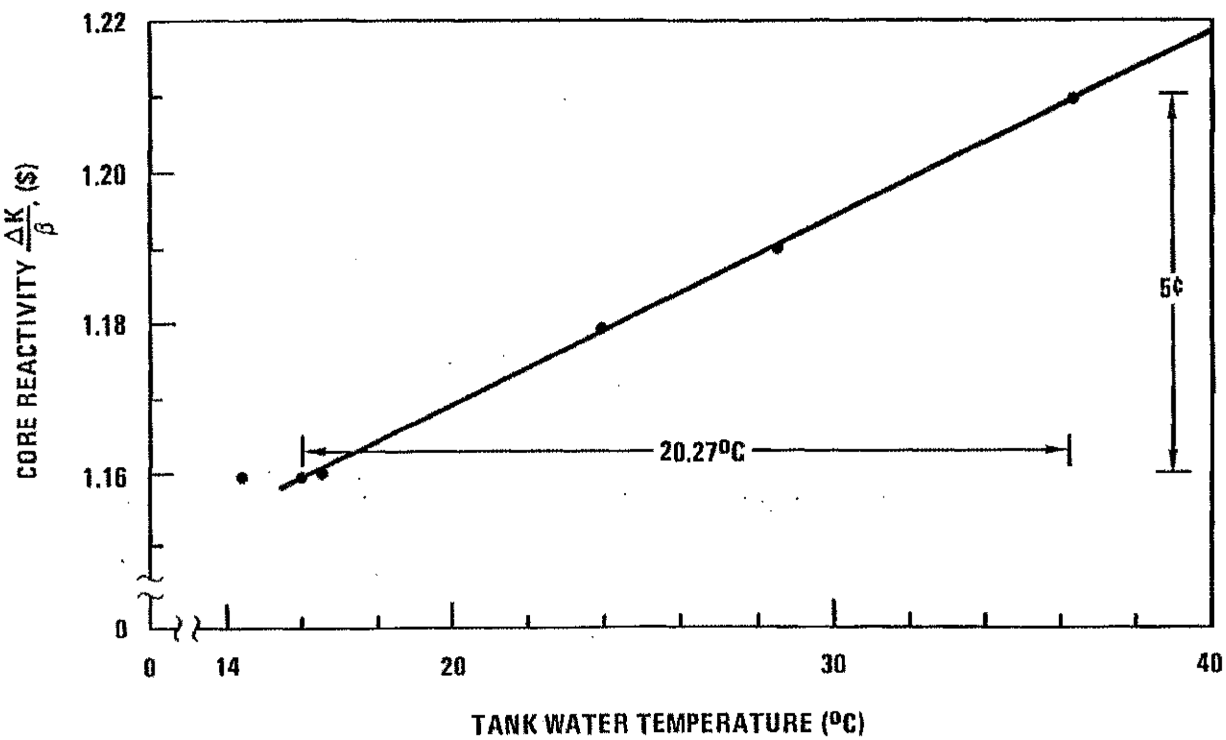

Figure 6. Core Reactivity ($\Delta K/\beta$) as a Function of Tank Water Temperatures for Graphite Reflected Core.

resulted in a 6.3 kW change in power. Thus

$$\frac{\Delta\phi}{\phi} = \frac{\Delta P}{P} = \frac{6.3}{200} \simeq 0.0315 \text{ or } 3.15\%$$

It may be noted that the temperature change of 20°C is probably larger than experienced during most operations of a Mark I or Mark II reactor. Consequently, the variation in actual power with water temperature when the out-of-core power monitors indications are maintained constant will be significantly less then 3%.

3.4 Temperature Probe Location

At the GA reactors the measurements of the pool water temperatures are made with thermistor probes read by a precision unit to one-hundredth of a degree (DIGITEC). Up to three separate probes are usually recorded at the end of each 2-minute time interval. The advantage of using several temperature probes positioned at different locations and depths in the tank is that it is quicker this way to identify a position that exhibits better mixing. Somewhat erratic mixing may lead to cyclical variation of the rising temperatures. Even in these cases, the time averaged slope of temperature versus time will usually agree to within 1 percent. The data in Figure 7 show the results for two probes. The least squares fit to each set of data points gives results that differ only 1.1%.

The exact location of a temperature probe to provide the smoothest set of data points is not usually predictable because of the variations of flow patterns within the tank. Once a suitable location is found, this location usually remains suitable for future calibration runs provided that other conditions remain the same including the position and angle of the motorized stirrer.

3.5 Accuracy and Precision of Results

In the above treatment much attention has been devoted to those conditions that improve the reproducibility and hence precision of the results.

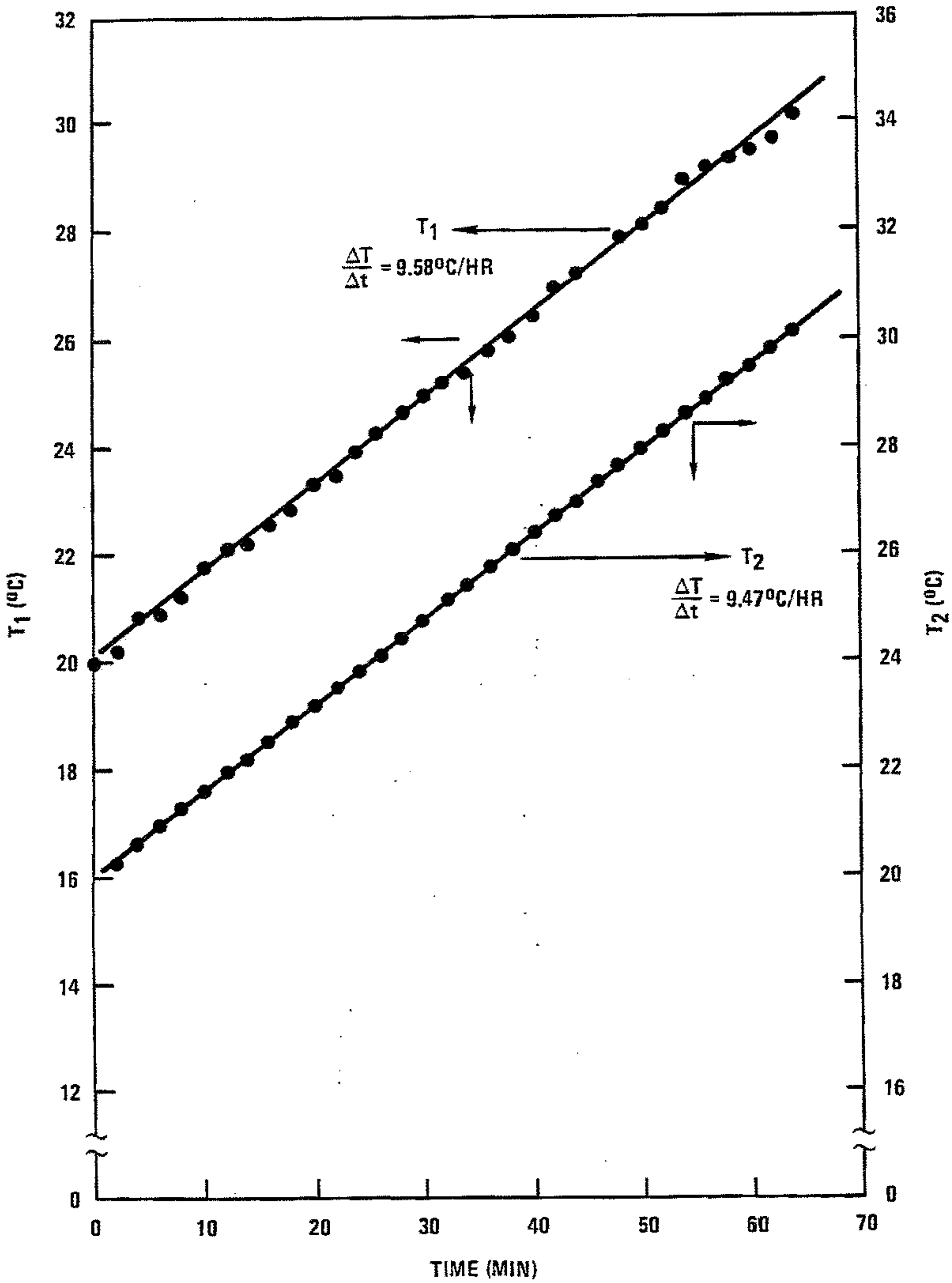

Figure 7. Plot of Two Temperature Probe Data Sets Illustrating Effect of Probe Location on Time Dependence of Temperature Data.

As shown herein, the precision of the results can approach ± 1.5%. To assure this degree of reproducibility, we have prepared a brief checklist of the items to be prepared prior to a calibration run. Table 2 presents these data for a typical calibration run at GA. Other facilities would require some alteration to accommodate differing circumstances.

We estimate that the <u>accuracy</u> of the power as determined by the calorimetric technique is worse than the demonstrated precision of the data. In earlier times, the accuracy was thought to be provided through use of electrical heaters to remove dependence on the actual heat content of the system. In that method the reactor was operated at the same 10-15 kW used for the electrical heaters. Temperature measurements were used to verify the same rate of temperature rise. Full power was then determined on the basis of an extrapolation by a factor of 10 to 20 or more. Careful current source measurements were then needed to determine accurately the linearity of the console power measurement instrumentation. However, unless careful pool stirring was used in the initial operation at 10-15 kW, the accuracy of determining the exact reactor power that matched the 10-15 kW electrical heating rate may not have been high. Without great care in all steps of the procedure, the overall accuracy at full power may have ranged between 5 and 10%.

An estimate of the accuracy of the calorimetric procedures described herein can be made from the factors in the basic formulation.

$$P = \frac{dE}{dt} = mC \frac{dT}{dt}$$

The measurement of temperature and time are made with considerable accuracy and precision. The rate of temperature change with time when calculated from the data obtained with several calibrated probes used simultaneously can also be determined with an accuracy of better than 1%. For the determination of power from the slope, the main source of error probably lies in the proper determination of the heat content of the system. Most of this comes from an uncertainty in the exact volume of the water in the system. As an example, an error of 1 inch in the tank diameter of 6.5 feet (20 foot

Table 2. Calorimetric Power Calibration Checklist

Date ____________________ Logbook No. ___________

Operator____________________

1. Prestart Checklist for Power Calibration

2.1 Reactor tank water level ____________ Add water __________

2.2 Experiments: In-core ________________ Out-of-core ____________________

2.3 Water treatment system off __________ Isolated ____________

2.4 Pit cooling system off ______________ Isolated ____________

2.5 Diffuser pump on ______ Canal pump on _______ Tank Stirrer on ______

2.6 Temperature probe locations: T1 ____

T2 __________________________________ T3 ____

2.7 Starting pit temperature (P<1kw) T1 _________ T2 _________ T3 _________ °C

3. Power Calibration Data

3.1 Reactor power ___________ kw on channel ______ reached at _______ hrs

3.2 Instrument readings during calibration (record temperature data on page 2)

Time	K1	K2	K3	SPD1	SPD2	SPD3

4. Power Calibration Results

4.1 Calculated Power from Least Squares Fit _______________ kw (9.6°C/hr/Mw)

4.2 Calculated Calibration Factors: K1 _____________ a/w K2 ______________ a/w

K3 _____________ a/w

4.3 Calibration factors verified and posted by __________ and _____________

5. Visual Inspection of Tank____

6. Remarks

depth) gives an uncertainty of 2.5%. Careful accounting of the metal components in the tank either by deducting their volume from the water or including them with their very small heat capacity can probably be done to an accuracy of 0.5%. Another quantity to estimate is the effect of heat leakage into or out of the tank through walls (0.5-0.6% for ballistic method; ≈ 1.3-2.6% for slope method) and from the various stirrer and pump motors. These motors may add on the order of 1 kW of heat which is a 0.1% correction for a 1 MW power level. As noted earlier, careful selection of the initial and final temperatures with proper regard for the ambient tank wall temperature can greatly minimize the already small heat flow through the walls. This is particularly important for the slope method of determination because no means normally exist to make this correction. To this end some TRIGA facilities have installed locations where probes can be used to measure the temperature of the material surrounding this water tank.

4.0 Conclusions.

With the information on estimated accuracies and precision discussed above and elsewhere in this paper, we make the following conclusions:

1. The major errors in the slope method of power calibration lie in the uncertainty: in the tank water volume (2-4%), in the tank wall heat losses (0-2.5%), and in the determination of the slope, °C/hr (0.5-1.0%).

2. The precision (repeatability) of the slope method has been demonstrated to be of the order 1.5%.

3. The major errors in the ballistic method of power calibration involve uncertainty: in tank water volume (2-4%), in the pre- and post- irradiation tank losses (0 - 1.0%); and in the total temperature change ΔT during the irradiation (≈ 0.2°C/10°C ≃ 2%).

This review has lent support to the belief that calorimetric power calibrations can be performed on a TRIGA reactor with an accuracy in the range 5-10% and with a precision of ≈ 1.5%.

REFERENCES

1. The calibration process was first described in Logbook Number 01 (April 29, 1958).
2. S. Glasstone, Principles of Nuclear Reactor Engineering, Van Nostrand (1960) p 22.
3. E. Blizzard, Editor; Reactor Handbook, Vol III Part B, Shielding, Interscience Publishers (1962) p 29.
4. H. Etherington, Editor, Nuclear Engineering Handbook. McGraw Hill Book Company (1958), Section 1-1, p 1-39.
5. J. Razvi and W. Whittemore, "In-Pile Testing on Thermionic Space Nuclear Power Fuel Using a TRIGA Reactor," presented at the 11th TRIGA Owner/User Conference, April 1988.

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1. PURPOSE AND DISCUSSION

   This exercise will demonstrate changes in power and temperature as a function of prompt reactivity.

2. EXERCISE OBJECTIVES

   Following completion of this exercise, participants should be able to:

   2.1 Enable “Reactor Pulse” or “Square Wave” mode

   2.2 Perform a TRIGA pulse to various power levels

   2.3 Explain reactor response to various pulsed reactivity additions

3. PREREQUISITES

   3.1 Preoperational Checks complete

4. BACKGROUND/REFERENCES

   4.1 UT Procedure OPER 1, Startup and Shutdown Checks

   4.2 UT Procedure OPER 2, Reactor Startup and shutdown

   4.2 UT Procedure OPER 3, Reactor operation Modes

   4.3 UT Procedure SURV 7, Comparison of Pulse Characteristics

Comment [#1]: Identify TS related to pulsing

5. INSTRUCTIONS

   5.1 In accordance with OPER 2, PERFORM reactor startup to 50 Watts

   5.2 POSITION the transient rod to produce the designated reactivity insertion

   5.3 PERFORM reactor pulse

   5.4 RECORD:

   - Peak Power
   - Integrated Power
   - Maximum Temperature

- Minimum Period

Reactivity Insertion	Peak Power	Integrated Power	Maximum Temperature	Minimum Period
1.20				
1.50				
1.80				
2.00				

6. REINFORCEMENT & REVIEW

   Plot maximum power level against reactivity insertion

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
2.1	Safety limit Fuel Temp	Cladd<500°C, <1150°C			
2.1	Safety limit Fuel Temp	Cladd>500°C, <950°C			
2.2.2	Power level	1100 kW			
3.2.3.b	Safety System	<2000 MW P	4.2.3	Annual or retest	
			4.2.3		
3.2.3.c	Safety System	HVPS	4.2.3	Annual or retest	
			4.2.3		
3.2.4.b	Power level	2 channels			
3.2.4.c	Pulse power	1 channel			
3.2.4.d	Pulse energy	1 channel			

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1. PURPOSE AND DISCUSSION

   This exercise will demonstrate experiment proposal, review, insertion, and removal.

2. EXERCISE OBJECTIVES

   Following completion of this exercise, participants should be able to:

   2.1 Propose an experiment

   2.2 Review an experiment

   2.3 Insert an experiment

   2.4 Remove an experiment using proper radiological precautions

3. PREREQUISITES

   None.

4. BACKGROUND/REFERENCES

   4.1 Basic Nuclear Principles 1 *Atomic Number Density*

   4.2 Basic Nuclear Principles 2 *Neutron Cross Sections*

   4.3 Basic Nuclear Principles 3 *Neutron Reactions*

   4.4 Basic Nuclear Principles 4 *Radioactive Decay*

   4.5 Basic Nuclear Principles 5 *Transmutation*

   4.6 UT Procedure OPER 2, Reactor Startup and shutdown

   4.7 UT Procedure OPER 3, Reactor operation Modes

   4.8 UT Procedure ADMIN 6, Authorization of Experiments

   4.9 UT Procedure FUEL 2, Experiment Movement

   4.10 ATTACHMENT 8.1: Relevant Technical Specifications

   4.11 ATTACHMENT 8.2: Experiment Request Form

5 INSTRUCTIONS

Comment [#1]: Provide an overview of activation

Comment [#2]: Review experiment TS requirements

Comment [#3]: Review request form; identify what information is required, how it is obtained or calcualted

5.1 COMPLETE experiment request form

5.2 INSERT experiment

5.7 OPERATE reactor

5.8 REMOVE Experiment

6 REINFORCEMENT & REVIEW

None.

SECT	DESCRIPTION	SPECIFICATION	LCO	FREQ	Procedure/record
3.4.1.a	Experiment reactivity	Moveable, <$1.00	4.4.1		
3.4.1.b	Experiment reactivity	Secured <$2.5	4.4.1		
3.4.1.c	Experiment reactivity	Total possible < $3.00	4.4.1		
3.4.2.a	Experiment Materials	Corrosive, reactive, explosive double encapsulated	4.2.2		
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, remove			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, inspect			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, corrective action			
3.4.b	Experiment Materials	Encapsulation failure with potential consequences, Director review prior to operation			
3.4.c	Experiment Materials	<25 mg explosives			
3.4.c	Experiment Materials	Pressure calculation (or exp) within capsule design			
3.4.d	Experiment Materials	<750 mCi I131-I135, <2.5 mCi Sr (fueled)			
3.4.e	Experiment Materials	<MPC for 100% release if volatile			
3.4.f	Experiment Materials	3.4.e calc assume 10% release			

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PA Format.doc		**Date:** 8/22/02
Attachment	**Number – Rev:**	OPER-1 1.01
a:\oper1-a2.doc	**Procedure Title:**	Startup – Shutdown Checks

OPERATION REQUEST

*Request No.*_________

*Exp #*______________

NETL Tracking No. _______
☐ *Research*
☐ *Service*
☐ *Internal*

Date: ____/____/____

Requested by: ____________________Phone:____________

Project Description:

Various operations for Remote Lab.

Mode of Operation ☐ Manual ☐ Auto ☐ Square

Power Level: _______kW

Time at Power: ______hr

☐ Pulse

Reactivity $'s______

Number: 0

Date Needed: ____/____/____

Expected Completion: ____/____/____

☐ **Class A or** ☐ **Class B, Experimenter** Steven Biegalski

☐ In-Core (Irradiation): ☐ CT ☐ RSR ☐ 3L (Cd) ☐ 3L (No Cd) ☐ ePNT (Cd) ☐ tPNT (No Cd) ☐ PNT-GA ☐ Other

☐ Ex-Core (Exposure): Beam Port #:_________ Other:_________

☐ New Experiment (Safety Evaluation Required)

☐ **Class C experiment, Operator:**________________ **Experimenter:**__________________

☐ Non-Reactor: Room Number ________________

☐ Experiment in Reactor ☐ Pool ☐ Area

NOTES/SPECIAL INSTRUCTIONS:

Reactor Operation Request		Stamp(Original-Red, Copy-Blue)
Date of Change	|8/22/02|_____|_____|_____|	
NETL Director Approval	|__SL__|_____|_____|_____|	**Page 1 of 2**

PA Format.doc		**Date:** 8/22/02
Attachment	**Number – Rev:**	OPER-1 1.01
a:\oper1-a2.doc	**Procedure Title:**	Startup – Shutdown Checks

Material Evaluation

☐ Exposure (ex-core)
☐ Irradiation (in-core)
☐ No Samples

*Request No.*_________

*Exp #*______________

NETL Tracking No. _______
☐ ***Research***
☐ ***Service***
☐ ***Internal***

Sample Description:
____
____

# of Samples: 0______ Composition: ☐ solid ☐ liquid ☐ gas AND ☐ Biological ☐ Geological ☐ Engineering (Act. Calc. Req'd)

Avg. Mass: _______ g (solid)
_______ ml (liquid)
_______ cc (gas)

HAZARDS: ☐ Toxic ☐ Explosive ☐ Gamma Activity ☐ Other (explain)
☐ Corrosive ☐ Combustible ☐ Neutron fission
☐ Volatile ☐ Beta Activity ☐ Neutron absorber

Element: Major Isotopes: ____
Trace Isotopes: ____

Encapsulation: (Liquids require double encapsulation. Other hazards require special consideration.)

Type	# of Samples	1539-LG	2/5 dram	2 dram poly	Heat seal	PNT rabbit	RSR Rabbit	Poly bag	Aluminum	Quartz	Glass	Other	None
Solid													
Liquid													
Gas													

OPERATIONS USE ONLY
*Time of operations (hrs):*____________
*Setup and breakdown time (hrs):*______
Total time (min. 1.0 hour):___________

LABORATORY USE ONLY
*Sample Preparation (hrs):*____________
*Irradiation/Counting (hrs):*___________
Analysis (hrs):_______________________
Lab Manager Approval______________

ADMIN USE ONLY:
Cost:____________________________
Account #: ________________________
Billed:____________________________
Payment Received: _________________

Customer Information
____
____
____
____

RESET

SAVE

SUBMIT

Operations Approvals and Review (Reactor Supervisor Signature)
Experiment Type: ☐ Authorization (New) ☐ Special ☐ Routine
Approval:____________________Date:_______Review:___________________Date:_______

Reactor Operation Request Stamp(Original-Red, Copy-Blue)

Date of Change	8/22/02			
NETL Director Approval	SL			

Page 2 of 2

NUCLEAR ENGINEERING TEACHING LABORATORY

PROCEDURE, Revision 1.00

AUTHORIZATION OF EXPERIMENTS

Approvals:

Thomas L Bauer — 7/14/92
Reactor Supervisor — Date

Bernard W. Wehring — 7/22/92
Director, NETL — Date

Randall J Charbeneau — 1/15/93
Chairperson, Reactor Committee — Date

[signature] — 1/27/93
Chairperson,
Radiation Safety Committee — Date

List of Pages: 1 2 3 4

Attachments:
Exp. Review Guide 1 2 3 4 5
Authorization Form
Operation Request
Sample Irradiation - Exposure
Non Reactor Experiments

BALCONES RESEARCH CENTER
THE UNIVERSITY OF TEXAS AT AUSTIN

ORIGINAL

Step	Action and Response	Comment or Correction

I. Purpose

The purpose of this procedure is to establish specific controls to review and analyze experiments. The process applies prior to the use of any experiment in the reactor and subsequent to initial operation to evaluate the routine application of the experiment.

II. Description

Reactor safety is a function of 3 basic physical conditions, (1) the reactivity available for changing the reactor criticality conditions, (2) the effects of temperature and hydraulic flow conditions that change coolant flow or neutron peak powers and (3) mechanical stress that might rearrange structures or components of the core configuration. An evaluation of each of the materials that will be in each experiment is done to identify both operational hazards and possible potential hazards. Limits will be set on experiments to assure that the proper safety conditions are met. Procedures may be necessary for some experiments to assure safe reactor and experiment operation.

III. References

Reg guide 2.2
ANS 15.1 Technical Specifications
Docket 50-602 Safety Analysis Report
Docket 50-602 Technical Specifications
10CFR 50.59 Changes, Tests, and Experiments

Date of Change:	4/4/02			
Change Approval: NETL Director	RSC			

II. PROCEDURE

A. Instructions

1. Submit experiment request to the Supervisory Operator (class A; SRO). All experiment requests involving materials placed in the pool or exposed to direct radiations from the pool require authorization.

2. Determine experiment description; operation requirements, class (A, B, C), facility, materials, estimate times, and the experiment type (special or routine).

3. Review the experiment:

3.1 Special Experiment - Nuclear Reactor Committee and Reactor Supervisor or class A operator (SRO) shall:
(a) Review experiment request for approval. Request is to be comparable to the guidance criteria.
(b) Refer to <u>Experiment Review</u>.
(c) Document review on Experiment Authorization form.
(d) Attach the analysis and any special procedures to the authorization form as a file record.
(e) Authorize approval as a special experiment by signature of the Supervisory Operator and by designated member of committee.

3.2 Routine Experiments - Reactor Supervisor or Class A operator (SRO) shall:
(a) Verify experiment conditions for approval. Conditions are to be equivalent to the experiment authorization.
(b) Refer to <u>Experiment Review</u>.
(c) Complete applicable Operation Request form, Sample
(d) Note any deviations from the authorization and any special safety hazards or instructions.
(e) Authorize experiment by signature of supervisory operator.

3.3 Minor deviations from the routine experiment may be approved although routine deviations shall require experiment amendment and reactor committee approval.

3.4 Operations for operator training, demonstrations, maintenance, or surveillance per reactor committee approved procedures does not require the existence of an experiment authorization. The SRO shall assign an appropriate experiment designation from the "Schedule of Experiments" for each activity.

4. Verify operator's and experimenter's knowledge of experiment and procedures.

5. Perform the experiment following procedures specified by the experiment authorization.

6. Review experimental results:

6.1 Special experiments - Nuclear Reactor Committee and Reactor Supervisor or class A operator (SRO) shall:
(a) Review experiment results by comparison to guidance criteria.
(b) Document comments on Experiment Authorization form.
(c) Authorize approval as a routine experiment by signature of the Supervisory Operator and by designated member of the committee.

6.2 Routine Experiments - Reactor Committee should:
(a) Verify experimental results are equivalent to the experiment authorization.
(b) Review should be noted by signature of the Supervisory Operator on applicable forms (Operation Request, etc.)

6.3 Reclassification as a routine experiment may not be appropriate for certain types of experiments that are not intended for periodic applications.

B. Experiment Classes:

1. Class A experiments require a senior operator (Class A, SRO) to direct an activity or experiment.
2.. Class B experiments require only an operator and if necessary an experimenter(Class B, RO) to perform the experiment, with an SRO available.
3.. Class C experiments are all non-reactor experiments.

C. Experiment Types:

1. A special experiment is an experiment which is authorized for one application.
2. A routine experiment is an experiment which is authorized for repeat applications.

Safety Analysis of Experiments

Descriptive Information

(1) Experiment title
(2) Description and purpose of experiment
(3) Experimental requirements:
  (a) Experiment facility and location
  (b) Maximum reactor power
  (c) Maximum operation time

The experiment review should evaluate each of the credible physical experiment effects and the possible material hazards. Document appropriate analysis for each experiment. Guidance of this review is similar to Regulatory Guide 2.2. Specific conditions of the Technical Specifications shall control all experiments. Experiments that do not meet the conditions of this review shall require reevaluation of the Safety Analysis Report and the Technical Specifications.

Physical Experiment Effects

(1) Reactivity
  (a) Evaluate magnitude of each experiment's reactivity
    (i) Static Reactivity (Measurable experiment reactivity resulting from normal experiment movement to or from reactor core).
      Limits: Compare estimate with actual measurement prior to functional acceptance of experiment.
    (ii) Potential Reactivity (Maximum experiment reactivity resulting from accident conditions such as abnormal movement, voiding, flooding, etc).
      License Limits: Single Moveable Experiment ≤ $1.00
      Single Secured Experiment ≤ $2.50
      Sum of all Experiments < $3.00
  (b) Positive step reactivity insertion of each secured or removeable experiment's potential reactivity will not cause transient leading to excess doses.
    Dose Limits: 10 CFR Part 20
  (c) Positive step reactivity insertion of each moveable or unsecured experiment's potential reactivity will not cause a safety limit or minimum shutdown margin violation.
    Safety Limit: Fuel Temp 1150°C at Clad Temp < 500°C
    Fuel Temp 950°C at Clad Temp > 500°C
    Min. Shutdown Margin: 0.2% ($0.14) with
    (i) Core in reference configuration
    (ii) Most reactive control rod fully withdrawn
    (iii) Highest worth experiment in most reactive state

Step	Action and Response	Comment or Correction

(d) Control system's ability to compensate for reactivity insertions resulting from intentionally moving any combination of unsecured or moveable experiments.
License Limit: One experiment ≤ \$1.00

(e) The sum of the static reactivity experiment worths of all unsecured experiments which coexist should not exceed the lesser of :
   (i) Maximum potential reactivity authorized for a single removeable experiment (sec (a) above)
   (ii) The minimum shutdown margin excluding items (ii) and (iii) (see (c) above)
License Limit: Sum of all experiments ≤ \$1.00

(2) Thermal Hydraulic:
(a) Actual and potential thermal effects on reactor safety
Limit; See 1 (c) above
(b) Flux peaking; flow blockage, redistribution, or phase changes
Limit See 1 (c) above
(c) Experiment boundary surface temperatures leading to:
(i) Reactor coolant phase change
(ii) Elevated corrosion rates
(iii)Material strength reduction
Limits: Dependent on experiment material properties

(3) Mechanical Stress:
(a) Potential storage and possible uncontrolled release of mechanical energy
Limits: Maintain reactor core and fuel element integrity
(b) Potential for projectiles or objects with substantial momentum
Limits: Maintain reactor core and fuel element integrity
(c) Structural ability to withstand external forces generated during installation, operation, or removal and internal forces generated by unintended but credible changes of confined materials
Limits: Capable of operation at twice normal stress anticipated
(d) Requirement for prototype tests
Limits: Experiment dependent

Material Evaluation

(1) Radioactivity:
(a) Quantities and types of materials
(b) Expected isotopes, quantities, and decay modes
(c) Radiation doses resulting from the accidental release of all gaseous, volatile, or particulate components (calculate per Tech. Specs. and Reg Guide 2.2) limit to:

Step	Action and Response	Comment or Correction

(i) For Singly Encapsulated Material - less than 10% of the equivalent annual doses stated in 10CFR20 for persons occupying (1) unrestricted areas continuously for 2 hours starting at the time of release or (2) restricted areas during the time required to evacuate the restricted area.

(ii) For Doubly Encapsulated or vented Materials - (1) 0.5 rem whole body or 1.5 rem thyroid to any person occupying an unrestricted area continuously for a period of 2 hours beginning at the time of release of (2) 5 rem whole body or 30 rem thyroid to any person occupying a restricted area during the time required to evacuated the restricted area.

(d) Presence of fissionable materials which when irradiated will produce isotopes in quantities greater than those specified in the Technical Specifications

Limits: Double encapsulation requirement

Isotopes of I^{131} thru I^{135}	< 750 mCi
Strontium	< 2.5 mCi

(2) Material Hazards:

(a) Trace element impurities which may represent a significant radiological hazard
Limits: Refer to exposure limits

(b) High cross section elements (fuels or absorbers)
Limits: Refer to reactivity limits

(c) Flammable, volatile, or liquid materials
Limits: Seal and test encapsulation

(d) Explosive chemicals
Limits: Less than 25 milligram quantity
Detonation pressure does not rupture container

(e) Corrosive chemicals
Limits: Double encapsulation requirement

(f) Chemicals highly reactive with water
Limits: Double encapsulation requirement

(g) Radiation sensitive materials which when exposed to radiation exhibit degradation of mechanical properties, decomposition, chemical changes, or gas evolution
Limits: Maintain integrity of encapsulation

(h) Toxic compounds
Limits: Personnel safety requirements

(i) Cryogenic liquids
Limits: Specific hazard authorization

(j) Unknown materials
Limits: No authorization for unknown materials

Experiment Classification:

Three classes of experiments will define the type of personnel requirements that are necessary to perform the tasks of each experiment class. Class A experiments will require performance or supervision of all experiment requirements by a Senior Operator. A Reactor Operator may perform the work of a class A experiment but a Senior Operator must review and approve each experiment task prior to continuation of operation. Class B experiments require a Senior Operator only for approval of startup, shutdown, significant changes in reactivity (power level changes that exceed 200 kilowatts) and recovery from any non-intentional scram condition. A Reactor Operator may perform the routine operation tasks of this experiment class. Class B experiments will include two subgroups of experiments that specify whether or not operation coordination is necessary with an experimenter. All other experiments, that do not require the presence of a Senior Operator or Reactor Operator, are class C experiments. A class C experiment may require approval by a Senior Operator or Reactor Operator if the experiment is in the reactor pool or the reactor bay.

The following schedule lists the general classification of experiments. Experiment reviews will document the safety analysis for each type of experiment. If necessary specific reviews or amendments will apply to special types of experiments. Any experiment that substantially deviates from the general classifications will become a new authorization within the appropriate category.

Step	Action and Response	Comment or Correction

Schedule of Experiments

A. Class A Experiment (Senior Operator Supervision)

A1.0 ICS Operation
- 1.1 ICS prestart checks
- 1.2 ICS system calibration
- 1.3 ICS system changes

A2.0 Core reactivity adjustments
- 2.1 Critical mass experiment
- 2.2 Fuel element movements
- 2.3 Control rod elements

A3.0 Radiation shield configurations/ Beam port experiments
- 3.1 Vertical beam ports
- 3.2 Beam ports 1,2, &4
- 3.3 Beam ports 3 & 5

A4.0 Special Projects
- 4.1 Texas Cold Neutron Source

B. Class B Experiments (Reactor Operator & Experimenter Tasks)

B1.0 Routing Operations - training (Reactor Operator Tasks)
- 1.1 Reactivity coefficients - voids and materials
- 1.2 Reactivity coefficients - power and temperature
- 1.3 Step reactivity insertion - positive and negative

B2.0 Routine operations - demonstration (Experimenter Participation)
- 2.1 Power operation
- 2.2 Pulse operation
- 2.3 Special projects

B3.0 Neutron Activation
- 3.1 Neutron activation (long-lived) Reactor Operator task
- 3.2 Neutron activation (short-lived) Experimenter participation
- 3.3 Special projects

B4.0 Isotope Production 3-element irradiator
- 4.1 Isotope production (long-lived) Reactor Operator task
- 4.2 Isotope production (short-lived) Experimenter participation
- 4.3 Special projects

B5.0 Reactor core exposures

B6.0 ~~Beam port exposures~~ Other

C. Class C Experiments (Non reactor experiment)

C1.0 Gamma irradiator

C2.0 Subcritical assembly

C3 0 Neutron generator

C4.0 Portable ~~x~~ray unit

Date of Change:	4/4/02			
Change Approval:	LBC			
NETL Director				

ORIGINAL

COPY

Number	Title	Rev. 1
ADMN-6	Authorization of Experiments	Date 12/90

Experiment Authorization

Date: _____/_____/_____ Class _______________

Requested by: ______________________________ Phone _______________ No. _______

Experiment Title:

Physical experiment effects:

Reactivity Extimates (in cents) ________ static ________ potential

Thermal Hydraulic ______________________________

____

Mechanical Stress ______________________________

____

Material Evaluation:

Radioactivity Material Hazards

Isotopes (major) _______________ _______________ Limits: ________

Activity (max) _______________ _______________ Limits: ________

Dose ( )_____mR/hr ____cm ____hr _______________ Limits: ________

Review of Safety Questions:

i) increases probability or consequence yes___ no___

ii) creates different type safety condition yes___ no___

iii) reduces margin of safety yes___ no___

Procedure Requirements:

Experiment Restrictions:

Special Experiment Results: Date Performed _____/_____/_____

____

Experiment Approvals:		Special	Routine
Reactor Supervisor	______________________	___/___/___	___/___/___
Health Physicist	______________________	___/___/___	___/___/___
Laboratory Director	______________________	___/___/___	___/___/___
Nuclear Reactor Comm.	______________________	___/___/___	___/___/___

NUCLEAR ENGINEERING TEACHING LABORATORY

FUEL-2, REV. 0

PROCEDURE
FOR MOVEMENT OF EXPERIMENTS

Approvals:

Thomas L. Bauer — 7/30/91
Reactor Supervisor — Date

Bernard W. Wehring — 7/30/91
Director, NETL — Date

7/30/91
Chairperson, Reactor Committee — Date

7/30/91
Chairperson,
Radiation Safety Committee — Date

List of Pages: 1 2 3

Attachments: None

BALCONES RESEARCH CENTER
THE UNIVERSITY OF TEXAS AT AUSTIN

I. PURPOSE

The purpose of this procedure is to control experiment facility or experiment movements that may cause reactivity changes to the reactor core.

II. DESCRIPTION

Setup or removal of reactor core experiment facilities and experiments can cause substantial changes in the core configuration reactivity. Knowledge of these reactivity changes, both magnitude and sign, and the measurement of these changes is necessary to approve any configuration for safe operation.

III. REFERENCE

Safety Analysis Report,
docket 50-602
Technical Specifications,
section 3.4 limitations on Experiments

IV. MATERIALS

Radiation Work Permits (RWP) - for work within the reactor pool access area, or for special experiments.

V. PROCEDURE

1. A licensed operator shall supervise all experiment facility or experiment movements in the reactor pool.

2. A careful examination of the reactivity consequences of any experiment or facility movement shall be reviewed.

3. Reactivity effects greater than $1.00 shall require supervision by a licensed senior operator; reactor startup checklist shall be performed and k-excess adjustments made as necessary.

4. Removal or replacement of experiment or facilities into or from the reactor core shall be recorded in the reactor logbook; a k-excess measurement shall be made at time of subsequent reactor criticality.

5. All experiments in the reactor tank shall be secured as required by reactivity constraints. Experiments or objects in the reactor pool that represent no reactivity effect shall be secured as necessary to prevent potential interference with reactor operation.

6. A beta-gamma survey shall be made of all objects or experiments removed from the pool; radiation tags and wipe tests should be used as necessary.

    a. Check the requirements of any extended or fixed RWP for work in the immediate area of the reactor pool access area.
    b. Special RWP's may apply to specific experiments.

Reactivity Estimates ($)

CTR	void vs. water	-0.50
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
RSR	poison 40 places	-0.40
PNT-G1	poison	-0.16
PNT-A1	poison	-0.90

poison is a significant neutron absorbing material

ATOMIC NUMBER DENSITY

Number of Atoms (n) and Number Density (N)

The number of atoms or molecules (n) in a mass (m) of a pure material having atomic or molecular weight (M) is easily computed from the following equation using Avogadro's number ($N_{Av} = 6.022\ x\ 10^{23}$ atoms or molecules per gram-mole):

$$n = \frac{mN_{Av}}{M} \tag{1}$$

In some situations, the *atomic number density* (N), which is the concentration of atoms or molecules per unit volume (V), is an easier quantity to find when the density (ρ) is given

$$N = \frac{n}{V} = \frac{\rho N_{Av}}{M} \tag{2}$$

Number Density for Compounds

For a chemical compound (mixture) Z, which is composed of elements X and Y, the number (atom) density of the compound is calculated from

$$N_Z = N_{mix} = \frac{\rho_{mix} N_{Av}}{M_{mix}} \tag{3}$$

In some cases, the desired quantity is the number density of the compound constituents. Specifically, if $Z = X_p Y_q$, then there are p atoms of X and q atoms of Y for every molecule of Z; hence

$$N_X = pN_Z$$

$$N_Y = pN_Z \tag{4}$$

Example: Calculate the number density of natural uranium in UO_2 with $\rho_{UO_2} = 10.5\ {}^{g}/_{cm^3}$

$$N_U = N_{UO_2} = \frac{\rho_{UO_2} N_{Av}}{M_{UO_2}} = \frac{\left(10.5\ {}^{g}/_{cm^3}\right)\left(6.022\ x\ 10^{23}\ {}^{atoms}/_{mole}\right)}{[238.0289 + 2(15.9994)]\ {}^{g}/_{mole}}$$

$$= 2.34\ x\ 10^{23}\ {}^{atoms}/_{cm^3}$$

Number Density Given Atom Fraction (Abundance)

Oftentimes, it is necessary to compute the concentration of an individual isotope j given its fractional presence (abundance) γ_j in the element

$$\gamma_j = \frac{Number\ of\ atoms\ of\ isotope\ j}{Total\ number\ of\ atoms\ of\ the\ element} \tag{4}$$

Many times, the fraction γ_j is stated as an atom percent, which is abbreviated a/o. The atomic number density of isotope j is then

$$N_j = \gamma_j N_{elem} = \frac{\gamma_j \rho_{elem} N_{Av}}{M_{elem}} \tag{5}$$

If the element has a non-natural abundance of its isotopes (that is, the elemental material is either *enriched* or *depleted*), then it is necessary to compute the atomic weight of the element (M_{elem}) from the sum of all the atomic weights of the isotopes (M_j) rather than use the tabulated M_{elem} value found in a reference

$$M_{elem} = \sum \gamma_j M_j \tag{7}$$

Example: Find the U-235 concentration for 3 a/o in UO_2.
Solution: To solve this example, Equations (4), (3) and (7) are progressively substituted into Eq. (6).

$$N_{U^{235}} = \gamma_{U^{235}} N_{U^{235}} = \gamma_{U^{235}} N_{UO_2}$$

$$= \gamma_{U^{235}} \frac{\rho_{UO_2} N_{Av}}{M_{UO_2}} = \frac{\gamma_{U^{235}} \rho_{UO_2} N_{Av}}{\gamma_{U^{238}} M_{U^{238}} + \gamma_{U^{235}} M_{U^{235}} + 2M_O}$$

$$= \frac{\left(10.5\ {}^{g}\!/_{cm^3}\right)\left(6.022\ x\ 10^{23}\ {}^{atoms}\!/_{mole}\right)}{[(238)(0.97) + (235)(0.03) + 2(16.0)]\ {}^{g}\!/_{mole}} 0.03 \frac{atoms\ U^{235}}{atoms\ U}$$

$$= 7.03\ x\ 10^{20}\ {}^{atoms}\!/_{cm^3}$$

Number Density Given Weight Fraction (Enrichment)

Other times, when working with nuclear fuels such as uranium, the *enrichment* may be specified in terms of weight percent or weight fraction, ω_i, of isotope i:

$$\omega_i = \frac{Mass\ of\ isotope\ i}{Total\ mass\ of\ the\ element} \tag{8}$$

The atomic number density of isotope i is

$$N_i = \frac{\rho_i N_{Av}}{M_i} = \frac{\omega_i \rho_{elem} N_{Av}}{M_i} \tag{9}$$

Clearly, if the material is enriched, then the atomic weight of the material differs from its natural reference value, and the enriched atomic weight, if needed, should be computed from

$$\frac{1}{M_{elem}} = \sum_i \frac{\omega_i}{M_i} \tag{10}$$

Example: Find the U-235 concentration for 4% enriched UO_2.
Solution: First, compute the molecular weight of the enriched uranium, which is basically 4% U-235 and 96% U-238 since the U-234 component is negligible.

$$\frac{1}{M_U} = \frac{\omega_{U^{235}}}{M_{U^{235}}} + \frac{\omega_{U^{238}}}{M_{U^{238}}} = \frac{0.04}{235.04} + \frac{0.96}{238.05}$$

$$M_U = 237.9\ {}^{g}/_{g*mol}$$

Next, use Equation (9) and the fact that $\rho_U = \rho_{UO_2} \frac{M_U}{M_{UO_2}}$

$$N_{U^{235}} = \frac{\omega_{U^{235}} \rho_U N_{Av}}{M_{U^{235}}} = \frac{\omega_{U^{235}} N_{Av}}{M_{U^{235}}} \rho_{UO_2} \frac{M_U}{M_{UO_2}}$$

$$= 0.04 \frac{g * U^{235}}{g * U} \frac{\left(6.022\ x\ 10^{23}\ {}^{atoms}/_{mole}\right)\left(10.5\ g * {}^{UO_2}/_{cm^3}\right)}{235.04\ g * {}^{U^{235}}/_{mole}} \frac{237.9\ g * U}{[237.9 + 2(16.00)] g * UO_2}$$

$$= 9.49\ x\ 10^{20}\ {}^{atoms}/_{cm^3}$$

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Basic Nuclear Principles, Article 2

NEUTRON CROSS SECTIONS

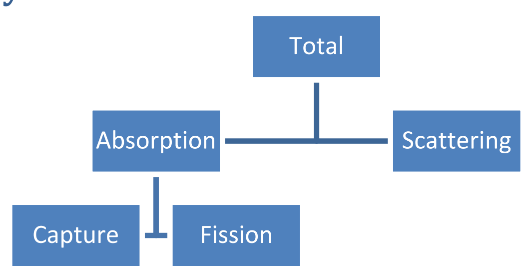
Cross Sections

The *microscopic cross section* (σ) is a property of a given nuclide; σ is the probability per nucleus that a neutron in the beam will interact with the nucleus; this probability is expressed in terms of an equivalent area that the neutron "sees." The *macroscopic cross section* (Σ) takes into account the number of those nuclides present

$$\Sigma = N \cdot \sigma \quad [\mathrm{cm}^{-1}] \tag{1}$$

The mean free path is *mfp* = λ = 1/Σ. The microscopic cross section is measured in units of barns (b): 1 barn equals 10^{-24} cm^2 = 10^{-28} m^2.

Cross Section Hierarchy



$$\begin{aligned} \sigma_t &= \sigma_s + \sigma_a = \sigma_s + (\sigma_c + \sigma_f) \qquad \text{where } \sigma_c = \sigma_\gamma \\ \Sigma_t &= \Sigma_s + \Sigma_a = \Sigma_s + (\Sigma_c + \Sigma_f) \end{aligned} \tag{2}$$

For mixtures of isotopes and elements, the Σ's add. For example

$$\begin{aligned} \Sigma_a^{H_2O} &= \Sigma_a^H + \Sigma_a^O = N_H \sigma_a^H + N_O \sigma_a^O \\ &= 2N_{H_2O} \cdot \sigma_a^H + N_{H_2O} \cdot \sigma_a^O = N_{H_2O} \cdot (2 \cdot \sigma_a^H + \sigma_a^O) \end{aligned} \tag{3}$$

$1/v$ Law

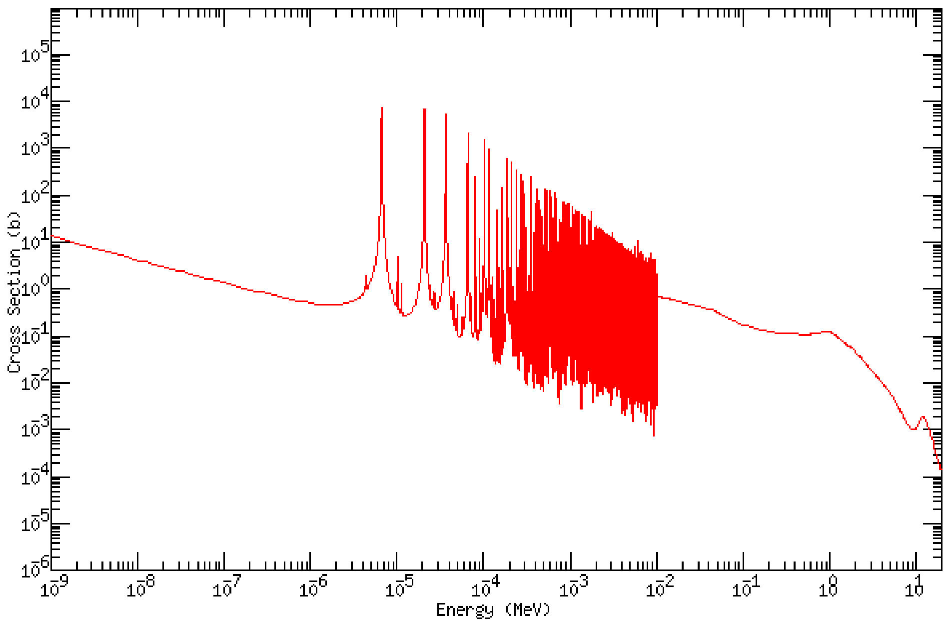
For very low neutron energies, many absorption cross sections are 1/*v* due to the fact the nuclear force between the target nucleus and the neutron has a longer time to interact

$$\sigma_a \propto \frac{1}{v} \propto \frac{1}{\sqrt{E}} \propto \frac{1}{\sqrt{T}} \tag{4}$$

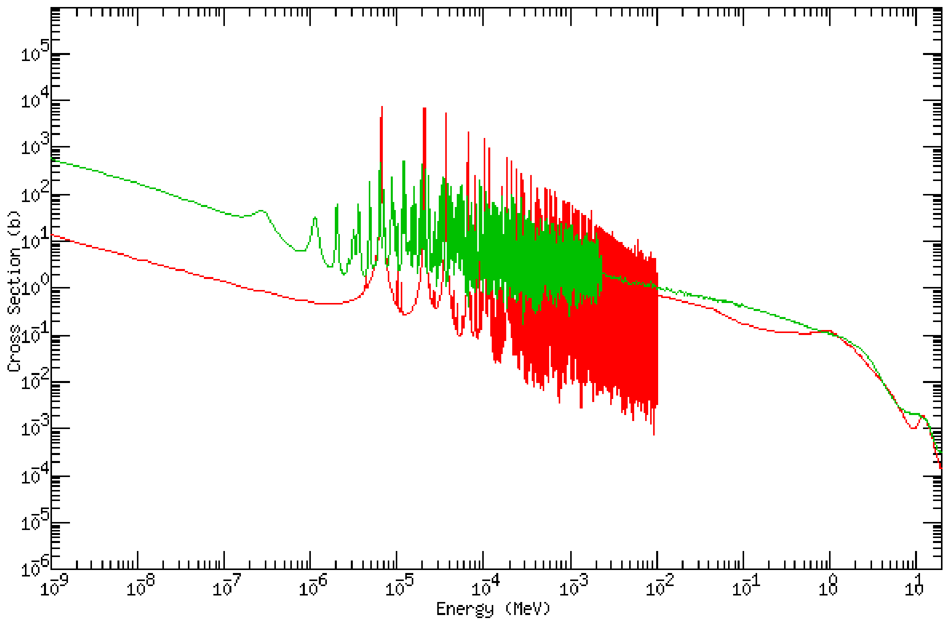
Energy dependence of cross sections

- σ_s is fairly independent of thermal energy (and temperature)
- σ_a (σ_f and σ_c) is highly energy dependent

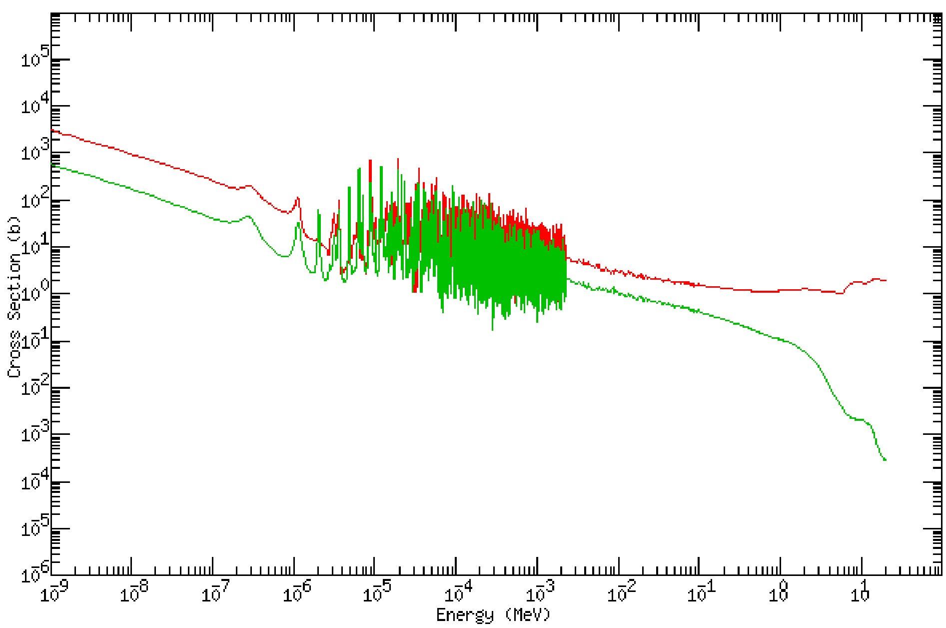
$$\frac{\sigma_a(E)}{\sigma_{a0}} = \frac{1}{v(E)} = \sqrt{\frac{E_0}{E}} = \sqrt{\frac{T_0}{T}}$$



Absorption cross section for U^{238}



Absorption cross sections for U^{235} (green)and U^{238} (red)



Absorption (red) and fission (green) cross sections for U^{235}

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Basic Nuclear Principles, Article 3

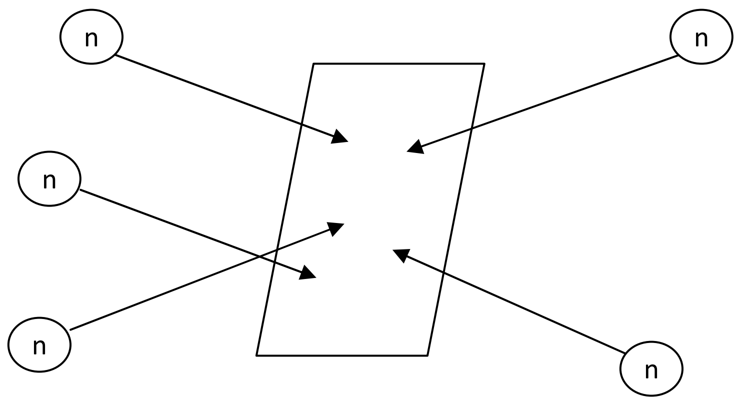
NEUTRON REACTIONS

Neutron Intensity (I) and Flux (φ)

When neutrons are monodirectional, we speak of neutron intensity (I) but when the neutrons are multidirectional we change the nomenclature to flux (φ). Where n is the number of neutrons per unit volume and v is the neutron speed,

$$I = n \cdot v$$

$$\varphi = n \cdot v$$



Fluence (Φ)

Fluence is the time-integrated flux (with unit sof neutrons/cm^3):

$$\Phi = \int \varphi(t)dt$$

The unit "nv" is an outdated measure of flux in n/cm^2-s, and similarly, the unit "nvt" is an archaic measure of fluence in n/cm^2.

Reaction (Interaction) Rate

Knowledge of the neutron flux (φ) and the material cross sections allows us to compute the rate of interactions. The interaction or reaction rate, *RR* [reactions/cm^{3}sec], is based on total number of neutron-nucleus interactions.

$$RR = (n \cdot v) \cdot N \cdot \sigma = \varphi \cdot (N \cdot \sigma) = \varphi \cdot \Sigma$$

The reaction rate for various types of interactions is found from the appropriate cross section type

Total reaction rate $= \Sigma_t \cdot \varphi$
Fission rate $= \Sigma_f \cdot \varphi$

Absorption rate $= \Sigma_a \cdot \varphi$
NOTE: $\Sigma_a \cdot \varphi \geq \Sigma_a \cdot \varphi$

The reaction rate in units of reactions/second can be computed by multiplying *RR* by the material volume. In similar fashion, the total number of reactions can be determined by using the neutron fluence (Φ), rather than the flux:

Number of Reactions $= \Sigma \cdot \Phi$

Neutron Attenuation through a Target

Because neutrons are uncharged, they have the ability to move long distances through matter. Neutron attenuation calculations are very similar to those for photons:

$$\frac{d\varphi}{dx} = -\Sigma_t \cdot \varphi(x)$$

$$\varphi(x) = \varphi_0 \cdot e^{-\Sigma_t \cdot x}$$

Fission Rate and Reactor Power

Where an energy release per fission is defined as E_R, the fission (reaction) rate is related to reactor power level by:

$$RR_f = \frac{P_{RX}}{E_R} = V_{fuel} \cdot \Sigma_{fuel}^{f} \cdot \varphi = V \cdot \left(N \cdot \sigma_f\right) \cdot (n \cdot v)$$

Basic Nuclear Principles, Article 4

RADIOACTIVE DECAY

Radio-Activity

The number of disintegrations or decays per unit time is the product of the probability for a single atom decaying and the total number of radioactive atoms.

$$\lambda \cdot N(t)$$

Fundamental Balance Equation

The rate of change of the amount of radioactive atoms is a function of production and losses. Where no production and some probability of loss that is constant (such as the probability of a set of radioactive atoms decaying, λ), this can be formulated as:

$$\frac{dN(t)}{dt} = -\lambda \cdot N(t)$$

Time dependent behavior can be found by integrating:

$$N(t) = \int_0^T -\lambda \cdot N(t)\ dt$$

Which resolves directly to:

$$ln\left[\frac{N(T)}{N(0)}\right] = -\lambda \cdot 0 - \lambda \cdot T = -\lambda \cdot T$$

or

$$N(T) = N(0) \cdot e^{-\lambda \cdot T} = N_0 \cdot e^{-\lambda \cdot T}$$

Half-Life

The time is takes for ½ of the radioactive atoms can be determined by:

$$ln\left[\frac{A}{2 \cdot A}\right] = ln\frac{1}{2} = -0.693 = -\lambda \cdot T_{½}$$

$$T_{½} = \frac{0.693}{\lambda}$$

Removal Rate

When the qualtity of concern is the amount of radioactive mateiral in some specific location, physcial transport (λ_p) may also contribute to losses:

$$\frac{dN(t)}{dt} = -\lambda_r \cdot N(t) - \lambda_p \cdot N(t)$$

$$\frac{dN(t)}{dt} = -(\lambda_r - \lambda_p) \cdot N(t) = -\lambda_{eff} \cdot N(t)$$

Which gives rise to the term "effective half life," calculated as above (the time it takes for ½ of the radioactive material to be removed).

Mean-Life

The average lifetime of the radionuclide is evaluated by taking a time-weighted average of the time dependent population:

$$\tau = \frac{1}{N(0)} = \int_0^T -t \cdot \lambda \cdot N(t)\ dt = \frac{1}{\lambda}$$

Compound Decay

If a radionuclide decays to another radionuclide, the neutron balance for the 2nd equation contains the 1st radionuclide decay term as a production term. There are a number of equally valid approaches to solving the resultant system of equations, but for one parent (radionuclide *A*) and one product radionuclide (radionuclide *B*) the time dependent behavior of the product is:

$$B(t) = A_0 \cdot \frac{\lambda_A}{\lambda_B - \lambda_A} \cdot \left\{e^{-\lambda_A} - e^{-\lambda_B}\right\}$$

In a similar way, activation of material followed by subsequent decay can be modeled. In the case of activation, the production term is $N_A \cdot \sigma \cdot \varphi$, and it may be necessary to include a term for removal by neutron absorption of $N_B \cdot \sigma \cdot \varphi$.

If $\lambda_A >> \lambda_B$, then $e^{-\lambda_B}$ will approach zero more rapidly than $e^{-\lambda_A}$, and $e^{-\lambda_B}$ can be approximated as 1. This is known as secular equilibrium, as the activity of both components converges to the same value.

If $\lambda_A \approx \lambda_B$, both exponential terms are required to provide correct time dependence

If $\lambda_A << \lambda_B$, the parent is substantially converted to the daughter before significant decay of *B* occurs and in approximation only the second exponential is retained, with the leading constant being the initial concentration of *A* at the decay probability λ_B. This is known as transient equilibrium as the tie dependence of both radionuclides converse, and the convergent values vary by a constant ratio. In a more complex serial decay scheme, a system of linear differential equations can be used to solve for the time dependent behavior of daughter products.

Basic Nuclear Principles, Article 5

TRANSMUTATION

During Irradiation$(t < t_I)$

A "target" with nuclei n_1 is placed in a reactor or similar facility and is exposed to a constant flux of particles, such as neutrons. Some of the target nuclides, n_1, absorb a neutron to form a radionuclide n_2 that here subsequently decays to a stable end product n_3:

$$n_1(target) \rightarrow n_2(radioactive) \rightarrow n_3(stable)$$

Considering the fundamental rate-of-change equation based on production minus losses, there is both
creation and decay of the activation product (n2) according to:

$$\frac{dn_2}{dt} = \sigma_{a,1} n_1(0)\phi - \lambda_2 n_2(t) \tag{1}$$

An assumption is made that the number of target nuclei, n1, remains constant and that there are few
(negligible) neutrons absorbed by the n2 or n3 nuclei. Laplace transforms provide a straightforward solution method, especially since there are zero initial conditions (i.e., there are no n_2 or n_3 atoms to begin with):

$$s n_2(s) - n_2(0) = \frac{\sigma_{a,1} n_1(0)\phi}{s} - \lambda_2 n_2(t)$$

$$n_2(s) = \frac{\sigma_{a,1} n_1(0)\phi}{s(s+\lambda_2)} \tag{2}$$

Inverse Laplace transforming gives

$$n_2(t) = \frac{\sigma_{a,1} n_1(0)\phi}{\lambda_2}\left(1 - e^{\lambda_2 t}\right) \tag{3}$$

The activity of the activation product, n2, is

$$A_2(t) \equiv \lambda_2 n_2(t) = \sigma_{a,1} n_1(0)\phi\left(1 - e^{\lambda_2 t}\right) \tag{4}$$

Where
ϕ = the particle (neutron) flux,
$\sigma_{a,1}$= the microscopic capture (absorption) cross section of the target nuclide,
λ_2= the decay constant of the activation product, and
t= the time since starting the irradiation.

The buildup of a stable decay product, n3, may be described by:

$$\frac{dn_3}{dt} = \lambda_2 n_2(t) \tag{5}$$

Laplace transforming and substituting the expression for $n_2(s)$ from Equation (2) yields

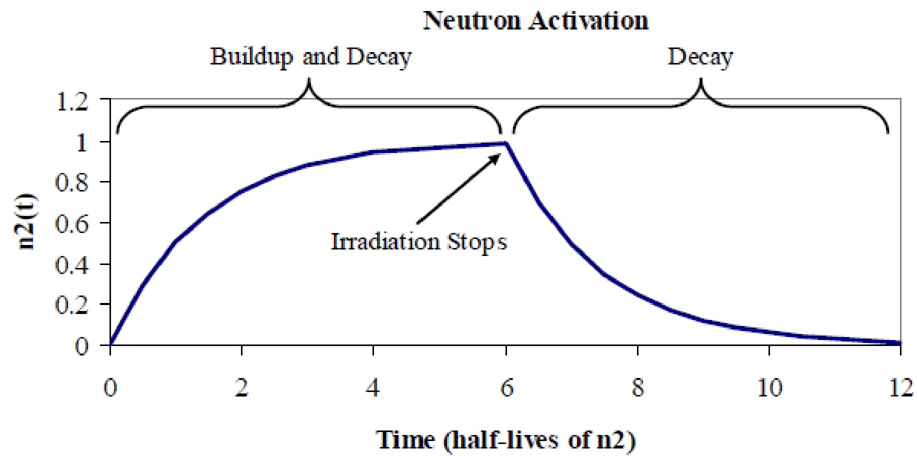
$$sn_3(s) - n_3(0) = \lambda_2 n_2(s)$$

$$n_3(s) = \frac{\lambda_2 n_2(s)}{s} = \sigma_{a,1} n_1(0)\phi\left[\frac{1}{s^2} - \frac{1}{s(s+\lambda_2)}\right] \tag{6}$$

The time domain solution is then

$$n_3(t) = \sigma_{a,1} n_1(0)\phi\left[t - \frac{1}{\lambda_2}\left(1 - e^{\lambda_2 t}\right)\right] \tag{7}$$

The buildup of the radioactive activation product, $n_2(t)$, during irradiation from t =0 to t =6, where the time scale is normalized to the number of half-lives of n_2. The number of activated nuclei (radionuclides) present at a particular time instant asymptotically approaches ${\sigma_{a,1} n_1(0)\phi}/{\lambda_2}$. After irradiation ceases, decay of n_2 continues.



After Irradiation$(t_I < t)$

Once the irradiation is stopped, there is no longer production of n2 rather only decay of the activation
product n2 and buildup of the decay product n_3:

$$\frac{dn_2}{dt} = -\lambda_2 n_2(t)$$
$$\frac{dn_3}{dt} = \lambda_2 n_2(t) \tag{8}$$

Again, Laplace transforms provide a natural solution, however, the initial condition are non-zero. The

"new" initial conditions are determined from the total irradiation time, t_I, (that is, $n_2(t_I)$and
$n_2(t_I)$are
computed from Equations (3) and (7), respectively).

$$sn_2(s) - n_2(t_I) = -\lambda_2 n_2(s)$$

$$n_2(s) = \frac{n_2(t_I)}{(s+\lambda_2)} \tag{9}$$

To retain the same time base, that is keeping t = 0 at the same reference point for both the irradiation and decay periods, it is necessary to make a slight adjustment during the inverse Laplace transform operation

$$n_2(t') = n_2(t_I)e^{\lambda_2 t'}$$

$$n_2(t') = \left[\frac{\sigma_{a,1} n_1(0)\phi}{\lambda_2}\left(1 - e^{\lambda_2 t}\right)\right] e^{-\lambda_2 (t - t_I)} \tag{10}$$

Likewise, the buildup of n3 may be determined via

$$sn_3(s) - n_3(t_I) = \lambda_2 n_2(s)$$

$$n_3(s) = \frac{n_3(t_I)}{s} + \frac{\lambda_2 n_2(s)}{s} \tag{11}$$

$$= \frac{n_3(t_I)}{s} + \frac{\lambda_2 n_2(t_I)}{s(s + \lambda_2)}$$

Inverse Laplace transforming similarly gives the buildup of n_3

$$n_3(t) = n_3(t_I) + n_2(t_I)\left[1 - \left(1 - e^{-\lambda_2 (t - t_I)}\right)\right] \tag{12}$$

Note how the simple relations in Equations (10) and (12) make physical sense.

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REACTOR MODES

Reactor behavior

The term "reactor behavior" refers to the response of neutron population to changes in physical parameters, usually a change in control rod position. A cycle of neutrons (a.k.a. generation) is considered to start with a thermal fission that produces neutrons and end when the neutron causes more neutrons to be produced from thermal fission. The ratio of the number of neutrons from thermal fission in one generation to the number of neutrons form thermal fission in the subsequent is called k_{eff}. Fractional deviations in k from an equal number per generation (k=1, or critical) are defined as reactivity (ρ).

$$\rho = \frac{k_{ref} - k}{k} = \frac{1 - k}{k}$$

k_{eff}	Reactivity	Definition
$k < 1$	$\rho < 0$	Subcritical
$k = 1$	$\rho = 0$	Critical
$k > 1$	$\rho > 0$	Supercritical

Viewed as a process, k_{eff} is the output of a cycle of neutron propagation normalized to the input, efficiency. Typical neutron populations in an operating reactor (weighted by the velocity of the neutron) are on the order of 10^9-10^{15} neutrons per cm^2 per second. In general this is adequate to understand reactor behavior; however, the fission process is not the only neutron production process. If efficiency is less than unity, there is a constant loss of neutrons. The loss fraction per cycle is calculated as $\{1\text{-}k_{eff}\}$.

Neutron Sources

There is a natural background of neutrons which are produced external to the thermal neutron life cycle, including cosmic radiation, boron/alpha reactions, deuterium/gamma reactions, and spontaneous fission (see table). With a detection rate at sea level typically less than 1 cpm, the neutron contribution from cosmic radiation is not large. The boron-alpha reaction is only significant where boron is used as a burnable poison (as in some power reactors). Light water reactors produce small amounts of heavy hydrogen, and the fuel itself has spontaneous fission decay constant. Transuranic isotope buildup in commercial reactors introduces other isotopes that will spontaneously fission. Fortunately, the magnitude of the neutrons is low enough that they are not detectable when the reactor is shutdown. Unfortunately the background signal in the neutron detectors calibrated to indicate reactor power level is low enough that electronic noise is comparable to the neutron signal; one cannot tell if the power level detectors are working. Therefore a separate neutron source may be installed to ensure the reactor power level detectors are operating.

Spontaneous Fission Rates[1]				
Nuclide	Half-life (Years)	Fission prob. (per decay)	Neutrons (per fission)	Neutrons ($g^{-1}s^{-1}$)
^{235}U	$7.04x10^{8}$	$7.0x10^{-11}$	1.86	$1.0x10^{-5}$
^{238}U	$4.47x10^{9}$	$5.4x10^{-7}$	2.07	0.0136
^{239}Pu	$2.41x10^{4}$	$4.4x10^{-12}$	2.16	$2.2x10^{-2}$
^{240}Pu	6569	$5.0x10^{-8}$	2.21	920
^{252}Cf	2.638	$3.09x10^{-2}$	3.73	$2.3x10^{12}$

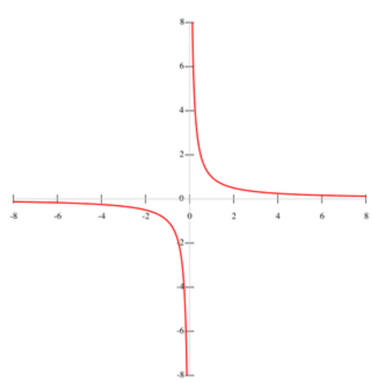
These sources generally provide activity on the order of 10^6 n s^{-1}, which distributed over even a small core like the UT TRIGA (core volume roughly 93,000 cm^3) does not intrinsically produce much signal. However, even neutrons from sources that do not originate from the thermal fission process can cause thermal fissions. This "source multiplication" in a subcritical reactor (a.k.a. subcritical multiplication) actually produces enough signal to prove power level detectors are working during a reactor startup.

Source Multiplication

The number of neutrons lost in a generation is the fractional loss {1-k_{eff}} times the number of number of neurons at the start of the generation. If neutron sources external to the fission process achieve appreciable fractions of the neutron population produced by thermal fission, then the neutron sources can make up for losses in the neutron life cycle. If a neutron source adds enough neutrons to make up for losses in the neutron life cycle, then the neutron population and the resultant rate of fissions is steady state. Using a convention of *CR* (count rate) for the neutron population, S^* for the non-fission source contribution per neutron generation and *SDM* (shutdown margin) for the fractional loss/difference from critical condition where k_{eff} is unity, the neutron population is steady state when:

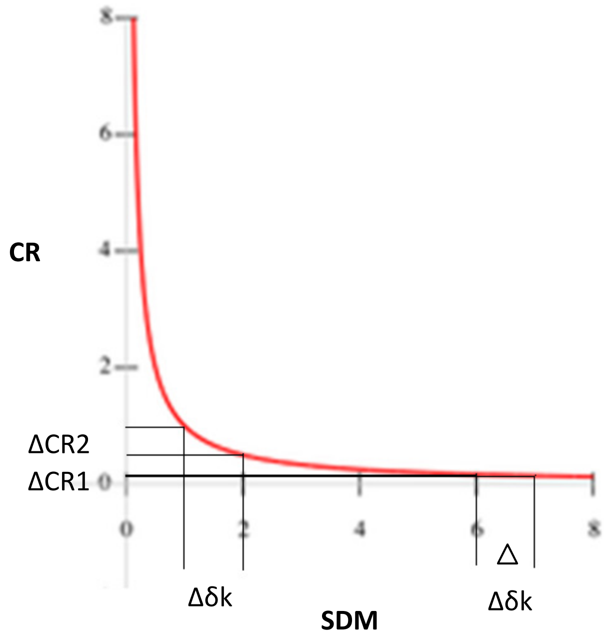
$$S^* = CR \cdot SDM$$

When the product of two variables is a constant, the function is a hyperbole. A hyperbole graphically is represented by:



[1] Fundamentals of Nuclear Science and Engineering. Shultis, J. Kenneth; Richard E. Faw (2002).

Restricting the graph to positive values for the neutron population:



Characteristic of a hyperbolic relationship, defined change in the shutdown margin has less effect on count rate as the shutdown margin is larger. Conversely, the closer to critical the larger the change in count rate resulting from a reactivity.

It would be easy to say that the larger change requires more time to come to equilibrium, but the analysis is based on a generation of fission neutrons and a fraction of the neutrons are not related to the fission process. If a source is inserted in a medium that will fission but is subcritical, each generation includes a constant contribution from the source as well as contributions from thermal fissions. As the source is multiplied, each subsequent generation will also have contributions from the multiplied source. In the table below, arrows are provided to illustrate how multiplied source neutrons contribute to subsequent generations

	Source				
Generation 0	S				
Generation 1	S	S· k			
Generation 2	S	S· k	S· k· k		
Generation 3	S	S· k	S· k· k	S· k· k· k	
Generation 3	S	S· k	S· k· k	S· k· k· k	S· k· k· k · k

At some arbitrary generation (I) the number of neutrons (n) is calculated:

$$n = S \cdot \sum_{i=0}^{I} k^{i}$$

Note that:

$$k \cdot n = k \cdot S \cdot \sum_{i=0}^{I} k^i = S \cdot \sum_{i=1}^{I} k^i$$

Subtracting the two equations:

$$n - k \cdot n = S \cdot \sum_{i=0}^{I} k^i - S \cdot \sum_{i=1}^{I} k^i = S$$

Which leads to:

$$n = \frac{S}{1-k}$$

This is the same equation previously noted to describe the behavior of a subcritical reactor. Source neutrons are therefore multiplied by a factor of $m = \frac{1}{1-k}$. For two separate values of *k* but the same source,

$$n_1 = \frac{S}{1-k_1}$$

And

$$n_2 = \frac{S}{1-k_2}$$

Can be divided as:

$$\frac{n_1}{n_2} = \frac{\frac{S}{1-k_1}}{\frac{S}{1-k_2}} = \frac{1-k_2}{1-k_1}$$

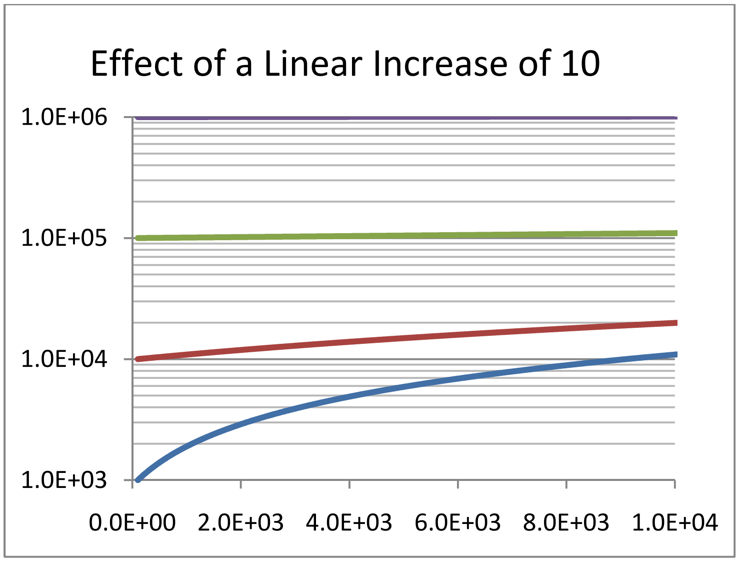
In terms of reactivity:

$$\frac{n_1}{n_2} = \frac{\rho_2 \cdot (\rho_1 - 1)}{\rho_1 \cdot (\rho_2 - 1)}$$

Significantly, the difference between the arbitrary generation *I* and the next (Δn) is $\Delta n = S \cdot k^{I+1}$. The system converges to but theoretically does not reach a set value. The number of iterations (generations) to the convergence value is a criteria; if the criteria is $\Delta n = 1$, the number of generations is a function of *k* only. Obviously, smaller *k* values require fewer generations.

Critical Operations

Source neutrons are introduced at a constant rate, regardless of the total number of neutrons, which determines reactor power level. Therefore, if the reactor is exactly critical there will be an increase of the same number of neutrons each generation, with no loss of neutrons – reactor power will rise at a linear rate. If the neutron flux is very low, the increase will be a large fraction of the neutron population and therefore detectable; at higher power levels, the increase may not be detectable. The graph shows how an increase of ten units appears on a log scale starting at different total values. When the reactor is exactly critical with a neutron source installed power level will be increasing at a linear rate; in a practical sense, the increase is not detectable when the ratio of source strength to neutron population exceeds about 0.1%.



Supercritical Operations

If the number of neutrons increases by a constant faction each generation, and a generation exists for a specified time, the rate of change in neutrons population is:

$$\frac{dn}{dt} = \frac{k}{\ell} \cdot n$$

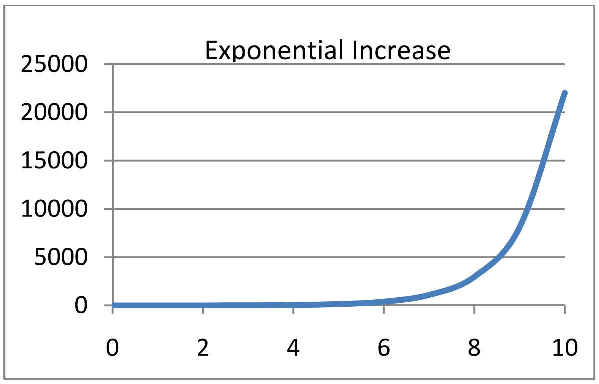
Which leads to the exponential equation:

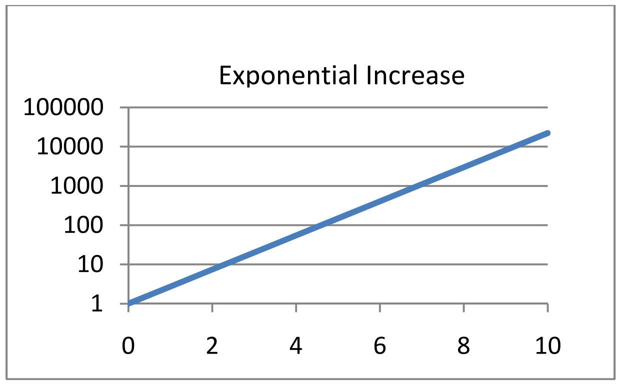
$$n(t) = n_0 \cdot e^{\frac{k}{\ell} \cdot t}$$

The exponential formula e^x is graphically displayed in two formats, one with a linear scale for the function and one with a logarithmic scale. Since reactor power generally increases exponentially, a line on the log-power scale illustrates

x	y
0	1
1	2.7

2	7.4
3	20.1
4	54.6
5	148.4
6	403.4
7	1096.6
8	2981.0
9	8103.1
10	22026.5





Basic Nuclear Principles, Article 7

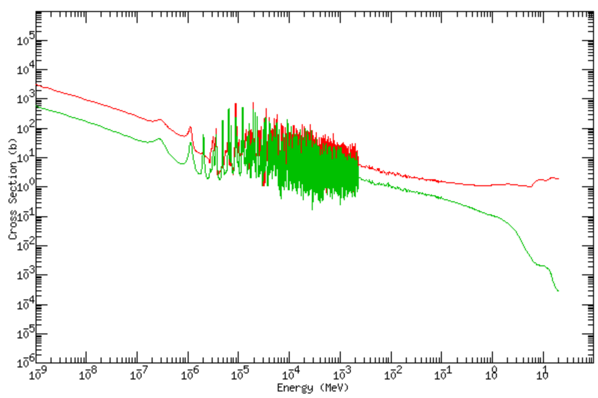
FOUR FACTOR FORMULA

Neutron Life Cycle

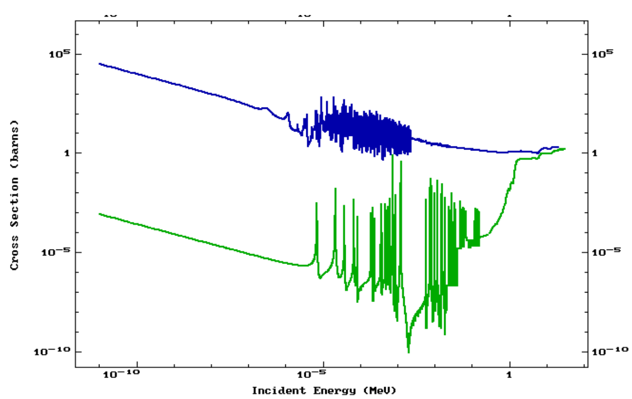
Modeling for reactor development or design analysis involves solutions to the Neutron Transport Equation, understanding how physical changes affect reactor behavior does not. The life cycle of reactor neutrons begins with high-energy neutrons released in the fission process. Fission energy neutrons are born with an average value 2.5 MeV kinetic in a Watt spectrum distribution (*N(E)* representing the number of neutrons at energy E):

$$N(E) = \sqrt{\frac{2}{\pi \cdot e}} \cdot \exp\left(-E\right) \cdot sinh\sqrt{2 \cdot E}$$

Decreasing neutron energy enhances the cross section for fission in ^{235}U; however, there is a fission cross section for high energy neutrons for both ^{235}U and ^{238}U, although the fission cross section for ^{238}U essentially vanishes below about 1 MeV.

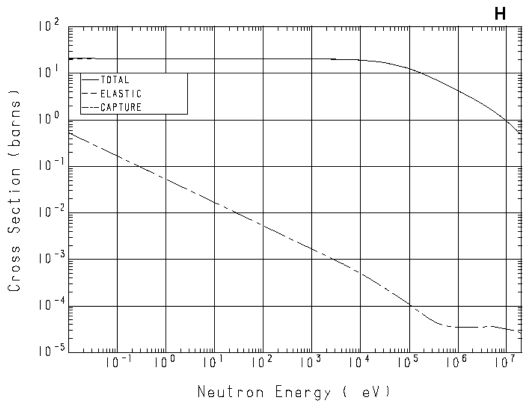


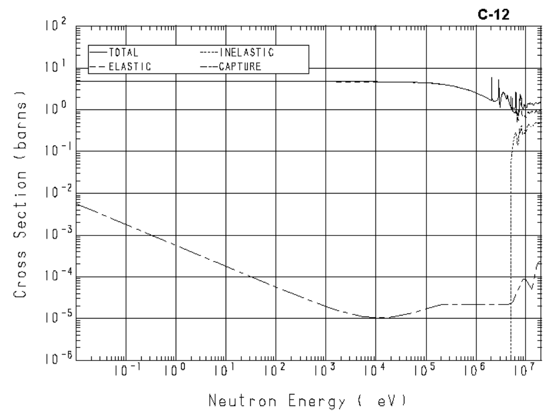
Absorption and Fission Cross Section for U^{235}



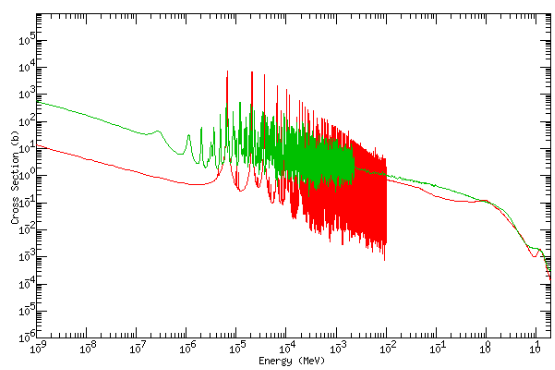
Fission Cross Sections for ^{235}U & ^{238}U

Fission energy neutrons lose energy by collision with atoms, transferring energy from neutrons to the "target" atoms. A relatively constant fraction of energy is transferred in each collision, leading to the concept of "average logarithmic energy decrement," with the symbol ξ. Straight forward analysis using conservation of momentum and energy demonstrates better transfer of energy when the mass of the target approaches the mass of the neutron. A neutron can transition from fission to 1 eV with 15 collisions with hydrogen, 92 collisions with carbon, 665 collisions with Zr, or 1700 collision with uranium. The scattering cross section for hydrogen and carbon are relatively constant over a wide range of energies.





Uranium exhibits elevated absorption cross section at very specific energies; if the neutron energy is slightly different than these "resonance" cross section peaks, then the absorption probability is orders of magnitude smaller. As neutrons lose energy, they will exist at various intermediate energies. If one of these energies matches a resonance, the neutron can easily be absorbed in ^{238}U in a non-fission absorption.



U^{238} and U^{235} Absorption Cross Sections (Energy in MeV)

Note that cross sections are experimentally measured. Neutron energy can be well-characterized, but individual atoms of the target material have kinetic energy that follows a Maxwell Boltzmann distribution. Individual target atoms cannot be measured or specified.

If a target atom is moving towards a neutron, the impact at interaction will be greater than the kinetic energy of the neutron and increase the energy of the collision. Therefore, an increase in material average kinetic energy (i.e., temperature) makes different neutron energies match the resonant energy. The sum of the cross section at energy across the whole resonance does not change. This effect is modeled as Doppler broadening of the resonance, with lower peak energy but the same total integral cross section value. If there is a distribution of neutron energies around a resonance, more neutrons at different energies can be resonantly absorbed as temperature increases. Therefore the spectrum-averaged macroscopic cross section increases with temperature, decreasing resonance escape probability.

With enough collisions, neutron energy will be on par with the kinetic energy of the target atoms. If a neutron has more energy that a target, it will lose energy in collision; if the neutron has less energy than a target, it will gain energy in collision. Therefore, the distribution of neutron energy will naturally parallel the energy of the target material. The Maxwell Boltzmann distribution in terms of velocity is:

$$f(v) = \sqrt{\frac{2}{\pi} \cdot \frac{m}{k \cdot T}} \cdot v^2 \cdot \exp\left(-\frac{m \cdot v^2}{k \cdot T}\right)$$

At low neutron energies, cross sections vary by $1/v$. For a neutron cross section of σ_0 at a reference 2200 m/s velocity (0.0253 eV), the cross section at a different velocity will be:

$$\sigma_x = \sigma_0 \cdot \frac{v_0}{v_x}$$

The cross section for all thermal neutrons will be:

$$\sigma_{th} = \int \left[\sigma_0 \cdot \frac{v_0}{v}\right] \cdot \left[\sqrt{\frac{2}{\pi} \cdot \frac{m}{k \cdot T}} \cdot v^2 \cdot \exp\left(-\frac{m \cdot v^2}{k \cdot T}\right)\right] dv = \sigma_0 \cdot \frac{\sqrt{2}}{\pi}$$

Using 20^0C, 293^0K, as the reference cross section measurement,

$$\sigma_x = \sigma_0 \cdot \frac{\sqrt{2}}{\pi} \cdot \sqrt{\frac{T_0}{T_x}}$$

Therefore, the cross section for thermal neutrons is reduced as the moderator temperature increases.

Multiplication factor, *k*

The multiplication factor (k) is a "gain" factor, describing whether neutron population from thermal fissions is increasing, constant, or decreasing.

$$k = \frac{Number\ of\ neutrons\ from\ thermal\ fission\ in\ current\ generation}{Number\ of\ neutrons\ from\ thermal\ fission\ in\ previous\ generation}$$

The multiplication factor has four components that are only functions of material, and not geometry. The fast fission factor term (ε) accounts for neutrons produced from high-energy neutron fissions. The resonance escape probability term (p) accounts for high energy neutrons absorbed by the fuel that do not produce neutrons. The thermal utilization factor term (f) accounts for absorptions that remove neutrons from potential neutron production. The reproduction factor term (η) accounts for the number of neutrons expected from an absorption in fuel.

$$k_\infty = \varepsilon \cdot p \cdot f \cdot \eta$$

Fast Fission Factor, ε

Although thermal neutrons only cause fission in U^{235}, high energy neutrons created in the fission process can cause both U^{235} and U^{238} to fission ("fast fission"). The fast fission factor (ε) accounts for the increase in neutrons because of fast fissions.

$$\varepsilon = \frac{Number\ of\ thermal\ and\ fast\ fission\ neutrons}{Number\ of\ thermal\ fission\ neutrons}$$

Where ν is the number of neutrons produced per thermal fission, σ is the cross section, and Φ is the neutron flux:

$$\varepsilon = \frac{\nu_{Th}^{235} \cdot N^{235} \cdot \sigma_{f,Th}^{235} \cdot \Phi_{Th} + \nu_{fast}^{235} \cdot N^{235} \cdot \sigma_{f,fast}^{235} \cdot \Phi_{fast} + \nu_{fast}^{238} \cdot N^{238} \cdot \sigma_{f,fast}^{238} \cdot \Phi_{fast}}{\nu_{Th}^{235} \cdot N^{235} \cdot \sigma_{f,Th}^{235} \cdot \Phi_{Th}}$$

A little simplification leads to:

$$\varepsilon = 1 + \left(\frac{\nu_{fast}^{235} \cdot \sigma_{f,fast}^{235}}{\nu_{Th}^{235} \cdot \sigma_{f,Th}^{235}} + \frac{\nu_{fast}^{238} \cdot \sigma_{f,fast}^{238}}{\nu_{Th}^{235} \cdot \sigma_{f,Th}^{235}} \cdot \frac{N^{238}}{N^{235}} \right) \cdot \frac{\Phi_{fast}}{\Phi_{Th}}$$

This can be simplified by assuming constant terms $\frac{\nu_{fast}^{235} \cdot \sigma_{f,fast}^{235}}{\nu_{Th}^{235}} = A$ and $\frac{\nu_{fast}^{238} \cdot \sigma_{f}^{238}}{\nu_{Th}^{235}} = B$:

$$\varepsilon = 1 + \left[\frac{\mathrm{A}}{\sigma_{f,Th}^{235}} + \frac{N^{238}}{N^{235}} \cdot \frac{\mathrm{B}}{\sigma_{f,Th}^{235}} \right] \cdot \frac{\Phi_{fast}}{\Phi_{Th}}$$

Resonance Escape Probability, *p*

High energy neutrons lose energy by scattering, which reduces neutron energy from about 2.5 MeV to about 0.0253 eV.

Absorption cross sections for U^{238} at energies less than about 10^4 eV and above about 1 eV have a series of "resonances," sharp peaks at discrete energies. The resonance escape probability (p) reflects those neutrons that are slowing down, but do not get absorbed in cross section resonances.

$$p = \frac{Number\ of\ neutrons\ slowed\ to\ thermal\ energy}{Number\ of\ neutrons\ from\ thermal\ and\ fast\ fission}$$

Capture resonances are complex discrete cross sections; development of an analytic form for the resonance escape probability is not trivial. One generally accepted approximation is:

$$p = exp\left[-\frac{1}{\xi}\cdot\int\frac{\Sigma_a^{Fuel}}{\Sigma_a^{Fuel}+\Sigma_s^{Fuel\ and\ Mod}}\right]$$

where ξ is the weighted average energy decrement. Note that this form implicitly includes some averaging processes in combining fuel and moderator scattering properties, and is concerned only with resonance absorptions (greater than thermal energies).

$$p = exp\left[-\frac{1}{\xi}\cdot\int_{F.En.}^{Th}\frac{1}{1+\dfrac{\bar{\Sigma}_s^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}\right]$$

With a resonance escape probability on the order of 60%, exponential expansion to the 1st order can describe the general direction (but not magnitude very well) of changes:

$$p \cong 1-\frac{1}{\xi}\cdot\int_{F.En.}^{Th}\frac{1}{1+\dfrac{\bar{\Sigma}_s^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}$$

Some qualitative simplification can be based on the relatively constant microscopic scattering cross sections. Therefore, variability in macroscopic scattering cross section is via changes in number density. It is also common to consider the average logarithmic energy decrement either as an average value or an average value across a few energy ranges.

Thermal Utilization, *f*

Of the neutrons that achieve thermal energy, only a fraction (the thermal utilization factor (*f*), will be absorbed in the fuel.

$$f = \frac{Number\ of\ thermal\ neutrons\ absorbed\ in\ fuel}{Number\ of\ thermal\ neutrons\ absorbed\ in\ core}$$

Where Σ_a is the macroscopic cross section for absorption,

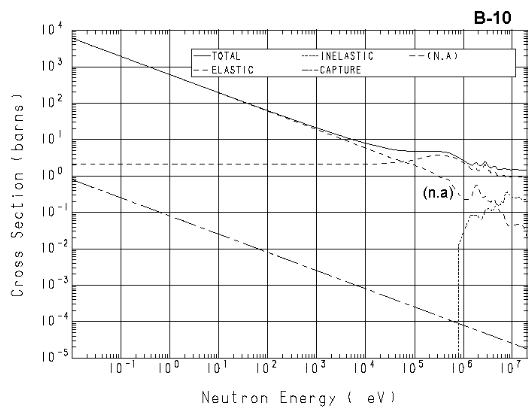
$$f = \frac{\Sigma_a^{Fuel}\cdot\Phi_{Th}^{Fuel}}{\Sigma_a^{Core}\cdot\Phi_{Th}^{Core}}$$

If the thermal neutron flux has the same value across the core components (fuel, cladding, water, control rods, and other components), then

$$f = \frac{\Sigma_{a,Th}^{Fuel}}{\Sigma_{a,Th}^{Fuel} + +\Sigma_{a,Th}^{Cladding} + \Sigma_{a,Th}^{H_2O} + \Sigma_{a,Th}^{Control\ Rods} + \Sigma_{a,Th}^{Other}}$$

The fuel matrix actually consists of ^{235}U, ^{238}U and $ZrH_{1.6}$ and boron (cross section for ^{10}B provided below) is the major component of the control rods so that:

$$f = \frac{\{N^{235}\cdot\sigma_{a,Th}^{235} + N^{238}\cdot\sigma_{a,Th}^{238} + N^{ZrH}\cdot\sigma_{a,Th}^{ZrH}\}}{\{N^{235}\cdot\sigma_{a,Th}^{235} + N^{238}\cdot\sigma_{a,Th}^{238} + N^{ZrH}\cdot\sigma_{a,Th}^{ZrH}\} + N^{H_2O}\cdot\sigma_{a,Th}^{H_2O} + N^{CR}\cdot\sigma_{a,Th}^{CR} + N^{Other}\cdot\sigma_{a,Th}^{Other}}$$



Reproduction Factor, η

The reproduction factor is the average of the number of neutrons released when a neutron is absorbed in fuel.

$$\eta = \frac{Number\ of\ neutrons\ produced\ by\ thermal\ fission}{Number\ of\ thermal\ neutrons\ absorbed\ in\ fuel}$$

Although most thermal neutrons absorbed in the fuel are absorbed in uranium, the fuel matrix includes zirconium-hydride, U^{238} and U^{235}. Since the absorption cross section for U^{235} is about 100 times greater than the absorption cross section for U^{238}, most of the neutrons absorbed in uranium are absorbed in U^{235}. However, only about 85% of absorptions in U^{235} result in fission. An average of 2.43 neutrons are released in U^{235} fissions. Where ν is the number of neutrons produced per thermal fission, σ is the cross section, and Φ is the neutron flux:

$$\eta = \frac{\nu\cdot N^{235}\cdot\sigma_{f,Th}^{235}\cdot\Phi_{Th}^{235}}{N^{Fuel}\cdot\sigma_{a,Th}^{Fuel}\cdot\Phi_{Th}^{Fuel}}$$

For homogenous fuel composed of U^{235}, U^{238}, and ZrH:

$$\eta = \frac{\nu \cdot N^{235} \cdot \sigma_f^{235}}{\nu \cdot N^{235} \cdot \sigma_{f,Th}^{235} + N^{235} \cdot \sigma_{a\neq f,Th}^{235} + N^{238} \cdot \sigma_{a,Th}^{238} + N^{ZrH} \cdot \sigma_{a,Th}^{2ZrH}}$$

A more useful form is:

$$\eta = \frac{\nu}{1 + \frac{\sigma_{a\neq f,Th}^{235}}{\sigma_{f,Th}^{235}} + \frac{N^{238} \cdot \sigma_{a,Th}^{238} + N^{ZrH} \cdot \sigma_{a,Th}^{ZrH}}{N^{235} \cdot \sigma_{f,Th}^{235}}}$$

Since $N^{238} \cdot \sigma_a^{238} \gg N^{ZrH} \cdot \sigma_a^{ZrH}$, defining $\left\{1 + \frac{\sigma_{a\neq f,Th}^{235}}{\sigma_{f,Th}^{235}}\right\} = C$ and $\frac{\sigma_{a,Th}^{238}}{\sigma_{f,Th}^{235}} = D$:

$$\eta \cong \frac{\nu}{C + D \cdot \frac{N^{238}}{N^{235}}}$$

Summary and conclusions

The effect of variations in physical parameters on the 4-factor formula can be understood by evaluating how each individual term is affected.

Term	Formula	Simplified	TRIGA Value
ε	$\frac{Number\ of\ thermal\ and\ fast\ fission\ neutrons}{Number\ of\ thermal\ fission\ neutrons^{gen\ 1}}$	$1 + \left[\frac{\mathrm{A}}{\sigma_{f,Th}^{235}} + \frac{N^{238}}{N^{235}} \cdot \frac{\mathrm{B}}{\sigma_{f,Th}^{235}}\right] \cdot \frac{\Phi_{fast}}{\Phi_{Th}}$	1.02-1.08
p	$\frac{Number\ of\ neutrons\ slowed\ to\ thermal\ energy}{Number\ of\ neutrons\ from\ thermal\ and\ fast\ fission}$	$1 - \frac{1}{\xi} \cdot \int_{F.En.}^{Th} \frac{1}{1 + \frac{\bar{\Sigma}_s^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}$	0.68
f	$\frac{Number\ of\ thermal\ neutrons\ absorbed\ in\ fuel}{Number\ of\ thermal\ neutrons\ absorbed\ in\ core}$	$\frac{1}{1 + \frac{N^{H_2O} \cdot \sigma_{a,Th}^{H_2O} + N^{CR} \cdot \sigma_{a,Th}^{CR} + N^{Other} \cdot \sigma_{a,Th}^{Other}}{N^{235} \cdot \sigma_{a,Th}^{235} + N^{238} \cdot \sigma_{a,Th}^{238} + +N^{ZrH} \cdot \sigma_{a,Th}^{ZrH}}}$	0.80
η	$\frac{Number\ of\ neutrons\ produced\ by\ thermal\ fission^{gen\ 2}}{Number\ of\ thermal\ neutrons\ absorbed\ in\ fuel}$	$\frac{\nu}{1 + \frac{\sigma_{a\neq f,Th}^{235}}{\sigma_{f,Th}^{235}} + \frac{N^{238} \cdot \sigma_{a,Th}^{238} + N^{ZrH} \cdot \sigma_{a,Th}^{ZrH}}{N^{235} \cdot \sigma_{f,Th}^{235}}}$	2.02

Enrichment

Enrichment is defined as:

$$w\% = \frac{m_{235}}{m_U}$$

Mass is related to number density by:

$$N = \frac{m \cdot N_{Av}}{A_{g \cdot mole}}$$

Since $\{m_U = m_{235} + m_{238}\}$, N^{235}, N^{238}, and/or the ratio of N^{235}:N^{238} can be developed in the factors as number density, mass, or enrichment.

Reactivity, ρ

Fractional deviation in k_{eff} from reference, stable, critical condition

$$\rho = \frac{k_{ref} - k}{k} = \frac{1 - k}{k}$$

$$\rho = \frac{\varepsilon_{ref} \cdot p_{ref} \cdot f_{ref} \cdot \eta_{\mathrm{ref}} - \varepsilon \cdot p \cdot f \cdot \eta}{\varepsilon \cdot p \cdot f \cdot \eta} = \frac{\varepsilon_{ref} \cdot p_{ref} \cdot f_{ref} \cdot \eta_{\mathrm{ref}}}{\varepsilon \cdot p \cdot f \cdot \eta} - 1$$

k_{eff}	Reactivity	Definition
$k < 1$	$\rho < 0$	Subcritical
$k = 1$	$\rho = 0$	Critical
$k > 1$	$\rho > 0$	Supercritical

Basic Nuclear Principles, Article 8

Neutron Non-Leakage

Neutron Path-length and Leakage

Not all neutrons interact within the core area, some escape the boundaries. As a neutron travels from fission to absorption, the neutron energy decreases by more than 8 orders of magnitude from an average birth energy 2.5 MeV to an average energy at absorption of 0.0253 eV. Cross sections change across this energy range dramatically, and at a minimum have to be managed in two groups: high-energy (fission and fast neutrons) and neutrons in thermal equilibrium with core materials (thermal neutrons). The distance a fast neutron travels is "Fermi Age." The distance a thermal neutron travels is "Diffusion Length." The total distance neutron travels ("Migration Length") is the Fermi Age and Diffusion Length combined in quadrature. Neutrons within a few migration lengths from a core boundary have the potential to escape. They may escape either as fast or thermal neutrons.

Calculation of leakage/non-leak probability is not trivial. The equations and terms provided here are intended to provide a framework for understanding what physically occurs in neutron leakage; more specifically to aid understanding of how physical changes in the reactor materials and geometry affect leakage. The models and equations provided are from intermediate steps and interpretations of the neutron transport equation (a special case of Boltzmann transport) and the Helmholtz equation.

Diffusion Coefficient

Simplistically, with Σ_a describing the number of absorptions per unit length, the mean free path (*mfp*) of a neutron is the average distance (λ_{mfp}) traveled until the neutron is absorbed:

$$\lambda_{mfp} = \frac{1}{\Sigma_a}$$

This is exact for the case where the only possible interaction is absorption. In the case where scattering occurs, the mean free path is modified by a "diffusion coefficient":

$$D = \frac{1}{3 \cdot (\Sigma_{total} - \Sigma_{scatter} \cdot \bar{\mu})}$$

For weakly absorbing material ($\Sigma_{total} = \Sigma_{scatter}$) and isotropic scattering ($\bar{\mu} = 0$), this reduces to:

$$\acute{D} = \frac{1}{3 \cdot \Sigma_{scatter}}$$

In the case of the transition of high energy neutrons to thermal neutrons, "birth" is the fission process with "death" being a loss of energy into the thermal range. At high energies most interactions are scattering, with little absorption. In the case of thermal neutrons, birth is the transition from high energy to thermal region, and death is absorption. In the thermal range both scattering and absorption must be considered.

Fast Neutron Path

The path length from birth to the point where some energy is achieved (for a high energy neutron with no absorptions, scattering cross section, Σ_S, a function of E, and ξ as the average logarithmic decrement per scatter) is calculated as:

$$\tau(E) = \int_E^{E_0} \frac{\acute{D}}{\xi \cdot \Sigma_S \cdot E}\, dE$$

Substituting the diffusion coefficient into the integral:

$$\tau(E) = \int_E^{E_0} \frac{1}{3 \cdot \Sigma_S^2 \cdot \xi \cdot E}\, dE$$

For weakly absorbing material, this is modified by the resonance escape probability:

$$p = exp\left[-\frac{1}{\xi} \cdot \int_E^{E_0} \frac{\Sigma_a^{Fuel}}{\Sigma_a^{Fuel} + \Sigma_S^{Fuel\ and\ Mod}}\right] \cong 1 - \frac{1}{\xi} \cdot \int_E^{E_0} \frac{1}{1 + \frac{\bar{\Sigma}_S^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}$$

Resonance absorptions are 1-p, So that:

$$\tau(E) \cong \int_E^{E_0} \frac{1}{3 \cdot \Sigma_S^2 \cdot \xi \cdot E}\, dE \cdot \left[\frac{1}{\xi} \cdot \int_E^{E_0} \frac{1}{1 + \frac{\bar{\Sigma}_S^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}\, dE\right]$$

This is not a friendly term, but the qualitative effects of changes in parameters are evident. Decreases in scattering cross sections and decreases in resonance absorption increase fast neutron path-length from birth to thermalization.

Thermal Neutron Path

The modification to mean free path in the thermal region is relatively straightforward (compared to high energy neutrons). Diffusion length of thermal neutrons (L) is calculated:

$$L = \sqrt{\frac{1}{3 \cdot \Sigma_a \cdot (\Sigma_a + \Sigma_{scatter} \cdot \{1 - \bar{\mu}\})}}$$

Geometric Buckling

Geometric buckling describes the propensity for neutrons to escape the core; neutrons more than 3-5 mean free paths from the core surface have no significant chance of escaping from the core. In simple terms, neutrons leak out of a surface, and interact within a volume. Symmetry and the physics of the diffusion process require specific flux distributions and specific geometric buckling factors for specific geometries.

Reactor Geometry and Buckling			
Geometry	Dimension	Flux Distribution	Buckling
Infinite slab	Thickness a	$\Phi_0 \cdot cos\dfrac{n \cdot x}{a}$	$\dfrac{\pi}{a}$
Sphere	Radius R	$\dfrac{\Phi_0}{r} \cdot cos\dfrac{\pi \cdot r}{R}$	$\dfrac{\pi}{R}$
Rectangular parallelepiped	a X b X c	$\Phi_0 \cdot cos\dfrac{\pi \cdot x}{a} \cdot cos\dfrac{\pi \cdot y}{b} \cdot cos\dfrac{\pi \cdot z}{c}$	$\sqrt{\left(\dfrac{\pi}{a}\right)^2 + \left(\dfrac{\pi}{b}\right)^2 + \left(\dfrac{\pi}{c}\right)^2}$
Finite cylinder	Radius R, height H	$\Phi_0 \cdot J_0\dfrac{2.405 \cdot r}{R} \cdot cos\dfrac{\pi \cdot z}{H}$	$\sqrt{\left(\dfrac{2.405}{R}\right)^2 + \left(\dfrac{\pi}{H}\right)^2}$

Buckling and k_{eff}

The ratio of neutrons from fission in one generation to neutrons in the previous generation is k_{eff}, or k. For k=1, the neutron population is stable. If neutrons are all absorbed in the reactor and none move to locations outside the reactor, there is no leakage. The absence of leakage implies an infinite geometry, with the ratio k (k_{eff}) labeled k_∞. Buckling relates k and k_∞ generally as:

$$\boldsymbol{k_\infty = k + k \cdot L^2 \cdot B^2 = k \cdot \left(1 + L^2 \cdot B^2\right)}$$

In general, neutrons are born at high energies and lose energy through scattering until they are in thermal equilibrium with their surroundings. Absorption cross sections at very high energies are much different than absorption cross sections, with scatter angle more forward directed. The ratio of scattering to absorption is different at high energies than low energies. Therefore the materials-based leakage of high and low energy neutrons is considered separately, with L reserved for low energy, and τ used to modify the neutron path length for high energy neutrons. Developing the form for both high energy and low energy neutron components that are coupled (high energy neutrons become low energy neutrons through scattering) creates a cross term that simplifies to:

$$k_\infty = k \cdot (1 + L^2 \cdot B^2) \cdot (1 + L^2 \cdot \tau^2)$$

or:

$$\boldsymbol{k}=\frac{\boldsymbol{k}_{\infty}}{(\mathbf{1}+\boldsymbol{L}^{2}\cdot\boldsymbol{B}^{2})\cdot(\mathbf{1}+\boldsymbol{L}^{2}\cdot\boldsymbol{\tau}^{2})}$$

The two terms in the denominator are called "nonleakage factors" and depending on convention may be written as $P_{nl,f}$, for fast neutrons and $P_{nl,th}$ for thermal neutrons or L_f and L_{th} (sometimes with a script L).

$$P_{nl,Th}=\frac{1}{1+\frac{B^2}{3\cdot\Sigma_a\cdot(\Sigma_a+\Sigma_{scatter}\cdot\{1-\ \bar{\mu}\})}}$$

$$P_{nl,fast}=\frac{1}{1+B^2\cdot\int_{0.1\ eV}^{2.5\ MeV}\frac{1}{3\cdot\Sigma_S^2\cdot\xi^2\cdot E}\cdot\frac{1}{1+\frac{\bar{\Sigma}_s^{Fuel\ and\ Mod}}{\bar{\Sigma}_a^{Fuel}}}\,dE}$$

As before, some qualitative simplification can be made based on microscopic scattering cross sections for moderating materials like hydrogen and carbon and highly absorbing materials like boron. As absorptions increase, nonleakage decreases, although only absorptions within the range of a few mean free paths from the core boundary contribute to the decrease in nonleakage.

Summary and Conclusions

Geometric buckling has the same form for all neutrons, regardless of neutron energy. Changes in the cross sections or number density affect nonleakage factors by changing the total path a neutron travels; in the case of fast neutrons, from birth to thermalization, and in the case of thermal neutrons from thermalization to absorption. In general, buckling is smaller for larger reactors; smaller buckling influences nonleakage toward unity.

Basic Nuclear Principles, Article 9

REACTOR KINETICS

Neutron Lifetime

- Since velocity is distance traveled over time, the lifetime of neutrons born directly from fission (prompt neutron lifetime, ℓ_p) is the distance traveled over the average neutron velocity (v)

$$l_p = \frac{\lambda_{mfp}}{v} = \frac{1}{v \cdot \Sigma_a}$$

- some fraction (β) of neutrons that cause fission occur as the decay of fission fragments, which have half lives from milliseconds to minutes; the life of these neutrons starts after the fission at the average (or mean) life of the radioisotope, then follows essentially the same lifetime as prompt neutrons.

$$\boldsymbol{\ell}_d = \frac{1}{\lambda_{\mathrm{DK}}} + l_p$$

Neutrons that result from the decay of radioactive fission products are called "delayed," and neutrons produced directly in fission process are called "prompt." Fission is a stochastic process, so that fission products from a single, specific fission can only be described statistically (i.e., as a probability of distribution of products); values for β and $\boldsymbol{\lambda}_{\mathbf{DK}}$ only apply in a broad, average sense. Experimental observations show that the half lives of delayed neutron precursors are close enough to be considered as several "groups" with a statistical spread around the half life of representative isotopes.

Group	Isotope	Half Life (s)	Decay Constant (s^{-1})	Energy (keV)	Neutrons per Fission	Fraction
1		55.72	0.0124	250	0.00052	0.000215
2		22.72	0.0305	560	0.00546	0.001424
3		6.22	0.111	405	0.00310	0.001274
4		2.30	0.301	450	0.00624	0.002568
5		0. 614	1.14	-	0.00182	0.000748
6		0.230	3.01	-	0.00066	0.000273

- a weighted average of the lifetimes of prompt and delayed neutrons represents the mean lifetime for all neutrons (τ):

$$l_{Avg} = (1-\beta) \cdot l_p + \sum_{i=1}^{6} \beta_i \cdot \left(l_p + \frac{1}{\lambda_{\mathbf{DK}_i}} \right)$$

In practice, the prompt neutron lifetime is orders of magnitude smaller than the mean lifetime of delayed neutron precursors, and:

$$l_{Avg} \cong (1-\beta) \cdot l_p + \sum_{i=1}^{6} \left[\frac{\beta_i}{\lambda_{\mathbf{DK}_i}}\right]$$

For convenience, this is frequently simplified to a single group of delayed neutrons:

$$l_{Avg} \cong (1-\beta) \cdot l_p + \frac{\beta}{\lambda_{\mathbf{DK}}} \cong \frac{\beta}{\lambda_{\mathbf{DK}}} = \beta \cdot \tau$$

- Similarly, a weighted average of the energies of the neutrons shows the average energy of delayed neutrons to be about 0.45 MeV. In contrast, the average energy of prompt neutrons at birth is about 2.5 MeV.

Typical Values

For thermal reactors, the time-to-absorption is mostly the time the neutron spends at thermal energy levels before being absorbed (i.e., slowing down time is comparatively short). Thermal lifetimes for pure moderators:

Moderator	H_2O	D_2O	Be	C
Thermal Lifetime (sec)	$2.1\ x\ 10^{-4}$	$1.4\ x\ 10^{-1}$	$3.7\ x\ 10^{-3}$	$1.8\ x\ 10^{-2}$

The delayed neutron fraction (β) and decay constant are a function of the fuel material and neutron energy.

Fuel Nuclide	Thermal Fission		Fast Fission	
	β	$\lambda\ (sec^{-1})$	β	$\lambda\ (sec^{-1})$
U_{233}	0.00266	0.0543	0.00267	0.0559
U_{235}	0.00650	0.0767	0.00642	0.0784
Pu_{239}	0.00212	0.0648	0.00204	0.0683

Reactor Flux/Power Increase

$$n(t) = n_0 e^{t/T} \qquad \phi(t) = \phi_0 e^{t/T} \qquad P(t) = P_0 e^{t/T}$$

Stable Reactor Period, Prompt T

- For very small reactivity insertions ($\rho << \beta$)

$$T \cong \frac{l_{Avg}}{\rho} \approx \frac{\beta}{\lambda\rho} = \frac{\beta}{\rho}$$

- For large positive reactivity insertions $(\rho >> \beta)$; i.e., prompt critical

$$T \cong \frac{l_p}{\rho - \beta}$$

Reactivity, ρ

$$\rho = \frac{k-1}{k} = \frac{\Delta}{k} \qquad \$ = \frac{\rho}{\beta}$$

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