Training Manual

Kansas State University Nuclear Reactor Facility

License R-88
Docket 50-188

2011 Revision

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General Characteristics of the Kansas State University TRIGA Mk II Reactor

INTRODUCTION

The Kansas State University (KSU) Nuclear Reactor Facility, operated by the Department of Mechanical and Nuclear Engineering, is located in Ward Hall on the Manhattan campus. The Department is also the home of the Tate Neutron Activation Analysis Laboratory, and several other supporting laboratories.

The TRIGA reactor was obtained through a grant from the United States Atomic Energy Commission and is operated under Nuclear Regulatory Commission License R-88 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. Criticality was fist achieved on October 16, 1962 at 8:25 pm. In 1968 pulsing capability was added and the maximum steady-state operating power was increased from 100 kilowatts (kW) to 250 kW. The aluminum clad fuel elements were replaced with stainless-steel clad elements in 1973. With support from the U.S. Department of Energy, coolant system replacement was completed in 1993, as was replacement of the reactor operating console, and enlargement and modernization of the reactor control room. All neutronic instrumentation was replaced in 1994. In 2008 the reactor license was renewed and the maximum power level was upgraded to 1250 kW.

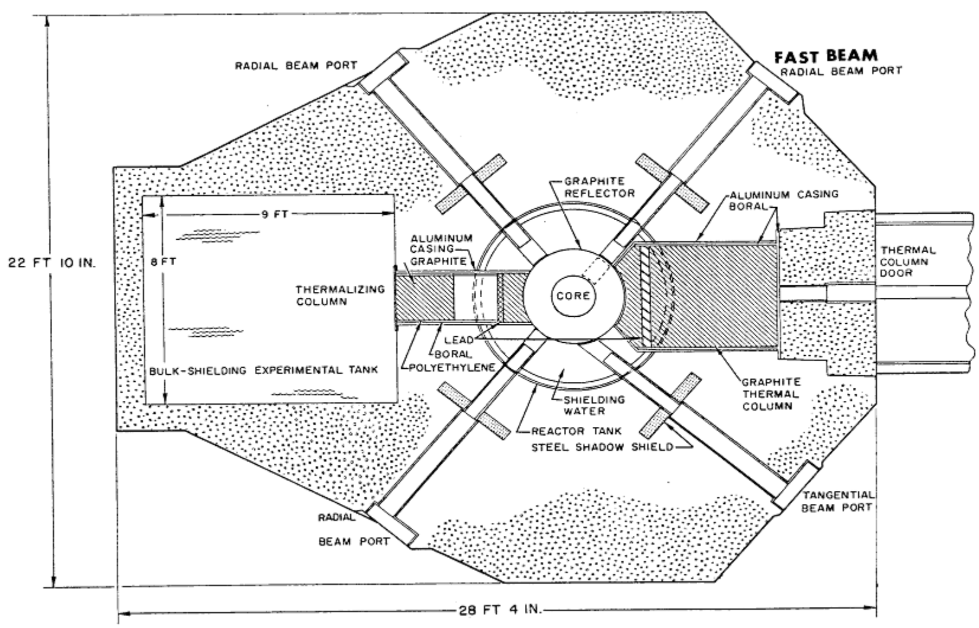


Figure 1 - Horizontal section through reactor

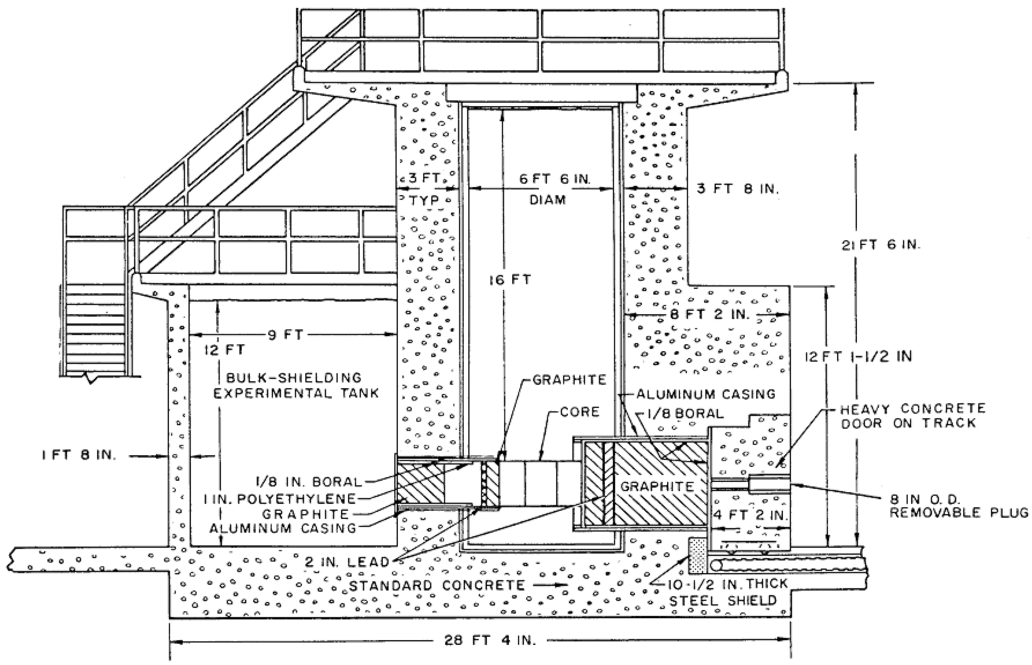


Figure 2 - Vertical section through reactor

The primary functions of the Nuclear Reactor Facility are to serve as:

- An educational facility for all students at KSU and nearby universities and colleges,
- An irradiation facility for researchers at KSU and for others in the central United States, and
- A facility for training nuclear reactor operators, and 4) a demonstration facility to increase public understanding of nuclear energy and nuclear reactor systems.

The reactor vendor was General Atomics division of General Dynamics Corporation, San Diego, California. Architect-Engineer was Uel. C. Ramey and Associates, Wichita, Kansas. Reactor constructor was Holmes and Narver, Inc. Orange, California.

Reactor design began in 1956. The General Atomics design philosophy was that the reactor be completely and inherently safe. Complete safety means that all the available excess reactivity of the reactor can be instantaneously introduced without causing an accident. Inherent safety means that an increase in the temperature of the fuel immediately and automatically results in decreased reactivity through a prompt negative temperature coefficient. These features were accomplished by using enriched uranium fuel in a zirconium hydride matrix. As power and temperature increase, matrix changes cause a shift in the neutron energy spectrum in the fuel to higher energies. The uranium exhibits lower fission cross sections for the higher energy neutrons, thus countering the power increase.

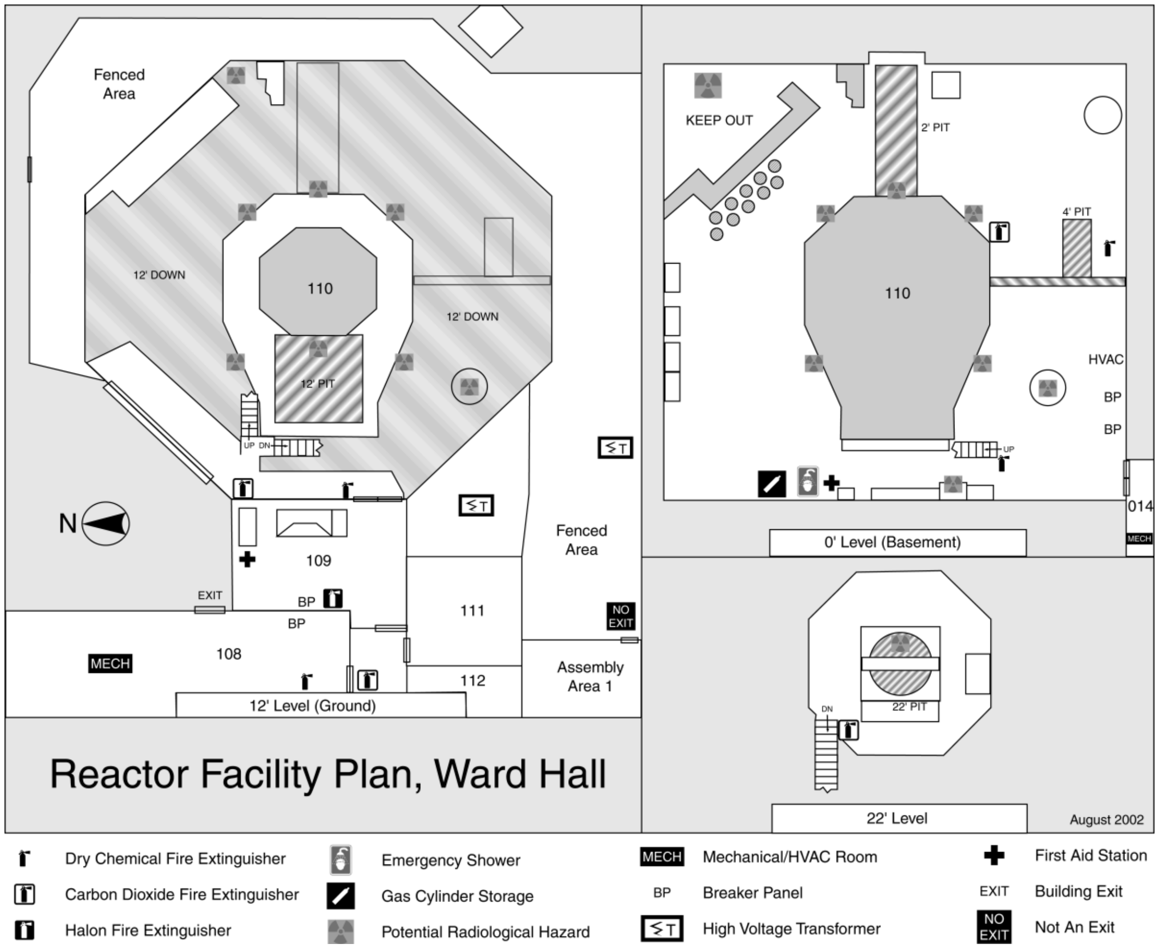


Figure 3 - Reactor facility layout

CONFINEMENT

The confinement building is a 144,000 ft^3 structure, made of reinforced concrete and structural steel, with composite sheathing and aluminum siding.

REACTOR TANK AND SHIELDING

The reactor tank is made of 0.25 in. thick aluminum. It is 6.5 ft (1.98 m) in diameter, and 22 ft (6.25 m) deep. It is surrounded at the base and on the sides by a reinforced concrete biological shield. The minimum thickness of the shielding at the core level is 8 ft 2 in. (2.5 m).

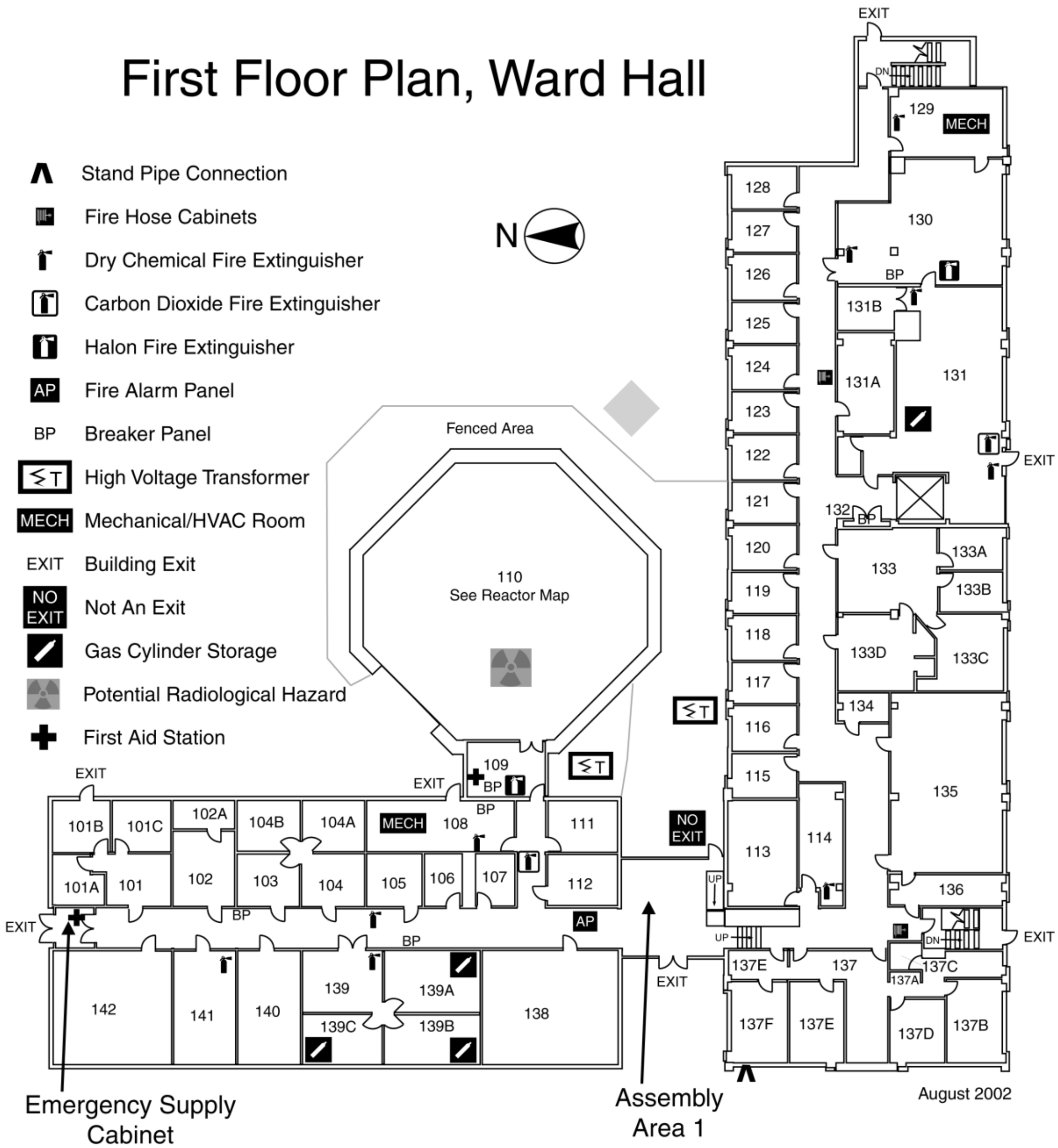


Figure 4 - Ward Hall layout, 1st Floor

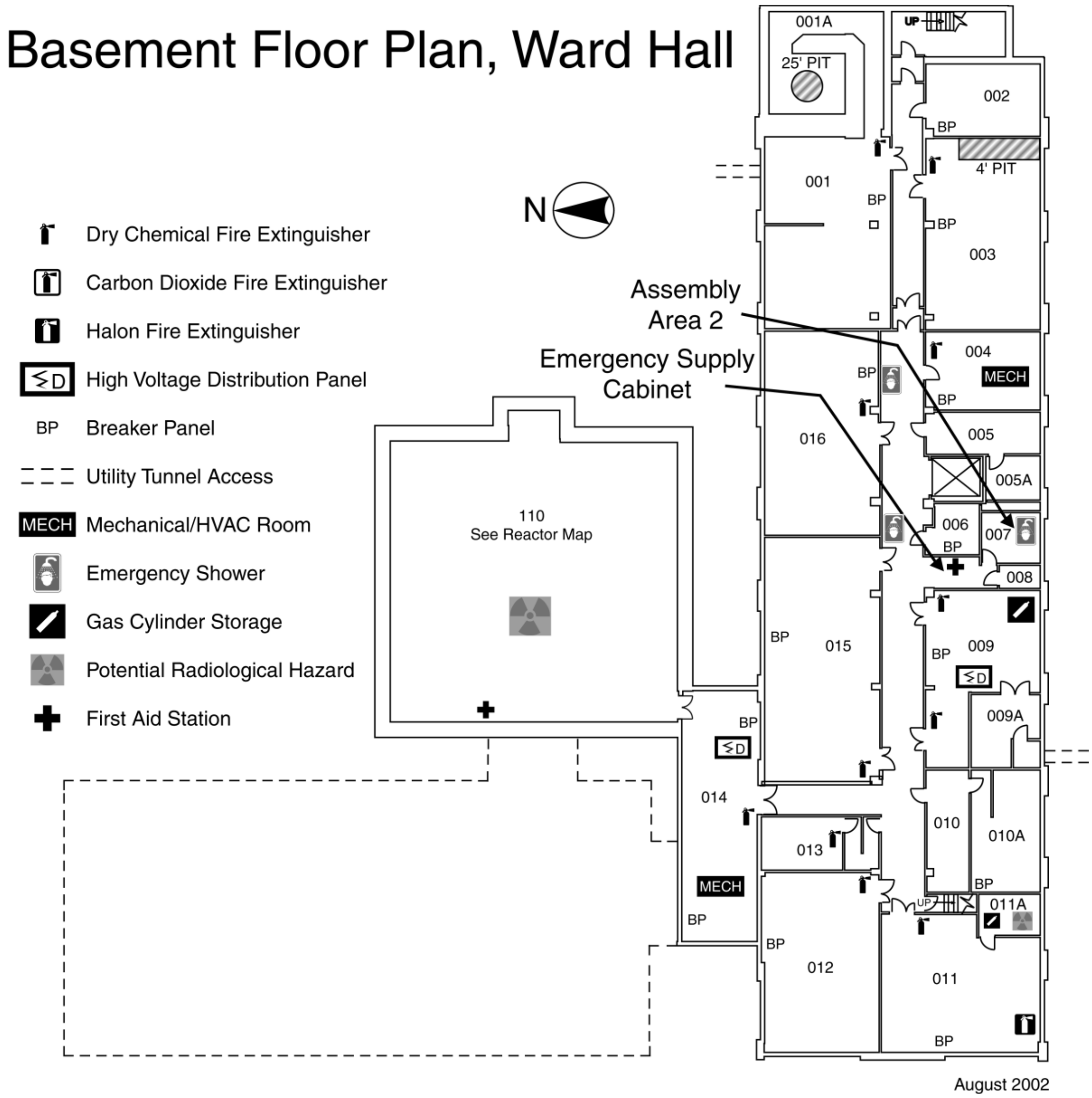


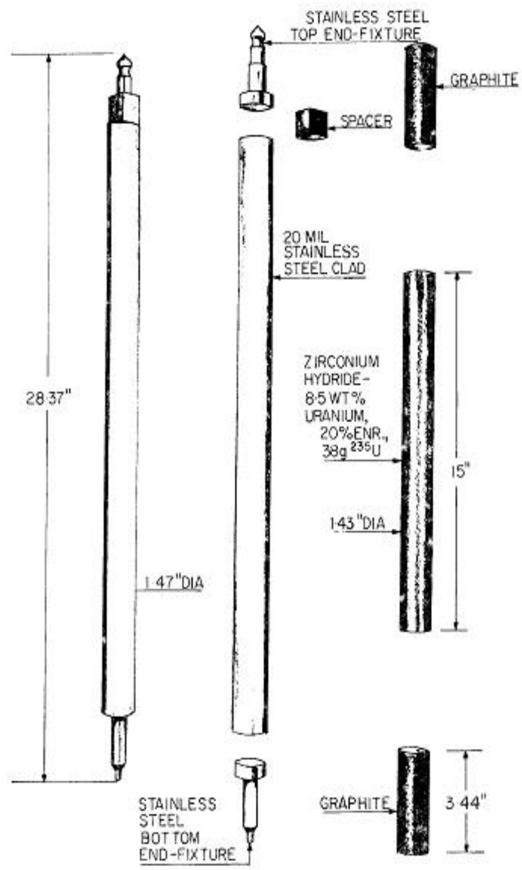
Figure 5 - Ward Hall layout, basement level

REACTOR CORE AND FUEL ELEMENTS

The TRIGA is designed to have a very small critical mass of uranium coupled with a ZrH moderator. Early calculation showed that the smallest ^{235}U mass was obtained using (i) 20% enriched U, (ii) H:^{235}U in active volume (meat) between 1:150-200, and (iii) core 40 volume % water.

The Mark III fuel elements use $ZrH_{1.7}$. The H:^{235}U in the meat is about 225:1. The volume % water in the core is 33%. All the fuel elements in the KSU reactor are of the Mark III type.

Fuel Elements



MIII Elements

- 28.37 in. overall length by 1.47 in. diameter
- Meat 15 in. x 1.43 in. $ZrH_{1.7}$ and 8.5 weight % U (20% enriched). A central 0.18 in. diameter hole (used in hydriding fuel is plugged by a Zr rod
- End reflectors 3.44 in. graphite
- Small volume for expansion in fuel element (.04 in. gap)
- Stainless steel top and bottom end fixtures and cladding (0.020 in.)
- No burnable poison

Graphite Elements

- 28.44 in. overall length by 1.47 in. diameter
- 22 in. (55.9 cm) graphite core
- Aluminum end fixtures and cladding

Lattice Geometry

Ring	Inner Radius (cm)	Outer Radius (cm)	Max. Fuel Vol. Fractions
A	0	2.19	0.729
B	2.19	6.11	0.642
C	6.11	10.08	0.651
D	10.08	14.05	0.655
E	14.05	18.02	0.657
F	18.02	22.86	0.529

Reflector

- 3.44 in. graphite reflector top and bottom
- 12 in. (30 cm) radial graphite reflector followed by water
- Rotating Specimen Rack (RSR) with 40 locations can be embedded in the reactor
- Graphite was chosen because for reflector due to better neutron economy than water
- Graphite gives longer neutron lifetime hence longer period for a given prompt reactivity insertion
- 30 cm (12 in.) followed by 7 cm of water gives about the same critical mass and core flux profile as does a 50 cm graphite reflector

Startup Source

- Am-Be (α, n) source
- Half life 458 y
- Strength $2x10^6$ neutrons/s
- Reactivity worth $0.025 for core II-19 (stainless steel encapsulated)

Control Rods

- Four control rods
  - Pulse (i.e., transient)
  - Shim
  - Safety
  - Regulating

- Pulse (or "transient" or "safety") rod is 20 in. long and 1.24 in. diameter; consists of solid graphite impregnated with boron

- Regulating, safety, and shim rods are boron carbide sealed in aluminum. Dimensions: regulating: 20 in. x 7/8 in., shim and safety: 20 in. x 1.25 in.

- Operated inside perforated aluminum guide tubes 1.495 in. in diameter

- Total control rod worth $6.40

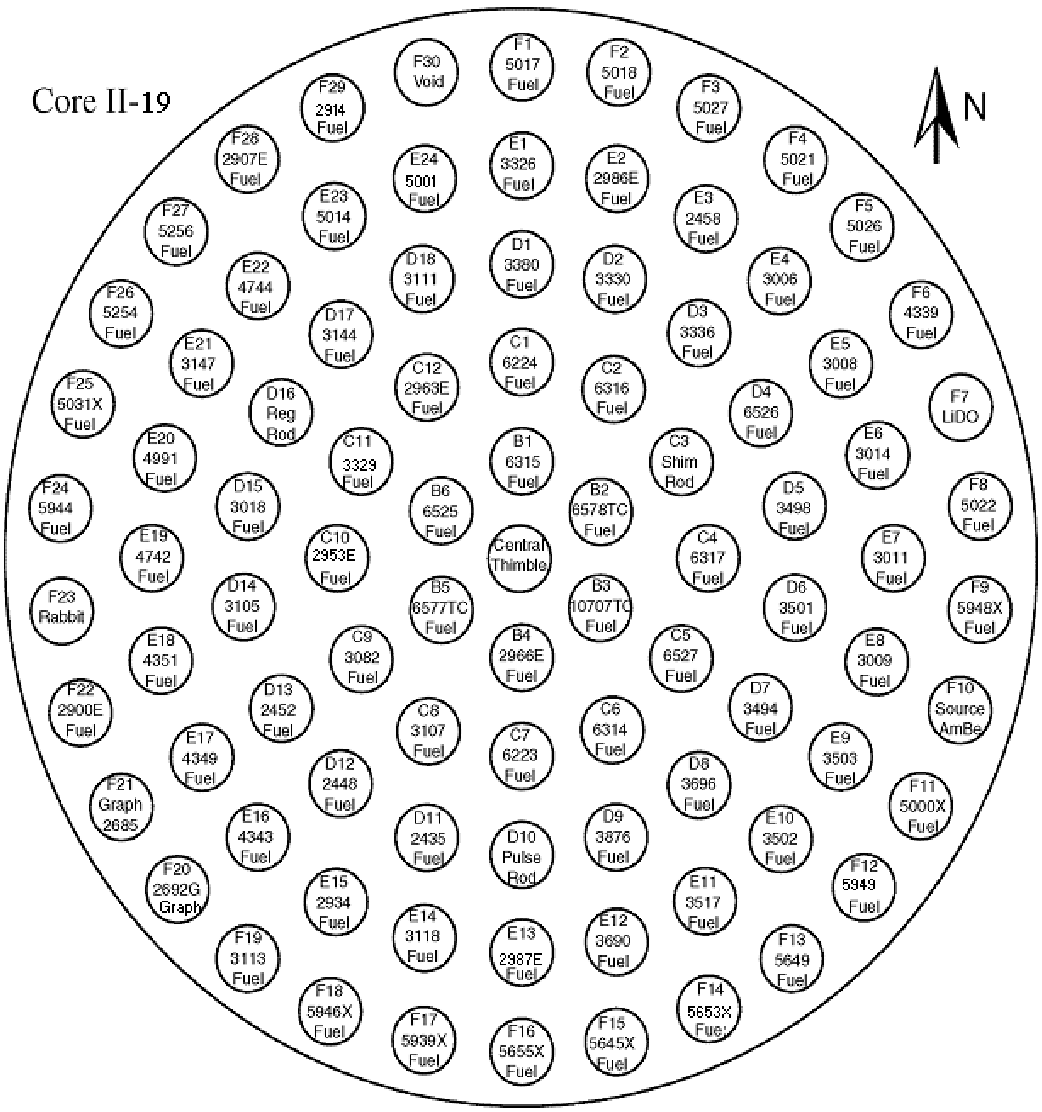


Figure 6 - Core lattice for Core II-19.

Miscellaneous Core Data

β_{eff} (effective delayed neutron fraction)	0.007
Prompt temperature coefficient	-\$0.017/°C at 250 kW (≈275°C)
Void coefficient of reactivity	-\$0.003/% void
Power coefficient (weighted average temperature coefficient)	-\$0.006 to -\$0.01/kW

IRRADIATION FACILITIES

Rotary Specimen Rack*
40 evenly spaced tubular containers, 1.25 in. I.D., 10.8 in. height, in a ring shaped aluminum housing located in a circular well in the reflector

*Note that the Rotary Specimen Rack is presently removed from service, but the reflector well that housed the RSR is available as an irradiation facility.

Central Thimble
Aluminum tube, 1.33 in. I.D., extending from the top of the reactor tank through central holes in the top and bottom grid plates, and terminating 7.5 in. below the bottom grid plate

Pneumatic Transfer System*
Helium driven "rabbits", 0.5 in. I.D., 4 in. long

Terminals in core, in reactor lower bay, and in neutron activation analysis laboratory

*Note that the pneumatic transfer system is currently removed from service

Bulk Shield Tank
8 ft x 9 ft x 12 ft deep

Available as shielded storage facility or thermal neutron irradiation facility

Fission plate can be installed at the end of the thermalizing column

THERMALIZING COLUMN

Installed between bulk shielding tank and reactor core

2 ft x 2 ft cross section, 4 ft 5 in. long, aluminum container internally lined with boral and polyethylene

From core outward: 8 in. graphite + 2 in. Pb + 19 in. void + 24 in. graphite

THERMAL COLUMN

Large boral and polyethylene lined, graphite filled aluminum container embedded in concrete shield

4 in. x 4 in. x 50 in. graphite blocks

Approximately 4 ft x 4 ft cross section; 5.5 ft long

2 in. lead shield curtain inside tank

Door recessed 3.5 in. thick

BEAM PORTS

Radial (two), NW & SW

Piercing radial, NE

Tangential, SE

Tubes; 6 in. diameter aluminum (inner section), 8 in. diameter cadmium-lined steel (outer sections)

Centerline: 2.75 in. below core centerline

Inner shield plug: 6 in. diameter x 48 in. long (180 lb) aluminum filled with, inner end to outer end: 0.125 in. boral, 4 in. Pb, 36 in. borated concrete, 5 in. steel

Outer shield plug: 8 in. diameter x 48 in. long (45 lb) wood

Safety shutter: steel lined lead, 9 in. square, 4.25 in. thick

Door: steel, line with 1.25 in. lead

RADIATION LEVEL AT EXPERIMENTAL LOCATIONS

	Dose Rates and Fluxes at 250 kW		
	Neutrons $cm^{-2}\ s^{-1}$		
Location	Fast (>10 keV)	Thermal (<0.21 eV)	Gamma (rad/s)
Central Thimble	1.2×10^{13}	1.0×10^{13}	2.5×10^{4}
"E" Ring	6.4×10^{12}	4.1×10^{12}	1.5×10^{4}
Pneumatic transfer system (F-ring rabbit terminus)	3.5×10^{12}	4.3×10^{12}	1.5×10^{4}
Piercing beam port next to core	2.0×10^{12}	2.0×10^{12}	1.0×10^{4}
Rotary Specimen Rack	1.5×10^{12}	1.8×10^{12}	4.0×10^{3}
Reactor pool outside reflector	6.8×10^{10}	6.8×10^{11}	4.5×10^{2}
Reactor pool inside reflector	1.1×10^{11}	3.4×10^{11}	---------

DOSE RATES AND FLUX DENSITIES AT BEAM PORTS

Measurements made at 50 cm from open face of port and normalized to 1 W of reactor thermal power. Lead shutter would reduce gamma ray exposure rate by factor of about 500.

Fast (Piercing) Beam Port

Total neutron flux (cm^{-2} s^{-1})	1700
Flux-average neutron energy (MeV)	0.5
Neutron dose rate (mrem/h)	64
Gamma-ray exposure rate (mR/h)	90

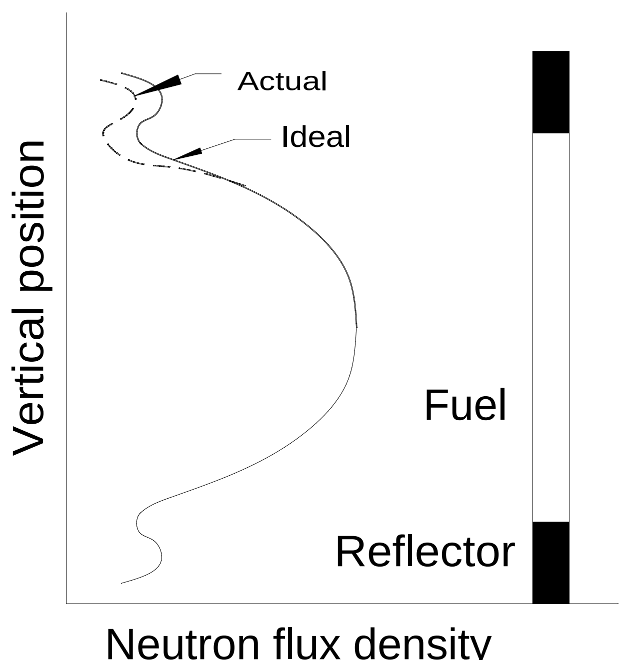
Tangential Beam Port

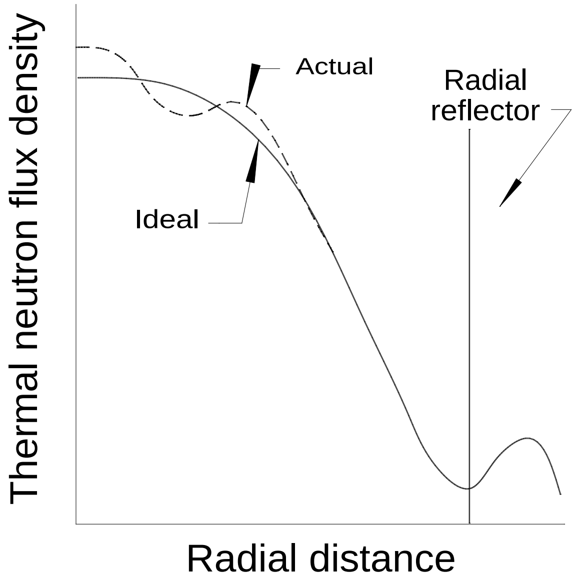
Total neutron flux (cm^{-2} s^{-1})	560
Flux-average neutron energy (MeV)	0.1
Neutron dose rate (mrem/h)	6.9
Gamma-ray exposure rate (mR/h)	1.7

Tangential Beam Port with 18 in Aluminum Filter

Total neutron flux (cm^{-2} s^{-1})	2.5
Flux-average neutron energy (MeV)	0.07
Neutron dose rate (mrem/h)	0.03
Gamma-ray exposure rate (mR/h)	0.02

Flux Distributions: (Thermal)





COOLANT SYSTEM

The TRIGA reactor cooling system serves four functions:

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

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

3. Maintains optical clarity of the water.

4. Provides a means of dissipating the heat generated in the reactor.

Primary Loop

The primary cooling system consists principally of a pump, a fiber cartridge filter, a mixed-bed type demineralizer, a flow meter, and a heat exchanger connected by suitable aluminum piping and valving. A surface skimmer is provided to assist in maintaining the cleanliness of the reactor tank water surface, and a monitor vessel contains probes for measuring the temperature, radioactivity, and conductivity of the water as it is passed through the system.

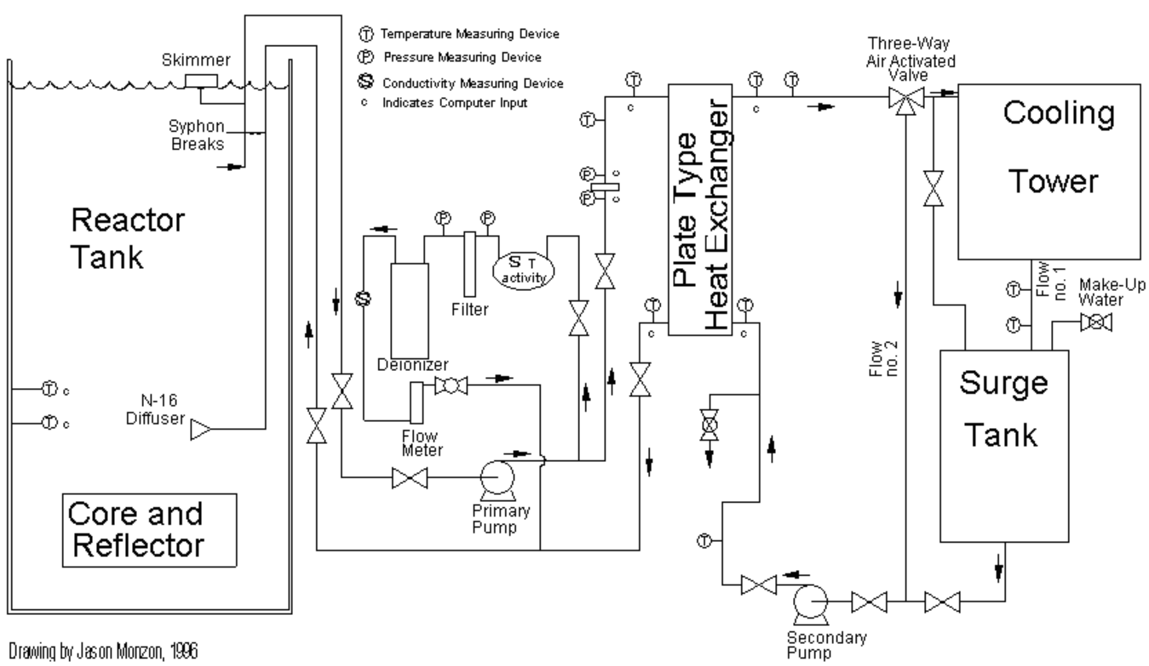


Figure 7 - Reactor cooling system

Flow path 1 is followed when the primary temperature exceeds the set point and the outside temperature is >-10°F. Flow path 2 is followed when the outside temperature is <-10°F or the primary temperature is less than the set temperature. Flow path 1 overrides path 2 when the outside temperature is <-10°F but the primary temperature is >110°F.

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

The water monitor vessel is made of aluminum and contains (1) a Geiger tube for detecting radioactivity in the water, (2) a conductivity probe for measuring water purity, and (3) a temperature prove for measuring water temperature. These three components are circuited to the reactor console for display purposes. The radioactivity of the primary water may also be measured with a Geiger-Mueller detector placed adjacent to the demineralizer and cartridge filter.

The water system pump is a centrifugal self-priming type with and aluminum body and impeller. A direct-coupled induction motor drives the pump.

The filter removes insoluble particulate matter from the reactor water system. It contains two replaceable filter cartridges of 5 micron rating. In addition to improving the optical clarity of the water in the reactor tank, the removal of solid particles from the water by the filter extends the life of the demineralizer resin.
Two pressure gages are provided to measure the pressure drop across the filter as an aid in determining the extent of filter clogging.

The demineralizer maintains the conductivity of the water at a sufficiently low level to prevent corrosion of the submerged reactor components, particularly the fuel elements. The demineralizer is of the mixed anion and cation resin bed type for removal of both positive and negative ions, and contains 3 ft^3 (0.085 m^3) of mixed resin. In addition to the conductivity probe in the monitor vessel, a second probe downstream from the demineralizer measures the conductivity of the water as it leaves the demineralizer, thus indicating whether the demineralizer is working properly.

The flow meter has a range of 1 to 28 gallons per minute. The meter is operated by the flow of water forcing a metal plug upward in a transparent tube; changes in flow rate will produce a corresponding change in the height of the plug. A scale behind the transparent tube is calibrated to indicate the flow in gallons per minute. The cleanup loop has a nominal flow rate of 10 GPM, however, the flow rate may be as low as 5 GPM depending on the amount and type of demineralizer resin.

A plate-type heat exchanger is used to remove heat from the reactor tank water. The heat exchanger consists of sandwiched stainless steel plates alternately carrying the primary and secondary cooling water. The heat exchanger has a capacity of 875,000 BTU/h. Four temperature transducers are located in inlet and outlet streams, and two in the reactor tank, for performance monitoring of the heat exchanger.

All piping and fittings in the primary system are of aluminum alloy. The piping is primarily of nominal 2.5 in. diameter pipe, except for the purification loop, which is of nominal 1 in. diameter pipe.

Secondary Loop

The secondary automatic control system has three basic control functions:

1. To maintain the primary water at a set temperature.

2. To control the cooling tower fan speed to maintain secondary water temperature within a set range.

3. To prevent cooling water freeze-up during cold weather operations (less than -10°F).

The secondary automatic control system performs these functions by pneumatically controlling the three-way valve and by electrically controlling the cooling tower fan. The secondary automatic control system is normally initiated by energizing the primary cooling pump. The three-way valve allows secondary to flow to the cooling tower under normal conditions. If the outside air temperature is less than -10°F (-23°C), then the three-way valve stops cooling tower flow. The outside air temperature limit of -10°F is bypassed if the primary water temperature exceeds 110°F (43.3°C) thus reestablishing cooling tower flow. The temperature of secondary water returning from the cooling tower controls the cooling tower fan speed. At 70°F (21.1°C), the cooling tower fan starts at low speed. At 90°F (32.2°C), the fan goes to high speed.

REACTOR INSTRUMENTATION

The KSU TRIGA instrumentation is a hybrid of original equipment and newer replacement equipment, with most of the equipment located within the control console. A line voltage conditioner, located in the lower reactor room, supplies all primary power to the TRIGA instruments. A loss of building power will de-energize the line conditioner until it is manually reset, thus allowing the orderly re-energizing of equipment. The control console is normally continuously energized.

Measurement

The control console provides four major measurement functions: neutron flux (power), radiation intensities, pool temperature, and pool water conductivity.

Three independent channels measure neutron flux: log wide range, linear multi-range, and percent power.

The log wide range channel uses a fission chamber in pulse mode for detecting thermal neutrons in the range of 1.4 to 1.4×10^5 nv, and provides approximately 0.7 counts/nv. The detector uses pulse height discrimination to distinguish neutron pulses from gamma pulses. A fission will produce a pulse of about 70-100 MeV due to the deposition of all of a fission product ion's energy into the detector gas. A gamma will deposit much less energy due to the lower energy of the gamma and the lower linear energy transfer of gammas relative to ions. Pulse height discrimination is used to eliminate the low-energy pulses, thereby removing the gamma signal. The detector has an aluminum case, an aluminum electrode, U_3O_8 (>90% enriched in ^{235}U) as the neutron sensitive material, and an argon-nitrogen mixture for a fill gas. A preamplifier is used to minimize noise and signal loss from the detector to the console, and is located at the upper level of the reactor room. The remainder of the channel circuitry is located in the NLW 1000 unit in the central console. The instrument switches from pulse mode operation to current mode as reactor power increases out of the source range, allowing the instrument to measure reactor power in the upper ranges. Three displays indicate reactor power, high voltage, and reactor period. The period meter has a scram at 3 seconds. There is also a high voltage scram on this channel. Both scrams are bypassed in pulse mode. This channel also provides a protective interlock which prevents rod withdrawal when indicated neutron flux is < 2 cps and a pulsing interlock when reactor power is above 10 kW.

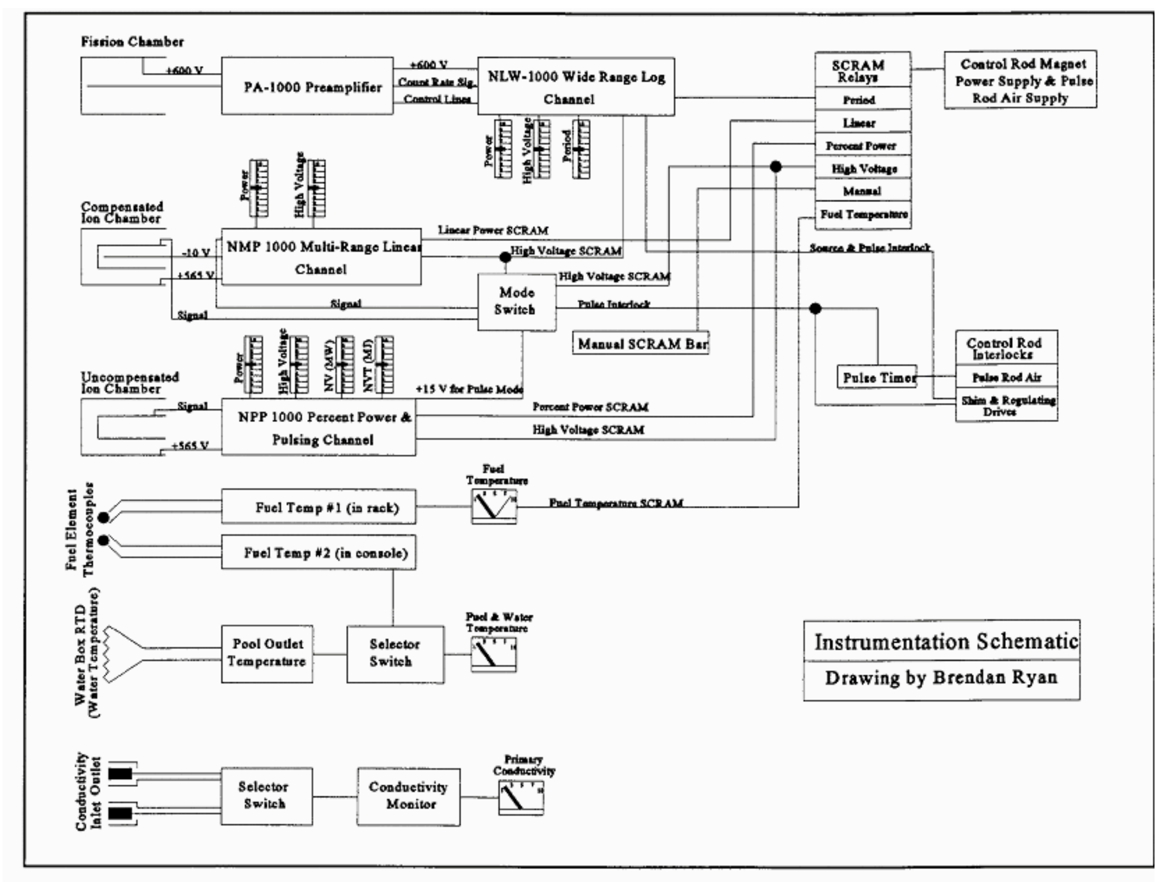


Figure 8 - Instrumentation functional block diagram

The second channel on the console provides a multi-range linear power indication. This channel uses a compensated ion chamber for detection of thermal neutrons. The linear channel detector signal goes directly to the NMP1000 unit, located in the center console. This unit features automatic or manual ranging. The instrument provides two outputs: a visual indication of power and high voltage. The power signal is also sent to the strip-chart recorder in the instrumentation rack. There are two scrams associated with this channel: a high power level scram (nominally 104% nominal rated power) and a high voltage scram. The signal from the detector, the high linear power scram, and the high voltage scram are bypassed in the pulsing mode.

Power range indication of neutron flux is provided by an uncompensated ion chamber signal, which indicates percentage of power in the upper two decades of the power range. The uncompensated ion chamber is virtually identical in construction to the compensated ion chamber, but no gamma compensation is provided in the circuitry.

The radiation monitor in the instrumentation rack provides an indication of radiation levels directly above the primary pool water surface. A meter on the right-hand section of the rack indicates a radiation dose rate of 100 mr/h full scale, utilizing a GM tube detector attached to the bridge section at the top of the primary water tank. A GM tube detector measuring coolant activity in the cleanup loop operates in parallel with this detector. Various other radiation measurement system are employed throughout the reactor facility, including 5 remote area monitors (4 permanent, 1 mobile), a 5 R/h evacuation alarm, several air activity monitors, and numerous portable radiation monitors.

Temperature indications for the primary water and specific B-ring fuel elements are provided on the front section of the control panel and in the instrumentation rack. The instrumented fuel elements have three chromel-alumel (i.e., K-type) thermocouples in the fuel meat. The average of the three thermocouples is displayed as temperature indication on the console. The temperature probe in the primary cleanup loop is a nickel alloy thermistor, and is displayed on a console meter. A reactor scram occurs if the measured fuel temperature exceeds 450°C.

Primary water conductivity is measured at the inlet and outlet of the purification loop by titanium electrode cells, which send signals to a bridge circuit in the instrumentation rack. The bridge circuit is automatically temperature compensated and nulled which provides a correct conductivity measurement. The inlet and outlet conductivities provide a good indication of the purity of the primary water and the effectiveness of the ion exchanger.

Safety Systems

Two major safety systems are incorporated in the control console: control interlocks and scrams.

Several interlocks are built into the control system of the reactor to prevent improper operation.

No control rod withdrawal is possible unless the count rate neutron channel is indicating >2 cps. This interlock prevents the possibility of a sourceless startup.

Air may not be applied to the pulse rod if the pulse rod shock absorber is above its full down position and the reactor is in the steady state mode. This interlock prevents the inadvertent pulsing of a reactor in the steady state mode.

There is no simultaneous withdrawal of two or more control rods when the reactor is in the steady state mode. This interlock was intended to prevent violation of the maximum reactivity insertion

rate of the reactor. At this time, however, there is not a maximum reactivity insertion rate specified in the facility Technical Specifications.
The pulse rod is the only control rod that can be moved when the reactor is in the PULSE mode (this does not prevent the scramming of any control rod). This interlock minimizes the possibility of pulsing a supercritical reactor.

Automatic and manual scrams are provided for reactor safety. All scram functions cause all four control rods to drop to the bottom of their travel. The manual scram is initiated by depressing the scram bar located above the rod drive buttons. The automatic scram is initiated by:

- Loss of high voltage for the nuclear instrumentation
- High linear power
- High percent power
- High fuel temperature
- Short reactor period

The short reactor period scram is not required for reactor operation. In the pulsing mode the linear, percent power, log high voltage, linear high voltage, and period scrams are not functional. (I.e.., only the fuel temperature scram and manual scram are functional in pulse mode). This allows peak power during a pulse to increase up to 1,000 times its nominal 100% power.

OPERATION OF THE SHIM, SAFETY, AND REGULATING RODS

The rod drive mechanism is an electric motor actuated linear drive equipped with a magnetic coupler. Its purpose is to adjust the reactor control rod positions.

Mechanical Operation

A 110 Volt, 60 cps, two-phase motor drives a pinion gear and a 10-turn potentiometer. The potentiometer may be employed to provide rod position indications. The pinion engages a rack attached to the magnet drawtube. An electromagnet mounted on the lower end of the drawtube engages an iron armature that screws into the end of a long connecting rod, which terminates (at its lower end) in the control rod. The magnet, armature, and upper portion of the connecting rod are housed in a tubular barrel that extends well below the reactor water line. Located part way down the connecting rod is a piston equipped with a stainless steel piston ring. Rotation of the motor shaft rotates the pinion, thus raising or lowering the magnet drawtube. If the magnet is energized, the armature and connecting rod will follow the drawtube so that the control rod is withdrawn from or inserted into the reactor core. In the event of a reactor scram, the magnet will be de-energized and release the armature. The connecting rod, piston, and control rod will then drop, thus reinserting the control rod into the reactor. Since the upper portion of the barrel is well ventilated the piston will move freely through this range. However, when the connecting rod is within 2 in. of the bottom of its travel the piston is restrained by the dashpot action of the restricted ports in the lower end of barrel. This restraint cushions bottoming impact.

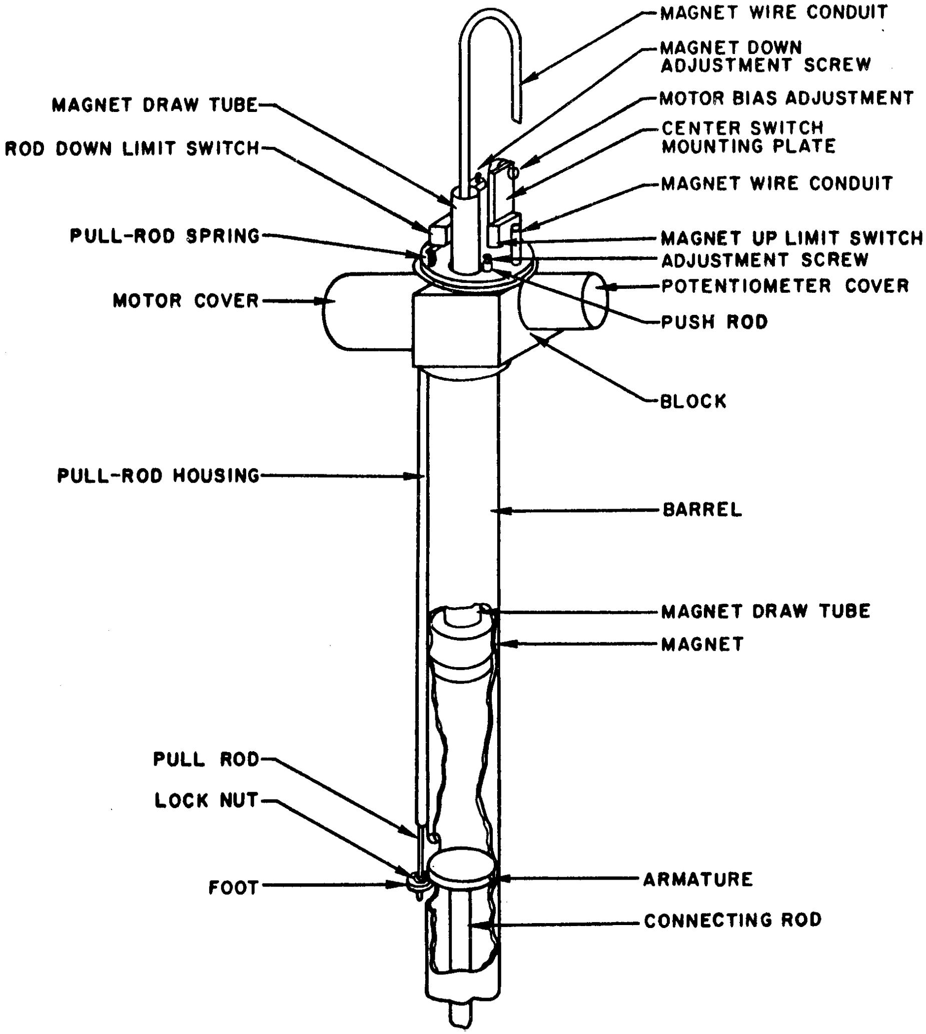


Figure 9 - Standard control rod drive mechanism

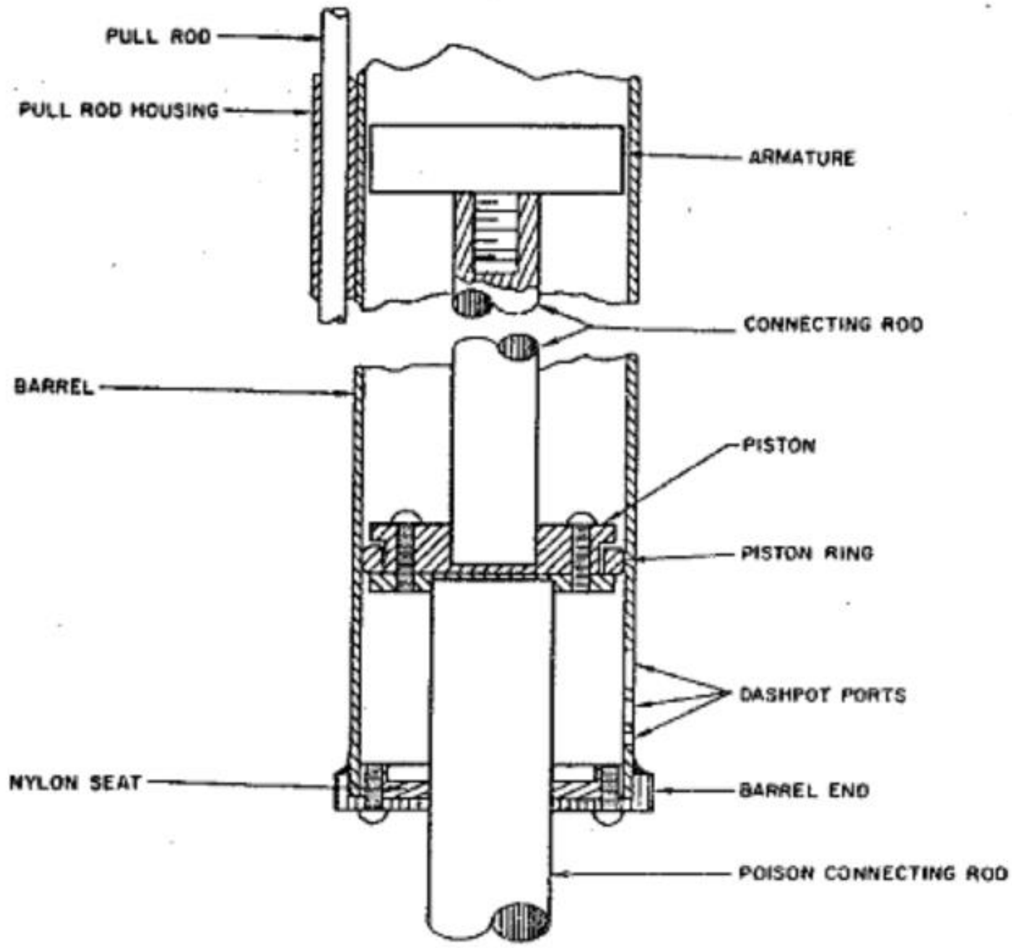


Figure 10 - Standard rod drive mechanism details

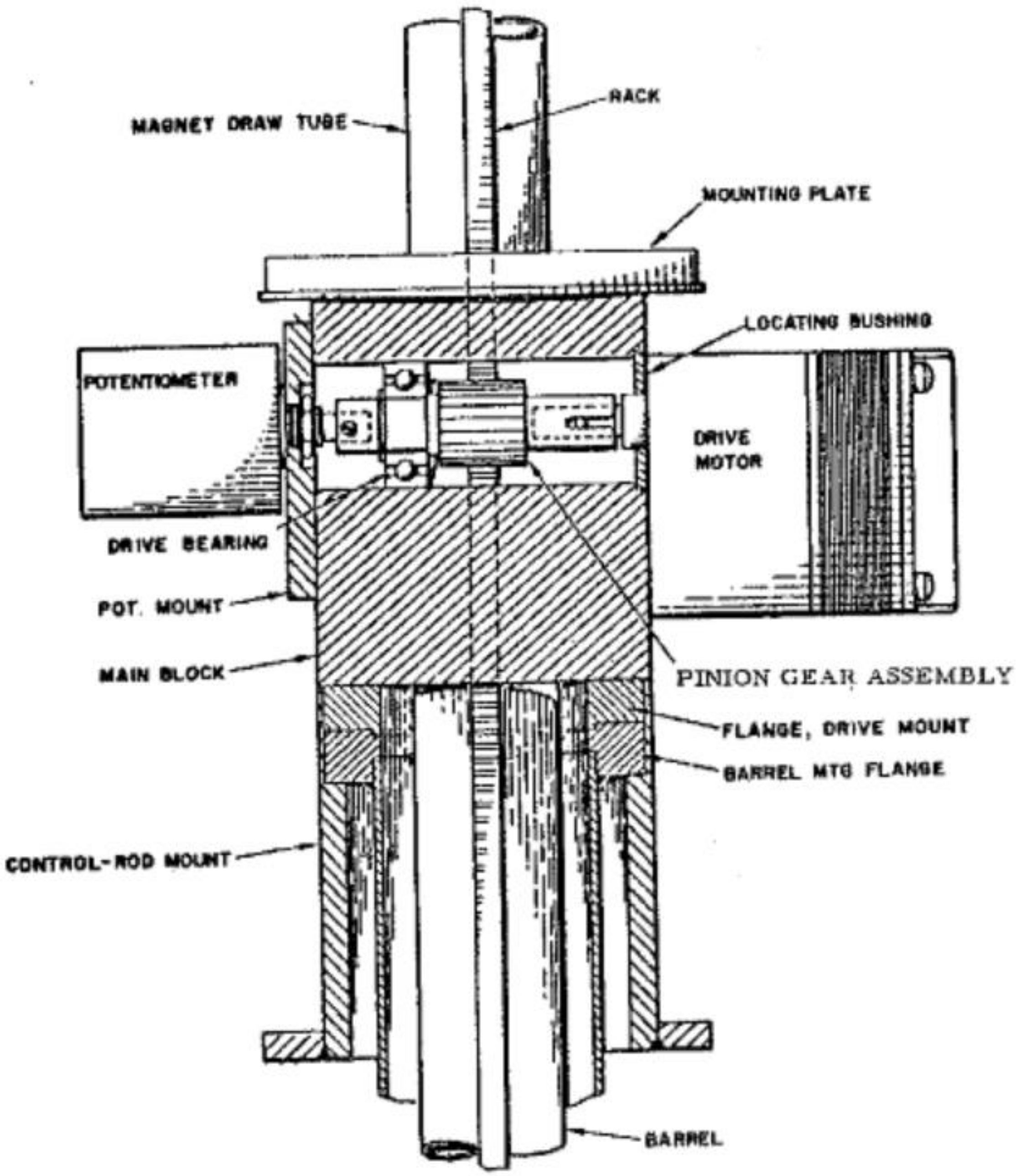


Figure 11 - Standard rod drive mechanism details

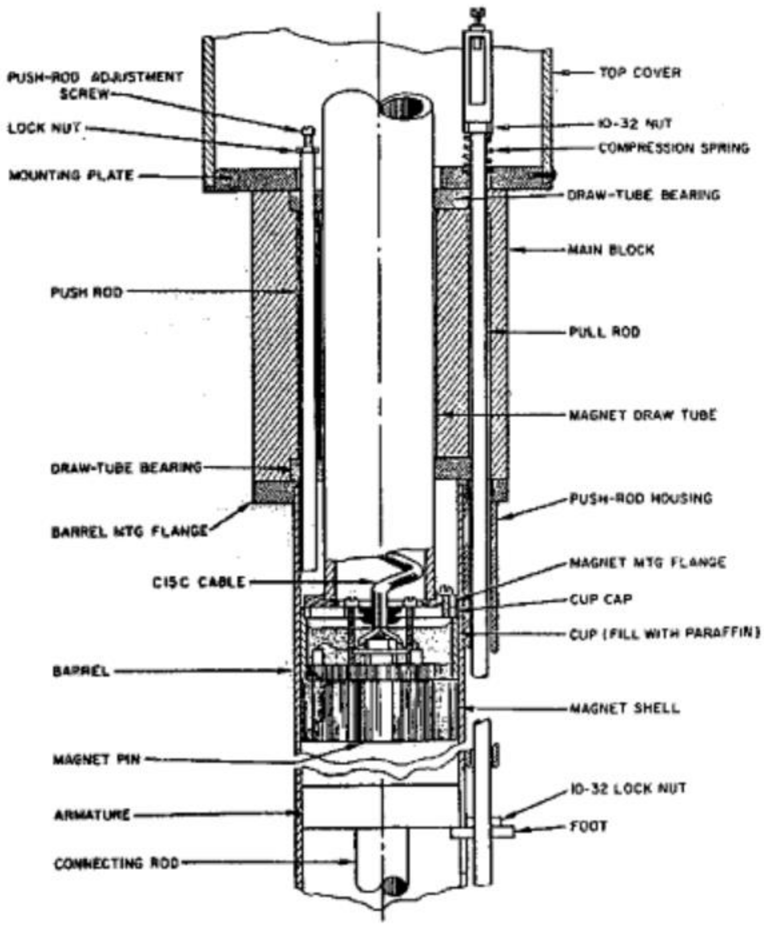


Figure 12 - Standard rod drive mechanism details

Circuit Operations

The rod drive motor is dynamically braked and held by an electrically locked motor. In the static condition both windings are energized with the same phase. Thus, the motor is electrically locked. In order to allow clockwise (up) or counter-clockwise (down) rotation, the phase between the windings is shifted by a 1 μF capacitor. Therefore, the motor control switches simply allow the appropriate phase shift.

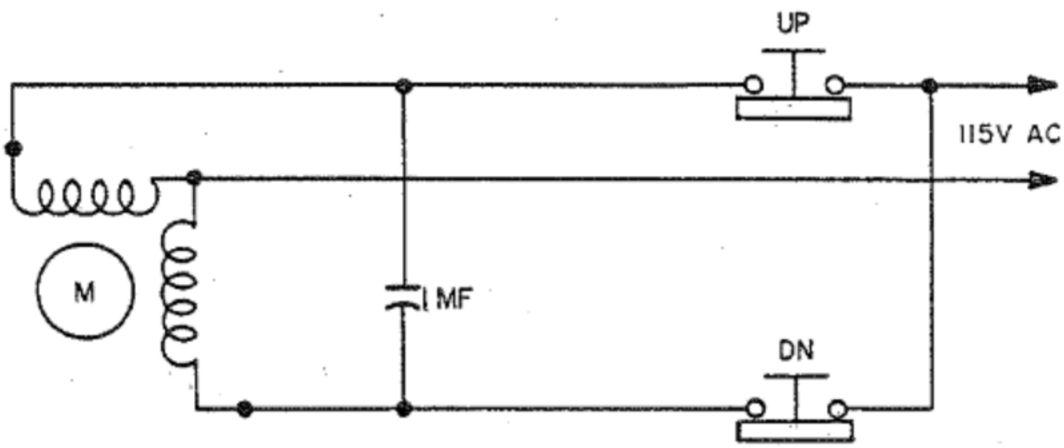


Figure 13 - Simplied rod drive motor control circuit

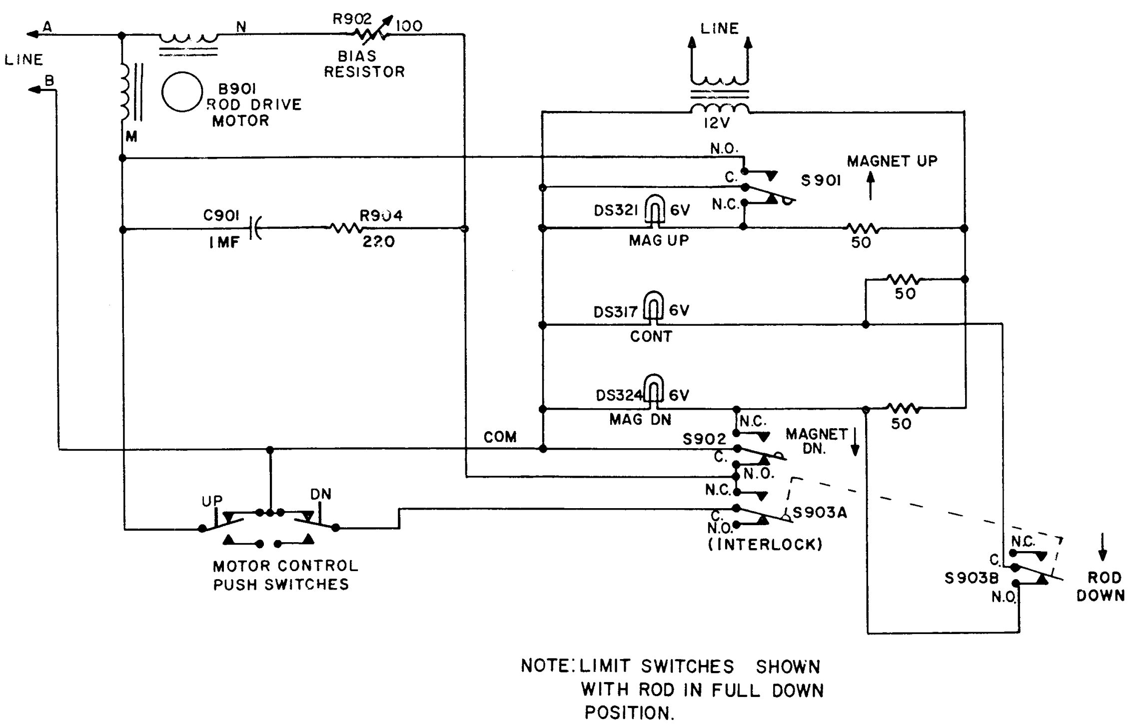


Figure 14 - Rod drive, motor control, and indicator lamp circuit

Three microswitches limit and control the travel of the magnet drawtube. Actuation of the magnet up limit microswitch (S901) applies line voltage to one winding therefore allowing only the phase shift that gives counter-clockwise rotation.

Actuation of the magnet down limit microswitch (S902) applies line voltage to the other winding therefore allowing only the phase shift that gives clockwise rotation. Actuation of the rod down microswitch (S903A) causes the phase shift for counter-clockwise rotation. Therefore, if the control rod drops, the magnet drawtube drives down until the magnet down limit microswitch locks the rotor. Since the rod down microswitch drives the magnet drawtube down, than the rod down microswitch must be open before the magnet down microswitch during coupled withdrawal of the control rod.

Three lights indicate that 1) the magnet drawtube is full up, 2) the magnet drawtube is full down, and 3) the armature and the magnet are coupled. When the magnet drawtube is full up, microswitch (901) is actuated opening the short across the mag up light (DS321). When the magnet drawtube is full down, microswitch (S02) is actuated opening the short across the mag down light (DS324). When the control rod drops, the unactuated magnet down microswitch (S902) and the actuated rod down microswitch (S903B) short the contact light (Ds317) indicating separation of the magnet and the armature.

Other features of the circuit are an adjustable bias resistor (R902), a 220 ohm surge resistor, and 50 ohm current limiting resistors. The adjustable bias resistor compensates for the torque applied by the weight of the control rod and the magnet drawtube. The 220 ohm surge resistor limits the capacitor current surge during switching. The 50 ohm current limiting resistors limit the currents in the 12 volt indication circuits when the indicating lamps are shorted.

The unconventional circuit employed in the rod-drive system minimizes the number of switch contacts required. Therefore relays with their attendant reliability problems are not required. It should be noted that the rod drive units are identical both mechanically and electrically they are, therefore, interchangeable.

Normal rod motion speed is about 12 in. per minute. Typically the standard control rod position indication is at 100 units with the rod fully inserted and 1100 units with the rod fully withdrawn.

Using this data along with the respective rod worth curves, the operator can determine the reactivity insertion rate for a given interval of rod motion.

The rod position indicators are 3½ digit LED display indicators, which receive a variable DC voltage input from 10-turn potentiometers that are driven by the respective rod drive motors. The digital display simply indicates a voltage, which is directly related to the control rod position. The position indicators have their own variable power supplies and are therefore completely independent. The indicator systems are located in the control console except for the 10-turn potentiometers and associated wiring.

TRANSIENT ROD DRIVE MECHANISMS

The transient rod-drive is an electrically controlled, pneumatically operated, mechanically limited system. The transient rod, and its aluminum extension rod, is mechanically connected to a pneumatically driven piston inside a worm-gear and ball-screw assembly. The system is housed on a steel support structure mounted above the reactor tank. Air to the piston is controlled by means of a three-way solenoid valve mounted below the support structure. The throw of the piston, and hence the amount of reactivity inserted into the core during pulsing operations, is regulated by adjusting the worm-gear and ball-screw assembly. The adjustment is made from the central console by actuating a reversible motor drive, which is coupled to a worm-gear and a 10-turn potentiometer for position indication. The operation of the position indicator is identical to that of the shim and regulating control rods.

The solenoid valve is actuated by means of the console mounted “rod fire” and “air” switches. When the rod fire switch is depressed, the solenoid valve opens, admitting air to the cylinder, coupling the piston and rod to the shock absorber. Depressing the air switch de-energizes the solenoid valve, which removes air from the cylinder and vents air to the atmosphere. In the event of a reactor scram, the solenoid will be de-energized via the scram circuitry, which will allow the transient rod to drop into the core after the air is removed. Microswitches are used to indicate the extreme positions, up or down, of the shock absorber.

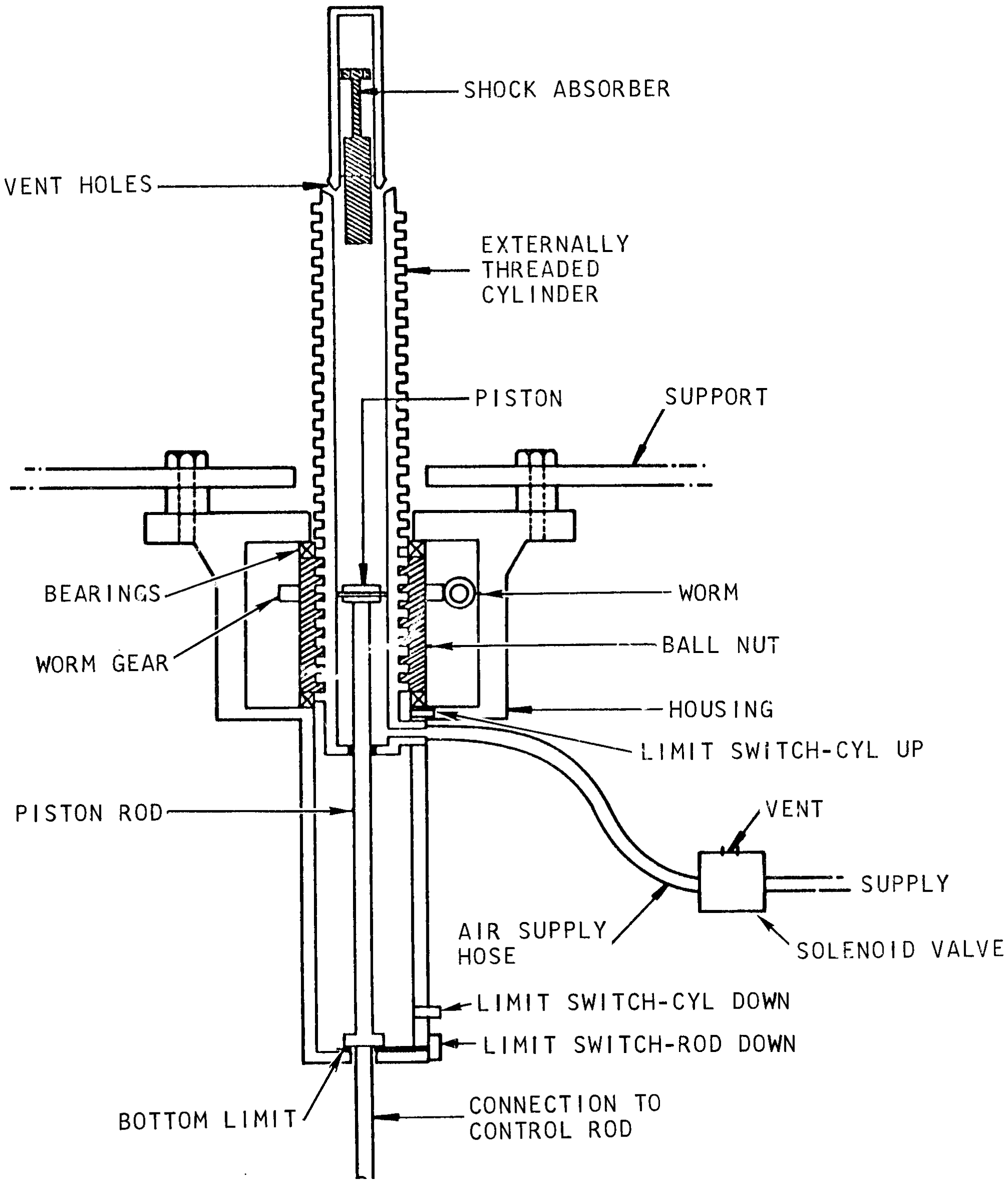


Figure 15 - Transient rod drive mechanism

In the pulse mode, a variable timer de-energizes the solenoid valve six seconds after the pulse is initiated. The shock absorber will remain in its preset position until the mode selector switch is taken to steady state. In steady state mode of operation, the six second timer is disengaged and the cylinder remains pressurized.

If the air supply for the pulse rod drive should drop below approximately 45 psi, an amber low air pressure warning light will be actuated on the control console.

Kansas State University TRIGA Mk II Reactor Facility Operations Manual

1.0 INTRODUCTION

The primary purpose of this manual is to provide explicit operating, maintenance, and experiment procedures which will minimize the probability of unintentional reactor excursions, unsafe practices involving the use or handling of radioactive materials within the reactor room, or unsafe practices involving radiation originating in the reactor core or fuel. A secondary purpose of the manual is to provide an operational guide to optimize security and maintenance of the reactor to the end that the useful life of the reactor shall be extended through the greatest possible time interval. Implicit in these instructions is the philosophy that the general public not be exposed to significant radiation and that no action be taken which might lead the general public to be anxious about the influence of this reactor on their safety. The Operations Manual is subject to amendment by proper authority. The method by which amendment is accomplished is set forth in the Administrative Plan. Integral parts of the Operations Manual, though published and amended separately, are all approved Procedures (including the Emergency Plan, the Physical Security Plan, and the Requalification Program), all approved Experiments, General Atomics Report GA-3399 “250-kW TRIGA MK II Pulsing Reactor Mechanical Maintenance and Operating Manual,” and General Atomics Report GA-9039 “TRIGA MK II Reactor Instrumentation and Maintenance Manual.” It is intended that the reactor always be operated in accordance with the provisions of the Operations Manual. The Manual, including all amendments thereto, shall reflect and is subject to the conditions and restrictions of the current KSUTMII Utilization Facility License and the associated Hazards Summary Report.

2.0 ADMINISTRATIVE STRUCTURE

The Reactor Supervisor is responsible for the safe and secure operation of the reactor, and as such is responsible for determining that the reactor is operated in accordance with the Operations Manual. The Reactor Supervisor is administratively responsible to the Nuclear Reactor Facility Manager. The Manager is responsible to the Head of the Department of Mechanical and Nuclear Engineering, who is responsible to the Dean of the College of Engineering. The Dean is responsible to the President of the University. Regarding matters of reactor safety[1], the Kansas State University Reactor Safeguards Committee shall advise the Nuclear Reactor Facility Manager and the Head of the Department of Mechanical and Nuclear Engineering. This advice shall be binding. Nominally, this advice shall be given in the form of review-for-approval of amendments and additions to the Operations Manual. Hence, only those operational procedures approved as part of this manual by the Reactor Safeguards Committee shall be permissible for execution. The authority of the Radiation Safety Committee in connection with reactor operation is specified in the Administrative Plan.

The Reactor Operators shall be directly responsible to the Reactor Supervisor and shall determine that the reactor is operated in strict accordance with the instructions of the Reactor Supervisor

1 “Reactor safety” is construed to mean prevention of accidental reactor power excursions, the prevention of accidental accumulation of critical or near-critical mass of fissile materials, and the avoidance of significant exposure of personnel to radiation caused by the operation of the reactor.

The Experimenter is responsible for planning an experiment, performing a preliminary investigation of the hazards involved and notifying the Reactor Manager of these hazards, designing necessary equipment, and for taking / evaluating the experimental data. The Experimenter may make recommendations to the Reactor Operator regarding reactor operation. However, the Reactor Operator is responsible for safe operation of the reactor, and the Experimenter has no authority to order changes to reactor power or reactivity, or to order the Reactor Operator to manipulate controls in a particular way. In case of a disagreement considering the safety of operation, the experiment must be stopped until an agreement is reached.
Administrative structure and the setting of the reactor within the structure is defined in greater detail in the reactor Administrative Plan.

3.0 Planning and Scheduling of Experiments

The Experimenter must prepare a written description of each proposed new experiment and submit this proposal to the Reactor Supervisor. This description must contain sufficient detail to enable evaluation of the safety of the experiment. It is recommended that the Experimenter provide their proposal at least one month prior to the planned experiment performance date to allow the proposal to be reviewed and approved. If possible, the experiment should be written in a format similar to existing experiments in the Operations Manual. The following minimum data must be given for the proposed experiment.

1. Objective of the experiment
2. Background (if appropriate)
3. Procedure – to include a description of the experimental methods and equipment to be used. A sketch of the physical layout and list of necessary equipment is recommended if appropriate.
4. Safety considerations, with explicit reference to the Technical Specifiations.
5. References.

If the Experimenter believes his proposed experiment to be an Approved Reactor Experiment, and the Reactor Supervisor concurs, then submission of an experiment proposal is not required. The University Radiation Safety Officer must concur in order for any approval of a reactor experiment on the part of the Reactor Safeguards Committee to be effective.
Form KSUTMII-2, “Request for Kansas State University TRIGA Mark II Operation,” must be submitted prior to execution of experiments involving potential changes to core reactivity of an absolute magnitude > $0.40, experiments with components internal to the reactor tank (excluding irradiation samples and their encapsulation), and experiments which produce radioisotopes for which no byproduct log entry exists.

4.0 OPERATIONAL PRACTICES

4.1 Admittance to the Reactor Bay

Visitor control is prescribed by Procedure 9. Normal entrance to the reactor bay is through the access door in the Control Room. Entrance through this door shall be controlled according to the requirements of the Reactor Operator on duty When the reactor is in operation, all persons wishing to enter the reactor bay must first obtain authorization for entry from the Reactor Operator on duty. It is the operator's responsibility to be aware of the existence of all radiation hazards in the reactor bay and to advise prospective visitors to the reactor bay of these hazards. The Reactor Operator on duty may, if he judges it to be necessary, prohibit entrance to the reactor bay for any or all visitors.

4.2 Key Control

Key control is prescribed by the Reactor Facility Physical Security Plan. Issuance of a key to the control room or reactor room requires the approval of the Reactor Supervisor and the Chairman of the Reactor Safeguards Committee.

4.3 Reactor Tours

For public relations ceremonies and reactor tours during which large numbers of casual visitors may seek admittance to the reactor room, the personnel monitoring requirements of Procedure 9 may be relaxed with prior approval of the Chairman of the Reactor Safeguards Committee, the Chairman of the Radiation Safety Committee, the Reactor Manager, or the Reactor Supervisor. During such ceremonies or tours, the reactor must be secured. Tour guides must carry self-reading pocket dosimeters. Conditions in the reactor room shall be such that visitors not be exposed to external dose rates in excess of those permitted for minors in 10CFR20.

5.0 OPERATIONAL LIMITATIONS

The operation of the reactor shall be subject to the following procedural requirements:

1. All reactor operations shall be performed in accordance with the Reactor Facility License (R-88) and Technical Specifications, as well as all applicable federal, state, and university regulations.
2. No experiment shall be performed until the experimental procedure has been approved by the Reactor Safeguards Committee.
3. A semi-annual inspection of the facility shall be made by the Reactor Safeguards Committee.
4. A semi-annual audit of reactor operations shall be performed by the Reactor Manager and reported to the Reactor Safeguards Committee.
5. All unexplained scrams must be reported to the Reactor Supervisor and entered in the log.
6. At the start of each operating day, or after a major interruption of operation, the Reactor Operator is responsible for checking out and certifying that the reactor electrical and mechanical systems are in proper working order, in accordance with the approved checkout procedure.
7. A reactor logbook must be kept and shall contain a complete record of all reactor operations, with a maximum of five minutes delay in recording.
8. The Reactor Manager may authorize temporary changes to a procedure, provided that:
   a) the changes do not alter the original intent of the procedure;
   b) the changes are noted in the operations logbook;
   c) all licensed individuals are informed of the changes;
   d) the changes are submitted for approval by the Reactor Safeguards Committee at the earliest possible time, as determined by the Reactor Manager.

6.0 OPERATIONS LOGBOOK

The Operations Logbook must contain a complete record of all operations and events which affect the reactor. It is the responsibility of the Reactor Operator to keep the Operations Logbook in a complete and concise manner. A sample page of the Logbook is attached.

Logbook entries will take the following form, if appropriate, with entries for each operating day beginning on a new page:

1. time of event;
2. description of event;
3. action taken;
4. person reporting.

All unintentional scrams and their cause, as well as operational malfunctions of essential equipment, must be entered in the logbook in red. Every attempt should be made to find the reason for such scrams. All unexplained scrams must be reported to the Reactor Supervisor.

All changes in reactor configuration, fuel loading, control rod position and worth, insertion of experimental apparatus, etc., shall be recorded.

In addition to reactor operations, the logbook should contain notes on equipment. The Reactor Operator shall note in the logbook the malfunction of any equipment associated with the reactor.

7.0 CHECKLISTS

7.1 Daily Checklist

Each day before the reactor is operated a series of observations and checks must be performed by the Reactor Operator to determine that the reactor is ready for operation. A copy of the daily checklist is attached. If any element of this check shows an unsatisfactory response, the circumstances must be reported to the Reactor Supervisor, even though the condition may not preclude operation of the reactor.

7.2 Additional Periodic Checklists

In addition to the daily checklist, periodic inspection of portions of the equipment must be made at less frequent intervals. Therefore, weekly, monthly, and semi-annual checklists have been compiled to facilitate these inspections. These checklists are incorporated into the Monthly Maintenance and Surveillance Report and the Monthly Reactor Utilization Report, copies of which are attached.

The Reactor Manager is required to carry out a semi-annual audit of reactor operations and to report the results to the Reactor Safeguards Committee. A copy of the audit report is attached.

8.0 SAFETY REGULATIONS

8.1 General Safety Regulations

Staff personnel working in the reactor bay shall have a knowledge of the biological effects of radiation, the maximum permissible doses, and the techniques for controlling radiation exposure as set forth in 10CFR20, "Standards for Protection Against Radiation."

The following are general safety rules which must be enforced by the Reactor Supervisor.

1. Two persons must be in the reactor bay before any fuel handling or maintenance work is done in the reactor tank on the reactor core, control rod drives, or control rods.
2. All irradiated samples must be monitored as they are removed from the reactor.
3. A life preserver will be placed within easy access of the top of the core tank and the top of the bulk shielding facility.
4. *Use of solvents* – Unless higher limits are specifically approved by the Reactor Supervisor, organic solvents in a volume greater than ¼ liter shall not be exposed in the reactor bay. The use of acetone, gasoline, or carbon tetrachloride is permitted only with specific approval of the Reactor Supervisor.
5. *Mercury* – Approval of the Reactor Supervisor must be obtained before exposed mercury is used in the reactor bay. Use of mercury-glass thermometers in the reactor pool is expressly forbidden. No exceptions may be granted.
6. *Volatile isotopes* – No radioisotopes in greater than exempt concentrations in liquid, gas, or suspension form may be produced or used in the reactor bay without the approval of the Reactor Supervisor and University Radiation Safety Officer.
7.

8.2 Radiation Warning Signs

Zones of potential radiation and radioactivity hazards shall be clearly marked so that personnel will not expose themselves unknowingly or unnecessarily. The posting and maintenance of warning signs is the responsibility of the Reactor Supervisor, who will be advised by the University Radiation Safety Officer. In accordance with 10CFR20, areas with radiation levels in excess of 5 mrem per hour shall be marked CAUTION, RADIATION AREA. Areas with radiation levels in excess of 100 mrem per hour shall be marked CAUTION, HIGH RADIAITON AREA.

Boundaries will also be established and maintained around any reactor facility areas from which radioactive contamination might spread. Monitoring instruments will be available for use at these zones.

8.3 Methods of Radiation Control (Also see Emergency Plan)

Prior to the occupancy of any radiation area, evaluation of the delivered dose rate with appropriate radiation survey instruments shall be made and exposure periods shall be limited, if necessary, to comply with the provisions of 10CFR20. Survey equipment used may include:

1. Neutron survey meter;
2. Geiger type beta-gamma survey meter;
3. Portable thin window ionization chambers.

The Reactor Facility shall be surveyed for surface contamination at intervals of not longer than once per month, and more frequently if so directed by the Reactor Supervisor. These surveys are in addition to the continuous radiation monitoring of the reactor core tank water and the reactor bay. Environmental radiation surveys, with the reactor at full power, shall be performed inside and outside the reactor building at semi-annual intervals.

8.4 Personnel Monitoring

Film badges[2] containing both gamma- and neutron-sensitive film will be issued to, and worn by, all staff personnel and students who regularly work at the reactor facility. These badges shall be processed monthly. Pocket ionization chambers sensitive to both thermal neutrons and gamma rays may also be used by students and staff.

It is the responsibility of the Reactor Supervisor, the Reactor Operator on duty, the Experimenter, or the person authorizing entry into the reactor bay to ensure that all individuals wear appropriate dosimeters while in the reactor bay.

Approval by the Reactor Safeguards Committee

________________________________ Date: _______________

2 Thermoluminescent dosimeters may be substituted at the discretion of the University Radiation Safety Officer.

Fitness for Duty Policy

Kansas State University Reactor Facility

The following policies apply to the use of illegal drugs and the abuse of legal drugs while employed by or utilizing the KSU Nuclear Reactor Facility.

1. The KSU Nuclear Reactor Facility operates in accordance with federal and state mandates for a work environment free from drugs, alcohol, and the effects of the use of these substances. Applicable federal regulations are contained in 10CFR, Parts 2, 26, and 55. Applicable state regulations are described in Department of Administration Personnel Services Policy Statement 32. Copies of these documents are available in the Facility for inspection.
2. Conviction for possession or distribution of a controlled substance will result in the permanent loss of access to the Nuclear Reactor Facility.
3. Consumption of alcohol is prohibited during and for 5 hours preceding any scheduled activities within the facility. However, consumption of alcohol during the abstinence period need not necessarily preclude an individual's responding to an emergency.
4. Extended use of prescription or over-the-counter drugs is to be reported to the examining physician during employment physical examinations. Long-term use of drugs initiated in the interim between physical examinations is to be reported to Facility management.
5. Personnel in need of advice or counseling in matters of drug or alcohol use may make use of the University's Employee Assistance Program, administered by the Department of Human Resources [phone (785)532-6657] or the Kansas "Lifeline" HealthQuest Program administered by the Department of Administration and the Kansas State Employees Health Care Commission [phone (800)284-7575]. Both services are confidential. Information is available in the Facility for inspection.

Jeffrey A. Geuther, Ph.D.
Nuclear Reactor Facilities Manager

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Kansas State University TRIGA Mk II Reactor Administrative Plan

Operation and use of the Kansas State University TRIGA MARK II Reactor, purchased with a grant from the Division of Reactor Development of the U.S. Atomic Energy Commission for the Mechanical and Nuclear Engineering Department of the University, is the responsibility of the Reactor Manager. Consistent with the terms of the request for reactor funds from the AEC, support of instruction and education in the Mechanical and Nuclear Engineering Department has primary call on reactor availability. Beyond this, the reactor is available for use by any department within the University. The Reactor Manager is administratively responsible to the Head of the Department of Mechanical and Nuclear Engineering (Chairman of the Reactor Safeguards Committee), who is responsible to the Dean of the College of Engineering. Details of duties and responsibilities follow in the Administrative Duties and Authority Associated with
the Kansas State University TRIGA Mark II Reactor section.

The Reactor Manager is directly responsible for the operation and use of the reactor, although these duties may be delegated to a Reactor Supervisor. Therefore, all duties associated with the position of Reactor Supervisor may be performed by the Reactor Manager. The Reactor Manager's nominal duties include reactor scheduling, responsibility for all records with regard to reactor operation as are required by appropriate federal licenses and regulations, laws and regulations of the State of Kansas and regulations of Kansas State University, including the Kansas State University TRIGA MARK II Reactor Operations Manual. The Reactor Manager is responsible for maintenance of a Reactor Operations Manual, the manual to include prescribed operating procedures for all routine modes of operation of the reactor, procedures for loading and unloading, start-up procedures, maintenance schedule, testing procedures, operational references, and other appropriate information as determined by the Reactor Manager. The Reactor Manager is responsible for determining that the reactor is operated in strict accordance with the Operations Manual and the Facility License.

All persons having direct responsibility for reactor operation have the concurrent responsibility for being aware of the geometry and location of all fissionable [3] materials in the Reactor Facility. To assure that the only fissionable material brought on Campus has been properly considered in regard to its potential for inducing a critical incident, the Radiation Safety Officer must approve in writing all orders, requests, or grants for fissionable material in quantity or type requiring Special Nuclear or Source Material License. A primary duty of the Reactor Manager is to continuously assess the degree of hazard associated with current and anticipated future conditions of the reactor core and all fissionable material in the Reactor Facility. Properly executed, these duties will prevent accidental and/or unintentional power excursions in the Reactor Facility and will prevent the accumulation of a critical mass or near critical mass in any Campus location unless the accumulation is intentional and has prior approval by the U.S. Nuclear Regulatory Commission (NRC), appropriate State of Kansas authority, and the Kansas State University Reactor Safeguards Committee. The Kansas State University Reactor Safeguards Committee shall be composed as indicated below.

Nothing in these regulations and instructions shall prevent licensed reactor operators from taking prompt remedial action to render harmless any situation concerning the reactor that, in their judgment, presents an immediate and significant threat to personnel or equipment. The Kansas State University Reactor Safeguards Committee will review changes in the Operations Manual for the purpose of passing judgment on proposed operating procedures. The Reactor Safeguards Committee must approve, disapprove, or approve-with-conditions proposed changes in operating procedures. If "approved-with-conditions," the

[3] Fissionable material shall be construed to mean, for purposes of this section, fissionable materials requiring NRC or State License.

conditions shall become part of the Operations Manual. Experiments involving operation of the reactor in a mode not previously described and approved in the Operations Manual shall be described in detail in a Proposed KSU TRIGA MK II Experiment, prepared by the Experimenter and the Reactor Supervisor. (See paragraph 3.0 of the Operations Manual.) Proposed experiments shall be submitted by the Reactor Manager to the Reactor Safeguards Committee for consideration. When approved by the Reactor Safeguards Committee, the experiment shall become part of the Operations Manual and may be scheduled by the Reactor Manager.

REACTOR SAFEGUARDS COMMITTEE COMPOSITION

Mechanical and Nuclear Department Chairman, ex officio Chairman
University Radiation Safety Officer, ex officio Member
Reactor Supervisor, ex officio Member
Faculty Member** Member
Faculty Member** Member
Faculty Member*, Dept. of Mechanical and Nuclear Engineering Member
Faculty Member*, Dept. of Mechanical and Nuclear Engineering Member

*Faculty members of the Department of Mechanical and Nuclear Engineering having experience in matters of reactor safety and operation.
**Kansas State University faculty members appointed from outside the Department of Mechanical and Nuclear Engineering, appointments to be made from those best qualified in matters of reactor safety.

Members of the Reactor Safeguards Committee, other than *ex officio*, shall be
appointed by the President of the University on recommendation of the Head of the Department of Mechanical and Nuclear Engineering.

In addition to the Reactor Manager, at least one Reactor Operator or Senior Reactor Operator shall be employed by the Department of Mechanical and Nuclear Engineering and will be available for operation and maintenance of the reactor. Operation in this sense includes start-up, shutdown, routine checkout of instrumentation and control systems, record keeping, routine maintenance and such other duties as may be described in the Operations Manual and/or as directed by the Reactor Manager.

The University Radiation Safety Officer who is also *ex officio* Chairman, Kansas State University Radiation Safety Committee, or an authorized representative shall be available, upon due notice, for advice and consultation regarding radiation surveys and radiation safety in connection with isotope production and radiation streaming problems as might arise in connection with reactor operation or experimentation.

ADMINISTRATIVE DUTIES AND AUTHORITY ASSOCIATED WITH THE KANSAS STATE UNIVERSITY TRIGA MARK II REACTOR

I. Reactor Safeguards Committee[4]

a. Evaluation, in accordance with the Facility License, of new and revised procedures, new and revised experiments, changes in facility configuration, and tests and specifications for additions, modification, or non-routine maintenance of the core and its associated support structure, the pool coolant system, the rod drive mechanisms, and the reactor safety system.
b. Periodic inspection of reactor operating records and audits performed by the Reactor Manager.

[4] Approval of the Operations Manual and its amendments shall require an affirmative vote of at least half of the Committee members. Approval by the University Radiation Safety Officer is also required.

II. Chairman, Reactor Safeguards Committee

a. Chair the Reactor Safeguards Committee.

b. Recommend to the President of the University faculty members for appointments, other than *ex officio*, to the Reactor Safeguards Committee.

c. Maintain liaison between the committee and the University Administration in matters of reactor operation.

III. Radiation Safety Committee

a. Exercise authority as defined by the KSU Radiation Safety Manual. Insofar as the reactor is a radiation source, and with regard to sources it may produce, the Radiation Safety Committee has authority in radiation safety matters over reactor operation.

IV. Chairman, Radiation Safety Committee

a. Authority to make or have made radiation surveys of all radiation sources including the reactor.

b. Responsibility for providing advice and consultation in connection with reactor operation and/or experimentation.

c. *Ex officio* member of the Kansas State University Reactor Safeguards Committee.

V. Chairman of the Department of Mechanical and Nuclear Engineering

a. Chairmanship of the Reactor Safeguards Committee.

b. Appointment of the Reactor Manager and, on advice of the Reactor Manager, the Reactor Supervisor and the Reactor Operators.

c. Fiscal administration of the Reactor Facility, including the provision of adequately trained support personnel.

VI. Reactor Manager

a. Operational administration of the Reactor Facility, including development and revision of administrative and technical guidance to ensure compliance with operational safety requirements

b. Management audit of reactor operations.

c. Supervision of training and requalification of Reactor Operators and Senior Reactor Operators.

d. Prepare proposals for changes in the Reactor Operations Manual.

e. Determine that all appropriate and necessary logs, inventories, and records required for the safe and secure operation and existence of a nuclear reactor on the Manhattan Campus of Kansas State University are prepared and kept.

f. Schedule reactor runs, experiments, assign reactor experimental space and coordinate reactor operation with other activities which might be affected by reactor operation.

g. Maintain continuous record of location, conditions or storage irradiation time (nvt) of all fissile materials in the Reactor Facility.

h. Assess current and future potential hazards as related to the probability of an unintentional nuclear excursion and direct reactor operations such that probability of such an excursion is minimized.

i. Maintain a current Nuclear Regulatory Commission Senior Reactor Operator's License for the Kansas State University TRIGA Mark II Reactor.

VII. Reactor Supervisor

a. Assess current and future potential hazards as related to the probability of an unintentional nuclear excursion and direct reactor operations such that probability of such an excursion is minimized.

b. Train Reactor Operators and prospective Reactor Operators when directed by the Reactor Manager.

c. Maintain a current Nuclear Regulatory Commission Senior Reactor Operator's License for the Kansas State University TRIGA Mark II Reactor.

d. Perform duties delegated by the Reactor Manager.

e. Be present at the facility or within 10 minutes travel time when supervising operations.

VIII. Reactor Operators

a. When assigned by the Reactor Manager, to be on duty in the reactor control room at all times when the reactor is unsecured.
b. Keep all records and logs as directed by the Reactor Manager.
c. Maintain a current Nuclear Regulatory Commission Reactor Operator's or Senior Reactor Operator's License for the Kansas State University TRIGA Mark II Reactor.

IX. Experimenter

The experimenter will be a faculty member, staff member, or graduate student who is responsible for planning an experiment, for the preliminary investigation of the hazards involved, for the design of the necessary equipment and, of course, for taking the experimental data and evaluating it. If appropriate, the request for operational time on the reactor will specify the minimum number of persons required to run the particular experiment. The Experimenter will be responsible for the preparation of a written proposal describing the experiment in sufficient detail, that the Radiation Safety Committee and/or the Reactor Safeguards Committee can evaluate the experiment from the standpoint of safety. (See paragraph 3.0 of the Operations Manual.) The Reactor Manager is available for consultation in the preparation of such proposals. There shall also be sufficient detail in the description to guide the Reactor Operator with respect to operating the reactor for any particular experiment. The Experimenter has the responsibility to make recommendations to the Reactor Operator regarding operations of the reactor, but *not* the authority to require the Reactor Operator to manipulate the controls of the reactor in a particular way. In case of disagreement concerning the safety of operation, the experiment must be stopped until agreement is reached.

Approved by Reactor Safeguards Committee:

___________________________________ Date: ________________
Chairman

Requalification Program

INTRODUCTION

The Kansas State University (K-State) Nuclear Reactor Facility is designed to maintain a continuing and improving level of proficiency for reactor operators and senior reactor operators licensed to operate the reactor. To meet this goal, the requalification program addresses:

- Medical certification
- On the job training and proficiency
- Examination
- Lecture
- Records

The requalification program must be conducted for each complete retraining cycle, to be completed biennially, and upon conclusion must be promptly followed, pursuant to a continuous schedule, by successive requalification programs.

For purposes of the requalification program, "biennially" means every two years, not to exceed 30 months and "annual" or "annually" means every 12 months not to exceed 18 months. The maximum period that could occur as part of an annual requirement is set specifically to allow flexibility to coordinate the requalification program with academic schedules.

MEDICAL CERTIFICATION

General

The USNRC grants license to reactor operators and senior reactor operators based formal, documented physician evaluation and facility management certification that the licensee's medical condition and general health will not adversely affect the performance of assigned operator job duties or cause operational errors endangering public health and safety.

ANSI/ANS15.4 (*American National Standard for the Selection, Training and Qualification of Personnel for Research Reactors*) 7.2.1, "Basis of Requirements," states that "The physical condition and the general health of research reactor operators shall be such that they are capable of properly operating under normal, abnormal and emergency conditions and able to perform the associated tasks." Specific criteria for medical examination are provided by the standard.

Medical evaluations are based on K-State Nuclear Reactor Facility operator responsibilities, jobs and tasks. Principle responsibilities of the K-State position are control of machinery and electrical equipment supporting reactor operation, occupational exposures, and (sedentary) monitoring & logging meter readings. Significantly, at the K-State reactor:

- There is occasional use of a pendant-controlled, overhead crane;
- There is routine access to facility areas via metal stairs;
- There is no routine heavy lifting (e.g., loads greater than 50 lbs.);
- K-State reactor operators do not require or use respiratory protection.

Periodic Requirements

Biennially, the Nuclear Reactor Facility Manager shall complete and sign Form NRC-396, ``Certification of Medical Examination by Facility Licensee," certifying medical fitness for each licensed operator/senior operator. Completion of this form requires that a physician conduct the medical examination of the licensee to determine that the licensee medical condition and general health will not adversely affect the performance of assigned operator job duties or cause operational errors endangering public health and safety.

Special Requirements

License Medical Condition Changes

The licensee shall immediately inform the Nuclear Reactor Facility Manager if (during the term of the license) the licensee develops a permanent physical or mental condition that invalidates the determination made by the physician. The Nuclear Reactor Facility Manager shall notify the Commission, within 30 days of learning of the diagnosis.

Responsibility, Task and Job Changes

As system and equipment modifications, procedure changes, and experiment program changes occur, reviews for personnel and facility safety will consider whether current medical evaluations could be affected. If a change could affect the ability of operator to perform (as described above), options for addressing changes include review of the changes by medical authorities, operational restrictions, additional safety measures, or additional medical evaluations for operators.

ON THE JOB TRAINING AND PROFICIENCY

General

Licensed operators shall participate in a sufficient set of operations over a minimum prescribed operating time to ensure their familiarity and competence to perform as operator at the controls. Changing conditions and emergent information are an essential part of maintaining operational proficiency; therefore, provisions are made to ensure this information is provided to licensed operators.

Periodic Requirements

Reactivity Manipulations

Biennially, each licensed operator (reactor operator and senior reactor operator) shall perform at least 10 reactivity manipulations (that demonstrate skill with reactivity control systems) including startups, shutdowns and other control manipulations.

Proficiency as Operator at the Controls

Every calendar quarter, each licensed reactor operator shall be the operator at the controls while the reactor is operating for at least 4 hours to maintain operational proficiency. Each reactor operator licensed to operate the K-State reactor shall be in direct control to meet proficiency requirements; reactor operators shall not meet proficiency requirements by directing activities of students or trainees. Each licensed senior operator either manipulates the controls or directs the activities of individuals during plant control manipulations to meet proficiency requirements.

In the event that facility conditions such as an extended shutdown do not allow licensed operators to complete at least 4-hours of operation within a quarter, operators shall complete a special Facility Retraining Program approved by the Reactor Safeguards Committee prior to resumption of normal licensed activities as operator at the controls.

Lessons Learned

The Nuclear Reactor Facility Manager shall review operating records at least annually to determine if there are operational issues that should be addressed by training, including formal lectures or "lessons learned" communications.

Special Requirements

Supervised Operation

Unless a Facility Retraining Program is required (as described above), any reactor operator and/or senior reactor operator who does not meet the periodic requirements previously described shall operate the reactor only under supervision of another licensed operator.

(a) Any operator who held a license (reactor operator and/or senior reactor operator) during the previous quarter and did not operate the reactor for at least 4 hours as described above shall operate the reactor under supervision of a licensed operator for at least 6 hours before operating without supervision.

(b) Any operator who held a license (reactor operator and/or senior reactor operator) during the previous requalification program cycle and has not completed at least 10 reactivity manipulations as described above shall complete at least 5 startups, at least 5 shutdowns and at least 5 changes in reactor power under supervision of a licensed operator before operating without supervision.

On the Job Training Information

(a) The Nuclear Reactor Facility Manager or Reactor Supervisor shall notify all operators of changes in facility design, operating procedures, facility license, abnormal procedures, and/or emergency procedures. Changes shall be indicated in the operating log, with explanatory or supporting material placed at the control console until all licensed operators have either been the operator at the controls or indicate they have reviewed the material (such as initialing the log entry indicating change).

(b) If operational issues that should be addressed by training are identified in annual review, the Nuclear Reactor Facility Manager shall initiate a formal lecture, incorporate the material in the next formal training, or initiate appropriate communication to all licensed personnel.

Facility Retraining Program

When a licensed operator is making reactivity manipulations for the first hour of retraining operations, a second person shall be in the control room. The second person in the control room shall be a licensed senior reactor operator or reactor operator, a previously licensed operator, Mechanical and Nuclear Engineering nuclear faculty, or a member of the Reactor Safeguards Review Committee (as approved by the Committee).

The Facility Retraining Program may consist of self-directed exercises following Committee approved written instructions to complete the items indicated above, or operation under direction of a licensed senior reactor operator to complete a set of operations, as approved by the Committee. The Facility Retraining Program shall include (as a minimum):

- Observation of subcritical behavior
- Approach to critical using subcritical multiplication to predict critical control rod positions
- Operation at power levels greater than 1 kW
- A minimum of 2 hours of training; at least one hour shall be as operator at the controls, and up to one hour may be as an observer

EXAMINATIONS

General

Content

Examinations should be based on a representative sample of questions covering areas in depth required to evaluate trainee understanding and capabilities. Examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Operating Examinations

Operating examinations shall be conducted by the Reactor Supervisor or Nuclear Reactor Facility Manager covering normal, abnormal and emergency operating procedures. Operating examinations shall be graded as Satisfactory (S) or Unsatisfactory (U). The person who prepares written examinations is exempt from participating in the written examination.

Written Examinations

Written examinations be prepared and graded (on a scale from 0 to 100%) by the Reactor Supervisor or Nuclear Reactor Facility Manager covering:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Periodic Requirements

Written and operating examinations will be given annually.

Written Examinations
Written examinations shall be given annually, covering the topics identified above.

Operating Examinations
Operating examinations shall be given annually, covering a selection of materials and operations that demonstrate skills and knowledge required for operation.

Certification of Examiner
The Reactor Safety Review Committee shall certify the person preparing examinations (Nuclear Reactor Facility Manager or Reactor Supervisor) annually. Certification shall consist of discussion and observation determined by the Committee adequate to assure the committee that the examinee is capable of safely operating the reactor.

Special Requirements

Requirements for Lecture Attendance
If an operator scores between 70% and 80% on any written examination, the examinee shall attend a lecture on the subject

Written Reexamination
If an operator scores less than 70% any written examination:

- The operator shall not act as operator at the controls

- The operator will attend a lecture on the subject area
- The operator will be reexamined
- The operator will be permitted to resume duties as operator at the controls when the operator scores at least 80% on a subsequent examination

4.3.3 Operating Reexamination

If an operator is evaluated Unsatisfactory (U) on operating examination,

- The operator shall not act as operator at the controls
- The examiner shall identify training required prior to reexamination
- The operator shall complete training required
- The operator shall be reexamined
- The operator will be permitted to resume duties as operator at the controls when the operator is determined to perform satisfactorily in a subsequent examination

4.3.4 Special Retraining Program

If an operator scores less than 70% any written examination and also Unsatisfactory on the operating examination, the Nuclear Reactor Facility Manager and/or Reactor Supervisor shall evaluate operator performance to determine lecture and reexamination requirements for reinstatement.

LECTURES

General

Lecture Topics

The requalification program will include preplanned lectures on a regular and continuing basis throughout the license period in those areas where operator and senior operator written examinations and facility operating experience indicate that emphasis in scope and depth of coverage is needed in the following subjects:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Lecture Content

Lecture material should be based on training objectives, including general and specific objectives. Lecture material should reflect operational needs, with objectives should be based on how the material relates to job performance.

Periodic Requirements

In general, lectures will be prepared based on need; however, training material will be prepared covering the Emergency Plan and emergency procedures, and all operators will participate in annual training either by lecture attendance or independent study of the training material.

Special Requirements

Lectures will be prepared when weaknesses in operator proficiency or understanding are identified through semi annual management review or examinations (written or practical).

REPORTS AND RECORDS

Control Room License, Technical Specification and Operator License Notebook
Copies of active reactor operator and senior reactor operator licenses are maintained in the control room.
Operator Training Folders
Records for individual reactor operators are maintained, including records of medical certification, license application, operator and senior operator licenses, and other information such as Suspensions and reinstatement of licenses

Reactor Safety Committee Audit/Review
Results of Annual RSC review are maintained, including certification of personnel preparing and grading examinations, review of training program status, and identification of problems or items that need to be communicated to operators.

Operating Logs
The operating logs contain records of reactor operator/senior reactor operator at the controls, including reactivity manipulations such as startup, shutdown and power changes as well as start and stop times for operators at the controls. This information is used to determine operator proficiency status in a monthly report.

Training Records
Training records include copies of lesson plans, written examinations (and keys), records of attendance and a summary of examination results.

Record Retention
All requalification program records shall be maintained for the current and previous cycle biennial requalification program.

Facility License

UNITED STATES NUCLEAR REGULATORY COMMISSION

RENEWAL OF FACILITY LICENSE NO. R-88

DOCKET NO. 50-188

KANSAS STATE UNIVERSITY NUCLEAR REACTOR FACILITY

Renewed License No. R-88

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

   A. The application for license renewal of Facility License No. R-88, filed by the Kansas State University, Manhattan, Kansas (the licensee) dated September 12, 2002, as supplemented on November 11, 2002; November 13, 2002; December 21, 2004; July 6, 2005; September 27, 2005; March 20, 2006; March 30, 2006; June 28, 2006; September 28, 2006; May 17, 2007; June 4, 2007; September 12, 2007; October 11, 2007; and February 6, 2008 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10, Chapter I, Code of Federal Regulations (10 CFR);

   B. The Kansas State Nuclear Reactor Facility (the facility) will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

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

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

H. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including Sections 30.33, 70.23 and 70.33.

Facility License No. R-88 is hereby renewed in its entirety to read as follows:

A. This license applies to the TRIGA research reactor (herein "the reactor"), owned by the Kansas State University and located on its campus in Manhattan, Kansas, and is described in the licensee's application for renewal dated September 12, 2002, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Kansas State University:

1. Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the reactor as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and in this license;

2. Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to 4.20 kilograms of contained uranium-235 at enrichments less than 20% in connection with operation of the reactor and up to 90 grams of uranium-235 at any enrichment for fission chambers and reactor experiments.

3. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use a 2-curie sealed americium-beryllium neutron source and a 7-curie sealed polonium-beryllium neutron source (Monsanto Item NS-1, Serial No. N-693) for reactor startup; and to possess, use, but not separate, such byproduct material and special nuclear material as may be produced by the operation of the reactor.

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

1. <u>Maximum Power Level</u>

The licensee is authorized to operate the reactor at steady-state power levels up to a maximum of 1,250 kW (thermal) and in the pulse mode with reactivity insertions not to exceed $3.00 with all stainless-steel clad fuel elements.

2. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license is effective as of March 19, 2008 and shall expire at midnight 20 years from that date.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Enclosure: Appendix A, Technical Specifications

Date of Issuance: March 13, 2008

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Technical Specifications

DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications. Capitalization is used in the body of the Technical Specifications to identify defined terms.

ACTION	Actions are steps to be accomplished in the event a required condition identified in a "Specification" section is not met, as stated in the "Condition" column of "Actions." In using Action Statements, the following guidance applies: • Where multiple conditions exist in an LCO, actions are linked to the (failure to meet a "Specification") "Condition" by letters and number. • Where multiple action steps are required to address a condition, COMPLETION TIME for each action is linked to the action by letter and number. • AND in an Action Statement means all steps need to be performed to complete the action; OR indicates options and alternatives, only one of which needs to be performed to complete the action. • If a "Condition" exists, the "Action" consists of completing all steps associated with the selected option (if applicable) except where the "Condition" is corrected prior to completion of the steps
ANNUAL	12 months, not to exceed 15 months
Channel Calibration	A channel calibration is an adjustment of the channel to that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures.
BIENNIAL	Every two years, not to exceed a 28 month interval
Channel Check	A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values, other independent channels, or other methods of measuring the same variable.
Channel Test	A channel test is the introduction of an input signal into a channel to verify that it is operable. A functional test of operability is a channel test.
Control Rod (Standard)	A standard control rod is one having an electric motor drive and scram capability.
Control Rod (Transient)	A transient rod is one that is pneumatically operated and has scram capability.
DAILY	Prior to initial operation each day (when the reactor is operated), or before

	an operation extending more than 1 day
ENSURE	Verify existence of specified condition or (if condition does not meet criteria) take action necessary to meet condition
EXHAUST PLENUM	The air volume in the reactor bay atmosphere between the pool surface and the reactor bay exhaust fan
Experiment	An EXPERIMENT is (1) any apparatus, device, or material placed in the reactor core region (in an EXPERIMENTAL FACILITY associated with the reactor, or in line with a beam of radiation emanating from the reactor) or (2) any in-core operation designed to measure reactor characteristics.
EXPERIMENTAL FACILITY	Experimental facilities are the beamports, thermal column, pneumatic transfer system, central thimble, rotary specimen rack, and the in-core facilities (including non-contiguous single-element positions, and, in the E and F rings, as many as three contiguous fuel-element positions).
IMMEDIATE	Without delay, and not exceeding one hour. NOTE: IMMEDIATE permits activities to restore required conditions for up to one hour; this does not permit or imply deferring or postponing action
INDEPENDENT EXPERIMENT	INDEPENDENT Experiments are those not connected by a mechanical, chemical, or electrical link to another experiment
LIMITING CONDITION FOR OPERATION (LCO)	The lowest functional capability or performance levels of equipment required for safe operation of the facility.
LIMITING SAFETY SYSTEM SETTING (LSSS)	Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit placed, the setting shall be chosen so that the automatic protective action will correct the abnormal situation before a safety limit is exceeded.
Measured Value	The measured value of a parameter is the value as it appears at the output of a MEASURING CHANNEL.
MEASURING CHANNEL	A MEASURING CHANNEL is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.
MOVABLE EXPERIMENT	A MOVABLE EXPERIMENT is one that may be moved into, out-of or near the reactor while the reactor is OPERATING.
NONSECURED EXPERIMENT	NONSECURED Experiments are these that should not move while the reactor is OPERATING, but are held in place with less restraint than a secured experiment.

Operable	A system or component is OPERABLE when it is capable of performing its intended function in a normal manner
OPERATING	A system or component is OPERATING when it is performing its intended function in a normal manner.
PULSE MODE	The reactor is in the PULSE MODE when the reactor mode selection switch is in the pulse position and the key switch is in the “on” position. NOTE: In the PULSE MODE, reactor power may be increased on a period of much less than l second by motion of the transient control rod.
REACTOR SAFETY SYSTEM	The REACTOR SAFETY SYSTEM is that combination of MEASURING CHANNELS and associated circuitry that is designed to initiate reactor scram or that provides information that requires manual protective action to be initiated.
REACTOR SECURED MODE	The reactor is secured when the conditions of either item (1) or item (2) are satisfied: (1) There is insufficient moderator or insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection (2) All of the following: a. The console key is it the OFF position and the key is removed from the lock b. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives (unless the drive is physically decoupled from the control rod) c. No experiments are being moved or serviced that have, on movement, a reactivity worth greater than $1.00
REACTOR SHUTDOWN	The reactor is shutdown if it is subcritical by at least $1.00 in the REFERENCE CORE CONDITION with the reactivity worth of all experiments included.
Ring	A ring is one of the five concentric bands of fuel elements surrounding the central opening (thimble) of the core. The letters B through F, with the letter B used to designate the innermost ring,
Reference CORE CONDITION	The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$0.30)
Safety Channel	A safety channel is a MEASURING CHANNEL in the REACTOR SAFETY SYSTEM
Secured Experiment	A secured EXPERIMENT is an EXPERIMENT held firm1y in place by a mechanical device or by gravity providing that the weight of the

	EXPERIMENT is such that it cannot be moved by force of less than 60 lb.
Secured EXPERIMENT with Movable Parts	A secured EXPERIMENT with movable parts is one that contains parts that are intended to be moved while the reactor is OPERATING.
SHALL (SHALL NOT)	Indicates specified action is required/(not to be performed)
SEMIANNUAL	Every six months, with intervals not greater than 8 months
Shutdown Margin	The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action
Standard Thermocouple Fuel Element	A standard thermocouple fuel element is stainless steel clad fuel element containing three sheathed thermocouples embedded in the fuel element.
Steady-State Mode	The reactor is in the steady-state mode when the reactor mode selector switch is in either the manual or automatic position and the key switch is in the "on" position.
TECHNICAL SPECIFICATION VIOLATION	A violation of a Safety Limit occurs when the Safety Limit value is exceeded. A violation of a Limiting Safety System Setting or Limiting Condition for Operation) occurs when a "Condition" exists which does not meet a "Specification" and the corresponding "Action" has not been met within the required "Completion Time." If the "Action" statement of an LSSS or LCO is completed or the "Specification" is restored within the prescribed "Completion Time," a violation has not occurred. NOTE "Condition," "Specification," "Action," and "Completion Time" refer to applicable titles of sections in individual Technical Specifications

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Fuel Element Temperature Safety Limit

2.1.1 Applicability

This specification applies when the reactor in STEADY STATE MODE and the PULSE MODE.

2.1.2 Objective

This SAFETY LIMIT ensures fuel element cladding integrity

2.1.3 Specification

(1)	Stainless steel clad, high-hydride fuel element temperature SHALL NOT exceed 1150°C.
(2)	Steady state fuel temperature shall not exceed 750°C.

2.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Stainless steel clad, high-hydride fuel element temperature exceeds 1150°C. OR Fuel temperature exceeds 750°C in steady state conditions	A.1 Establish SHUTDOWN condition AND A.2 Report per Section 6.8	A.1 IMMEDIATE A.2 Within 24 hours

2.1.5 Bases

Safety Analysis Report, Section 3.5.1 (Fuel System) identifies design and operating constraints for TRIGA fuel that will ensure cladding integrity is not challenged.

NUREG 1282 identifies the safety limit for the high-hydride ($ZrH_{1.7}$) fuel elements with stainless steel cladding based on the stress in the cladding (resulting from the hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding with fuel temperatures below 1,150°C. A change in yield strength occurs for stainless steel cladding temperatures of 500°C, but there is no scenario for fuel cladding to achieve 500°C while submerged; consequently the safety limit during reactor operations is 1,150°C.

Therefore, the important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification, and it is readily measured. During operation, fission product gases and dissociation of the hydrogen and zirconium builds up gas inventory in internal components and spaces of the fuel elements. Fuel temperature acting on these gases controls fuel element internal pressure. Limiting the maximum temperature prevents excessive internal pressures that could be generated by heating these gases.

Fuel growth and deformation can occur during normal operations, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, steady state fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Applicability

This specification applies when the reactor in STEADY STATE MODE

2.2.2 Objective

The objective of this specification is to ensure the safety limit is not exceeded.

2.2.3 Specifications

(1)	Power level SHALL NOT exceed 1,250 kW (th) in STEADY STATE MODE of operation

2.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Steady state power level exceeds 1,250 kW (th)	A.1 Reduce power to less than 1,250 kW (th) OR A.2. Establish REACTOR SHUTDOWN condition	A.1 IMMEDIATE A.2. IMMEDIATE

2.2.5 For a core containing an aluminum-clad, low-hydride thermocouple fuel element (i.e., for restricted-mode operation) the limiting safety system setting for that element shall be 230°C with the element located in the B-ring.
Bases

Analysis in Chapter 4 demonstrates that if operating thermal (th) power is 1,250 kW, the maximum steady state fuel temperature is less than the safety limit for steady state operations by a large margin. For normal pool temperature, calculations in Chapter 4 demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, steady state operations at a maximum of 1,250 kW meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin. Steady state operation of 1,250 kW was assumed in analyzing the loss of cooling and maximum hypothetical accidents. The analysis assumptions are protected by assuring that the maximum steady state operating power level is 1,250 kW.

In 1968 the reactor was licensed to operate at 250 kW with a minimum reactor safety system scram set point required by Technical Specifications at 110% of rated full power, with the scram set point set conservatively at 104%. In 1993 the original TRIGA power level channels were replaced with more reliable, solid state instrumentation. The actual safety system setting will be chosen to ensure that a scram will occur at a level that does not exceed 1,250 kW._

For the aluminum-clad, low-hydride element the margin of 300°C between the safety limit of 530°C and the limiting safety system setting of 230°C in the B-ring was selected to assure that conditions would not arise which would allow the fuel element temperature to approach the safety limit. The margin is large enough to allow for differences in properties of all aluminum-clad, all stainless-steel-clad, and mixed cores and for uncertainty in temperature channel calibration.

3 LIMITING CONDITIONS FOR OPERATION (LCO)

3.1 Core Reactivity

3.1.1 Applicability

These specifications are required prior to entering STEADY STATE MODE or PULSING MODE in OPERATING conditions; reactivity limits on experiments are specified in Section 3.8.

3.1.2 Objective

This LCO ensures the reactivity control system is OPERABLE, and that an accidental or inadvertent pulse does not result in exceeding the safety limit.

3.1.3 Specification

(1)	The maximum available core reactivity (excess reactivity) with all control rods fully withdrawn is less than $4.00 when: 1. reference core conditions exists 2. No experiments with net negative reactivity worth are in place
(2)	The reactor is capable of being made subcritical by a SHUTDOWN MARGIN more than $0.50 under REFERENCE CORE CONDITIONS and under the following conditions: 1. The highest worth control rod is fully withdrawn 2. The highest worth NONSECURED EXPERIMENT is in its most positive reactive state, and each SECURED EXPERIMENT with movable parts is in its most reactive state.

3.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactivity with all control rods fully withdrawn exceeds $4.00	A.1 ENSURE REACTOR SHUTDOWN AND A.2 Configure reactor to meet LCO	A.1 IMMEDIATE A.2 Prior to continued operations

B. The reactor is not subcritical by more than $0.50 under specified conditions	B.1.a ENSURE control rods fully inserted AND B.1.b Secure electrical power to the control rod circuits AND B.1.c Secure all work on in-core experiments or installed control rod drives AND B.2 Configure reactor to meet LCO	B.1 IMMEDIATE B.2 Prior to continued operations

3.1.5 (3) When operating in restricted mode operation, the reactivity with all control rods fully withdrawn is less than 2.00$ with or without experiments in place.
Bases

The value for excess reactivity was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis. Since the fundamental protection for the KSU reactor is the maximum power level and fuel temperature that can be achieved with the available positive core reactivity, experiments with positive reactivity are included in determining excess reactivity. Since experiments with negative reactivity will increase available reactivity if they are removed during operation, they are not credited in determining excess reactivity.

Analysis (Chapter 13) shows fuel temperature will not exceed l,150°C for the stainless-steel-clad fuel in the event of inadvertent or accidental pulsing of the reactor. Section 13.2 demonstrates that a $3.00 reactivity insertion from critical, zero power conditions leads to maximum fuel temperature of 746°C, while a $1.00 reactivity insertion from a worst-case steady state operation at 107 kW leads to a maximum fuel temperature of 869°C, well below the safety limit.

The limiting SHUTDOWN MARGIN is necessary so that the reactor can be shut down from any operating condition, and will remain shut down after cool down and xenon decay, even if one control rod (including the transient control rod) should remain in the fully withdrawn position.
At power levels of 10 kW or below, the steady-state fuel temperature is small compared to the temperature rise caused by a pulse of 3.00$ or less.

When our test program for initial operation above 100 kW power was conducted in February-March 1984 thermocouple measurements showed that at 200 kW the B-ring fuel temperature for the all stainless-steel-clad high-hydride core was 160°C. Even when allowing for differences resulting from the substitution of an aluminum-clad low-hydride element, the fuel temperature at 200 kW will be far below the safety limit of 530°C.

3.2 Pulsed Mode Operations

3.2.1 Applicability

These specifications apply to operation of the reactor in the PULSE MODE.

3.2.2 Objective

This Limiting Condition for Operation prevents fuel temperature safety limit from being exceeded during PULSE MODE operation.

3.2.3 Specification

(1)	The transient rod drive is positioned for reactivity insertion (upon withdrawal) less than or equal to $3.00

3.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With all stainless steel clad fuel elements, the worth of the pulse rod in the transient rod drive position is greater than $3.00 in the PULSE MODE	A.1 Position the transient rod drive for pulse rod worth less than or equal to $3.00 OR A.2 Place reactor in STEADY STATE MODE	A.1 IMMEDIATE OR A.2 IMMEDIATE

3.2.5 (3) When operating in restricted mode operation, the reactor shell not be operated in pulse mode. The pulse mode circuit shall be disconnected to assure that pulsing is not possible with the key-operated switch.
Bases

The value for pulsed reactivity with all stainless steel elements in the core was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis.

3.3 Measuring Channels

3.3.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during STEADY STATE MODE and PULSE MODE operations.

3.3.2 Objective

The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor

3.3.3 Specifications

(1)	The MEASURING CHANNELS specified in TABLE 1 SHALL be OPERATING
(2)	The neutron count rate on the startup channel is greater than the minimum detector sensitivity

TABLE 1: MINIMUM MEASURING CHANNEL COMPLEMENT		
MEASURING CHANNEL	Minimum Number Operable	
	STEADY STATE MODE	PULSE MODE
Reactor power level[1]	2	1
Primary Pool Water Temperature	1	1
Reactor Bay Differential Pressure	1	1
Fuel Temperature	1	1
22 foot Area radiation monitor	1	1
0 or 12 foot Area monitor	1	1
Continuous air radiation monitor[2]	1	1
EXHAUST PLENUM radiation monitor[2]	1	1

NOTE[1]: One "Startup Channel" required to have range that indicates <10 W
NOTE[2]: High-level alarms audible in the control room may be used

3.3.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.1 Reactor power channels not OPERATING (min 2 for STEADY STATE, 1 PULSE MODE)	A.1.1 Restore channel to operation OR A.1.2 ENSURE reactor is SHUTDOWN	A.1.1 IMMEDIATE A.1.2 IMMEDIATE
A.2 High voltage to reactor power level detector less than 90% of required operating value	A.2.1 Establish REACTOR SHUTDOWN condition AND A.2.2 Enter REACTOR SECURED mode	A.2. IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Primary water temperature, reactor bay differential pressure or fuel temperature CHANNEL not operable	B.1 Restore channel to operation OR B.2 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE A.2 IMMEDIATE
C. 22 foot Area radiation monitor is not OPERATING	C.1 Restore MEASURING CHANNEL OR C.2 ENSURE reactor is shutdown OR C.3 ENSURE personnel are not on the 22 foot level OR C.4 ENSURE personnel on 22 foot level are using portable survey meters to monitor dose rates	C.1 IMMEDIATE C.2 IMMEDIATE C.3 IMMEDIATE C.4 IMMEDIATE
D. 0 or 12 foot Area monitor is not OPERATING	D.1 Restore MEASURING CHANNEL OR D.2 ENSURE reactor is shutdown OR D.3 ENSURE personnel are not in the reactor bay OR D.4 ENSURE personnel entering reactor bay are using portable survey meters to monitor dose rates	D.1 IMMEDIATE D.2 IMMEDIATE D.3 IMMEDIATE D.4 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Continuous air radiation monitor is not OPERATING	E.1 Restore MEASURING CHANNEL OR E.2 ENSURE reactor is shutdown OR E.3.a ENSURE EXHAUST PLENUM radiation monitor is OPERATING AND E.3.b Restore MEASURING CHANNEL	E.1 IMMEDIATE E.2. IMMEDIATE E.3.a. IMMEDIATE E.3.b Within 30 days
F. Exhaust plenum radiation monitor is not OPERATING	F.1 Restore MEASURING CHANNEL OR F.2 ENSURE reactor is shutdown OR F.3.a ENSURE continuous air radiation monitor is OPERATING AND F.3.b Restore MEASURING CHANNEL	F.1 IMMEDIATE F.2. IMMEDIATE F.3.a. IMMEDIATE F.3.b Within 30 days
G. The neutron count rate on the startup channel is not greater than the minimum detector sensitivity	G.1 Do not perform a reactor startup OR G.2 Perform a neutron-source check on the startup channel prior to startup	G.1 IMMEDIATE G.2 IMMEDIATE

3.3.5 Bases

Maximum steady state power level is 1,250 kW; neutron detectors measure reactor power level. Chapter 4 and 13 discuss normal and accident heat removal capabilities. Chapter 7 discusses radiation detection and monitoring systems, and neutron and power level detection systems.

According to General Atomics, detector voltages less than 90% of required operating value do not provide reliable, accurate nuclear instrumentation. Therefore, if operating voltage falls below the minimum value the power level channel is inoperable.

Primary water temperature indication is required to assure water temperature limits are met, protecting primary cleanup resin integrity. The reactor bay differential pressure indictor is required to control reactor bay atmosphere radioactive contaminants. Fuel temperature indication provides a means of observing that the safety limits are met.

The 22-foot and 0-foot area radiation monitors provide information about radiation hazards in the reactor bay. A loss of reactor pool water (Chapter 13), changes in shielding effectiveness (Chapter 11), and releases of radioactive material to the restricted area (Chapter 11) could cause changes in radiation levels within the reactor bay detectable by these monitors. Portable survey instruments will detect changes in radiation levels.

The air monitors (continuous air- and exhaust plenum radiation-monitor) provide indication of airborne contaminants in the reactor bay prior to discharge of gaseous effluent. Iodine channels provide evidence of fuel element failure. The air monitors provide similar information on independent channels; the continuous air monitor (CAM) has maximum sensitivity to iodine and particulate activity, while the air monitoring system (AMS) has individual channels for radioactive particulate, iodine, noble gas and iodine.

When filters in the air monitoring system begin to load, there are frequent, sporadic trips of the AMS alarms. Although the filters are changed on a regular basis, changing air quality makes these trips difficult to prevent. Short outages of the AMS system have resulted in unnecessary shutdowns, exercising the shutdown mechanisms unnecessarily, creating stressful situations, and preventing the ability to fully discharge the mission of the facility while the CAM also monitors conditions of airborne contamination monitored by the AMS. The AMS detector has failure modes than cannot be corrected on site; AMS failures have caused longer outages at the K-State reactor. The facility has experienced approximately two-week outages, with one week dedicated to testing and troubleshooting and (sometimes) one-week for shipment and repair at the vendor facility.

Permitting operation using a single channel of atmospheric monitoring will reduce unnecessary shutdowns while maintaining the ability to detect abnormal conditions as they develop. Relative indications ensure discharges are routine; abnormal indications trigger investigation or action to prevent the release of radioactive material to the surrounding environment. Ensuring the alternate airborne contamination monitor is functioning during outages of one system provides the contamination monitoring required for detecting abnormal conditions. Limiting the outage for a single unit to a maximum of 30 days ensures radioactive atmospheric contaminants are monitored while permitting maintenance and repair outages on the other system.

Chapter 13 discusses inventories and releases of radioactive material from fuel element failure into the reactor bay, and to the environment. Particulate and noble gas channels monitor more routine discharges. Chapter 11 and SAR Appendix A discuss routine discharges of radioactive gasses generated from normal operations into the reactor bay and into the environment. Chapter 3 identifies design bases for the confinement and ventilation system. Chapter 7 discusses air-monitoring systems.

Experience has shown that subcritical multiplication with the neutron source used in the reactor does not provide enough neutron flux to correspond to an indicated power level of 10 Watts. Therefore an indicated power of 10 Watts or more indicates operating in a potential critical condition, and at least one neutron channel is required with sensitivity at a neutron flux level corresponding to reactor power levels less than 10 Watts ("Startup Channel"). If the indicated neutron level is less than the minimum sensitivity for both the log-wide range and the multirange linear power level channels, a neutron source will be used to determine that at least one of the channels is responding to neutrons to ensure that the channel is functioning prior to startup.

3.4 Safety Channel and Control Rod Operability

3.4.1 Applicability

This specification applies to the reactor MEASURING Channels during STEADY STATE MODE and PULSE MODE operations.

3.4.2 Objective

The objectives are to require the minimum number of REACTOR SAFETY SYSTEM channels that must be OPERABLE in order to ensure that the fuel temperature safety limit is not exceeded, and to ensure prompt shutdown in the event of a scram signal.

3.4.3 Specifications

(1)	The SAFETY SYSTEM CHANNELS specified in TABLE 2 are OPERABLE
(2)	CONTROL RODS (STANDARD) are capable of 90% of full reactivity insertion from the fully withdrawn position in less than 1 sec.

TABLE 2: REQUIRED SAFETY SYSTEM CHANNELS				
Safety System Channel or Interlock	Minimum Number Operable	Function	Required OPERATING Mode	
			STEADY STATE MODE	PULSE MODE
Reactor power level	2	Scram	YES	NA
Manual scram bar	1	Scram	YES	YES
CONTROL ROD (STANDARD) position interlock	1	Prevent withdrawal of standard rods in the PULSE MODE	NA	YES
Pulse rod interlock	1	Prevent inadvertent pulsing while in STEADY STATE MODE	YES	NA

3.4.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any required SAFETY SYSTEM CHANNEL or interlock function is not OPERABLE	A.1 Restore channel or interlock to operation OR A.2 ENSURE reactor is SHUTDOWN	A1. IMMEDIATE A2. IMMEDIATE

3.4.5 (3) When operating in restricted mode operation, one reactor power level scram set point shall be set at 250 kW.

Bases

The power level scram is provided to ensure that reactor operation stays within the licensed limits of 1,250 kW, preventing abnormally high fuel temperature. The power level scram is not credited in analysis, but provides defense in depth to assure that the reactor is not operated in conditions beyond the assumptions used in analysis (Table 13.2.1.4).

The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

The CONTROL ROD (STANDARD) interlock function is to prevent withdrawing control rods (other than the pulse rod) when the reactor is in the PULSE MODE. This will ensure the reactivity addition rate during a pulse is limited to the reactivity added by the pulse rod.

The pulse rod interlock function prevents air from being applied to the transient rod drive when it is withdrawn while disconnected from the control rod to prevent inadvertent pulses during STEADY STATE MODE operations. The control rod interlock prevents inadvertent pulses which would be likely to exceed the maximum range of the power level instruments configured for steady state operations.

The power leve1 scram trip point specified for restricted mode operation is an added protection against fuel temperature exceeding the safety limit for aluminum-clad, low-hydride fuel and ensures that the reactor power will not exceed 200 kW.

3.5 Gaseous Effluent Control

3.5.1 Applicability

This specification applies to gaseous effluent in STEADY STATE MODE and PULSE MODE.

3.5.2 Objective

The objective is to ensure that exposures to the public resulting from gaseous effluents released during normal operations and accident conditions are within limits and ALARA.

3.5.3 Specification

(1)	The reactor bay ventilation exhaust system SHALL maintain in-leakage to the reactor bay
(2)	Releases of Ar-41 from the reactor bay exhaust plenum to an unrestricted environment SHALL NOT exceed 30 Ci per year.

3.5.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor bay ventilation exhaust system is not OPERABLE	A.1 ENSURE reactor is SHUTDOWN OR A.2.a Do not OPERATE in the PULSE MODE AND A.2.b Secure EXPERIMENT operations for EXPERIMENT with failure modes that could result in the release of radioactive gases or aerosols. A.2.c ENSURE no irradiated fuel handing AND A.2.d Restore the reactor bay ventilation exhaust system to OPERABLE	A.1 IMMEDIATE A.2.a IMMEDIATE A.2.b IMMEDIATE A.2.b IMMEDIATE A.2.d Within 30 days
Calculated releases of Ar-41 from the reactor bay exhaust plenum exceed 30 Ci per year.	Do not operate.	IMMEDIATE

3.5.5 Bases

The confinement and ventilation system is described in Section 3.5.4. Routine operations produce radioactive gas, principally Argon 41, in the reactor bay. If the reactor bay ventilation system is secured, SAR Chapter 11 Appendix A demonstrates reactor bay concentration of 0.746 Bq ml^{-1} (2.01×10^{-5} μCi ml^{-1}), well below the 10CFR20 annual limit of 2000 DAC hours of Argon 41 at 6×10^{-3} μCi h/mL. Therefore, the reduction in concentration of Argon 41 from operation of the confinement and ventilation system is a defense in depth measure, and not required to assure meeting personnel exposure limits. Consequently, the ventilation system can be secured without causing significant personnel hazard from normal operations. Thirty days for a confinement and ventilation system outage is selected as a reasonable interval to allow major repairs and work to be accomplished, if required. During this interval, experiment activities that might cause airborne radionuclide levels to be elevated are prohibited.

It is shown in Section 13.2.2 of the Safety Analysis Report that, if the reactor were to be operating at full steady-state power, fuel element failure would not occur even if all the reactor tank water were to be lost instantaneously.

Section 13.2.4 addresses the maximum hypothetical fission product inventory release. Using unrealistically conservative assumptions, concentrations for a few nuclides of iodine would be in excess of occupational derived air concentrations for a matter of hours or days. ^{90}Sr activity available for release from fuel rods previously used at other facilities is estimated to be at most about 4 times the ALI. In either case (radio-iodine or -Sr), there is no credible scenario for accidental inhalation or ingestion of the undiluted nuclides that might be released from a damaged fuel element. Finally, fuel element failure during a fuel handling accident is likely to be observed and mitigated immediately.

SAR Appendix A shows the release of 30 Ci per year of Ar-41 from normal operations would result in less than 10 mrem annual exposure to any person in unrestricted areas.

3.6 Limitations on Experiments

3.6.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.6.2 Objectives

These Limiting Conditions for Operation prevent reactivity excursions that might cause the fuel temperature to exceed the safety limit (with possible resultant damage to the reactor), and the excessive release of radioactive materials in the event of an EXPERIMENT failure

3.6.3 Specifications

(1)	If all fuel elements are stainless steel clad, the reactivity worth of any individual EXPERIMENT SHALL NOT exceed \$2.00
(2)	If two or more experiments in the reactor are interrelated so that operation or failure of one can induce reactivity-affecting change in the other(s), the sum of the absolute reactivity of such experiments SHALL NOT exceed \$2.00.
(3)	Irradiation holders and vials SHALL prevent release of encapsulated material in the reactor pool and core area

3.6.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. INDEPENDENT EXPERIMENT worth is greater than \$2.00	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Remove the experiment	A.1 IMMEDIATE A.2 Prior to continued operations
C. An irradiation holder or vial releases material capable of causing damage to the reactor fuel or structure into the pool or core area	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Inspect the affected area AND C.3 Obtain RSC review and approval	C.1 IMMEDIATE C.2 Prior to continued operation C.3 Prior to continued operation

3.6.5 Bases

Specifications 3.7(1) through 3.7(3) are conservatively chosen based on prior operation at 250 kW to limit reactivity additions to maximum values that are less than an addition which could cause temperature to challenge the safety limit.

Experiments are approved with expectations that there is reasonable assurance the facility will not be damaged during normal or failure conditions. If an irradiation capsule which contains material with potential for challenging the fuel cladding or pool wall, the facility will be inspected to ensure that continued operation is acceptable.

The additional limitations for restricted mode operation are chosen to limit the temperature excursion in a pulse initiated by possible malfunctions in experiments.

3.7 Fuel Integrity

3.7.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.7.2 Objective

The objective is to prevent the use of damaged fuel in the KSU TRIGA reactor.

3.7.3 Specifications

(1)	Fuel elements in the reactor core SHALL NOT be elongated more than 1/8 in. over manufactured length
(2)	Fuel elements in the reactor core SHALL NOT be laterally bent more than 1/8 in.

3.7.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any fuel element is elongated greater than l/8 in. over manufactured length, or bent laterally greater than l/8 in.	Do not insert the fuel element into the upper core grid plate.	IMMEDIATE

3.7.5 Bases

The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at TRIGA installations. Fuel cladding integrity is important since it represents the only process barrier for fission product release from the TRIGA reactor.

3.8 Reactor Pool Water

3.8.1 Applicability

This specification applies to operations in STEADY STATE MODE, PULSE MODE, and SECURED MODE.

3.8.2 Objective

The objective is to set acceptable limits on the water quality, temperature, conductivity, and level in the reactor pool.

3.8.3 Specifications

(1)	Water temperature at the exit of the reactor pool SHALL NOT exceed 130°F with flow through the primary cleanup loop
(2)	Water conductivity SHALL be less than 5 μmho/cm
(3)	Water level above the core SHALL be at least 13 ft from the top of the core

3.8.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Water temperature at the exit of the reactor pool exceeds 130°F	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Secure flow through the demineralizer AND A.3 Reduce water temperature to less than 130°F	A.1 IMMEDIATE A.2 IMMEDIATE A.3 IMMEDIATE
B. Water conductivity is greater than 5 μmho/cm	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Restore conductivity to less than 5 μmho/cm	B.1 IMMEDIATE B.2 Within 4 weeks
C. Water level above the core SHALL be at least 13 ft from the top of the core for all operating conditions	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Restore water level	C.1 IMMEDIATE C.2 IMMEDIATE

3.8.5 Bases

The resin used in the mixed bed deionizer limits the water temperature of the reactor pool. Resin in use (as described in Section 5.4) maintains mechanical and chemical integrity at temperatures below 130°F (54.4°C).

Maintaining low water conductivity over a prolonged period prevents possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding. Although fuel degradation does not occur over short time intervals, long-term integrity of the fuel is important, and a 4-week interval was selected as an appropriate maximum time for high conductivity.

The top of the core is 16 feet below the top of the primary coolant tank. The lowest suction of primary cooling flow into the forced cooling loop is 3.5 feet below the top of the primary coolant tank (water level is maintained about 6 inches below the top of the tank).

The principle contributor to radiation dose rates at the pool surface is Nitrogen 16 generated in the reactor core and dispersed in the pool. Calculations in Chapter 11 show the pool surface radiation dose rates from Nitrogen 16 with 13 feet of water above the core are acceptable.

For normal pool temperature, calculations in Chapter 4 assuming 16 feet and 13 feet above the core demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, pool levels greater than 13 feet above the core meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin.

Therefore, a minimum pool level of 13 feet above the core is adequate to provide shielding and support the core cooling.

3.9 Maintenance Retest Requirements

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.9.2 Objective

The objective is to ensure Technical Specification requirements are met following maintenance that occurs within surveillance test intervals.

3.9.3 Specifications

Maintenance activities SHALL NOT change, defeat or alter equipment or systems in a way that prevents the systems or equipment from being OPERABLE or otherwise prevent the systems or equipment from fulfilling the safety basis

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
Maintenance is performed that has the potential to change a setpoint, calibration, flow rate, or other parameter that is measured or verified in meeting a surveillance or operability requirement	Perform surveillance OR Operate only to perform retest	Prior to continued, normal operation in STEADY STATE MODE or PULSE MODE

3.9.5 Bases

Operation of the K-State reactor will comply with the requirements of Technical Specifications. This specification ensures that if maintenance might challenge a Technical Specifications requirement, the requirement is verified prior to resumption of normal operations.

4 SURVEILLANCE REQUIREMENTS

4.1 Core Reactivity

4.1.1 Objective

This surveillance ensures that the minimum SHUTDOWN MARGIN requirements and maximum excess reactivity limits of section 3.1 are met.

4.1.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
SHUTDOWN MARGIN Determination	SEMIANNUAL
Excess Reactivity Determination	SEMIANNUAL
	Following Insertion of experiments with measurable positive reactivity
Control Rod Reactivity Worth determination	BIENNIAL

4.1.3 Basis

Experience has shown verification of the minimum allowed SHUTDOWN MARGIN at the specified frequency is adequate to assure that the limiting safety system setting is met

When core reactivity parameters are affected by operations or maintenance, additional activity is required to ensure changes are incorporated in reactivity evaluations.

4.2 Pulse Mode

4.2.1 Objectives

The verification that the pulse rod position does not exceed a reactivity value corresponding to $3.00 assures that the limiting condition for operation is met.

4.2.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
ENSURE Transient Pulse Rod position corresponds to reactivity not greater than $3.00	Prior to pulsing operations

4.2.3 Basis

Verifying pulse rod position corresponds to less than $3.00 ensures that the maximum pulsed reactivity meets the limiting condition for operation.

4.3 Measuring Channels

4.3.1 Objectives

Surveillances on MEASURING CHANNELS at specified frequencies ensure instrument problems are identified and corrected before they can affect operations.

4.3.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
Reactor power level MEASURING CHANNEL	
CHANNEL TEST	DAILY
Calorimetric calibration	ANNUAL
CHANNEL CHECK high voltage to required power level instruments	DAILY
Primary pool water temperature CHANNEL CALIBRATION	ANNUAL
Reactor Bay differential pressure CHANNEL CALIBRATION	ANNUAL
Fuel temperature CHANNEL CALIBRATION	ANNUAL
22 Foot Area radiation monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
0 or 12 Foot Area Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Continuous Air Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
EXHAUST PLENUM Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Startup Count Rate	DAILY

4.3.3 Basis

The DAILY CHANNEL CHECKS will ensure that the SAFETY SYSTEM and MEASURING CHANNELS are operable. The required periodic calibrations and verifications will permit any long-term drift of the channels to be corrected.

4.4 Safety Channel and Control Rod Operability

4.4.1 Objective

The objectives of these surveillance requirements are to ensure the REACTOR SAFETY SYSTEM will function as required. Surveillances related to safety system MEASURING CHANNELS ensure appropriate signals are reliably transmitted to the shutdown system; the surveillances in this section ensure the control rod system is capable of providing the necessary actions to respond to these signals.

4.4.2 Specifications

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
Manual scram SHALL be tested by releasing partially withdrawn CONTROL RODS (STANDARD)	DAILY
CONTROL ROD (STANDARD) drop times SHALL be measured to have a drop time from the fully withdrawn position of less than 1 sec.	ANNUAL
The control rods SHALL be visually inspected for corrosion and mechanical damage at intervals	BIENNIAL
CONTROL ROD (STANDARD) position interlock functional test	SEMIANNUAL
Pulse rod interlock functional test	SEMIANNUAL
On each day that PULSE MODE operation of the reactor is planned, a functional performance check of the CONTROL ROD (TRANSIENT) system SHALL be performed.	Prior to pulsing operations each day a pulse is planned
The CONTROL ROD (TRANSIENT) rod drive cylinder and the associated air supply system SHALL be inspected, cleaned, and lubricated, as necessary.	SEMIANNUAL

4.4.3 Basis

Manual and automatic scrams are not credited in accident analysis, although the systems function to assure long-term safe shutdown conditions. The manual scram and control rod drop timing surveillances are intended to monitor for potential degradation that might interfere with the operation of the control rod systems. The verification of high voltage to the power level monitoring channels assures that the instrument channel providing an overpower trip will function on demand.

The control rod inspections (visual inspections and transient drive system inspections) are similarly intended to identify potential degradation that lead to control rod degradation or inoperability.

A test of the interlock that prevents the pulse rod from coupling to the drive in the state state mode unless the drive is fully down assures that pulses will occur only when in pulsing mode. A test of the interlock that prevents standard control rod motion while in the pulse mode assures that the interlock will function as required.

The functional checks of the control rod drive system assure the control rod drive system operates as intended for any pulsing operations. The inspection of the pulse rod mechanism will assure degradation of the pulse rod drive will be detected prior to malfunctions.

4.5 Gaseous Effluent Control

4.5.1 Objectives

These surveillances ensure that routine releases are normal, and (in conjunction with MEASURING CHANNEL surveillances) that instruments will alert the facility if conditions indicate abnormal releases.

4.5.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
Perform CHANNEL TEST of air monitor	ANNUAL
Verify negative reactor bay differential pressure	DAILY

4.5.3 Basis

The continuous air monitor provides indication that levels of radioactive airborne contamination in the reactor bay are normal.

If the reactor bay differential pressure gage indicates a negative pressure, the reactor bay exhaust fan is controlling airflow by directing effluent out of confinement.

4.6 Limitations on Experiments

4.6.1 Objectives

This surveillance ensures that experiments do not have significant negative impact on safety of the public, personnel or the facility.

4.6.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
Experiments SHALL be evaluated and approved prior to implementation.	Prior to inserting a new experiment for purposes other than determination of reactivity worth
Measure and record experiment worth of the EXPERIMENT (where the absolute value of the estimated worth is greater than $0.40).	Initial insertion of a new experiment where absolute value of the estimated worth is greater than $0.40

4.6.3 Basis

These surveillances allow determination that the limits of 3.7 are met.

Experiments with an absolute value of the estimated significant reactivity worth (greater than $0.40) will be measured to assure that maximum experiment reactivity worths are met. If an absolute value of the estimate indicates less than $0.40 reactivity worth, even a 100% error will result in actual reactivity less than the assumptions used in analysis for inadvertent pulsing at low power operations in the Safety Analysis Report (13.2.3, Case I).

4.7 Fuel Integrity

4.7.1 Objective

The objective is to ensure that the dimensions of the fuel elements remain within acceptable limits.

4.7.2 Applicability

This specification applies to the surveillance requirements for the fuel elements in the reactor core.

4.7.3 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
The standard fuel elements SHALL be visually inspected for corrosion and mechanical damage, and measured for length and bend	500 pulses of magnitude equal to or less than a pulse insertion of 3.00$ AND Following the exceeding of a limited safety system set point with potential for causing degradation
B, C, D, E, and F RING elements comprising approximately 1/3 of the core SHALL be visually inspected annually for corrosion and mechanical damage such that the entire core SHALL be inspected at 3-year intervals, but not to exceed 38 months	ANNUAL

4.7.4 Basis

The most severe stresses induced in the fuel elements result from pulse operation of the reactor, during which differential expansion between the fuel and the cladding occurs and the pressure of the gases within the elements increases sharply.

Triennial visual inspection of fuel elements combined with measurements at intervals determined by pulsing as described is considered adequate to identify potential degradation of fuel prior to catastrophic fuel element failure.

4.8 Reactor Pool Water

This specification applies to the water contained in the KSU TRIGA reactor pool.

4.8.1 Objective

The objective is to provide surveillance of reactor primary coolant water quality, pool level, temperature and (in conjunction with MEASURING CHANNEL surveillances), and conductivity.

4.8.2 Specification

Surviellance RequiremeNts

SURVEILLANCE	FREQUENCY
Verify reactor pool water level above the inlet line vacuum breaker	DAILY
Verify reactor pool water temperature channel operable	DAILY
Measure reactor Pool water conductivity	DAILY
	At least every 20 days

4.9.3 Bases

Surveillance of the reactor pool will ensure that the water level is adequate before reactor operation. Evaporation occurs over longer periods of time, and daily checks are adequate to identify the need for water replacement.

Water temperature must be monitored to ensure that the limit of the deionizer will not be exceeded. A daily check on the instrument prior to reactor operation is adequate to ensure the instrument is operable when it will be needed.

Water conductivity must be checked to ensure that the deionizer is performing properly and to detect any increase in water impurities. A daily check is adequate to verify water quality is appropriate and also to provide data useful in trend analysis. If the reactor is not operated for long periods of time, the requirement for checks at least every 20 days will ensure water quality is maintained in a manner that does not permit fuel degradation.

4.9 Maintenance Retest Requirements

4.9.1 Objective

The objective is to ensure that a system is OPERABLE within specified limits before being used after maintenance has been performed.

4.9.2 Specification

Surviellance Requirements

SURVEILLANCE	FREQUENCY
Evaluate potential for maintenance activities to affect operability and function of equipment required by Technical Specifications	Following maintenance of systems of equipment required by Technical Specifications
Perform surveillance to assure affected function meets requirements	Prior to resumption of normal operations

4.9.3 Bases

This specification ensures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

Surveillance of the sealed plutonium source material will ensure that the total-body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material.

5 DESIGN FEATURES

5.1 Reactor Fuel

5.1.1 Applicability

This specification applies to the fuel elements used in the reactor core.

5.1.2 Objective

The objective is to ensure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their mechanical integrity.

5.1.3 Specification

(l) The high-hydride fuel element shall contain uranium-zirconium hydride, clad in 0.020 in. of 304 stainless steel. It shall contain a maximum of 9.0 weight percent uranium which has a maximum enrichment of 20%. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

(2) For the loading process, the elements shall be placed in a close packed array except for experimental facilities or for single positions occupied by control rods and a neutron startup source.

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5.1.4 Bases

These types of fuel elements have a long history of successful use in TRIGA reactors.

5.2 Reactor Fuel and Fueled Devices in Storage

5.2.1 Applicability

This specification applies to reactor fuel elements in storage

5.2.2 Objective

The objective is to ensure fuel elements or fueled devices in storage are maintained Subcritical in a safe condition.

5.2.3 Specification

(1) All fuel elements or fueled devices shall be in a safe, stable geometry

(2) The k_{eff} of all fuel elements or fueled devices in storage is less than 0.8

(3) Irradiated fuel elements or fueled devices will be stored in an array which will permit sufficient natural convection cooling by air or water such that the fuel element or fueled device will not exceed design values.

5.2.4 Bases

This specification is based on American Nuclear Society standard 15.1, section 5.4.

5.3 Reactor Building

5.3.1 Applicability

This specification applies to the building that houses the TRIGA reactor facility.

5.3.2 Objective

The objective is to ensure that provisions are made to restrict the amount of release of radioactivity into the environment.

5.3.3 Specification

(l) The reactor shall be housed in a closed room designed to restrict leakage when the reactor is in operation, when the facility is unmanned, or when spent fuel is being handled exterior to a cask.

(2) The minimum free volume of the reactor room shall be approximately 144,000 cubic feet.

(3) The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 30 ft. above ground level.

5.3.4 Bases

To control the escape of gaseous effluent, the reactor room contains no windows that can be opened. The room air is exhausted through an independent exhaust system, and discharged at roof level to provide dilution.

5.4 Experiments

5.4.1 Applicability

This specification applies to the design of experiments.

5.4.2 Objective

The objective is to ensure that experiments are designed to meet criteria.

5.4.3 Specifications

(1) EXPERIMENT with a design reactivity worth greater than $1.00 SHALL be securely fastened (as defined in Section l, Secured Experiment).

(2) Design shall ensure that failure of an EXPERIMENT SHALL NOT lead to a direct failure of a fuel element or of other experiments that could result in a measurable increase in reactivity or a measurable release of radioactivity due to the associated failure.

(3) EXPERIMENT SHALL be designed so that it does not cause bulk boiling of core water

(4) EXPERIMENT design SHALL ensure no interference with control rods or shadowing of reactor control instrumentation.

(5) EXPERIMENT design shall minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials.

(6) Each fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 5 millicuries except as the fueled experiment is a standard TRIGA instrumented element in which instance the iodine inventory limit is removed.

(7) Where the possibility exists that the failure of an EXPERIMENT (except fueled EXPERIMENTS) could release radioactive gases or aerosols to the reactor bay or atmosphere, the quantity and type of material shall be limited such that the airborne concentration of radioactivity averaged over a year will not exceed the limits of Table II of Appendix B of 10 CFR Part 20 assuming 100% of the gases or aerosols escape.

(8) The following assumptions shall be used in experiment design:

a. If effluents from an experimental facility exhaust through a hold-up tank which closes automatically at a high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

b. If effluents from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the aerosols produced will escape.

c. For materials whose boiling point is above 130ºF and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, at least 10% of these vapors will escape.

(9) Use of explosive solid or liquid material with a National Fire Protection Association Reactivity (Stability) index of 2, 3, or 4 in the reactor pool or biological shielding SHALL NOT exceed the equivalent of 25 milligrams of TNT without prior NRC approval.

5.4.4 Basis

Designing the experiment to reactivity and thermal-hydraulic conditions ensure that the experiment is not capable of breaching fission product barriers or interfering with the control systems (interferences from other - than reactivity - effects with the control and safety systems are also prohibited). Design constraints on industrial hazards ensure personnel safety and continuity of operations. Design constraints limiting the release of radioactive gasses prevent unacceptable personnel exposure during off-normal experiment conditions.

Irradiated fuel storage is described in the Safety Analysis Report, Section 5.3.4.

6 ADMINISTRATIVE CONTROLS

6.1 Organization and Responsibilities of Personnel

a) Structure.

The reactor organization is related to the University structure as shown in SAR Figure 12.1 and Technical Specifications Figure TS.1 below.

Kansas State University (KSU) holds the license for the KSU TRIGA Reactor, located in the KSU Nuclear Reactor Facility in Ward Hall on the campus of Kansas State University. The chief administrating officer for KSU is the President. Environment, safety and health oversight functions are administered through the Vice President for Administration and Finance, while reactor line management functions are through the Provost Chief Academic Officer.

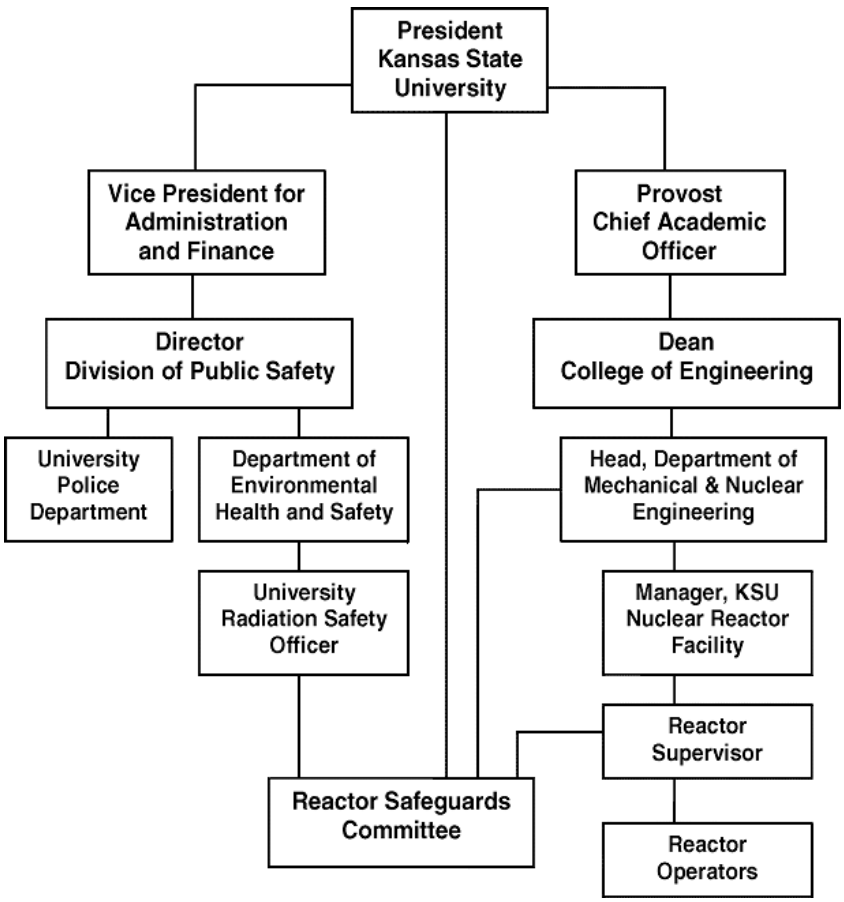

Figure TS. 1 - Organization and management structure for the K-State reactor

Radiation protection functions are divided between the University Radiation Safety Officer (RSO) and the reactor staff and management, with management and authority for the RSO separate from line management and authority for facility operations. Day-to-day radiation protection functions implemented by facility staff and management are guided by approved administrative controls (Reactor Radiation Protection Program or RPP, Facility Operating Manual, operating and experiment procedures); these controls are reviewed and approved by the RSO as part of the Reactor Safeguards Committee (with specific veto authority). The RSO has specific oversight functions assigned though the RPP. The RSO provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The RSO provides guidance on request for non-routine operations such as transportation and implementation of new experiments.

b) Responsibility.

The President of the University shall be responsible for the appointment of responsible and competent persons as members of the TRIGA Reactor Safeguards Committee upon the recommendation of the *ex officio* Chairperson of the Committee.

The KSU Nuclear Reactor Facility shall be under the supervision of the Nuclear Reactor Facility Manager, who shall have the overall responsibility for safe, efficient, and competent use of its facilities in conformity with all applicable laws, regulations, terms of facility licenses, and provisions of the Reactor Safeguards Committee. The Manager also has responsibility for maintenance and modification of laboratories associated with the Reactor Facility. The Manager shall have education and/or experience commensurate with the responsibilities of the position and shall report to the Head of the Department of Mechanical and Nuclear Engineering.

A Reactor Supervisor may serve as the deputy of the Nuclear Reactor Facility Manager in all matters relating to the enforcement of established rules and procedures (but not in matters such as establishment of rules, appointments, and similar administrative functions). The Supervisor should have at least two years of technical training beyond high school and shall possess a Senior Reactor Operator's license. The Supervisor shall have had reactor OPERATING experience and have a demonstrated competence in supervision. The Supervisor is appointed by the Nuclear Reactor Facility Manager and is responsible for enforcing all applicable rules, procedures, and regulations, for ensuring adequate exchange of information between OPERATING personnel when shifts change, and for reporting all malfunctions, accidents, and other potentially hazardous occurrences and situations to the Reactor Nuclear Reactor Facility Manager. The Nuclear Reactor Facility Manager may also serve as Reactor Supervisor.

The Reactor Operator shall be responsible for the safe and proper operation of the reactor, under the direction of the Reactor Supervisor. Reactor Operators shall possess an Operator's or Senior Operator's license and shall be appointed by the Nuclear Reactor Facility Manager.

The University Radiation Safety Officer (RSO), or a designated alternate, shall (in addition to other duties defined by the Director of Environmental Health and Safety, Division of Public Safety) be responsible for overseeing the safety of Reactor Facility operations from the standpoint of radiation protection. The RSO and/or designated alternate shall be appointed by the Director of Environmental Health and Safety, Division of Public Safety, with the approval of the University Radiation Safety Committee, and shall report to the Director of Environmental Health and Safety, whose organization is independent of the Reactor Facility organization, as shown on SAR Figure 12.1.

The Nuclear Reactor Facility Manager, with the approval of the Reactor Safeguards Committee, may designate an appropriately qualified member of the Facility organization as Reactor Facility Safety Officer (RFSO) with duties including those of an intra-Facility Radiation Safety Officer. The University Radiation Safety Officer may, with the concurrence of the Nuclear Reactor Facility Manager, authorize the RFSO to perform some of the specific duties of the RSO at the Nuclear Reactor Facility.

c). Staffing.

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator (designated as Reactor Supervisor). The Supervisor shall be on call, within twenty minutes travel time to the facility.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations.

In addition to the above requirements, during fuel movement a senior operator shall be inside the reactor bay directing fuel operations.

6.2 Review and Audit

a) There will be a Reactor Safeguards Committee which shall review TRIGA reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public and of persons within the Laboratory.

b) The responsibilities of the Committee include, but are not limited to, the following:

1. Review and approval of rules, procedures, and proposed Technical Specifications;

2. Review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and Technical Specifications, in accordance with procedures in Section 6.3;

3. Review and approval of experiments using the reactor in accordance with procedures and criteria in Section 6.4;

4. Determine whether changes in the facility as described in the safety analysis report (as updated), changes in the procedures as described in the final safety analysis report (as updated), and the conduct of tests or experiments not described in the safety analysis report (as updated) may be accomplished in accordance with 10 CFR 50.59 without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

5. Review of abnormal performance of plant equipment and OPERATING anomalies;

6. Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR 20 and 10 CFR50;

7. Inspection of the facility, review of safety measures, and audit of operations at a frequency not less than once a year, including operation and operations records of the facility;

8. Requalification of the Nuclear Reactor Facility Manager and/or the Reactor Supervisor,

9. Review of container failures where released materials have the potential for damaging reactor fuel or structural components including:

   a) results of physical inspection

   b) evaluation of consequences

   c) need for corrective actions

c) The Committee shall be composed of:

1. one or more persons proficient in reactor and nuclear science or engineering,

2. one or more persons proficient in chemistry, geology, or chemical engineering,

3. one person proficient in biological effects of radiation,

4. the Nuclear Reactor Facility Manager, *ex officio*,

5. the University Radiation Safety Officer, *ex officio*, and,

6. The Head of the Department of Mechanical and Nuclear Engineering, *ex officio*, or a designated deputy, to serve as chairperson of the Committee.

The same individual may serve under more than one category above, but the minimum membership shall be seven. At least five members shall be faculty members. The Reactor Supervisor, if other than the Nuclear Reactor Facility Manager, shall attend and participate in Committee meetings, but shall not be a voting member.

d) The Committee shall have a written statement defining its authority and responsibilities, the subjects within its purview, and other such administrative provisions as are required for its effective functioning. Minutes of all meetings and records of all formal actions of the Committee shall be kept.

e) A quorum shall consist of not less than a majority of the full Committee and shall include all *ex officio* members.

f) Any permissive action of the Committee requires affirmative vote of the University Radiation Safety Officer as well as a majority vote of the members present.

g) The Committee shall meet a minimum of two times a year. Additional meetings may be called by any member, and the Committee may be polled in lieu of a meeting. Such a poll shall constitute Committee action subject to the same requirements as for an actual meeting.

6.3 Procedures

a) Written procedures, reviewed and approved by the Reactor Safeguards Committee, shall be followed for the activities listed below. The procedures shall be adequate to assure the safety of the reactor, persons within the Laboratory, and the public, but should not preclude the use of independent judgment and action should the situation require it. The activities are:

1. Startup, operation, and shutdown of the reactor, including

   (a) startup checkout procedures to test the reactor instrumentation and safety systems, area monitors, and continuous air monitors,

   (b) prohibition of routine operations with failed (or leaking) fuel except to find leaking elements, and

   (b) shutdown procedures to assure that the reactor is secured before OPERATING personnel go off duty.

2. Installation or removal of fuel elements, control rods, and other core components that significantly affect reactivity or reactor safety.

3. Preventive or corrective maintenance activities which could have a significant effect on the safety of the reactor or personnel.

4. Periodic inspection, testing or calibration of auxiliary systems or instrumentation that relate to reactor operation.

b) Substantive changes in the above procedures shall be made only with the approval of the Reactor Safeguards Committee, and shall be issued to the OPERATING personnel in written form. The Nuclear Reactor Facility Manager may make temporary changes that do not change the original intent. The change and the reasons thereof shall be noted in the log book, and shall be subsequently reviewed by the Reactor Safeguards Committee.

c) Determination as to whether a proposed activity in categories (1), (2) and (3) in Section 6.2b above does or does not have a significant safety effect and therefore does or does not require approved written procedures shall require the concurrence of

1. the Nuclear Reactor Facility Manager, and

2. at least one other member of the Reactor Safeguards Committee, to be selected for relevant expertise by the Nuclear Reactor Facility Manager. If the Manager and the Committee member disagree, or if in their judgment the case warrants it, the proposal shall be submitted to the full Committee, and

3. the University Radiation Safety Officer, or his/her deputy, who may withhold agreement until approval by the University Radiation Safety Committee is obtained.

The Rector Safeguards Committee shall subsequently review determinations that written procedures are not required. The time at which determinations are made, and the review and approval of written procedures, if required, are carried out, shall be a reasonable interval before the proposed activity is to be undertaken.

d) Determination that a proposed change in the facility does or does not have a significant safety effect and therefore does or does not require review and approval by the full Reactor Safeguards Committee shall be made in the same manner as for proposed activities under (c) above.

6.4 Review of Proposals for Experiments

a) All proposals for new experiments involving the reactor shall be reviewed with respect to safety in accordance with the procedures in (b) below and on the basis of criteria in (c) below.

b) Procedures:

1. Proposed reactor operations by an experimenter are reviewed by the Reactor Supervisor, who may determine that the operation is described by a previously approved EXPERIMENT or procedure. If the Reactor Supervisor determines that the proposed operation has not been approved by the Reactor Safeguards Committee, the experimenter shall describe the proposed EXPERIMENT in written form in sufficient detail for consideration of safety aspects. If potentially hazardous operations are involved, proposed procedures and safety measures including protective and monitoring equipment shall be described.

2. If the experimenter is a student, approval by his/her research supervisor is required. If the experimenter is a staff or faculty member, his/her own signature is sufficient.

3. The proposal is then to be submitted to the Reactor Safeguards Committee for consideration and approval. The Committee may find that the experiment, or portions thereof, may only be performed in the presence of the University Radiation Safety Officer or Deputy thereto.

4. The scope of the EXPERIMENT and the procedures and safety measures as described in the approved proposal, Including any amendments or conditions added by those reviewing and approving it, shall be binding on the experimenter and the OPERATING personnel. Minor deviations shall be allowed only in the manner described in Section 6 above. Recorded affirmative votes on proposed new or revised experiments or procedures must indicated that the Committee determines that the proposed actions do not involve changes in the facility as designed, changes in Technical Specifications, changes that under the guidance of 10 CFR 50.59 require prior approval of the NRC, and could be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

5. Transmission to the Reactor Supervisor for scheduling.

c) Criteria that shall be met before approval can be granted shall include:

1. The EXPERIMENT must meet the applicable Limiting Conditions for Operation and Design Description specifications.

2. It must not involve violation of any condition of the facility license or of Federal, State, University, or Facility regulations and procedures.

3. The conduct of tests or experiments not described in the safety analysis report (as updated) must be evaluated in accordance with 10 CFR 50.59 to determine if the test or experiment can be accomplished without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

4. In the safety review the basic criterion is that there shall be no hazard to the reactor, personnel or public. The review SHALL determine that there is reasonable assurance

that the experiment can be performed with no significant risk to the safety of the reactor, personnel or the public.

6.5 Emergency Plan and Procedures

An emergency plan shall be established and followed in accordance with NRC regulations. The plan shall be reviewed and approved by the Reactor Safeguards Committee prior to its submission to the NRC. In addition, emergency procedures that have been reviewed and approved by the Reactor Safeguards Committee shall be established to cover all foreseeable emergency conditions potentially hazardous to persons within the Laboratory or to the public, including, but not limited to, those involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity.

6.6 Operator Requalification

An operator requalification program shall be established and followed in accordance with NRC regulations.

Physical Security Plan

Administrative controls for protection of the reactor plant shall be established and followed in accordance with NRC regulations.

6.8 Action To Be Taken In The Event A Safety Limit Is Exceeded

In the event a safety limit is exceeded:

a) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

b) An immediate report of the occurrence shall be made to the Chair of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.11 of these specifications.

c) A report shall be made to include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to Reactor Safeguards Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.9 Action To Be Taken In The Event Of A Reportable Occurrence

a) A reportable occurrence is any of the following conditions:

1. any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings;

2. VIOLATION OF SL, LSSS or LCO;

NOTES

Violation of an LSSS or LCO occurs through failure to comply with an “Action” statement when “Specification” is not met; failure to comply with the “Specification” is not by itself a violation.

Surveillance Requirements must be met for all equipment/components/conditions to be considered operable.

Failure to perform a surveillance within the required time interval or failure of a surveillance test shall result in the /component/condition being inoperable

3. incidents or conditions that prevented or could have prevented the performance of the intended safety functions of an engineered safety feature or the REACTOR SAFETY SYSTEM;

4. release of fission products from the fuel that cause airborne contamination levels in the reactor bay to exceed 10CFR20 limits for releases to unrestricted areas;

5. an uncontrolled or unanticipated change in reactivity greater than $1.00;

6. an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor;

7. an uncontrolled or unanticipated release of radioactivity.

b) In the event of a reportable occurrence, the following actions shall be taken:

1. The reactor shall be shut down at once. The Reactor Supervisor shall be notified and corrective action taken before operations are resumed; the decision to resume shall require approval following the procedures in Section 6.3.

2. A report shall be made to include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safeguards Committee for review.

3. A report shall be submitted to the NRC in accordance with Section 6.11 of these specifications.

6.10 Plant Operating Records

a) In addition to the requirements of applicable regulations, in 10 CFR 20 and 50, records and logs shall be prepared and retained for a period of at least 5 years for the following items as a minimum.

1. normal plant operation, including power levels;

3. principal maintenance activities;

4. reportable occurrences;

5. equipment and component surveillance activities;

6. experiments performed with the reactor;

7. all emergency reactor scrams, including reasons for emergency shutdowns.

b) The following records shall be maintained for the life of the facility:

1. gaseous and liquid radioactive effluents released to the environs;

2. offsite environmental monitoring surveys;

3. fuel inventories and transfers;

4. facility radiation and contamination surveys;

5. radiation exposures for all personnel;

6. updated, corrected, and as-built drawings of the facility.

6.11 Reporting Requirements

All written reports shall be sent within the prescribed interval to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the US. Nuclear Regulatory Commission (NRC) as follows:

a) A report within 24 hours by telephone and fax or electronic mail to the NRC Operations Center and the USNRC Region IV of;

1. any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;

2. any violation of a safety limit;

3. any reportable occurrences as defined in Section 6.9 of these specifications.

b) A report within 10 days in writing of:

1. any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury or exposure; the written report (and, to the extent possible, the preliminary telephone and telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;

2. any violation of a safety limit;

3. any reportable occurrence as defined in Section 1.1 of these specifications.

c) A report within 30 days in writing of:

1. any significant variation of a MEASURED VALUE from a corresponding predicted or previously MEASURED VALUE of safety-connected OPERATING characteristics occurring during operation of the reactor;

2. any significant change in the transient or accident analysis as described in the Safety Analysis Report.

3. a change in personnel for the Department of Mechanical and Nuclear Engineering Chair, or a change in reactor manager

d) A report within 60 days after criticality of the reactor in writing to the US Nuclear Regulatory Commission, resulting from a receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the MEASURED VALUE of the OPERATING conditions or characteristics of the reactor under the new conditions.

e) A routine report in writing to the US. Nuclear Regulatory Commission within 60 days after completion of the first calendar year of OPERATING and at intervals not to exceed 12 months, thereafter, providing the following information:

1. a brief narrative summary of OPERATING experience (including experiments performed), changes in facility design, performance characteristics, and OPERATING procedures related to reactor safety occurring during the reporting period; and results of surveillance tests and inspections;

2. a tabulation showing the energy generated by the reactor (in megawatt-hours);

3. the number of emergency shutdowns and inadvertent scrams, including the reasons thereof and corrective action, if any, taken;

4. discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required;

5. a summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;

6. a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge;

7. a description of any environmental surveys performed outside the facility;

8. a summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

Radiation Protection Program

I. INTRODUCTION

This Radiation Protection Program has been prepared by personnel of the Kansas State University TRIGA Mk II Nuclear reactor facility in response to the requirements of Title 10, Part 20, Code of Federal Regulations (10CFR20). The FACILITY is operated under FEDERAL LICENSE R-88 (Docket 50-188) issued by the U.S. Nuclear Regulatory Commission (NRC). The Program is executed in coordination with the Office of Radiation Safety, Department of Environmental Health and Safety, Kansas State University. It has been reviewed and approved by the Reactor Safeguards Committee for the Reactor Facility. Certain aspects of the Program deal with radioactive materials regulated by the State of Kansas (an Agreement state) under STATE LICENSE 38-C011-01. The University Radiation Safety Committee has reviewed the Radiation Protection Program with respect to State license activities.

This program is a part of the Operations Manual for the Reactor Facility, although it is published separately. A closely related part of the Operations Manual, also published separately, is the Emergency Plan. Appendix A is a glossary of terms used in the Radiation Protection Program. Appendices B and C contain lists of operational and emergency procedures referred to in the Radiation Protection Program.

The Radiation Protection Program is designed to meet requirements of 10CFR20. It has been developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* [1] and Regulatory Guides issued by the NRC [2-7].

2. MANAGEMENT AND ADMINISTRATION

Radiation Protection Program preparation, audit, and review are the responsibilities of the Nuclear Reactor Facility Manager. The Reactor Safeguards Committee chaired by the Head of the Department of Mechanical and Nuclear Engineering reviews the activities of the Nuclear Reactor Facility Manager and semi-annual audits prepared under the direction of the Nuclear Reactor Facility Manager. The Reactor Safeguards Committee examines records required by the Radiation Protection Program as well as audit reports by the Nuclear Reactor Facility Manager during their semi-annual inspections.

Training, surveillance and record keeping are the responsibility of the Nuclear Reactor Facility Manager. ALARA activities, for which record keeping is the particular responsibility of the Nuclear Reactor Facility Manager, are incumbent upon all radiation workers associated with the reactor facilities.

Substantive changes to the Radiation Protection program require approval of the Reactor Safeguards Committee. Changes approved by the Reactor Safeguards Committee for operating or emergency procedures apply automatically to the Radiation Protection Program and corresponding changes may be made without further consideration of the Reactor Safeguards Committee.

Changes approved automatically through approval of other procedures, editorial changes, or changes to appendices may be incorporated into the Radiation Protection Program on the authority of the Nuclear Reactor Facility Manager. These changes SHALL be processed through individual change pages identified with revision level and change date. An index of changes SHALL be maintained with a summary of the reason for the change, a summary of the change, and a copy of the superceded page. The Reactor Safeguards Committee SHALL review all changes implemented since the previous review.

The Reactor Supervisor or the Nuclear Reactor Facility Manager may deviate from elements of the Program on a temporary basis for reasons of facility or personnel safety; the deviation SHALL be brought promptly to the attention of the Reactor Safeguards Committee.

2.1 Radiation Units

The traditional units of Curie, rad, rem and roentgen are to be used in record keeping. SI units of Becquerel, gray and sievert may be used in calculations, DOSE assessments and reports, so long as final results, conclusions, etc. are given in traditional units as well.

EXTERNAL DOSE is to be recorded in terms of DEEP or SHALLOW DOSE EQUIVALENT (index). According to the ICRP [8], the DEEP DOSE EQUIVALENT (in rem units) is within 4% of the free-field exposure rate (in roentgen units) for gamma rays with energies between 0.6 and 8.0 MeV. Therefore, survey or area monitoring instruments calibrated in roentgen units may be used for assessment of DEEP DOSE EQUIVALENT from gamma radiation in routine surveillances. A neutron survey meter must be used to measure the dose equivalent from neutrons.

The TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) is the sum of the DEEP DOSE EQUIVALENT for external exposure and the COMMITTED DOSE EQUIVALENT for internal exposure. Internal exposure associated with the Reactor Facility has never been a source of significant radiation exposure to workers or MEMBERS OF THE PUBLIC. Should significant exposure be considered possible (such as in connection with planned special exposures or in the conduct of ALARA reviews), evaluation should follow the guidance of 10CFR20, Regulatory Guides [3-7] or the ICRP [9-11].

2.2 Radiation Limits

Occupational dose limits (except for planned special exposures, as described in Section 5.3), are given by 10CFR20.1201 as follows. Annual limits for adults, in summary, are the more limiting of the following:

RADIATION DOSE LIMITS		
POPULATION	EXPOSURE	LIMIT
Radiation Workers (OCCUPATIONAL EXPOSURE)	TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE)	5 rem in one year
	the lens of the eye	15 rem in one year
	SHALLOW DOSE EQUIVALENT to the skin or any extremity	50 rem in one year
	combined DEEP DOSE EQUIVALENT and COMMITTED DOSE EQUIVALENT to any organ other than the eye	50 rem in one year

RADIATION DOSE LIMITS		
POPULATION	EXPOSURE	LIMIT
MEMBER OF THE PUBLIC (NON OCCUPATIONAL EXPOSURE)	TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) in one year	0.1 rem in one year
	TEDE	0.002 rem in one hour

2.3 Radiological Event Reporting

NOTE: If reporting is required as indicated below, review the requirements of 10CFR30.50.

Notify the USNRC as soon as possible after the event is discovered, not to exceed 4 hours, as required by 10CFR30.50 if immediate protective action is required to avoid exposures to radiation or radioactive materials or releases of licensed material that could exceed Regulatory limits (events may include fires, explosions, toxic gas releases, etc.).

If any of the following conditions occur, notify the USNRC within 24 hours after the discovery of the incident as required by 10CRF30.50:

1. An unplanned contamination event that

    - Requires additional radiological controls or results in prohibited access for more than 24 hours

    - Involves a quantity of any isotope greater than five times the lowest annual limit on intake specified in Appendix B of § 20.1001-20.2401 of 10 CFR part 20 for the material

      AND

      Has the restrictions imposed as above, EXCEPT where the restrictions are in place only to permit isotopes with half life less than 24-hours to decay prior to decontamination.

2. Equipment is disabled or fails to function as designed when:

    - The equipment is required by regulation or license condition to prevent releases of radiological effluents exceeding regulatory limits, to prevent exposures to radiation and radioactive materials exceeding regulatory limits, or to mitigate the consequences of an accident;

    - The equipment is required to be available and operable when it is disabled or fails to function; and

    - No redundant equipment is available and operable to perform the required safety function.

3. An event that requires unplanned medical treatment at a medical facility of an individual with spreadable radioactive contamination on the individual's clothing or body.

4. An fire or explosion damaging any licensed material or any device, container, or equipment containing licensed material when the integrity of the licensed material or its container is affected and the quantity of material involved is greater than five times the lowest annual limit on intake specified in appendix B of §§ 20.1001-20.2401 of 10 CFR part 20 for the material

As required by 10CFR30.50, all reports shall include (if available):

- The name of the person making notification and call back number;
- A description of the event, including date and time;
- The exact location of the event;
- The isotopes, quantities, and chemical and physical form of the licensed material involved; and
- Any personnel radiation exposure data available.

Review 10CFR30.50 for any followup reporting requirements.

3. TRAINING

Implementation of training for radiation protection is the responsibility of the Nuclear Reactor Facility Manager. Training guidance, a syllabus, and a sample examination are provided in Appendix D. All persons granted unescorted access to the Reactor Facility must receive the training and must complete without assistance a written examination over radiation safety and emergency preparedness. An examination score of at least 70 percent is required.

Re-training for holders of unescorted access privileges must be administered biennially except for Reactor Operators and Senior Reactor Operators taking part in the annual Reactor Facility Requalification Program.

4. SURVEILLANCE AND MONITORING

The KSU Reactor Technical Specifications and the KSU Reactor Facility Emergency Plan independently impose other surveillance requirements related to radiation protection. Periodic surveillance requirements related to radiation protection and imposed only via the KSU Nuclear Reactor Radiation Protection Program by the Reactor Safeguards Committee are tabulated in table “*Radiation Protection Program – Periodic Surveillance Activities*.”

NOTE: Surveillances related to radiation protection and required in other formally approved documents are not specified herein, except by reference.

RADIATION PROTECTION PROGRAM - PERIODIC SURVIELLANCE ACTIVITIES	
FREQUENCY	SURVIELLANCE
Monthly	Wipe test reactor bay and control room

	Occupational Dose Record Review (when delivered)
Quarterly	Source inventory report
	Source inventory and leak test
	Special nuclear material inventory
	Emergency equipment inventory
	Review extremity monitoring report, when provided
Semi-annually	Environmental surveillance (radiation levels at full power)
	Radiation Protection Program Implementation Audit
Annually	Calibration of the pool surface monitor
	Calibration of the AMS II air monitor
	Special nuclear material reports
	Calibration of PIC SRPDs
	Evacuation alarm response test
Biennially	Radiation Protection Program review

4.1 Radioactive Materials Accountability

Radioactive materials accountability is assured by a quarterly inventory report, quarterly source inventory and leak test, and annual inventories of special nuclear materials.

The quarterly inventory report is initiated by request of the Radiation Safety Officer, and returned to the RSO to ensure the byproduct material on the Kansas State University campus meets LICENSE restrictions. The quarterly source inventory and leak test is a physical check of storage location and a leak test of all sources on inventory. Annual inventories of SPECIAL NUCLEAR MATERIALS include a report on the status of material leased from DOE, nuclear material transaction report indicating fuel burnup and other transfers of SNM, and inventory of SNM at the Facility. Facility MANAGEMENT and Facility Staff prepare the reports and submit to the University Radiation Safety Officer and the Department of Energy.

4.2 EFFLUENT MONITORING

Liquid EFFLUENT Surveillance

Radioactive liquid waste is collected in the reactor bay sump (typically condensate from the air handling unit, sometimes contaminated with low levels of tritium). The sump is batch-discharged to sewerage when water quality meets permitted discharge requirements.

MONITORING of liquid EFFLUENTS to sewerage assures compliance with 10CFR20.2003. Facility Procedures 19, 20, and 21 guide assays for radionuclides emitting gamma rays, beta particles, and alpha particles.

Gaseous EFFLUENTS

Per 10CFR20.1101, air EFFLUENTS are limited to prevent a member of the public from receiving in excess of 10 mrem per year due to licensed activities at the facility. Although normal, steady state operations are not capable of discharging effluent concentrations high enough to challenge this limit; an air monitor system was installed to sample air representative of reactor bay effluent stream. This monitor provides relative indication that conditions of air effluent are normal, and has an annual CALIBRATION requirement.

4.3 CONTAMINATION MONITORING and SURVEYS

MONITORING

At exits of known or suspect CONTAMINATION areas, personnel shall monitor at least hands and feet. Personnel shall also monitor any other body or clothing surface that made contact with a surface in the CONTAMINATION area. If CONTAMINATION on the body or clothing is detected, then a check of exposed areas of the body and clothing should be made. Materials, tools and equipment shall be monitored for CONTAMINATION before removal from contaminated or RESTRICTED AREAS likely to be contaminated.

SURVEYS

Wipe tests of the reactor bay and control room are required monthly. Alpha and beta particle assay for radionuclides is performed using Facility Procedures 20 and 21.

Limits for Removable and Fixed CONTAMINATION

The USNRC has provided guidance for evaluating potential radioactive contamination and determining appropriate methods of control for release of potentially contaminated material from restricted areas to unrestricted areas in the following documents:

- IE Circular No. 81-07: Control of Radioactively Contaminated Material Information Notice No. 89-35: Loss and Theft of Unsecured Licensed Material
- Information Notice 94-16: Recent Incidents Resulting in Offsite Contamination
- Information Notice No. 85-92: Surveys of Waste Before Disposal From Nuclear Reactor Facilities

Prior to removal from contaminated or likely contaminated areas, items, equipment and material SHALL be subject to a SURVEY for radioactive contamination over all surfaces, with hand-carried personal items subject to personal monitoring requirements. All waste SHALL be monitored prior to removal from the reactor bay for disposal.

Contamination monitoring using portable survey instruments or laboratory measurements should be performed with instrumentation and techniques (survey scanning speed, counting times, background radiation levels) necessary to detect 5000-dpm (decays per minute)/100 cm^2 total and 1000 dpm/100 cm^2 removable beta/gamma contamination. If alpha contamination is suspected, appropriate surveys and/or laboratory measurements capable of detecting 100-dpm/100 cm^2 fixed and 20 dpm/100 cm^2 removable alpha activity should be performed. Based on the studies of residual radioactivity limits for decommissioning (NUREG-06132 and NUREG-07073), surfaces uniformly contaminated at levels of 5000 dpm/100cm2 (beta-gamma activity from nuclear power reactors) would result in potential doses that total less than 5 mrem/yr.

Surveys conducted with portable survey instruments using pancake G.M. probes SHALL be performed:

- At probe distances approximately ½ inches from the surface
- At scan rates less than 2 inches per second.
- In an area with less than 300 cpm background counts.

Levels greater than 100 cpm above background require decontamination.

Surveys of very large items should not consist solely of portable survey instruments using pancake G.M. tubes because of the loss of detection sensitivity created by moving the probe and the difficulties in completely scanning large areas.

Measurements at appropriate access points to inaccessible surfaces (e.g., pipes, drain lines and duct work) may be used to evaluate contamination of enclosed surfaces, provided contamination levels at accessible locations can be demonstrated to represent the potential contamination at the inaccessible surfaces. Such measurements should be supplemented with a measurement using a sodium iodide detector or other gamma spectrometer to ensure that the interior surfaces of the object are not contaminated with high concentrations of specific isotopes. The material should not be released for unrestricted use if is shown to be contaminated with one or more isotopes at concentrations exceeding the limits specified in 10CFR-20, Appendix D. These guidelines for radiation surveillance of materials prior to release for unrestricted use are documented in procedural form in Reactor Management Order SOR-5.

SURVEYS SHALL be conducted for aggregate material (e.g., bag, drum or pallet) after collection of all material and prior to final removal to ensure there has hot been an accumulation of licensed material resulting from a buildup of multiple quantities below detectable levels (e.g., final measurements using sensitive scintillation detectors in low-background areas). These surveys shall be conducted using a gamma spectrometer to ensure that the aggregate material does not contain specific radionuclides in concentrations exceeding the limits specified in 10CFR-20, App. D.

If activities require that licensed materials be used or stored in unrestricted areas, personnel SHALL maintain immediate control and constant surveillance of the materials, or secure the materials against unauthorized removal.

4.4 Environs MONITORING

Environs MONITORING is required to assure compliance with 10CFR20, Subpart F (SURVEYS and MONITORING), specific operating requirements, CALIBRATION frequency, and set point verification within the Technical Specifications for the FACILITY OPERATING LICENSE including:

a. Technical Specifications, Section C. Reactor Pool requires:

Pool surface monitor

b. Technical Specifications, Section F. Radiation Monitoring requires:

Area radiation monitor located on or near the pool bridge

Area radiation monitors in the reactor bay

Continuous air monitor

Additional MONITORING imposed by the Reactor Safeguards Committee is as follows:

a. An evacuation alarm (high radiation level) is required at the 22-ft level of the reactor. Response testing of the alarm is performed annually following Facility Procedure 18.

b. Semi-annual environmental MONITORING, involving measurement of both gamma-ray and neutron DOSE rates at the Facility operations boundary with the reactor at full-power operation

c. When shielding is changed from normal configuration:

(1) MONITORING for potential neutron and gamma exposures is required at the AREA OF INTEREST under the following conditions:

- Each time a new, previously untested, shielding configuration is established or a tested configuration is modified
- During initial operation at significantly higher power (i.e., at least 10% higher) than previous MONITORING

(2) SURVEYS of the area affected by the shielding change are required IF personnel will have access to the area.

(3) MONITORING is not required for a well-defined shielding configuration that previously met radiological LIMITS as demonstrated by MONITORING (or on restoration to normal shielding), but may be performed at the discretion of the Operator at the Controls.

(4) During operations following shielding changes, the operator at the controls should be aware of and attentive to area radiation monitor indications as potential indicators of unanticipated anomalies in shielding effectiveness.

4.5 Personnel Exposure

INTERNAL DOSE MONITORING is required only for (1) adults likely to receive in one year in excess of 10% of the applicable ANNUAL LIMIT ON INTAKE for ingestion and inhalation, or (2) minors or DECLARED PREGNANT WOMEN likely to receive in excess of 0.05 rem COMMITTED EFFECTIVE DOSE in one year. The KSU Nuclear Reactor Facility does not have potential for exceeding a DOSE that could require INTERNAL DOSE MONITORING.

Regulation 10CFR-20.1502 requires MONITORING of workers likely to receive, in one year from sources external to the body, a DOSE in excess of 10 percent of the limits given in Section 2.2 of this program, and Individuals entering a HIGH or VERY HIGH RADIATION AREA.

According to Regulatory Guide 8.7 [2], if a prospective evaluation of likely DOSES indicates that an individual is not likely to exceed 10 percent of any applicable DOSE LIMIT, then there are no requirements for record keeping or reporting. Likewise, Regulatory Guide 8.34 [3] indicates that, if INDIVIDUAL MONITORING results serve as confirmatory measures, but INDIVIDUAL MONITORING is not required by 10CFR20.1502, then such results are not subject to the record keeping requirements of 10CFR20.2106(a) even though they may be used to satisfy 10CFR20.1501 requirements. The regulation also requires MONITORING of any individuals entering a HIGH RADIATION AREA, i.e., areas accessible to major portions of the whole body within which an individual could receive a TOTAL EFFECTIVE DOSE EQUIVALENT of 0.1 rem in one hour.

As shown in Appendix E, which lists OCCUPATIONAL DOSES for a period of 12 years, there have been no instances of any OCCUPATIONAL DOSE in excess of 10 percent of the above limits at the KSU Reactor Facility. Thus, retrospectively, only confirmatory MONITORING would be required and 10CFR20.2106(a) record keeping requirements would not apply, so long as there are no significant changes in the Facility operating procedures or occupational expectations. If, in the view of supervisory personnel (Reactor Supervisor, Facility Manager, or Radiation Safety Officer), any action under consideration might lead to DOSE in excess of 10 percent of any applicable limit, then the ALARA program is triggered. A consequence of ALARA program planning, which is described in Section 6, might be the imposition of federally required record keeping procedures.

MONITORING of workers and MEMBERS OF THE PUBLIC for RADIATION EXPOSURE is required by the Reactor Safeguards Committee and is described in Facility Procedure 9. Objectives implemented through Procedure 9 to ensure control of personnel RADIATION EXPOSURE include:

a. Personnel who enter the control room or the reactor bay will either hold authorization for unescorted access, or be under direct supervision of an escort (i.e., escorted individuals can be observed by and hear instructions of the escort) who holds authorization for unescorted access.

b. When the reactor is not secured, the licensed reactor operator (or senior reactor operator) at the controls SHALL be responsible for controlling access to the control room and the reactor bay.

c. Personnel who enter the reactor bay during reactor operation SHALL have a record of accumulated DOSE measured by a gamma sensitive INDIVIDUAL MONITORING DEVICE; at the discretion of the reactor operator at the controls, a single INDIVIDUAL MONITORING DEVICE may be used for INDIVIDUAL MONITORING of two people who agree to stand together in the reactor bay.

d. If there is potential for EXPOSURE of personnel to neutrons within the reactor bay, personnel who enter the reactor bay SHALL have neutron sensitive INDIVIDUAL MONITORING, assigned only to individuals, or SHALL continuously monitor neutron dose rates with portable survey instrumentation.

e. Personnel who enter the reactor bay while the reactor is secured SHALL have a record of accumulated DOSE either by measurement through INDIVIDUAL MONITORING or based on assessment of data from INDIVIDUAL MONITORING DEVICES or SURVEY.

f. It there is potential for extremity shallow dose in excess of 10% of the annual limit set forth in 10CFR-20, personnel SHALL be assigned extremity (e.g., ring) dosimeters in addition to other required dosimetry.

The Radiation Safety Officer distributes records of INDIVIDUAL MONITORING DEVICES used to record OCCUPATIONAL DOSE monthly for whole body monitors and quarterly for extremity monitors. These records are reviewed as specified in Section 8, *Reviews and Audits,* and posted so that individuals may be kept aware of their OCCUPATIONAL DOSE.

5. RECORD KEEPING

5.1 Administrative Records

Personnel EXPOSURE Records

The Facility is exempt from Federal record keeping requirements (see Section 4.5), of 10CFR20.2106(a) as long as OCCUPATIONAL DOSES and PUBLIC DOSES are controlled to less than 10% of the limiting personnel DOSE (previously noted) and as long as personnel do not enter HIGH or VERY HIGH RADIAITON AREAS. However, certain records are required to confirm that personnel exposures are less than 10 percent of applicable limits.

Records of Prior OCCUPATIONAL EXPOSURE are initially obtained, then maintained, by the Office of Radiation Safety. A sample form (NRC Form 4) is provided in Appendix F.

Training and Qualification Records

Unescorted Access Records are maintained at the Facility. A list of persons with unescorted access will be maintained on file. Results of unescorted access training examinations SHALL be maintained on file for at least 3 years. A review and assessment of persons with unescorted access, and copies of notification of individuals requiring retraining SHALL be recorded with the semi-annual radiation protection program audit.

Radiation Protection Program Review and Audit Records

Monthly Reviews of Personnel EXPOSURE Records are recorded by completion in the Maintenance and Surveillance Report. Reports not delivered to schedule will be reviewed on receipt. If investigation of cause and circumstances is required based on OCCUPATIONAL DOSE exceeding 1/2 the annual ALARA limit, the report SHALL be submitted to the RSC and file copy maintained in the RSC Notebook.

Radiation Protection Program Semi-annual Audits of implementation (Appendix F, Illustration F-16) SHALL be submitted to the RSC.

Biennial Review of the Radiation Protection Program provisions SHALL be submitted to the RSC.

5.2 Routine Operational Records

Personnel Exposure Records

Records of Occupational INDIVIDUAL MONITORING are maintained by the Office of Radiation Safety. Illustrated in Appendix F is a sample form (NRC Form 5) and samples of forms in use, namely, monthly report for the University as a whole, monthly summary report for the Nuclear Reactor Facility, and quarterly report on EXTREMITY DOSES for the University as a whole.

Records of DOSES to Individual MEMBERS OF THE PUBLIC are maintained in dosimeter records maintained at the Facility. Self-reading and electronic pocket dosimeter records are kept in a logbook. Such records are kept permanently. A sample page is illustrated

Radioactive Material Accountability

Radioactive Source Material Inventory is conducted for the Office of Radiation Protection.

Source Inventory and Leak Check is conducted to control inventory and integrity of radioactive material associated with the Facility. Records are maintained at the Facility.

Special Nuclear Material Records is conducted as required by the Department of Energy and the Nuclear Regulatory Commission. Records are maintained by DOE, NRC and at the Facility.

Survey Instrument and Self-Reading Personal Dosimeter CALIBRATIONCALIBRATION of these instruments is performed according to Procedures 13 and 14. Separate CALIBRATION records are kept for each instrument, and for 3 years at the Facility. Sample records are included in Appendix F.

Environs MONITORING

Monthly swipe SURVEYS and water sample tests are performed according to Procedure 20. Records are kept on file in the Reactor Facility for 3 years. Semi-annual SURVEYS of gamma ray and neutron DOSE RATES are required along the operations boundary with the reactor at full power. Sample records are included in Appendix F.

The results of special, non-scheduled SURVEYS at LOCATIONS OF INTEREST conducted to verify the adequacy of shielding installations are recorded in the Operations Log.

Waste Disposal

When liquid wastes are released from the Reactor Facility to sanitary sewerage, both gamma ray and alpha-particle assay are required to assure compliance with 10CFR20. Assay records and records of releases are kept on file in the Reactor Facility for 3 years. Sample records are included in Appendix F.

Records of the transfer wastes from the Reactor FACILITY are kept on file in the Reactor FACILITY for 3 years. Procedure 22 may be followed in estimation of activities transferred. At the discretion of the Reactor Supervisor, a detailed report of estimated activities may be filed with the transfer records. Examples of such records and such a report are included in Appendix F.

Records are not required for effluent concentrations when discharging reactor bay HVAC condensate to sanitary. Years' worth of assays have shown that the concentration of radionuclides in the HVAC condensate is at background level except for levels of tritium far below release limits. HVAC may be discharged to sanitary without assay EXCEPT following emergencies involving airborne radionuclides, which may contaminate the airborne moisture collected in the condensate tank.

Emergency Equipment Inventories are maintained according to requirements in the *KSU Reactor Emergency Plan*.

5.3 Planned Special Exposures

10CFR20.106 allows ADULT workers (excluding DECLARED PREGNANT WOMEN) to receive DOSES above 10CFR20.101 limits under special circumstances, when the following conditions are satisfied:

a. Alternatives to higher exposure are unavailable or impractical

b. Exposures are pre-authorized, in writing

c. Individuals involved are informed of risks and instructed in procedures

d. Individual's DOSES in excess of annual DOSE LIMITS (and from prior special exposures) are known

e. Special exposures and marginal occupational exposures over annual limits do not exceed 10CFR20.1201 limits in any one year

f. Special exposures and marginal occupational exposures (i.e., exposures above an annual limit) do not exceed 5 times the annual DOSE LIMITS specified in 10CFR20.1201 DOSE LIMITS over the course of a lifetime

g. Records are maintained and submitted to the NRC according to 10CFR20.1201 and 10CFR20.1206

h. The exposed individual is informed of the dose received during the planned special exposure within 30 days, per 10CFR-20.1206.

Any planned special exposures must receive full ALARA consideration. Documents related to planned special exposures, including measurements and calculations used to assess INTERNAL DOSES SHALL be kept permanently at the Reactor Facility.

6. ALARA PROGRAM

SUMMARY OF ALARA GOALS		
Applies to:	10CFR20 Annual Limit	ALARA Goal (annual)
Workers	5 rem TEDE	< 500 mrem annual TEDE
	50 rem combined deep dose equivalent (DDE) and committed dose equivalent (CDE) to any organ other than the eye	< 5 rem annual DOSE EQUIVALENT to any organ except the lens of the eye
	15 rem lens of the eye	< 1.5 rem annual DOSE EQUIVALENT to the lens of the eye
	50 rem SHALLOW DOSE EQUIVALENT to the skin or any extremity	< 5 rem annual DOSE EQUIVALENT to the skin
	100 mrem TEDE for DECLARED PREGNANT WOMAN workers	< 50 mrem DOSE EQUIVALENT to the fetus during pregnancy
MEMBER OF THE PUBLIC	100 mrem TEDE	< 50 mrem annual TEDE

6.1 Policies and Objectives

MANAGEMENT of the Reactor Facility is committed to keeping both OCCUPATIONAL WORKERS and MEMBERS OF THE PUBLIC radiation exposure AS LOW AS REASONABLY ACHIEVABLE (ALARA). The specific goal of the ALARA program is to assure that actual exposures result in DOSES no greater than 10 percent of the occupational limits and no greater than 50 percent of the MEMBER OF THE PUBLIC limits prescribed by 10CFR20

6.2 Implementation of the ALARA Program

Planning and scheduling of operations and experiments, education and training, and facility design are the responsibilities of the Reactor Supervisor and/or the Nuclear Reactor Facility Manager. Any action that, in either of their opinions, might result in personnel exposure to one-half the annual ALARA DOSE goal (Section 6.1) to any one individual in one calendar quarter requires a formal ALARA review and report. Any staff member or experimenter, or any member of the Reactor Safeguards Committee may call for an ALARA review of a proposed action. Under any of these circumstances, it is the responsibility of the Reactor Supervisor to conduct an ALARA review and report. The report shall be issued to the Reactor Safeguards Committee and the University Radiation Safety Committee. Only with the approval of the Reactor Supervisor and the endorsement of the Nuclear Reactor Facility Manager may the action proceed.

6.3 Elements of the ALARA Review and Report

The following topics SHALL be considered, if applicable. The report SHALL include discussion of how these topics affect personnel exposure and specific actions recommended, categorized by topic:

Features for External Radiation Control

Shielding and construction materials
Radioactive material storage and disposal
MONITORING systems
Facility layout
Control of access to HIGH and VERY HIGH RADIATION AREAS

CONTAMINATION Control

Ventilation and filtration
Containment of CONTAMINATION
Confinement of CONTAMINATION spread
Construction materials to facilitate decontamination
Facility layout

EFFLUENT Control

Gaseous EFFLUENTS
Liquid EFFLUENTS
EFFLUENT MONITORING

Operations and Operations Planning

Assessment of potential individual and collective exposures
Application of *shielding, time, and distance* for DOSE reduction
Use of ventilation and decontamination to reduce COLLECTIVE DOSE
Provision of special radiation survey or communications instrumentation
Provision of special personnel training and practice
Provision of special supervision and surveillance
Provision of special clothing or other protective gear

6.4 Reviews and Audit

The ALARA Program SHALL be audited by the Nuclear Reactor Facility Manager integral to the general audit of the Radiation Protection Program.

7. CALIBRATIONS AND QUALITY ASSURANCE

CALIBRATION requirements related to radiation protection and imposed by the Reactor Safeguards Committee are as follows:

Semi-annually	Survey meters Pocket dosimeters
Annually	Continuous air monitor Neutron "rem" meters Alpha & beta particle efficiencies for surveillance probes Self-reading pocket ion chambers

CALIBRATION procedures are prescribed in the following Facility Procedures:

No. 3	Annual Remote Air Monitor Calibration
No. 8	(Continuous) Air Monitor Calibration
No. 13	Portable Radiation SURVEY Meter Calibration
No. 14	Personnel Pocket Dosimeter Calibration
No. 19	Gamma-Ray Assay of Reactor Samples
No. 20	Liquid Scintillator Assay Methods
No. 21	Alpha-particle Assay of Reactor Liquids

8. REVIEW AND AUDIT

8.1 Occupational Dose Record Reviews

The Reactor Supervisor SHALL review personnel DOSE records monthly. If personnel DOSES exceed 1/2 annual ALARA limit, causes and circumstances SHALL be investigated and reported to the Nuclear Reactor Facility Manager. The report SHALL be reviewed and submitted to the RSC.

SUMMARY OF DOSE LIMITS, GOALS AND LEVELS FOR INVESTIGATION			
Applies to:	10CFR20 Annual Limit	ALARA Goal (annual)	Investigation Trigger (quarter)
Workers	5 rem TEDE	500 mrem annual TEDE	250 mrem
	50 rem combined DDE & CDE to any organ other than the eye	5 rem DOSE EQUIVALENT to any organ except the lens of the eye	2.5 rem DOSE EQUIVALENT to any organ except the lens of the eye
	15 rem lens of the eye	1.5 rem DOSE EQUIVALENT to the lens of the eye	0.75 rem DOSE EQUIVALENT to the lens of the eye
	50 rem SHALLOW DOSE EQUIVALENT to the skin or any extremity	5 rem l DOSE EQUIVALENT to the skin	2.5 rem DOSE EQUIVALENT to the skin
	100 mrem TEDE for DECLARED PREGNANT WOMAN workers	50 mrem DOSE EQUIVALENT to the fetus during pregnancy	25 mrem DOSE EQUIVALENT to the fetus during pregnancy
MEMBER OF THE PUBLIC	100 mrem TEDE	50 mrem annual TEDE	25 mrem TEDE

8.2 Radiation Protection Program Implementation Audits

The Nuclear Reactor Facility Manager SHALL review implementation of the KSU Nuclear Reactor Radiation Protection Program semi-annually. As a minimum, the Nuclear Reactor Facility Manager SHALL review (1) instrument CALIBRAITONS and surveillance performance and record keeping (2) results of INDIVIDUAL MONITORING and record keeping; and (3) planned special exposures and ALARA reviews. Appendix F, Illustration F-16 provides guidance for performing the audit. The Nuclear Reactor Facility Manager may delegate the audit, but must review and sign the audit record to verify that all of the necessary items were covered.

8.3 Radiation Protection Program Reviews

The Nuclear Reactor Facility Manager SHALL review the Radiation Protection Program provisions biennially. As a minimum, the Nuclear Reactor Facility Manager shall review the Radiation Protection Program, 10CFR20, and Facility implementing procedures.

9. EMERGENCY EQUIPMENT

Equipment and supplies required to support emergency operations are identified in the *KSU Nuclear Reactor Emergency Plan*. An inventory of equipment in two storage lockers is conducted in accordance with the Plan to ensure readiness at all times.

10. REFERENCES

1. American National Standard *Radiation Protection at Research Facilities*, ANSI/ANS-15.11 (Final Draft), American Nuclear Society, La Grange Park, Illinois, October 1992.

2. *Instructions for Recording and Reporting Occupational Radiation Exposure Data*, Regulatory Guide 8.7 (Rev. 1), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

3. Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, Regulatory Guide 8.34, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

4. A*ir Sampling in the Workplace*, Regulatory Guide 8.25 (Rev. 1), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

5. *Planned Special Exposures*, Regulatory Guide 8.35, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

6. *Radiation Dose to the Embryo/Fetus*, Regulatory Guide 8.36, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

7. *Interpretation of Bioassay Measurements*, Draft Regulatory Guide 8.9 (DG-8009), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.

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Emergency Plan

1. INTRODUCTION

1.1. Reactor Facility Description

The Kansas State University (K-State) Nuclear Reactor Facility houses a General Atomics TRIGA Mk. II Nuclear Reactor (Reactor). The Reactor was obtained in 1960-1962 through a grant from the United States Atomic Energy Commission and is currently operated under Nuclear Regulatory Commission (NRC) License R-88 and the regulations of Section 1, Title 10, Code of Federal Regulations (CFR). Chartered functions of the Nuclear Reactor Facility are to serve as: 1) an educational facility for all students at K-State and nearby universities and colleges, 2) an irradiation facility for researchers at K-State and for others in the central United States, 3) a facility for training nuclear reactor operators, and 4) a demonstration facility to increase public understanding of nuclear energy and nuclear reactor systems.

The Reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. The current facility license (approved March 2008) permits steady state operation at thermal powers up to 1,250 kW with pulsing to $3.00 reactivity insertion.

The Reactor is a water-moderated, natural convection water-cooled thermal reactor operated in an open pool and fueled with heterogeneous elements consisting of nominally 20 percent enriched uranium in a zirconium hydride matrix and clad with stainless steel.

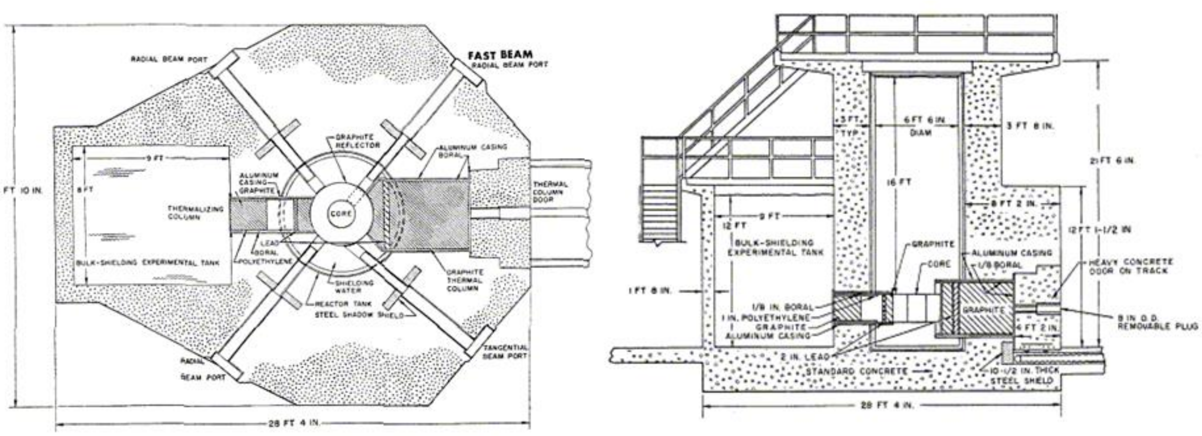

Figure EP. 1 - Sectional views of the K-State TRIGA reactor

Nuclear reactors of the TRIGA type are inherently safe. No credible chain of events can lead to melting of the fuel and gross releases of radioactivity. In normal operation, the Reactor is in its most sensitive configuration. Any mechanical disturbance or disruption of the Reactor would physically prevent its continued operation. The fuel is designed in such a way that causes large power increases to be automatically reduced to a safe level without human intervention.

Because of the inherent characteristics of the Reactor, the design of the Reactor Facility, administrative controls within the University and by the NRC, no event is ever expected to occur that would impact the health and safety of the public or disrupt domestic, education, agricultural, or commercial activities outside the Reactor Facility. While off-site interventions, or emergencies leading to off-site effects, are never expected, this Emergency Plan nevertheless makes provisions for dealing with such situations and is to be used as a plan of action to follow in the event of a radiological accident.

1.2 Emergency Plan and Procedures Objectives

This Emergency Plan and its associated Procedures are meant to prepare for and assist the Reactor Staff and outside agencies during abnormal circumstances that may lead to an emergency. The Plan is designed to establish a means of organizing what the different agencies are responsible for and how they should be contacted should the Plan be activated. It outlines a basic structure for identifying a situation and the necessary steps to return the Reactor Facility to normal status. Requirements for maintaining emergency preparedness are also defined in the Plan.
The Emergency Plan Procedures are to be held within the definitions of the Plan and are designed to assist persons in dealing with the present situation quickly and effectively. Deviations from the Procedures may be required since not all emergency situations can be foreseen and therefore planned for.

1.3 Reactor Facility Owner/Operator

The Reactor is owned by Kansas State University and is operated by the Department of Mechanical and Nuclear Engineering.

1.4 Reactor Facility Location and Maps

The Reactor is located on the campus of Kansas State University, in the City of Manhattan, in Riley County, Kansas. It is located in the north wing of Ward Hall, which faces onto 17th St., about 180 meters south of Claflin Rd. Figure EP. 2 shows the location of Riley County in relation to the other counties in Kansas. Figure EP. 3 shows road maps into and through Manhattan with specific driving directions from neighboring cities listed below. Figure EP. 4 shows a map and an isometric layout of the K-State campus.

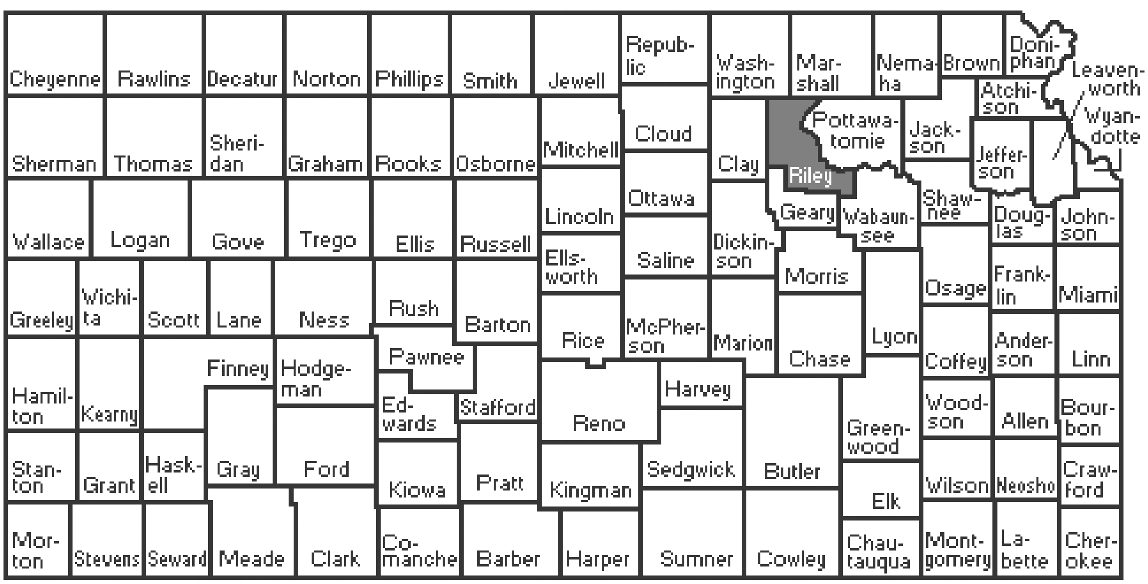

Figure EP. 2 - Riley County in relation to other counties in Kansas.

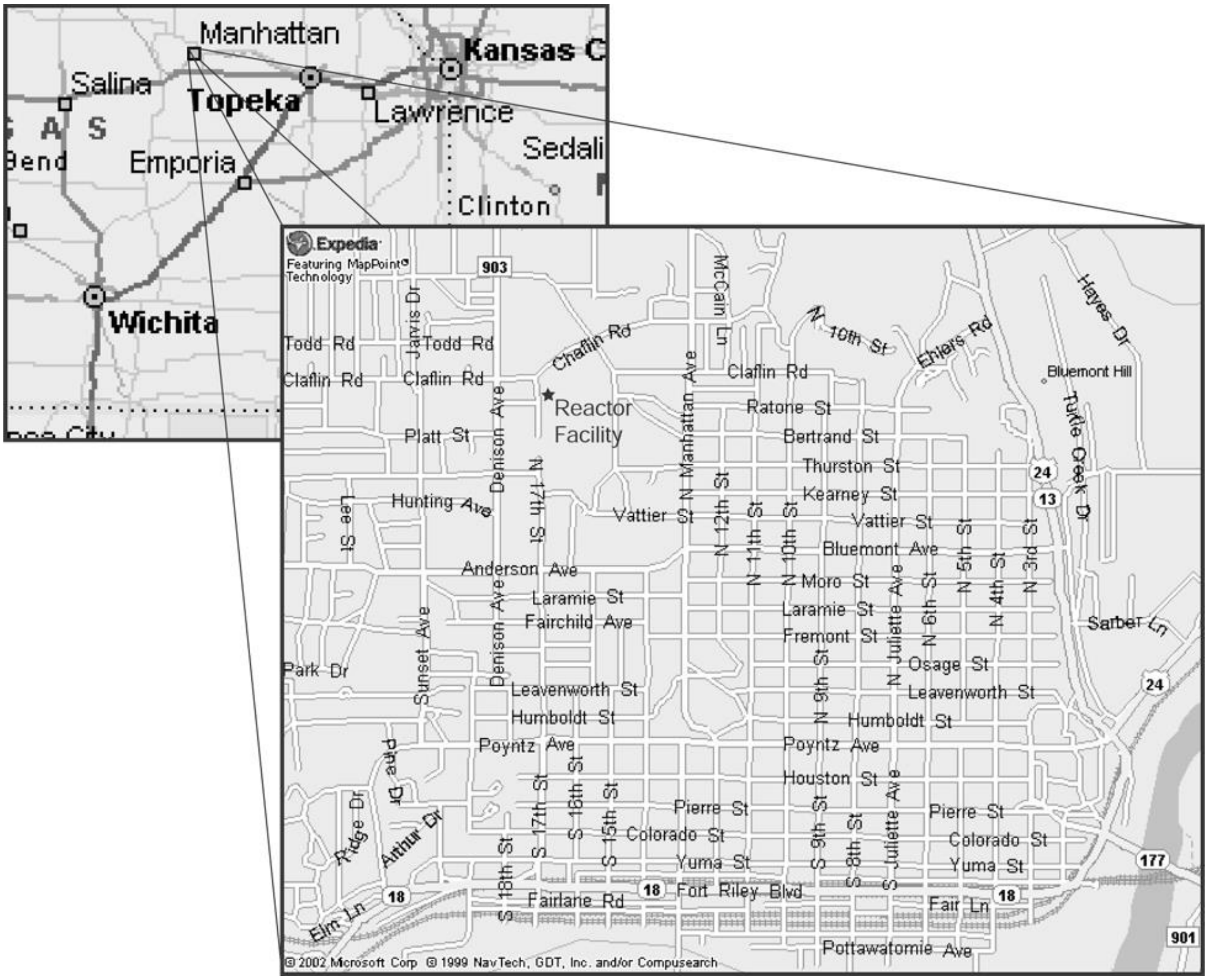

Figure EP. 3 - Maps showing access routes to Manhattan, Kansas and the Reactor Facility.

Directions from Kansas City, Lawrence, Topeka:

- Take I-70 West to Exit 313
- Go North on Highway 177 (stay in Right lane)
- Turn Right onto Fort Riley Blvd. (Highway 18)
- Turn Right onto 17th St.
- Turn Left onto Anderson Ave.
- Turn Right onto Denison Ave.
- Turn Right into the first large parking lot

Directions from Salina, Fort Riley:

- Take I-70 East to Exit 303
- Go North on Highway 18 (turns into Fort Riley Blvd.)
- Turn Left onto 17th St.
- Turn Left onto Anderson Ave.
- Turn Right onto Denison Ave.
- Turn Right into the first large parking lot

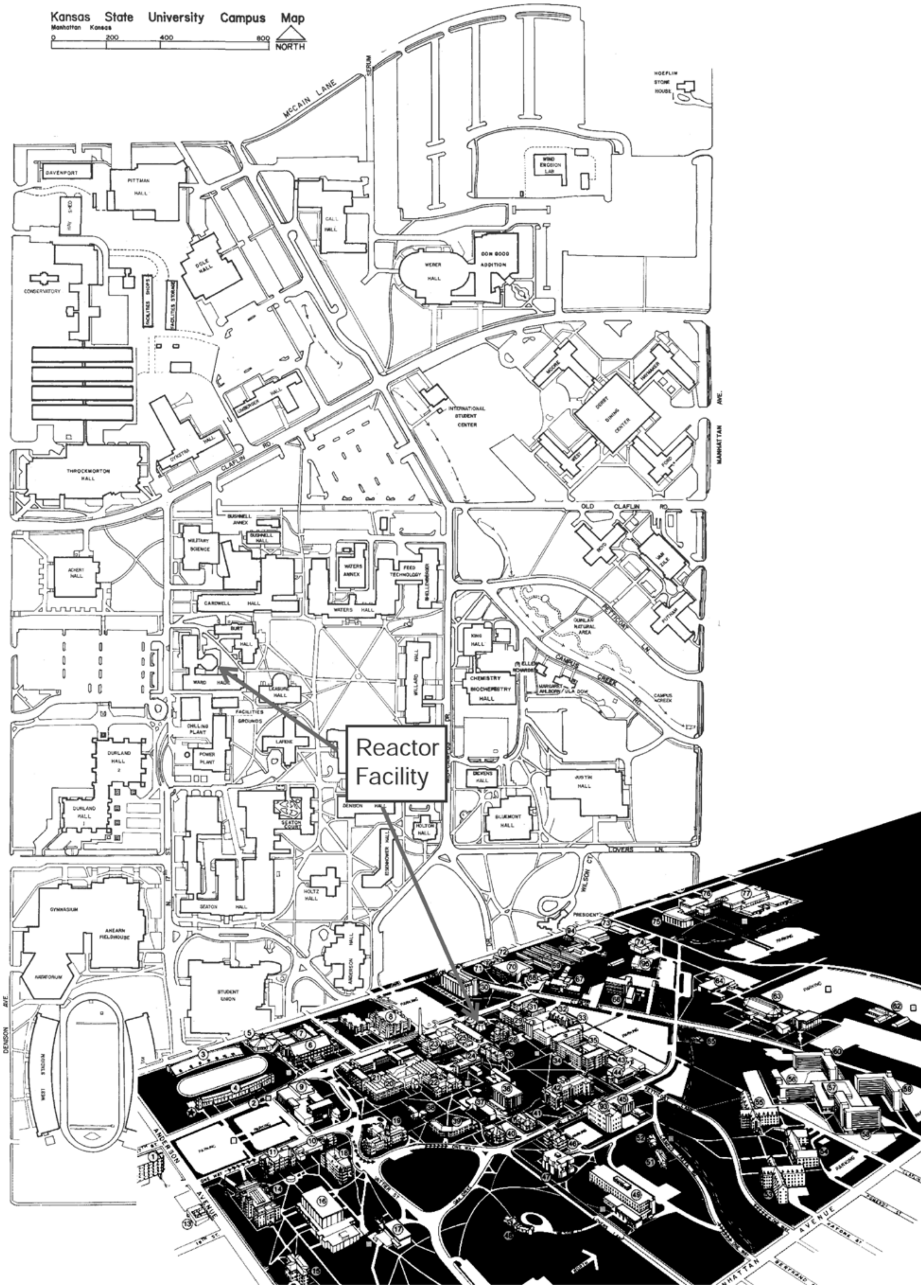

Figure EP. 4 - Map and Isometric Layout of the Kansas State University campus.

1.5 Reactor Facility Layout

Ward Hall is home to several groups not associated with the Reactor or even Nuclear Engineering. The following three maps are provided to assist personnel not familiar with the building. Items such as fire extinguishers and electrical panels are shown to assist fire fighters in the absence of Reactor Staff. Table 1.1 gives a brief description of what each room in Ward Hall is used for along with the potential hazards that may exist in that room.

Table 1.1 Ward Hall room descriptions.

Room	Description	Hazard
101-102	Office	
102A	Storage	
103-105	Office	
106	Women's Restroom	
107	Men's Restroom	
108	Mechanical HVAC	Machinery
109	Reactor Control Room	
110	Reactor Bay	Radiation, 4' Pit, Gas Bottles
111-112	Office	
113	Computer Lab	
114	Computer Admin Storage	
115-128	Office	
129	Mechanical HVAC	Machinery
130	Electronics Shop	
131	Machine Shop	Gas Bottles, Machinery
131A	Office	
131B	Machine Shop	Machinery
132	Electrical Closet	Electrical Panels
133-133D	Office	
134	Janitor's Closet	Chemicals
135	Classroom	
136	Storage	
137-137F	Office	
138	Lab	
139-139C	Laser Lab	Flammable Gases, Lasers
140-142	Lab	Flammable Gases, High Voltage, Radiation

Room	Description	Hazard
001	SMART Lab	Storage, 25' pit
001A	SMART Lab	Radiation, high temperature furnaces, compressed gas
002	Panoramic Irradiator and radiation detector research facility	Radiation
003	General nuclear research lab	4' Deep Pit
004	Mechanical HVAC	Machinery
005-005A	Equipment Storage and machine shop	Machinery, flammable material
006	Extension Storage	
007	Bathroom/Shower	
008	Janitor's Closet	Chemicals
009	Lab	Gas Bottles
009A	Storage, Tunnel Access	Utility Tunnel
010-010A	SMART Lab	Chemicals, High temperature furnaces
011	NAA Lab	Radiation, cryogenic liquid
011A	Source Storage	Radiation
012	Wind/Water Tunnel	
013	Dark Room	Radiation, Chemicals
014	Mechanical HVAC (Dungeon)	Machinery, Electrical Panels
015	SMART Lab	Chemicals, compressed gas
016	SMART Lab	Chemicals, compressed gas
110	Reactor Bay	Radiation, 4' Pit, Gas Bottles
Outside	Fenced Area	High Voltage
Outside	South wing, NE	Cryogenic liquid

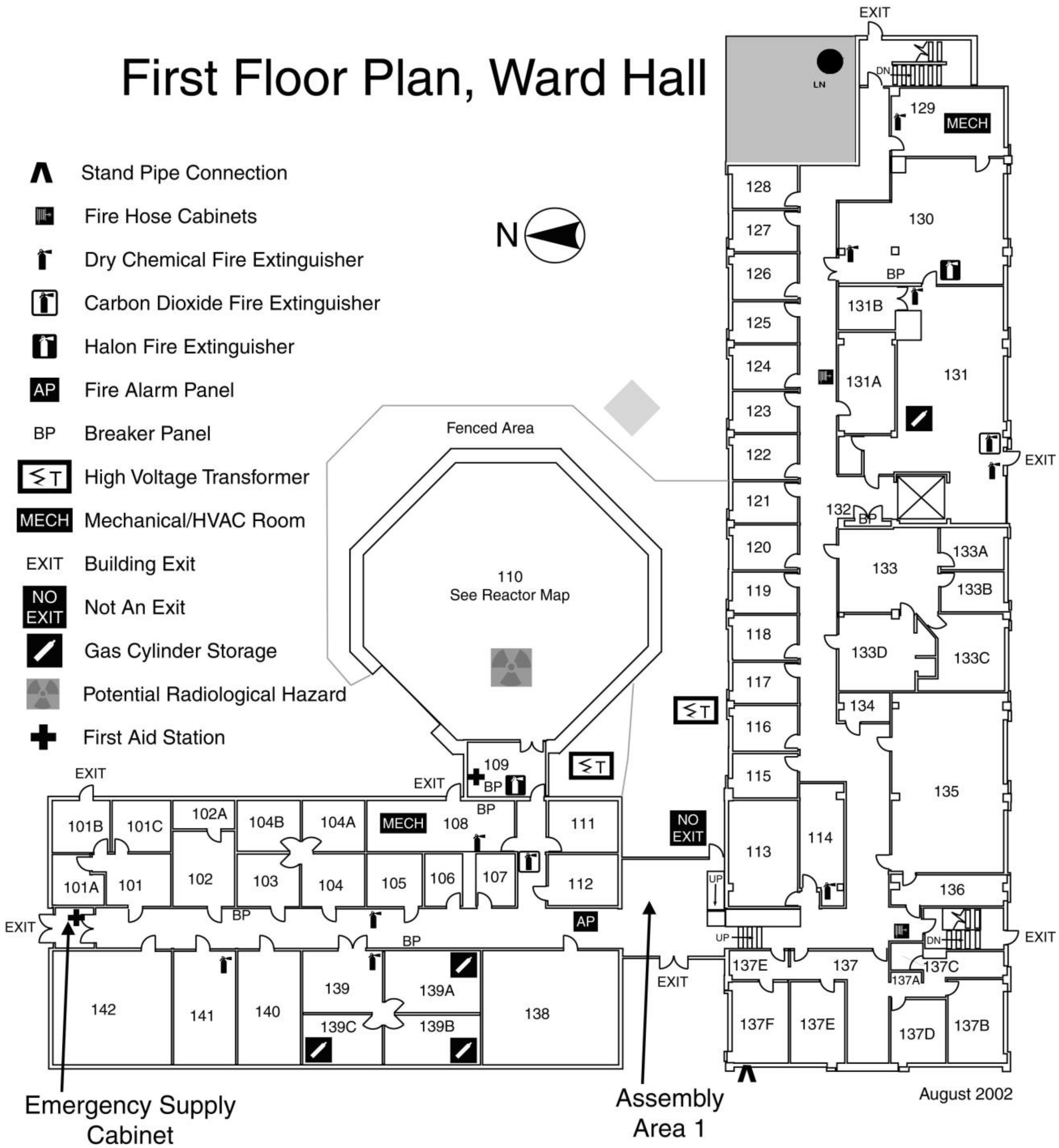

Figure EP. 5 - First floor level of Ward Hall and the Reactor Facility.

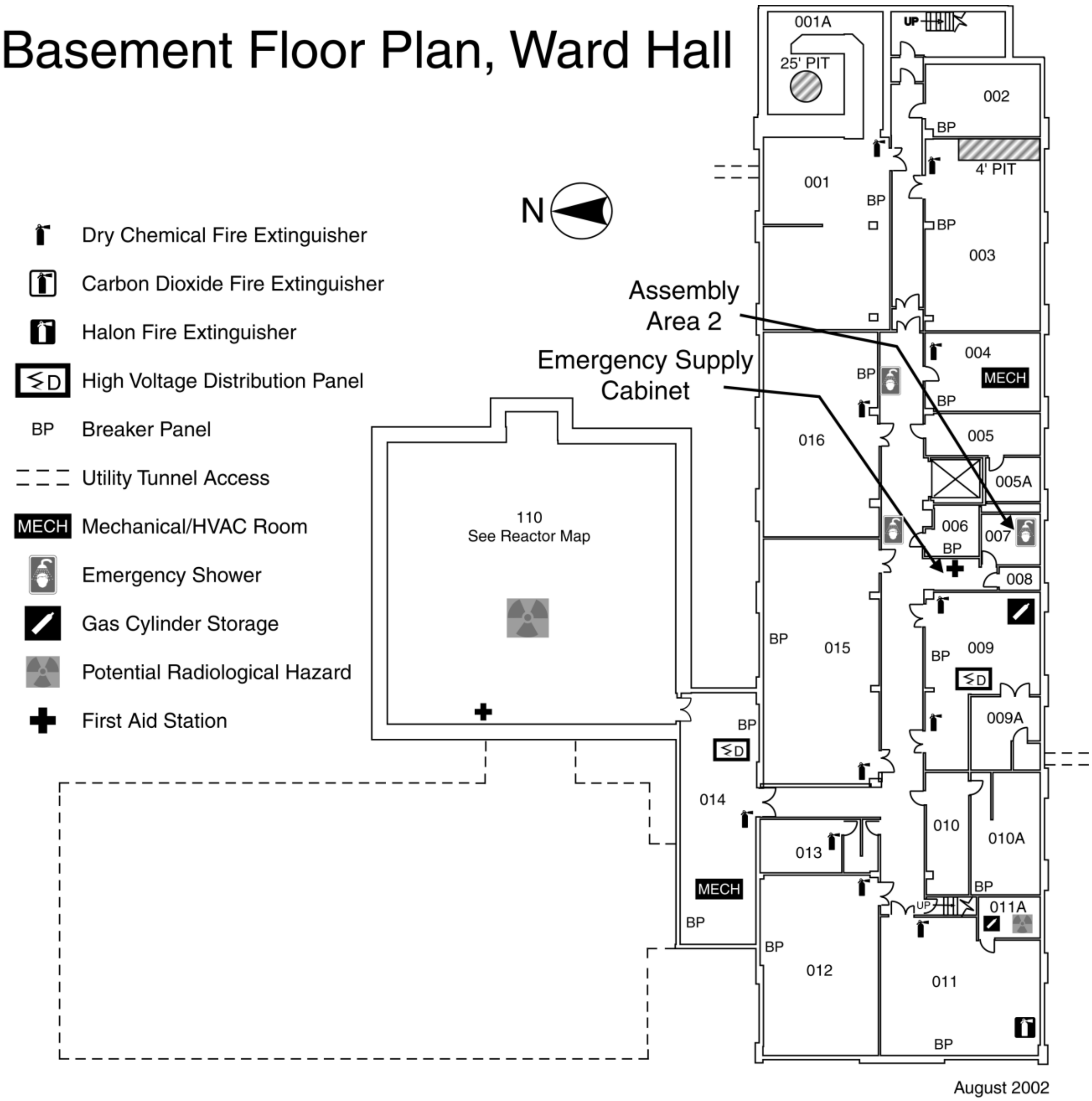

Figure EP. 6 - Basement level of Ward Hall and the Reactor Facility.

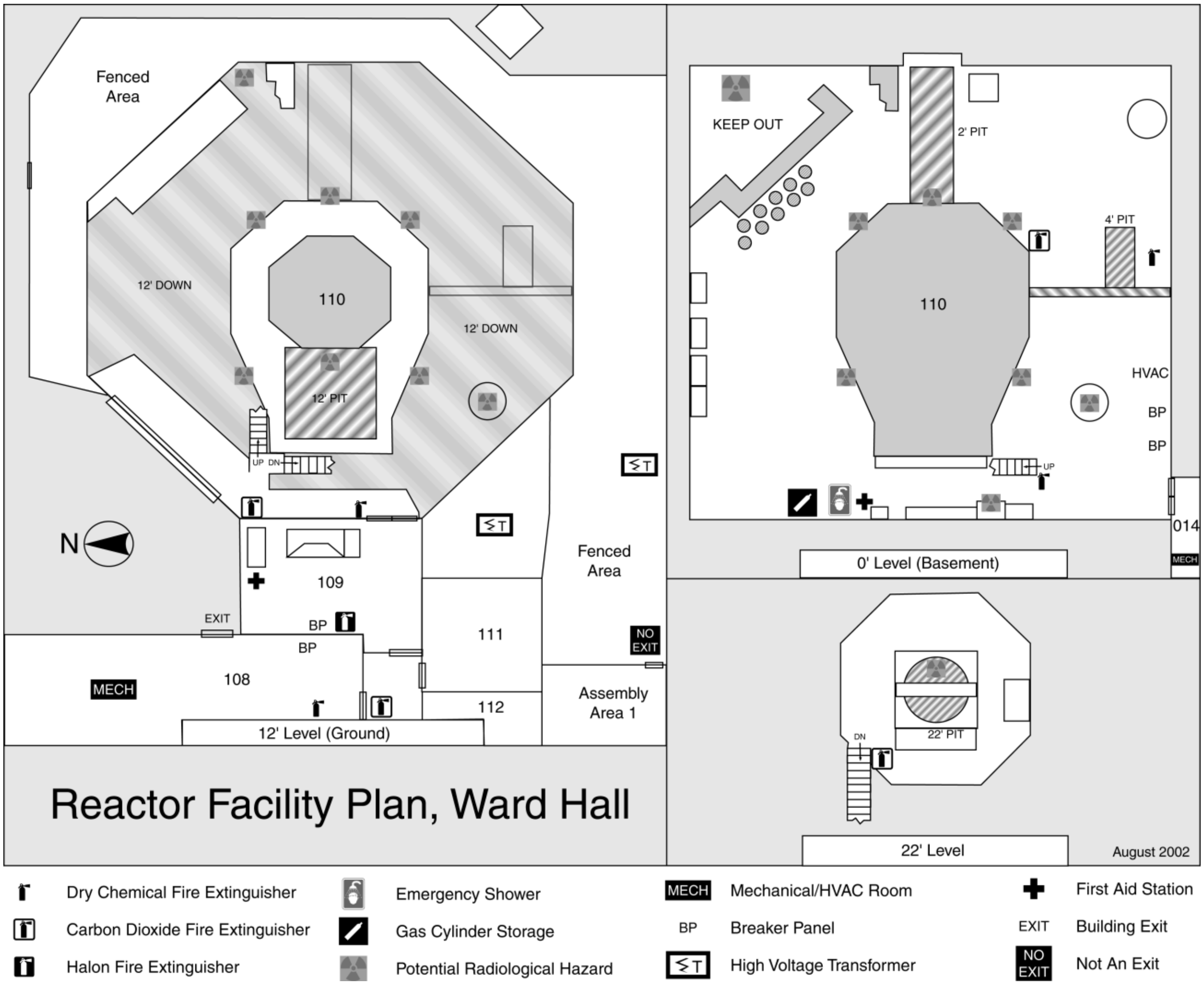

Figure EP. 7 -Detailed map of Reactor Facility.

<u>1.5.1 Boundary Definitions</u>

- Operations Boundary is defined to be Room 109 (Control Room) and Room 110 (Reactor Bay).
- Site Boundary encompasses all of Ward Hall and the adjacent Fenced Area.
- Emergency Planning Zone is equivalent to the Operations Boundary (Rooms 109 and 110).
- The Hot Zone shall be defined as the area affected directly by the emergency. This will normally be equivalent to the Operations Boundary but may be extended to beyond the Site Boundary when necessary (i.e. bomb threat). This zone should be indicated through the use of "Caution – Crime Scene – Do Not Enter" tape or magenta and yellow "Radiation Area" rope or tape. Absolutely no personnel are to be allowed in this area without approval of the Emergency Director. Entrance and exit shall be through Access Control Points.
- The Warm Zone shall be defined as the area beyond the Hot Zone that is needed to treat the situation. Entrance and exit across this boundary shall be through Access Control Points only. A

decontamination area shall be directly on the Warm Zone side of the Access Control Point. The following shall be a minimum requirement at this point.

- Low range β-γ monitor
- Absorbent paper at step-off point
- Plastic bag for collecting contaminated wastes
- Any protection clothing necessary for entry into the Exclusion Zone.

The minimum boundary shall extend to a point that is less than 2 mR/h and should be indicated through the use of "Police Line – Do Not Cross" tape or equivalent. Only necessary personnel and vehicles shall be allowed in this zone.

- The Cold Zone shall be defined as the area beyond the Hot and Warm Zones. The Emergency Operations Center (EOC) shall be located in this area upwind from the Hot Zone. All members of the public and press shall remain in this area along with all nonessential personnel.
- The Emergency Operations Center (EOC) is where all emergency coordination shall take place. This is also where the Emergency Director and his/her necessary advisors should reside. This area shall be closed to the press and public.

2. DEFINITIONS, ABBREVIATIONS, AND SYMBOLS

2.1 Definitions

The following frequently used terms are defined to aid in the uniform interpretation of this plan.

DOSE	A general term denoting the quantity of radiation or energy absorbed. For special purposes it must be appropriately qualified. For purposes of the latter mentioned Protective Action Guides (PAG) it refers specifically to dose equivalent.
DOSE EQUIVALENT	A quantity that expresses all radiation on a common scale for determining the biological effect. It is defined as the product of the absorbed dose in rads and the appropriate scaling factor. The unit of dose equivalent is the rem.
DOSE COMMITMENT	The radiation dose equivalent received by an exposed individual or a specific organ cited over a lifetime from a single event.
EMERGENCY	An emergency is a condition that calls for immediate action, beyond the scope of normal operating procedures, to avoid an accident or to mitigate the consequences of one.
EMERGENCY ACTION LEVELS	Specific instrument readings, or observations, radiological dose or dose rates, or specific contamination levels of airborne, waterborne, or surface-deposited radioactive materials that may be used as thresholds for establishing emergency classes and initiating appropriate emergency measures.
EMERGENCY CLASSES	Emergency classes are classes of accidents grouped by severity level for which predetermined emergency measures should be taken or considered.
EMERGENCY PLAN	An emergency plan is a document that provides the basis for actions to cope with an emergency. It outlines the objectives to be met by the emergency plan procedures and defines the authority and responsibilities to achieve such objectives.
EMERGENCY PLANNING ZONE (EPZ)	Area for which offsite emergency planning is performed to assure that prompt and effective actions can be taken to protect the public in the event of an accident. The EPZ size depends on the distance beyond the site boundary at which the Protective Action Guide (PAG) could be exceeded.
EMERGENCY PLAN PROCEDURES	Emergency plan procedures are documented instructions that detail the implementation actions and methods required to achieve the objectives of the emergency plan.
OFFSITE	The geographical area that is beyond the site boundary.
ONSITE	The geographical area that is within the site boundary.
OPERATIONS BOUNDARY	The area with the site boundary such as the reactor building (or the nearest Physical personnel barrier in cases where the reactor building is not a principal physical personnel barrier) where the reactor chief administrator has direct authority over all activities. The area with this boundary shall have prearranged evacuation procedures known to personnel frequenting the area.

PROJECTED DOSE EQUIVALENT	The dose equivalent that is projected to be received by individuals in a population group from a contaminating event if no protective actions were taken.
PROTECTIVE ACTION	An action taken to avoid a portion of the exposure to radiation that would result in the projected dose equivalent.
PROTECTIVE ACTION GUIDES (PAG)	Projected radiological dose or dose commitment values to individuals that warrant protective action following a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the protection action is not offset by excessive risks to individual safety in taking the protective action. The projected dose does not include the dose that has unavoidably occurred prior to the assessment.
RAD	The unit of absorbed dose equal to 100 ergs/gram in any medium.
REM	The unit of dose equivalent. The dose equivalent in rems is numerically equal to the absorbed dose (rads) multiplied by the quality factor, the distribution factor, and any other necessary modifying factors. The rem represents the quantity of radiation that is equivalent in biological damage, of a specific sort, to one (1) rad of 250 kVp x rays.
RESEARCH REACTOR	A device designed to support a self-sustaining neutron chain reaction for research, developmental, educational, training, or experimental purposes, and which may have provisions for production of nonfissile radioisotopes.
SITE BOUNDARY	The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities. The area within the site boundary may be frequented by people unacquainted with the reactor operations.
SHALL, SHOULD AND MAY	The word “shall” is used to denote a requirement; the word “should” to denote a recommendation; and the word “may” to denote permission, neither a requirement nor a recommendation.
SUITABLE SAMPLE	A sample of the population consisting exclusively of children less than one year of age, or if children less than one year of age are absent or have been removed from the exposed population, a suitable sample is considered to consist of adults exclusively. The population of K-State students, faculty, and staff is taken to be exclusively adults.

2.2 Abbreviations

The following frequently used abbreviations are listed to aid in the uniform interpretation of this plan.

CFR	U. S. Code of Federal Regulations
EOC	Emergency Operations Center
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
Reactor Facility	Kansas State University Nuclear Reactor Facility
K-State	Kansas State University
kW	Kilowatt
MW	Megawatt
NRC	U. S. Nuclear Regulatory Commission
PAG	Protective Action Guides
Reactor	General Atomics TRIGA Mk. II Nuclear Reactor
University	Kansas State University
U. S.	United States

2.3 Symbols

3. ORGANIZATION AND RESPONSIBILITIES

In order to properly respond to an emergency situation, and those abnormal circumstances that may lead to one, a well-defined administrative structure is needed. During a declared emergency (defined in Section 0), or those events which if not attended to would lead to an emergency, the Emergency Director shall activate the necessary agencies to provide assistance for coping with the emergency situation, recovery from the emergency, and maintaining emergency preparedness. Written confirmations from the agencies shall be stored with the Reactor Manager with copies placed in **Error! Reference source not found.**.

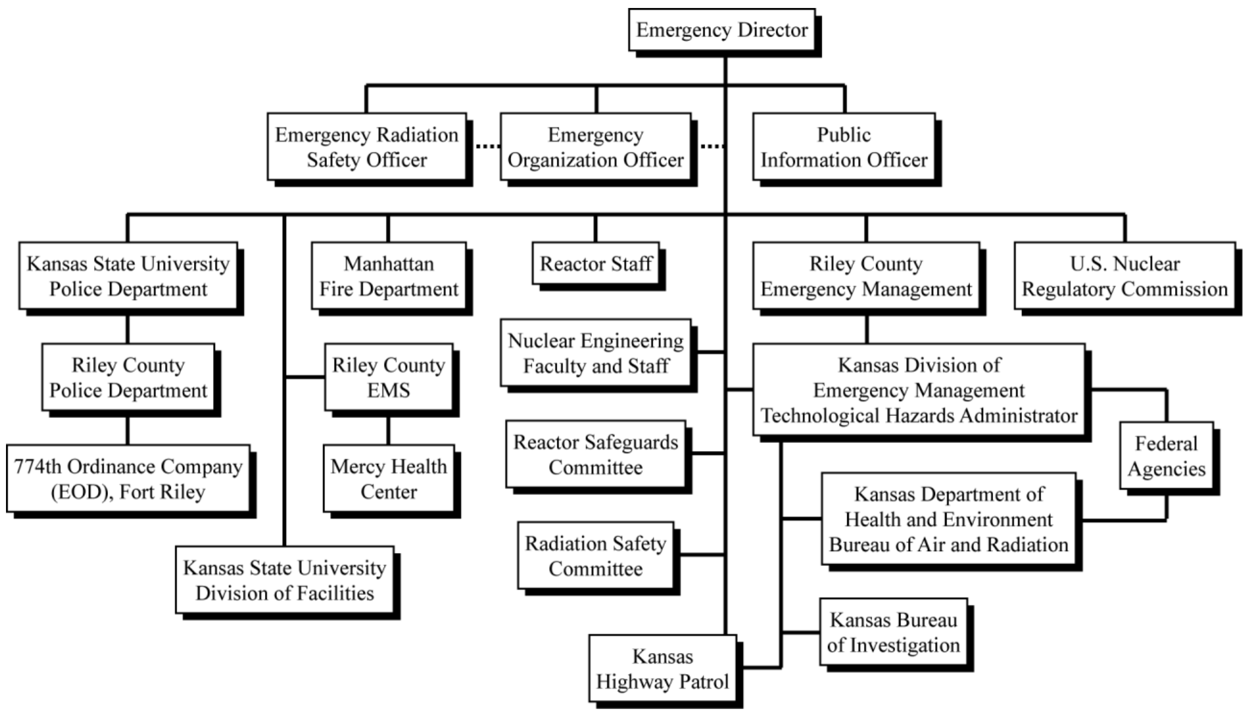

Figure EP. 8 - Emergency Organization Chart

3.1 Emergency Director

The Emergency Director shall be the only one to declare and terminate an Unusual Event or Alert and shall initiate recovery actions. He may also be the one to declare a Medical Incident. All emergency operations shall be under the direct supervision of the Emergency Director. This person should be in or near the Emergency Operations Center (EOC) at all times. The Emergency Director should provide periodic reports for command post/EOC responders to ensure all coordinating elements in the EOC are aware of status and response actions.

Line of succession:

1) Reactor Manager
2) Reactor Supervisor
3) Senior Reactor Operators
4) Reactor Operators working at least 1 year prior
5) Reactor Safeguards Committee
6) Reactor Operators working less than 1 year prior

In the event this list is exhausted, University Management (Dean of Engineering, then Provost) will select an Emergency Director.

Shall be notified by:
3.5.3 Kansas State University Police Department (**UPD)**
3.6.3 Riley County Police Department (**RCPD)**
or by onsite events

Notification of Emergency Agencies:
The following agencies must be immediately notified either directly or at the direction of the Emergency Director if an Unusual Event or Alert is declared.

1) 3.5.3 Kansas State University Police Department (UPD)
2) 3.6.1 Riley County Emergency Management
3) 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
4) 3.8.1 U.S. Nuclear Regulatory Commission (NRC)

The Radiation Control Section of the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator shall be notified if a Medical Incident is declared.

3.2 Emergency Radiation Safety Officer

The Emergency Radiation Safety Officer may be the one to declare a Medical Incident. This person shall be responsible for coordinating radiation-monitoring activities and advising the Emergency Director concerning recommended protective actions in order to ensure that all personnel and the public receive minimal doses. They shall also be the one to authorize volunteer emergency workers to incur radiation exposures in excess of normal occupational limits.

Line of succession:
1) University Radiation Safety Officer

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)
3.6.3 Riley County Police Department (RCPD)
or by onsite events

3.3 Emergency Organization Officer (EOO)

The Emergency Organization Officer is responsible maintaining organizational coherence between the different responding agencies and the Emergency Team. They shall assist in guiding people to the Emergency Operations Center and shall relay messages between the EOC and the other emergency personnel at the scene. They shall also ensure that proper dosimetry is worn and recorded, that security procedures are followed, and shall record an approximate timeline as events occur. The EO is also responsible for providing courtesy notification to KSU management (i.e., Head of Mechanical and Nuclear Engineering Department, Office of the Dean of Engineering) not otherwise notified during communications from the Emergency Director or response agencies.
The Emergency Organization Officer shall be designated by the Emergency Director preferably as one of the next members in line of succession for Emergency Director.

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)
3.6.3 Riley County Police Department (RCPD)
or by onsite events

3.4 Public Information Officer (PIO)

The Public Information Officer (PIO) shall deal with all press releases under the direction of the Emergency Director. Only authorized statements may be disclosed. This person (if on scene) is to stay within the Cold Zone or the Emergency Operations Center. The assumption is to be made that they have absolutely no radiological training and shall not be used outside of dealing with the press and public. The Emergency Operations Officer should ensure that copies of information passed from the Emergency Director to the PIO are also distributed to the Department Head or his delegate and to the Office of the Dean of Engineering.

Shall be from:
3.5.5 Kansas State University Media Relations and Marketing

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD) *or*
Emergency Director

3.5 University Resources

<u>3.5.1 Reactor Staff</u>
The Reactor Staff shall assist the Emergency Director, the Emergency Radiation Safety Officer, and the Emergency Operations Officer in assessing and resolving the situation at hand. They shall be used as needed and where needed. They shall also maintain watch to make sure that security and radiation protection procedures are followed.

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)
3.6.3 Riley County Police Department (RCPD)
or by onsite events

<u>3.5.2 Nuclear Engineering Faculty and Staff, Reactor Safeguards Committee, Radiation Safety Committee</u>
These individuals shall assist the Emergency Director, the Emergency Radiation Safety Officer, and the Emergency Operations Officer in assessing and resolving the situation at hand. They shall be used as needed and where needed. They shall also maintain watch to make sure that security and radiation protection procedures are followed.

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)
3.6.3 Riley County Police Department (RCPD)
or by onsite events

<u>3.5.3 Kansas State University Police Department (UPD)</u>
The Kansas State University Police Department may be notified through Dispatch either by means of telephone, security patrol, security/fire alarms, or by any other means identifying a potential problem at the Reactor Facility. Dispatch shall send officers to assist in securing the area, question witnesses, gather

evidence, and maintain proper boundaries through the use of radiological monitoring. If needed, they may contact the necessary agencies for assistance. 3.6.3 Riley County Police Department (RCPD) Dispatch may be used to assist in notifying proper personnel and agencies. The Reactor Staff shall be contacted in the event that explosive or toxic gas, fire or other serious event is found in or near one of the following buildings or in the area encompassed by them.

Areas defined as being Near the Reactor:
Ward Hall
Cardwell Hall
Burt Hall
Leasure Hall
Lafene
Facilities Grounds
Chilling Plant
Power Plant
Rathbone, Durland, Fiedler Complex
Parking Lot A28
Ackert Hall

Shall be notified by:
Reactor Staff
Security/Fire Alarm
Security Patrol
Any means identifying a potential problem

Shall notify:
Personnel in Emergency Plan Procedure EPP-2 (page EPP-2-1)

If assistance is needed, may notify:
3.6.3 Riley County Police Department (RCPD)
3.6.2 Manhattan Fire Department (MFD)
3.6.4 Riley County Emergency Medical Service (EMS)
3.7.3 Kansas Highway Patrol (KHP)
3.5.4 Kansas State University Division of Facilities

<u>3.5.4 Kansas State University Division of Facilities</u>
Certain circumstances such as building damage, fire, or flood requires that the Kansas State University Division of Facilities be notified. They shall assist with situations involving building electricity, plumbing, and HVAC systems. A letter of agreement is not required.

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)

<u>3.5.5 Kansas State University Media Relations and Marketing</u>
The personnel of Media Relations and Marketing shall be used as the Public Information Officer. They shall issue press releases and public notices concerning the events. They shall work directly under the Emergency Director as stated in Section 0 (page 131).

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)

3.6 Local Resources

The local resources present in Manhattan and Riley County shall be used as the primary responders when assistance is needed beyond that of the University's capabilities.

3.6.1 Riley County Emergency Management

The Riley County Emergency Management shall be contacted directly by the Emergency Director in the event that either an Unusual Event or Alert is declared. They may be used to assist in coordinating local and state agencies in order to resolve the situation. They will work directly with the Emergency Director to ensure that public safety is maintained and that resources are available as needed.

Shall be notified by:

General

Content

Examinations should be based on a representative sample of questions covering areas in depth required to evaluate trainee understanding and capabilities. Examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Operating Examinations

Operating examinations shall be conducted by the Reactor Supervisor or Nuclear Reactor Facility Manager covering normal, abnormal and emergency operating procedures. Operating examinations shall be graded as Satisfactory (S) or Unsatisfactory (U). The person who prepares written examinations is exempt from participating in the written examination.

Written Examinations

Written examinations be prepared and graded (on a scale from 0 to 100%) by the Reactor Supervisor or Nuclear Reactor Facility Manager covering:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Periodic Requirements

3.6.2 Manhattan Fire Department (MFD)

The Manhattan Fire Department shall provide assistance with emergency situations including fire, explosive or toxic gases, radiological monitoring, and rescue. Liaison shall be provided by the Reactor Director in order to safeguard against radiological hazards and other hazards inherent to the Reactor Facility.

May be notified by:

3.5.3 Kansas State University Police Department (UPD)

3.6.3 Riley County Police Department (RCPD)
The Riley County Police Department shall provide assistance for the Kansas State University Police Department as requested. They may be involved in securing the area, questioning witnesses, gathering evidence, and/or maintaining proper boundaries through the use radiological monitoring. The RCPD Dispatch may be requested to assist in notifying personnel and agencies by the UPD Dispatch.

May be notified by:
3.5.3 Kansas State University Police Department (UPD)

3.6.4 Riley County Emergency Medical Service (EMS)
The Riley County EMS shall be used when transporting injured or contaminated victims to Mercy Health Center.
May be notified by:
3.5.3 Kansas State University Police Department (UPD)

3.6.5 Mercy Health Center
Mercy Health Center shall assist in the treatment of injured personnel, students, and visitors of the Reactor Facility and the associated laboratories. If the victim cannot be fully decontaminated due to life threatening injury then Mercy Health Center shall be advised by the Emergency Team about the situation. Victims shall be transported by Riley County EMS.

3.7 State Resources

During an emergency State Resources may be requested for assistance and guidance. The below abstracts are from the Kansas Emergency Procedures Handbook which defines the overall responsibilities of the Government Agencies. Additional details are provided in the State of Kansas Nuclear Facilities Response Plan.
3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
The Kansas Division of Emergency Management, Technological Hazards Administrator, shall provide coordination in the majority of cases as the responding State Agency. They shall be immediately contacted by the Emergency Director upon declaration of an Unusual Event, Alert, or Medical Incident. KDEM shall contact the 3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation for assistance.

Nuclear blackmail, theft, or sabotage is coordinated by the KDEM and they shall immediately notify the Federal Bureau of Investigation, the 3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation, the Attorney General, and the Kansas Bureau of Investigation.

Shall be notified by:
General
Content

Examinations should be based on a representative sample of questions covering areas in depth required to evaluate trainee understanding and capabilities. Examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Operating Examinations

Operating examinations shall be conducted by the Reactor Supervisor or Nuclear Reactor Facility Manager covering normal, abnormal and emergency operating procedures. Operating examinations shall be graded as Satisfactory (S) or Unsatisfactory (U). The person who prepares written examinations is exempt from participating in the written examination.

Written Examinations

Written examinations be prepared and graded (on a scale from 0 to 100%) by the Reactor Supervisor or Nuclear Reactor Facility Manager covering:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Periodic Requirements *(automated messaging service, requires call back for verification)*
3.7.3 Kansas Highway Patrol (KHP)

Shall notify:
3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation

Shall notify the additional following agencies in the event of nuclear blackmail, theft, or sabotage:
Federal Bureau of Investigation (FBI)
3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation
Attorney General
Kansas Bureau of Investigation (KBI)

3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation
The Kansas Department of Health and Environment, Bureau of Air and Radiation may provide technical guidance and approval of monitoring and clean-up operations associated with spills, discharges, or other types of dispersion of radioactive material resulting from a nuclear emergency. They shall also provide coordination assistance with the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator.

Shall be notified by:
3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
3.7.3 Kansas Highway Patrol (KHP)

3.7.3 Kansas Highway Patrol (KHP)
Upon notification by local authorities, the Kansas Highway Patrol shall contact the Governor, the 3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation, and the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator. The KHP shall provide radiological monitoring assistance, the securing of the area, and evacuation of communities, when necessary.

May be notified by:
3.5.3 Kansas State University Police Department (UPD)
General
Content

Examinations should be based on a representative sample of questions covering areas in depth required to evaluate trainee understanding and capabilities. Examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Operating Examinations

Operating examinations shall be conducted by the Reactor Supervisor or Nuclear Reactor Facility Manager covering normal, abnormal and emergency operating procedures. Operating examinations shall be graded as Satisfactory (S) or Unsatisfactory (U). The person who prepares written examinations is exempt from participating in the written examination.

Written Examinations

Written examinations be prepared and graded (on a scale from 0 to 100%) by the Reactor Supervisor or Nuclear Reactor Facility Manager covering:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Periodic Requirements

Shall notify:
Governor
3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation
3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator

3.8 Federal Resources

Additional Federal Resources may be necessary but shall be coordinated through the State Agencies except for those mentioned below.

3.8.1 U.S. Nuclear Regulatory Commission (NRC)
The U.S. Nuclear Regulatory Commission is the licensing agency for the Reactor Facility and shall be notified of any declared emergency by the Emergency Director. They may provide guidance and resources to effectively solve the situation.

Shall be notified by:
General
Content

Examinations should be based on a representative sample of questions covering areas in depth required to evaluate trainee understanding and capabilities. Examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Operating Examinations

Operating examinations shall be conducted by the Reactor Supervisor or Nuclear Reactor Facility Manager covering normal, abnormal and emergency operating procedures. Operating examinations shall be graded as Satisfactory (S) or Unsatisfactory (U). The person who prepares written examinations is exempt from participating in the written examination.

Written Examinations

Written examinations be prepared and graded (on a scale from 0 to 100%) by the Reactor Supervisor or Nuclear Reactor Facility Manager covering:

- Theory and principles of operation.
- General and specific plant operating characteristics.
- Plant instrumentation and control systems.
- Plant protection systems.
- Engineered safety systems.
- Normal, abnormal, and emergency operating procedures.
- Radiation control and safety.
- Technical specifications.
- Applicable portions of title 10, chapter I, Code of Federal Regulations.

Periodic Requirements *(initial call requires call back for verification)*

3.8.2 774th Ordinance Company (EOD), Fort Riley
The 774th Ordinance Company (EOD) from Fort Riley shall assist the 3.6.3 Riley County Police Department (RCPD) Emergency Response Unit in dealing with explosives and hazardous materials.

Shall be notified by:
3.5.3 Kansas State University Police Department (UPD)
3.6.3 Riley County Police Department (RCPD)

3.9 Additional Resources

The following groups have agreed to provide radiological and technical emergency assistance. They shall all be notified through the Emergency Director or Emergency Radiation Safety Officer.

- University of Kansas, Radiation Safety Officer
- University of Missouri, Columbia
- Radiation Safety Officer
- Reactor Health Physics
- Reactor Facility

4. EMERGENCY CLASSIFICATION SYSTEM

The State of Kansas Nuclear Facilities Incidents Response Plan identifies, in increasing severity, four classes of emergencies: 1) Unusual Events, 2) Alert, 3) Site Emergency, and 4) General Emergency. The following three reports have reviewed each of these classifications and how they pertain to the Reactor.

1. Hazards Summary Report for the KSU TRIGA Mk. II Reactor, January 1961.
2. Application for Amendment of License R-88, submitted January 3, 1968 (revised February 7, 1968).
3. Application for Amendment of License R-88, submitted September, 2002.

In summary, neither total loss of reactor coolant nor any credible power transient would cause damage to the reactor fuel or release of fission products. This notwithstanding, failure of a fuel element and release of fission products would not lead to unacceptable potential radiation doses within the Reactor Facility or in unrestricted areas outside the Reactor Facility. Thus, only the first two emergency classifications apply. They are described in more detail below.

4.1 Non-Classified Events

Certain circumstances, while not presenting a threat to safety or security of the reactor, do call for immediate response on the part of the Reactor Staff. These circumstances do not require notification of offsite agencies. In all cases, the reactor operator on duty will promptly notify the Reactor Manager who may, if appropriate, notify the University Radiation Safety Officer.

4.2 Medical Incident

A Medical Incident may arise from a laboratory accident involving radiation exposure or radioactive contamination accompanied by bodily injury. The initiating accident may have no potential for escalation to a nuclear emergency and may have no effect on reactor operations. The Radiation Control Section of the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator requests notification if a Medical Incident occurs. Except for local medical assistance that may be required, no other offsite emergency agencies must be contacted.

4.3 Unusual Event

This class describes abnormal conditions as a result of an incident or the threat of an incident. An unusual event must reasonably have the potential to increase in significance if proper action is not taken or if circumstances beyond the control of the operating staff render the situation more serious in terms of safety. The entire onsite emergency organization may not be activated but individual emergency teams might be required. No significant offsite release of radiation has occurred.

Offsite response agencies may be alerted but will not be activated. Depending on the type of Unusual Event, modified increased readiness measures will be considered for the State Emergency Operations Center (EOC) for the duration of the emergency situation.

4.4 Alert

The alert class is designed to provide early and prompt notification of minor events that could lead to more serious consequences as the Reactor Staff assesses and responds to the incident. Small releases of radioactivity may occur. These releases will not be life-endangering in that they will be small fractions of the EPA protective action guides. Although this type of incident is associated with a management judgment that the emergency situation can be corrected and controlled by the Reactor Staff, notification of appropriate offsite agencies to alert them as to the nature, extent and scope of the incident shall be accomplished. This is the lowest class level where emergency offsite response may be necessary. This response will mainly be for State and Local agencies to establish an organization to monitor the incident and to make preparations for active response should the situation degrade. The State Emergency Operations Center (EOC) will be activated and if necessary the Governor will proclaim a State of Disaster

Emergency. State agencies will be alerted and placed on standby. Local EOCs may be activated or may assume an increased readiness posture. Thus government agencies will remain on alert status until the situation is corrected or active operations are initiated.

5. 4.1 NON-CLASSIFIED EVENTS

There are several circumstances that require the immediate response of the Reactor Staff even though they do not present a threat either to the safety or the security of the Reactor. These circumstances do not require the notification of offsite agencies. In all cases, the reactor operator on duty shall promptly notify the Reactor Manager who may, if appropriate, notify the University Radiation Safety Officer.

5.1 Fire Outside the Reactor Facility

The Reactor shall be secured immediately if a fire near the building is discovered. The exhaust fan shall be turned off. The Manhattan Fire Department shall be called. The MFD shall be consulted to ensure that methods taken to combat the fire do not lead to any additional hazards. Liaison shall be established with the MFD to ensure that firefighting personnel are familiar with the geography and procedures to be followed. Before arrival of the MFD the staff present should take steps to keep the fire from reaching the building.

If a fire exists near the Reactor Facility (Ward, Cardwell, Burt, Leasure, Lafene, Facilities Grounds, Chilling Plant, Power Plant, Rathbone, Durland, Fiedler, Parking Lot A28, or Ackert) the UPD Dispatch shall contact the Reactor Facility (532-6657) and notify them of the situation as soon as possible.

5.2 Fire Inside the Reactor Facility

In general, remedial action shall be the same as that detailed above for a fire outside the Reactor Facility. Particular care should be used in fighting fire in the electrical equipment of the control console. Carbon dioxide or Halon extinguishers should be used. All further operation shall be suspended until the Reactor Safeguards Committee approves restart. Any non-safety system related fire within the Reactor Facility not extinguished within 15 minutes shall be reported as an Unusual Event. Fire that causes damage to safety-related reactor components or loss of security or surveillance capabilities for longer than 15 minutes shall be reported as an Alert.

5.3 Tornado

The Reactor shall be secured immediately if a tornado is sighted in the Manhattan area or if a tornado warning exists for the city of Manhattan, KS. If the tornado exists near the Reactor Facility (Ward, Cardwell, Burt, Leasure, Lafene, Facilities Grounds, Chilling Plant, Power Plant, Rathbone, Durland, Fiedler, Parking Lot A28, or Ackert) the event shall be classified as an Unusual Event. If there is structural damage to the Reactor Facility the event shall be classified as an Alert. Either classification requires the Reactor Safeguards Committee to approve restart.

5.4 Earthquake

The Reactor shall be secured immediately if an earthquake is felt. If structural damage to the Reactor Facility is incurred the situation shall be classified as an Alert otherwise it shall be classified as an Unusual Event. All further operation shall be suspended until the Reactor Safeguards Committee approves restart.

5.5 Floods, Snow and Ice

While the Reactor Facility is geographically not susceptible to flood waters, some of the surrounding areas could be. In addition, Manhattan occasionally receives heavy snow and ice storms that can make traveling difficult. This could pose a problem for the agencies trying to assist during an emergency. Due

to this issue, during times of flooding and poor winter conditions the Reactor Manager should keep advised as to the condition of the major access routes to Manhattan and the Reactor Facility. If conditions exist where it would be difficult to reach the Reactor Facility from locations outside of Manhattan the Reactor Manager should suspend operations until conditions improve.
In the event of high water in the Reactor Bay, not associated with the primary coolant, the Reactor Manager shall suspend operations while the source of the water is investigated. All electrical systems (pumps, meters, etc.) potentially affected by the water shall be operationally verified before Reactor operations may continue. The Reactor Manager must verify restart before operations may continue.

5.6 Loss of Power and Electrical Storms

The Reactor is designed so that a loss of power will not cause any safety concerns. However, to make sure that all systems are functioning properly, a walkthrough of the checkout procedure should be completed with any system changes from the original checkout noted and investigated. The reactor should be secured during an electrical storm.

5.7 High Radiation Levels

The Reactor shall immediately be secured and the exhaust fan and the cooling system shall be shut down in the event that high radiation levels are indicated through the Reactor's monitors, the sounding of the evacuation alarm, by visible symptoms from the Reactor or through the perception of other conditions which would appear to threaten personnel safety or threaten compliance with appropriate guides for protection against radiation.

The Operations Boundary shall be evacuated upon sounding of the evacuation alarm (5 R/h at the 22' level). Persons believed to be contaminated should be evacuated to the shower-equipped rest room in the basement of Ward Hall (Room 007) also known as Emergency Assembly Area #2. All other personnel should evacuate to the lobby of Ward Hall, also known as Emergency Assembly Area #1.

The Site Boundary shall be evacuated if the radiation level at the door between the Control Room and the Reactor Bay is in excess of 100 mR/h. The Emergency Director shall give this order. Evacuation should be to an upwind sheltered location where the exposure rate is less than 2 mR/h. The Emergency Director, in concert with the Emergency Radiation Safety Officer will evaluate the hazard, declare an emergency, if appropriate, and take appropriate corrective steps to the end that life and health will be protected, equipment and facilities will be protected, and the spread of radioactive contamination will be minimized.

The Emergency Director and the Emergency Radiation Safety Officer shall evaluate the probability of successful recovery with available aids against the probable consequences of recovery if immobilized personnel remain in the Reactor Bay. All appropriate aids will be used for such recovery, including protective clothing and dust masks. A Medical Incident (Section 0, page 142) shall be initiated if serious personnel exposure or contamination is believed to have resulted. The Reactor Facility will be re-entered and decontaminated when possible.

6. MEDICAL INCIDENT

A Medical Incident may arise from a laboratory accident involving radiation exposure or radioactive contamination accompanied by bodily injury. The initiating accident may have no potential for escalation to a nuclear emergency and may have no effect on reactor operations. The Radiation Control Section of the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator requests notification if a Medical Incident occurs. Except for local medical assistance that may be required, no other offsite emergency agencies must be contacted.

6.1 Responsibilities

The following gives a brief description of how to coordinate the responsibilities when dealing with a Medical Incident. A full definition of who is responsible for what and the overall emergency organization is explained in detail in Section 0 (page 129).

<u>6.1.1 Declaration</u>

The recognition of a Medical Incident is a judgment matter for supervisory personnel (Reactor Manager, Reactor Supervisor, Laboratory Instructor, etc.). A Medical Incident may exist under conditions of 1) over-exposure to radiation or 2) bodily injury requiring medical treatment accompanied by radioactive contamination. If it is suspected that the action levels listed in Section 0 (page 142) are exceeded, the Emergency Radiation Safety Officer and the Emergency Director will be notified. If assessment reveals that action levels have been exceeded, the Emergency Radiation Safety Officer will declare that a Medical Incident has occurred.

<u>6.1.2 Notification</u>

The Emergency Radiation Safety Officer and the Emergency Director shall be notified if a suspected Medical Incident has occurred. The Radiation Control Section of the 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator shall be notified if a Medical Incident is declared.

<u>6.1.3 Resources</u>

No Local, State, or Federal Resources should be needed. If the contaminated person must be transferred to a hospital while still contaminated the situation shall be classified as an Unusual Event (Section 0, page 144).

<u>6.1.4 Assessment</u>

The Emergency Radiation Safety Officer, assisted by the Emergency Director, will assess the degree of radiation exposure and/or contamination and will advise medical personnel. The Emergency Director will determine the incident cause and scope of involvement and assess the possibility for escalation of the incident to a more severe emergency condition.

6.2 Initiating Conditions

The recognition of a Medical Incident is a judgment matter for supervisory personnel (Reactor Manager, Reactor Supervisor, Laboratory Instructor, etc.). The following action levels are meant to assist with that judgment and will be considered as the more limiting case:

1) Known or suspected exposure greater than:
   a) Total Effective Dose Equivalent of 5.0 rems.
   b) Sum of Deep Dose Equivalent and Committed Dose Equivalent of 50 rems to any organ or tissue other than the lens of the eye.
   c) Eye Dose Equivalent of 15 rems.
   d) Shallow Dose Equivalent of 50 rems to the skin or any extremity.
2) External contamination of skin surface, with β-γ dose rates greater than 10 mR/h as measured with an open window GM survey meter, with injury, after rinsing wounds with running water, and scrubbing skin surface with detergent and water.

6.3 Corrective and Protective Actions

Unless life is threatened, first aid as well as monitoring the decontamination will take place in the Reactor Facility prior to the release of an injured person to medical authorities. The primary decontamination station is at the shower and sink on the 0' Level (Basement) of the Reactor Bay (Room 110 – see Figure Figure EP. 7, page 124). A secondary, or overflow, decontamination area is in Room 007 also known as Emergency Assembly Area #2 (see Figure Figure EP. 6, page 123). This room contains a shower and is conveniently located nearan emergency supply cabinet as well as an elevator.

6.4 Recovery and Re-Entry

After release of an injured person to medical authorities, all potentially contaminated work areas, equipment, and personnel will be monitored and decontaminated as necessary. Documentation of the Medical Incident is the responsibility of the University Radiation Safety Officer. If the Emergency Radiation Safety Officer and the University Radiation Safety Officer are not one in the same person then they will work together to complete the necessary paperwork.

All recovery operations require written damage assessment reports, radiological survey reports, and specific recovery procedures. All recovery operations involving the reactor core, the shielding tank, the biological shield or other safety-related components require approval of the Emergency Director and the Reactor Safeguards Committee. All other operations require the approval of the Emergency Director. Applicable Emergency Plan Procedures

For consistency within Plan, the Emergency Plan Procedures applicable to a Medical Incident are listed in Appendix B in page EPP-2.

7. UNUSUAL EVENT

Unusual Events are circumstances which are in process or have occurred and which indicate a potential degradation of the safety of the Reactor Facility. No release of radioactive material requiring offsite response or monitoring is expected unless further degradation of safety systems occurs. Offsite notification is required to 1) assure that the first step in any response later found to be necessary has been carried out, 2) bring the Reactor Staff to a state of readiness, and 2) provide systematic handling of Unusual Events information and decision-making.

7.1 Responsibilities

The following gives a brief description of how to coordinate the responsibilities when dealing with an Unusual Event. A full definition of who is responsible for what and the overall emergency organization is explained in detail in Section 0 (page 129).

7.1.1 Declaration
Notification or cancellation of an Unusual Event is the responsibility of the Emergency Director as defined by Section 0 (page 40).

7.1.2 Notification
Verbal notification of an Unusual Event will be made by the Emergency Director to following agencies:

1) 3.5.3 Kansas State University Police Department (UPD)
2) 3.6.1 Riley County Emergency Management
3) 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
4) 3.8.1 U.S. Nuclear Regulatory Commission (NRC)

Cancellation (closeout) of the Unusual Event shall be made to the same agencies with a written summary to be transmitted to these agencies within 24 hours of cancellation. The 3.4 Public Information Officer (PIO) may make a press release if the situation warrants one with the approval of the Emergency Director.

7.1.3 Resources
All necessary agencies shall be assembled as per Section 0 (page 129) in order to alleviate the situation.

7.1.4 Assessment
The principle responsibilities of the Emergency Director and the Emergency Team are to:

1) Assess the need for and initiate corrective and protective actions to mitigate exposure of personnel, the magnitude of radiological releases within the Reactor Facility, and damage to equipment.
2) Assess and advise off-site emergency organizations of the threat of escalation of the Unusual Event to an Alert condition.

7.2 Initiating Conditions

Declaration of an Unusual Event requires a judgment by the Emergency Director that the potential exists for degradation of the safety of the Reactor Facility. Listed below are action levels and guidance for that judgment.

Security Compromise

Threats, warnings, or other indications of an impending breach of the physical security of the Reactor Facility.

Sabotage
Threats, warnings, or other indications of an impending attempt at sabotage of the Reactor Facility.

Fire
Fire within the Reactor Facility lasting longer than 15 minutes that causes damage to non-safety-related Reactor components.

Medical Incident
Any Medical Incident (Section 0, page 142) requiring transportation of a contaminated accident victim to an off-site hospital.

External Hazards
Report or observation of a severe natural phenomenon affecting the Reactor Facility. Near or onsite life threatening release of toxic or flammable gases, with potential requirement for Reactor Facility evacuation.

Radiological Releases
Actual or projected radiological effluent at the site boundary which is calculated or measured to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:

1. A deep dose equivalent of 15 mrem (0.15 mSv)
2. A committed effective dose equivalent of 15 mrem based on the following considerations:
    - 100 EC x 24 hr = 2.4×10^3 EC-hr = 15 mrem (for radionuclides other than noble gasses)
    - 50 EC x 24 hr = 1.2×10^3 EC-hr = 15 mrem (for noble gases)

Suspected Fuel Damage
The consequences of breached cladding on a fuel element depends on the irradiation history of the element. The release of radioactive noble gases will be evidenced only by increased readings of area radiation monitors. Radioactive iodine would be observed in air samples. Radioactive cesium as well as radioactive iodine may be observed in primary coolant samples or indicated by increased radiation levels at the cleanup demineralizer. Statistically significant and increasing levels of radioactive iodine or radioactive cesium, confirmed by gamma ray spectroscopy, call for notification of an Unusual Event. Unambiguous evidence, calling for notification of an Unusual Event, would be an otherwise unexplained exposure rate in excess of 5 mrem/h at the reactor pool surface, or a ^{137}Cs specific activity in the primary coolant in excess of 8000 pCi/cm^3.

7.3 Corrective and Protective Actions

All operating and research personnel using the Reactor Facility will be advised in the event of external hazards with the potential requirement for Reactor Facility evacuation. Emergency assessment and health physics programs will be invoked in the event of suspected fuel damage or radiological releases.

Only personnel essential for assessment and health physics programs will be permitted within the Hot Zone. Self-reading dosimetry, as a minimum, will be issued for all these persons. Concentrations and dose commitments outside the Operations Boundary but within the Site Boundary in excess of action levels call for partial or total evacuation of onsite members of the public.

7.4 Recovery and Re-Entry

Specific recovery and re-entry operations depend on actual conditions within the Reactor Facility. Although comprehensive advanced planning is not possible, guidance for corrective actions and criteria for restoring the Reactor to a safe status is provided in the Emergency Plan Procedures. All recovery operations require written damage assessment reports, radiological survey reports, and specific recovery procedures. All recovery operations involving the reactor core, the shielding tank, the biological shield or other safety-related components require approval of the Emergency Director and the Reactor Safeguards Committee. All other operations require the approval of the Emergency Director. Criteria for re-entry into the Reactor Facility or other onsite areas that have been evacuated are also provided.

Recovery operations involving repair or modification of the reactor core, tank, biological shield, or other safety-related components require written procedures approved by the Reactor Safeguards Committee. All recovery operations require written procedures approved by the Emergency Director and the Emergency Radiation Safety Officer. Such procedures will consider radiation levels as well as personnel monitoring and decontamination. Emergency personnel exposure may be incurred to a limit of 25 rem TEDE for lifesaving efforts (see Table 0, page 151). Every effort will be made to limit emergency exposure only to personnel over 45 years of age. Authorization to incur such exposure is given by the Emergency Radiation Safety Officer.

Re-entry of the Reactor Facility for normal use and resumption of reactor operations following an emergency require approval of the Reactor Safeguards Committee.

7.5 Applicable Emergency Plan Procedures

For consistency within Plan, the Emergency Plan Procedures applicable to an Unusual Event are listed in Appendix B**Error! Reference source not found.**.

8. ALERT

An alert shall be declared when events are in process or have occurred that involve an actual or potential substantial degradation of the level of safety of the Reactor Facility. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. Offsite notification is required to 1) assure that emergency personnel are available to respond or to perform confirmatory radiation monitoring if required, and 2) provide offsite authorities current status information.

8.1 Responsibilities

The following gives a brief description of how to coordinate the responsibilities when dealing with an Alert. A full definition of who is responsible for what and the overall emergency organization is explained in detail in Section 0 (page 129).

<u>8.1.1 Declaration</u>
Notification or cancellation of an Alert is the responsibility of the Emergency Director as defined by Section 0 (page 40).

<u>8.1.2 Notification</u>
Verbal notification of an Alert will be made by the Emergency Director to following agencies:

1) 3.5.3 Kansas State University Police Department (UPD)
2) 3.6.1 Riley County Emergency Management
3) 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
4) 3.8.1 U.S. Nuclear Regulatory Commission (NRC)

Cancellation (closeout) of the Alert shall be made to the same agencies with a written summary to be transmitted to these agencies within 24 hours of cancellation. The 3.4 Public Information Officer (PIO) may make a press release if the situation warrants one with the approval of the Emergency Director.

<u>8.1.3 Resources</u>
All necessary agencies shall be assembled as per Section 0 (page 129) in order to alleviate the situation.

<u>8.1.4 Assessment</u>
The principle responsibilities of the Emergency Director and the Emergency Team are to:

1) Assess the need for and initiate corrective and protective actions to mitigate exposure of personnel, the magnitude of radiological release within the Reactor Facility and offsite, and damage to equipment.
2) If the Alert involves the potential for radiological releases, monitoring teams will be dispatched to determine offsite involvement, if any. Periodic status reports will be provided to offsite emergency agencies.

8.2 Initiating Conditions

Declaration of an Alert requires a judgment by the Emergency Director that a substantial degradation of the Reactor Facility has occurred or may occur, with the possible need for assistance by off-site agencies. Listed below are action levels and guidance for that judgment.

Security Compromise

Confirmed attempt or actual breach of the security of the Reactor Facility.

Sabotage
Damage due to confirmed or apparent sabotage of the Reactor Facility.

Fire
Fire within the Reactor Facility causing damage to safety-related Reactor components or loss of security or surveillance capabilities for longer than 15 minutes.

External Hazards
Damage to the Reactor Facility due to aircraft crash, missile impact, tornado, earthquake, etc. Entry into the Reactor Facility of life-threatening toxic or flammable gases, requiring evacuation of the Reactor Facility. An explosion causing damage to the Reactor Facility.

Loss of Shielding Water
Abnormal loss of shielding water (evidenced by alarm) from the reactor tank at a rate exceeding makeup capabilities.

High Radiation Levels
Loss of radioactive material control leading to:

1. Exposure rate at the door between the Control Room and the Reactor Bay in excess of 100 mR/h for 1 hour.
2. Exposure rates at the site boundary in excess of 20 mR/h for 1 hour or 100 mrem to the thyroid (committed dose equivalent).
3. Actual or projected radiological effluent at the site boundary which is calculated or measured to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:
4. A deep dose equivalent of 75 mrem
5. A committed effective dose equivalent of 75 mrem based on the following considerations:
   - 500 EC x 24 hr = 1.2×10^4 EC-hr = 75 mrem (for radionuclides other than noble gasses)
   - 250 EC x 24 hr = 6.0×10^3 EC-hr = 75 mrem (for noble gases)
6. Potential off-site radiation exposures in excess of 10% of the EPA Protective Action Guides for the general public (Section 0)

Confirmed Fuel Damage
Confirmed breach of cladding of multiple fuel elements.

8.3 Corrective and Protective Actions

All operating and research personnel using the Reactor Facility will be advised in the event of external hazards with the potential requirement for Reactor Facility evacuation. Emergency assessment and health physics programs will be invoked in the event of suspected fuel damage or radiological releases.
Only personnel essential for assessment and health physics programs will be permitted within the Hot Zone. Self-reading dosimetry, as a minimum, will be issued for all these persons. Concentrations and dose commitments outside the Operations Boundary but within the Site Boundary in excess of action levels call for partial or total evacuation of onsite members of the public.

8.4 Recovery and Re-Entry

Specific recovery and re-entry operations depend on actual conditions within the Reactor Facility. Although comprehensive advanced planning is not possible, guidance for corrective actions and criteria for restoring the Reactor to a safe status is provide in the Emergency Plan Procedures. All recovery operations require written damage assessment reports, radiological survey reports, and specific recovery procedures. All recovery operations involving the reactor core, the shielding tank, the biological shield or other safety-related components require approval of the Emergency Director and the Reactor Safeguards Committee. All other operations require the approval of the Emergency Director. Criteria for re-entry into the Reactor Facility or other onsite areas that have been evacuated are also provided.

Recovery operations involving repair or modification of the reactor core, tank, biological shield, or other safety-related components require written procedures approved by the Reactor Safeguards Committee. All recovery operations require written procedures approved by the Emergency Director and the Emergency Radiation Safety Officer. Such procedures will consider radiation levels as well as personnel monitoring and decontamination. Emergency personnel exposure may be incurred to a limit of 25 rem TEDE for lifesaving efforts (see Table 0, page 151). Every effort will be made to limit emergency exposure only to personnel over 45 years of age. Authorization to incur such exposure is given by the Emergency Radiation Safety Officer.

Re-entry of the Reactor Facility for normal use and resumption of reactor operations following an emergency require approval of the Reactor Safeguards Committee.

8.5 Applicable Emergency Plan Procedures

For consistency within Plan, the Emergency Plan Procedures applicable to an Alert are in Appendix B.

9. PROTECTIVE ACTION GUIDES

Discussed below are Kansas Protective Action Guides. The information is extracted from the Nuclear Facilities Incidents Response Plan of the Kansas Division of Emergency Management, edited to apply to the Reactor Facility.

9.1 Introduction

In the event of a nuclear incident, a contamination event could have public health implications. If such an incident occurs, an estimate shall be made of the radiation dose which affected population groups could receive. This dose estimation is called the projected dose. A protective action is an action taken to avoid or reduce this projected dose when the benefits derived from such an action are sufficient to offset any undesirable effects of the protective action. The Protective Action Guide (PAG) is the projected dose to individuals in the population which warrants taking protective actions.

The decision to initiate a protective action may be a complex process with the benefits of taking such action being weighed against the risks and constraints involved in taking the action. In addition, the decision will most likely be made under difficult emergency conditions with little time available in which to act. PAGs have therefore been developed to reduce to manageable levels the decisions that must be made to protect the public in the event of a nuclear incident. The response for a given situation will be based on the State Protective Action Guides and the spectrum of possible protective action options available at the time.

A Protective Action Guide under no circumstances implies an acceptable dose. Since PAGs are based on projected dose, they are used only in an after-the-fact effort to minimize the risk from an event that is occurring or has already occurred. In some situations, protective actions may be indicated at levels lower than the PAGs.

9.2 Application of Protective Action Guides for Exposure to Airborne Radioactive Materials

Following a nuclear incident involving a release of airborne materials, urgent action is required to protect the population from inhalation of radioactive materials and from direct whole body exposure to gamma radiation. Therefore, it may be necessary to initiate early protective actions on the basis of dose projections provided by the Emergency Radiation Safety Officer, followed by appropriate adjustments to these actions based on more detailed environmental measurements.

The gaseous portion of a radioactive release may consist of noble gases and/or vapors such as radioiodines. Inhalation of noble gases will not result in a dose as large as that from whole body external exposure and therefore need not be considered as a separate contributor to inhalation exposure. The principal inhalation dose will be from the iodines and particulate material in the plume. Due to the ability of the thyroid to concentrate iodine, the thyroid dose resulting from inhalation of radioiodines may be several times greater than the corresponding whole body external gamma dose that would be received.

The 3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation may recommend any of the actions mentioned below or others not mentioned. Such recommendations will be based upon consideration of the best available information relative to projected dose and the fiscal and societal cost of taking action.

Table EP-1: Protective Actions to be Considered for Whole Body and Thyroid Dose from Exposure to Airborne Radioactive Materials

Projected Dose (rem)[1]	Actions Considered	Comments
General Population: Whole Body ≤ 1 Thyroid ≤ 1	No specific action required. Considerations may be given to: 1. An advisory to seek shelter and await further instructions. 2. Voluntary evacuation. 3. Monitor environmental radiation levels.	Monitoring will be continued until the extent of the incident can be determined and the situation fully evaluated. *KDHE 28-35-193 and EPA guidance indicates up to 5 rem thyroid in this PAG; KDHE SOP DHE/REP40 (9/2002) specifies 1 rem thyroid threshold*
General Population: Whole Body 1 to 5 Thyroid 1 to 5	1. An advisory to seek shelter and await further instructions. 2. Evacuation of children and pregnant women. 3. Control access. 4. Monitor environmental radiation levels.	Monitoring will be initiated so the projected dose can be more accurately determined and initial actions reevaluated for possible adjustments. The situation will be evaluated continually as more data becomes available.
General Population: Whole Body > 5 Thyroid > 5	1. Mandatory evacuation of populations in a predetermined area. 2. Seeking shelter if evacuation is not immediately possible. 3. Control access.	Monitoring will be used to confirm the evacuation area or as a basis for necessary adjustments in the evacuation area.
Emergency Team Workers: Whole Body ≤ 5 Thyroid ≤ 10	Control exposure of emergency team workers through the use of time, distance, shielding, job assignment, prophylaxis and respirators.	Monitoring will be used to regulate exposure levels.
Emergency Team Workers: Whole Body ≤ 10 Thyroid ≤ 10	Control exposure of emergency team workers through the use of time, distance, shielding, job assignment, prophylaxis and respirators.	For the protection of valuable property and only if lower dose is not practicable.
Emergency Team Workers: (Life Saving) Whole Body[2] ≤ 25	Control exposure of emergency team workers through the use of time and task assignment.	For life saving or protection of large populations and only if lower dose is not practicable.
Emergency Team Workers: (Volunteer Life Saving) Whole Body[2] > 25	This is only on a voluntary basis to persons fully aware of the risks involved.	For life saving or protection of large populations. This is a last resort only.

[1] Whole Body dose is to be calculated in rem as TEDE. Committed dose equivalents to the skin may be 50 times larger. Thyroid is to be the committed dose equivalent from radioiodine in rem.

[2] No specific upper limit is given for thyroid dose since in lifesaving activities, complete thyroid loss might be an acceptable sacrifice if a life can be saved. However, this should not be necessary if respirators and/or thyroid protection for rescue personnel are available as a result of adequate planning.

10. EMERGENCY FACILITIES AND EQUIPMENT

This is a listing of the facilities and equipment that should be available in the event of an emergency.

10.1 Emergency Operations Centers

During an Unusual Event or an Alert, the Emergency Operations Center (EOC) should be the Reactor Control Room, located in Room 109 Ward Hall (see Figure EP. 5 through Figure EP. 6 for maps). If the situation warrants, the alternate location is the Conference Room located in Room 137 Ward Hall. If necessary, the location shall be moved to a sheltered location upwind from the Hot Zone and shall be maintained outside of the Warm Zone. In an emergency involving smoke or fire, the EOC shall be located in an appropriate location outside of Ward Hall. An Emergency Supply Cart is located in Room 1143A (changing from 143A) in Rathbone Hall. UPD shall have a key for access to this room.
The Kansas EOC is located in the lower level of the State Defense Building in Topeka, Kansas. It is a facility from which key State officials can exercise direction and control in all emergencies. Within the EOC is an Emergency Communications Center that includes all those communications facilities (telephone, radio, teletype) necessary for the State to communicate with subordinate entities as well as with echelons of State government, and with the Federal Government at regional and national levels.

10.2 Communications

The primary communications link between the Reactor Facility Control Center and offsite emergency agencies is by telephone. The telephone system is used on a daily basis and is used to perform security system testing. A secondary link is by telephone through UPD, RCPD, and the State Highway Patrol. Telephones equipped with long distance access shall be located in the Reactor Control Room (109 Ward Hall) and in the Conference Room (137 Ward Hall). Emergency Notification lists shall be provided at each telephone.

The following agencies shall invite immediate call back via telephone for message verification and emergency authenticity.

- 3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
- 3.8.1 U.S. Nuclear Regulatory Commission (NRC)

The additional following information should be recorded and/or transmitted during the telephone call.

- Agency notified
- Person contacted
- Caller’s name, title, and telephone number
- Description of event, date, time, and location
- Emergency classification
- Expected release of radionuclides (onsite and offsite)
- Projected offsite dose rates and/or dose commitments

10.3 Radiation Monitoring Equipment

<u>10.3.1 Reactor Instrumentation and Monitors</u>
Area radiation monitors continuously monitor the Reactor Facility. These are G-M detectors, with some set to alarm at exposure rates of 10 mR/h and some at 100 mR/h. The Reactor tank is monitored at the water surface by a G-M detector set to alarm at 100 mR/h. Area radiation monitors are function-tested each operating day, response-tested quarterly, and calibrated semiannually. All the aforementioned alarms annunciate in the Control Room. A high level G-M detector at the reactor 22’ level, alarms at an

exposure rate of 5 R/h. A continuous air monitor, sensitive to radioiodine, alarms at high effluent concentrations in air of iodine-131. Thin-window G-M detectors are used at each exit to the Reactor Facility to monitor for a β-γ skin contamination. A special area monitor, located at the main exit to the Reactor Bay, alarms locally at an exposure rate of 2.5 mR/h. Three independent channels, one fission chamber and two boron-lined ion chambers monitor reactor power. Reactor fuel temperature and primary coolant activity are each monitored by two independent channels. An abnormally low level of shielding water is indicated by a visual alarm located so as to be seen at either the entrance to the Control Room or the Reactor Bay. Primary coolant temperature is monitored by one channel. All power-related data is displayed in the Control Room. Correlation of monitor readings with quantities of released radionuclides and methods for offsite dose projections are described in the Emergency Plan Procedures. In the event that fixed monitors are inoperable or off-scale, readings should be taken using portable monitors.

10.3.2 Portable Radiation Monitors

A wide variety of portable radiation monitors are available for use within the Reactor Facility both routinely and in the event that fixed monitors are inoperable or off-scale. Representative type and functions are summarized below.

Type	Radiation Detected	Range
G-M Detectors (Low Range)	β-γ	0.01-200 mR/h (γ)
Scintillation Detector	γ	0.01-10.0 mR/h (γ)
G-M Detectors (High Range)	β-γ	1-1000 mR/h (γ)
Ionization Chambers	β-γ	10 μR/h-500 R/h (γ)
Neutron Scintillator	n	0-1 rem/h
Rem Ball	n	0-500 mrem

The portable radiation monitors designated for emergency use shall be inspected, calibrated, and checked for operability semi-annually.

10.3.3 Sampling Equipment

An air sampler is available, when needed, to monitor separately for radioiodine and airborne particulate radionuclides. Filters (for particulate) and charcoal cartridges (for radioiodine) may be removed for radioassay (see below). An air tube leading from the Control Room into the Reactor Bay is also installed for taking air samples.

10.3.4. Instrumentation for Specific Radionuclide Identification and Analysis

Laboratories of the University Radiation Safety Office and the Mechanical and Nuclear Engineering Department are well equipped for the following types of analysis:

α-β Counting	Gas-Flow Proportional Counters
α-β-γ Analysis	Liquid Scintillation Spectrometer
γ Spectroscopy	Pulse-Height Analyzers with HPGe and NaI Detectors
α Spectroscopy	Si Surface Barrier Detector

10.3.5 Personnel Monitoring Equipment

Personal dosimetry available at the Reactor Facility consists of, but is not limited to, electronic dosimeters, pocket ionization chambers, film badges, and Luxel personal dosimbers.

Dosimeters for use during an emergency located in the Emergency Supply Cabinets and Cart shall be response checked biennially.

10.4 Emergency Indication and Protection Equipment

10.4.1 Fire

Fire within the operations boundary would be sensed by two systems. There are combustion-product and photo-ionization detectors located in the Control Room and at several locations in the Reactor Bay. Activation of the detectors is annunciated, as is an alarm at UPD Dispatch. Upon an alarm, security personnel are dispatched immediately to the Reactor Facility.

A sprinkler system is installed throughout Ward Hall along with a separate system in the Reactor Bay. The Reactor's system has a main shutoff valve located on the west side of the 0' level above the emergency shower. This valve shall be checked in the open position on a quarterly basis and after any maintenance involving the sprinkler system.

Several fire extinguishers are also available throughout the Reactor Facility. Carbon Dioxide (CO_2) extinguishers are located in the hallway outside the Control Room, in the Northwest corner of the 22' level, in the Southeast side of the 0' level, and in the Northwest corner of the 12' level. Dry Chemical extinguishers are located on the West side of the 12' level, in the Southwest side of the 0' level, and at the base of the stairway on the 0' level. There is also a Halon extinguisher located in the Control Room. These extinguishers shall be checked monthly to ensure that they are fully charged. The Department of Environmental Health and Safety shall be responsible for recharging, inspection, and testing. The Reactor Staff shall ensure that the fire extinguishers are weighed and inspected on a nominal annual basis. Figure EP. 5 through Figure EP. 6 (pages 122-123) show where these and additional extinguishers are located throughout Ward Hall.

10.4.2 Security

A Security Intrusion system should sound an audible alarm inside the Reactor Facility and shall notify UPD Dispatch upon alarm. The system shall be tested weekly between the Reactor Facility and UPD Dispatch. with a monthly test occurring with the presence of a UPD Officer. Additional details are included in the Physical Security Plan.

10.4.3 Earthquake Sensors

The Reactor is not equipped with any kind of earthquake sensor. If an earthquake is felt during operations the Reactor will be immediately secured. UPD will notify the Reactor Staff if they receive any notifications concerning earthquakes in the area.

10.5 FACILITIES

10.5.1 Decontamination

The Reactor Bay has an emergency shower and a sink located on the west side of the 0' level (see Figure EP. 7, page 124). An additional shower is located in the basement restroom (Ward 007 – see Figure 122, page 123). Additional supplies are located in the Emergency Supply Cabinets.

10.5.2 Emergency Supply Cabinets and Cart

The Emergency Supply Cabinets and Cart contains supplies necessary for entering a contaminated area, restricting access, and monitoring personnel. An Emergency Supply Cabinets are located at the end of the hallway at the North exit and in the basement hallway of Ward Hall. An Emergency Supply Cart is located in Room 1143A (changing from 143A) of Rathbone Hall which contains additional supplies for setting up a remote Emergency Operations Center. These cabinets and cart shall be inventoried quarterly. A revised list of the contents shall be posted on the Emergency Supply Cabinet or Cart with an additional copy placed with the Control Room copy of the Emergency Plan. The respirators shall be inspected quarterly for tightness of connections and the quality of the rubber. Minimum contents and quantities of emergency supply caches are listed below.

Cart	1st Floor	Basement	Item
1	1	1	Dosimeter Chargers
3	3	3	Pocket Dosimeters (0-200mR)
--	--	1	High Volume Air Sampler with Spare Filters
1	1	1	Beta/Gamma Survey Meter
1	1	1	Water Sample Bottle
1	1	1	#1 Swipes (box)
1	1	1	Full-Face Respirator with Spare Filters
1	1	1	Dust Mask with Spare Filters
4	4	8	High Top Polylaminated Booties (pr)
4	4	8	Low Top Tyvek Booties (pr)
--	--	4	Cloth Booties (pr)
--	--	4	Tyvek Lab Coats
2	2	4	Tyvek Pants (pr)
2	2	4	Tyvek Shirts (pr)
2	2	4	Tyvek Shoe Covers (pr)
2	2	4	Tyvek Hoods
--	1	1	Wet Suit (Top and Bottom with Hood)
2	2	2	Disposable Coveralls
Cart	1st Floor	Basement	Item
--	2	4	PVC Boots (pr)
2	4	4	Rubber Gloves (pr)
2	4	4	Glove Liners (pr)
1	1	1	“Radiation Area” Barrier Tape (roll)
1	1	1	Duct Tape (roll)
1	1	1	Radiation (sticky) Tape (roll)
1	1	1	Radiation Rope, Warning Signs, and Labels (set)
2	2	2	Large Plastic Bags
1	1	1	Disposable Emergency Blanket
--	--	1	Stretcher
3	2	2	Emergency Plan and Emergency Plan Procedures
3	1	1	Clipboard with Pen
1	1	1	First Aid Kit

10.5.3 First Aid Kits

First Aid Kits are to be located in the Emergency Supply Cabinets and Cart along with one located next to the sink on the 0' level of the Reactor Bay and another located in the Control Room. The contents are to be checked annually with expired or aging contents replaced. Minimum contents and quantities are listed below.

Quantity	Item
1	Assorted Plasters (Packet of 20)
2	No. 16 Sterile Dressings (Eye Pads)
6	No. 8 Sterile Dressings (Medium)
2	No. 9 Sterile Dressings (Large)
2	Triangular Bandages
1	Safety Pins (Packet of 6)
10	Nitrile Disposable Gloves (Not Latex)
1	Resusci Face Shield
1	Tincture of Benzoin
1	Neosporin
1	Tweezers
1	Hydrogen Peroxide
2	Ice Packs
2	Hot Packs

11. EMERGENCY PREPAREDNESS

The Reactor Manager is responsible for emergency preparedness planning, updating of emergency plans and procedures, and coordination of plans with other applicable organizations.

11.1 Training, Drills and Exercises

11.1.1 Reactor Staff

The Reactor Staff (Reactor Operators and Senior Reactor Operators) shall undergo training in Emergency Plan Procedures as part of their preparation for initial NRC operator licensing and shall review the Emergency Plan Procedures as part of the Reactor Facility Requalification Program. Training and evaluation are the responsibility of the Reactor Manager.

Training for Reactor Staff incorporates annual lectures, drills, and written examinations over the Emergency Plan and Emergency Plan Procedures, with special attention given to the following topics:

Emergency notification
Radiation surveys
Decontamination
Evacuation procedures
Personnel accountability
Re-entry and recovery operations

As part of the requalification training program, drills shall be carried out to test, on a rotating basis, at least two of the following:

Radiological Surveys and Accident Assessment
Contamination Control, Decontamination and Personnel Monitoring
Evacuation of Onsite Areas and Personnel Accountability
Medical Incidents
Fire Control

These drills shall be executed by the Reactor Manager and observed and evaluated by the University Radiation Safety Officer. Drills shall test onsite communications as appropriate, emergency supplies, and record keeping procedures. Drills shall be on a nominal annual basis, with no more than 15 months, and may be held in conjunction with training sessions of offsite support personnel.

Emergency exercises, which may incorporate drills, shall be carried out to simulate an Alert situation. The exercise must test, as a minimum, communication links and notification procedures with offsite agencies. Preparation of the scenario and execution of the exercise is the responsibility of the Reactor Manager who must coordinate with the Riley County Emergency Preparedness Coordinator and the Technological Hazards Administrator (with, if possible, observers from each organization). Exercises shall be on a nominal biennial basis, with no more than 30 months between exercises. They may be held in conjunction with training sessions for offsite personnel.

Within one week of the execution of a drill or exercise, observers and participating supervisory personnel shall confer to critique the operation. The Reactor Manager shall prepare a written record of the critique. The record shall be transmitted to the Reactor Safeguards Committee and shall be taken into account in the revision of Emergency Plan and Emergency Plan Procedures.

11.1.2 Offsite Emergency Personnel

Offsite emergency personnel shall be invited to take part in appropriate drills and exercises. Agencies invited shall be those listed in Sections 0 and 0 as well as emergency-room personnel of local hospitals.

Training for offsite personnel incorporates lectures, drills and exercises, with special attention given to the following topics:

- External and internal radiation hazards
- Radiation surveys of personnel and equipment
- Contamination control and decontamination
- Protective actions
- First-aid and patient treatment

These training sessions shall be on a biennial basis with duration of approximately one-half day. The drills and exercised should be designed to maintain proficiency in fire, security, and medical incidents, such that exercises requiring similar response do not occur in subsequent years. (I.e., a year with an emergency exercise featuring a fire should be followed by an exercise featuring a medical or security incident).

11.2 Emergency Plan and Procedures Review

The Emergency Plan shall be reviewed biennially (30 month maximum interval) by the Reactor Manager with the review being reported to the Reactor Safeguards Committee in the Manager's Audit. Revisions will take into account changes in the Reactor Facility and its surroundings as well as comments by participants and observers in drills and exercises. The Reactor Safeguards Committee must approve any substantive changes in the Emergency Plan. The same policy applies to agreements with offsite agencies and to the Emergency Plan Procedures, except for the contact points on the Emergency Notification List. The Emergency Notification List shall be reviewed annually (15 month maximum) or whenever Reactor Staff incurs personnel changes. Reactor Safeguards Committee approval is not required for changes in names or telephone numbers.

11.2.1 Distribution

Copies of the Emergency Plan and Procedures shall be issued to the list on the following page in order to maintain consistency between emergency agencies and the Reactor Facility. This shall coincide within 75 days of the biennial review and shall include copies of all current agreement letters. A cover letter shall be included indicating any changes made along with a request to discard any previous versions. Changes to the Emergency Notification List (EPP-2) outside of the review period shall be issued.

All Reactor Staff shall be issued a minimum of two (2) copies of the current Emergency Plan and Procedures. One copy should remain in their residence with the other either in their vehicle or in their book bag. It is not recommended that these be stored in their office.

Distribution List

Copies	Agency/Location
2	Reactor Control Room
2	1st Floor Emergency Supply Cabinet
2	Basement Emergency Supply Cabinet
3	Mobile Emergency Cart – Rathbone
1	Emergency Telephone – Reactor Control Room
1	Emergency Telephone – Conference Room (137 Ward Hall)
7	Reactor Safeguards Committee
2 each	Reactor Staff
5	3.5.3 Kansas State University Police Department (UPD)
1	3.6.1 Riley County Emergency Management
6	3.6.2 Manhattan Fire Department (MFD)
2	3.6.3 Riley County Police Department (RCPD)
1	3.6.4 Riley County Emergency Medical Service (EMS)
1	3.6.5 Mercy Health Center
3	3.7.1 Kansas Division of Emergency Management (KDEM), Technological Hazards Administrator
3	3.7.2 Kansas Department of Health and Environment (KDHE), Bureau of Air and Radiation
	3.7.3 Kansas Highway Patrol (KHP)
1	Emergency Operations (Topeka)
1	Troop C Headquarters (Salina)
1	℅ Carl Jones (Salina)
	3.8.1 U.S. Nuclear Regulatory Commission (NRC)
1	Office of Richard Wessman
1	Document Control Desk
1	Non-power Licensing Division
2	3.8.2 774th Ordinance Company (EOD), Fort Riley
1	University of Kansas, Lawrence – Radiation Safety Officer
	University of Missouri, Columbia
1	MURR
1	Campus Radiation Safety

12. NOTES TO REACTOR STAFF

Listed below are some general notes for the Reactor Staff.

STAY CALM, Use Your Head
Certain situations can be very stressful but if you are not calm then they just get worse. Remember to breath and to enter the situation using your head. We got this job because we are smart and we can handle the odd situations thrown at us. Solve the problem!

Good Operating Procedures
When alone or while operating at night it is advised that the Control Room door remain closed. If someone unknown shows up don't hesitate to ask for identification and their reason for being there.

People/Situations That Make You Uncomfortable
Don't call the council – Call 911. Anyone or anything that doesn't sit perfectly well with you should be reported immediately. If you hesitate then the police may not be able to help since the person may be long gone. Let them determine whether there is a problem or not. Better safe than sorry!

Bomb Threats
Don't use radios or cell phones until you are clear of the building! Inform people around you but don't cause hysteria. It is the job of the Emergency Director to decide whether or not to evacuate the building.

Arriving on the Scene
If you are called to an emergency don't park next to the building. It is preferred that you park as far away as possible since other emergency vehicles may need your parking space. In addition, if something does happen to the building then at least your vehicle won't be harmed. Also, don't rush in – find out where the Emergency Operations Center (EOC) is located and ask what you can do.

Radiation Knowledge
The Manhattan Fire Department does have basic radiological training but should still be given guidance as to limitations to observe. They should have at least one Civil Defense (CD) G-M detector in each truck. The University Police also have a couple of detectors available but do not have radiological training. In order to make sure the boundaries are established properly tell them to watch the meter and if it reaches 2 mR/h then they need to move the line further back. EMS has enough understanding of radiation to keep away from it so be sure to explain any contamination or exposure to them thoroughly so proper medical attention can be given to the patient.

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Multiplication and Criticality

CRITICALITY

Nuclear fuels must have the ability to multiply neutrons in order to achieve a critical configuration. A critical configuration is reached when the quantity k_{eff} reaches unity. This quantity is commonly expressed in the form of the six-factor formula:

$k_{eff} = \eta \epsilon p f P_f P_{th}$

Before we consider each of these terms in detail, it is instructive to examine the neutron life-cycle in the reactor.

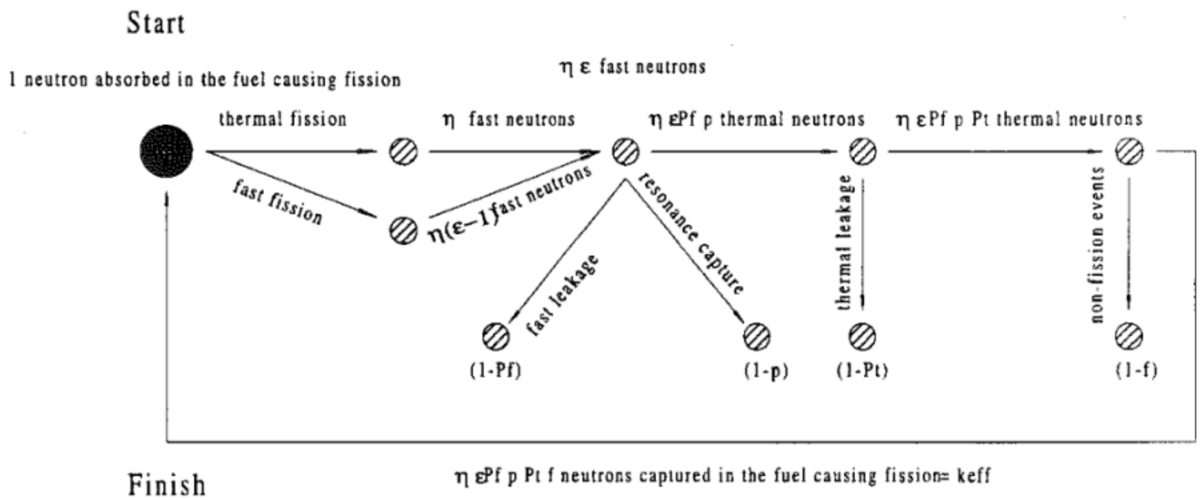

Figure 16- Neutron life cycle in a reactor

Following this cycle through, we start with one neutron absorbed in the fuel. Thermal fission leads to η fission neutrons per neutron capture. Correction for fast neutron-induced fission reveals that a total of ηϵ fission neutrons are produced, where η is the fast fission factor. These fast neutrons must then slow down to thermal energies. P_f fast neutrons escape leakage and p neutrons escape resonance capture. At this point we have $\eta\epsilon P_f p$ thermal neutrons. The fraction P_{th} of the thermal neutrons then escapes leakage. A fraction f of these neutrons is then captured in the fuel. Thus the overall change from generation to generation, given by the effective multiplication factor k_{eff}, is found by solving the six factor formula. Each of the terms in the six factor formula can be individually defined in terms of other quantities.

The **reproduction factor η** is the number of neutrons yielded per thermal neutron absorbed in the fuel and can be computed as

$$\eta = \frac{\nu \Sigma_f}{\Sigma_a}$$

The reproduction factor is typically much higher than unity (η = 2.07 for pure ^{235}U), and is determined by the properties of the fissile isotope and the enrichment of the fuel.

The **fast fission factor** ϵ accounts for neutrons yielded by fast neutron absorption in the fuel. Its calculation requires knowledge of the energy dependence of the neutron flux, fission cross section, and number of neutrons per fission. The fast fission factor is always slightly greater than one. For TRIGA reactors, $\epsilon \approx 1.05$.

The **fast non-leakage probability** P_f is the probability that fast neutrons will reach thermal energies without leaking from the core. For a bare homogeneous reactor using the age diffusion model, P_f is related to the geometric buckling B^2 (a measure of the second derivative of the reactor flux distribution) and the Fermi age to thermal τ (a measure of the area traveled by a neutron during slowing down):

$$P_f = e^{-B^2 \tau}$$

The **resonance escape probability** p accounts for the number of neutrons that escape capture in the many resonance peaks in the ^{238}U cross section. To evaluate p, one needs knowledge of the energy dependent resonance functions. For TRIGA reactors, p is approximately 0.88.

The **thermal non-leakage probability** P_{th} is the probability that thermal neutrons will not escape from the reactor core before being captured. For a bare homogeneous reactor, P_{th} is related to the buckling and diffusion area L^2 for thermal neutrons as

$$P_{th} = \frac{1}{1 + L^2 B^2}$$

The **thermal utilization factor** f is the ratio of thermal neutrons absorbed in the fuel to total thermal neutron absorptions. For a homogeneous reactor:

$$f = \frac{\Sigma_a(fuel)}{\Sigma_a(fuel) + \Sigma_a(non-fuel)}$$

The **reactivity ρ** of the core is the fractional deviation of k_{eff} from unity:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

Reactivity can be expressed in terms of dollars, where one dollar is equal to the effective delayed neutron fraction β_{eff}, 0.007 for the TRIGA. The reactivity in units of dollars is found by dividing ρ by β_{eff}.

MULTIPLICATION

Neutron multiplication also leads to an essential tool for determining reactivity changes in subcritical assemblies, where $k_{eff}<1$. Unexpected criticalities can have disastrous results. Fortunately, we can measure how close an assembly is to a critical configuration by simply using a neutron source and a detector. By placing the source and detector on opposite sides of the assembly, we can obtain a relative measure of the neutron multiplication from the detector count rate.

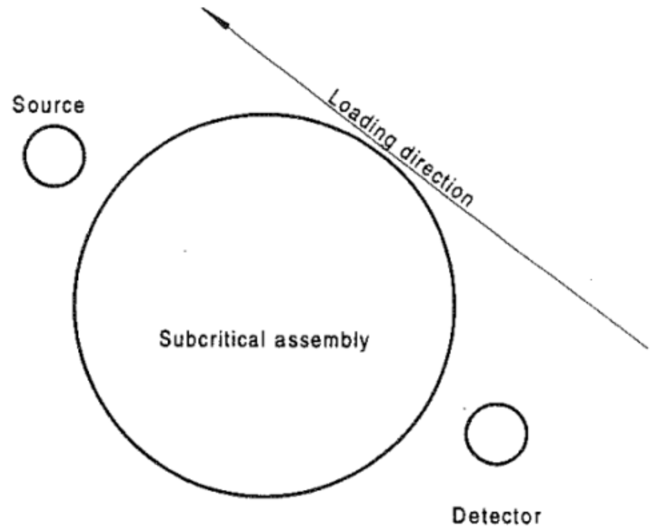

Figure 17 - Measurement of neutron multiplication in a subcritical assembly

All fissile isotopes multiply neutrons. For ^{235}U, fission releases an average of $\nu = 2.42$ neutrons for every fission reaction, which are then multiplied further if these neutrons then cause fission of other ^{235}U atoms. If a source is present resulting in Q neutrons being thermalized in the assembly every second, then there will be a fission rate F (s^{-1}). This fission rate is dependent on k_{eff} and provides a measure of how much multiplication we have in the reactor core. Therefore, we can define the neutron multiplication M as

$$M = \frac{\nu F + Q}{Q}$$

where the quantity νF is the number of neutrons per second appearing at any one instant, all arising from chains begun in all previous generations. The most recent generation produces Qk_{eff} neutrons, the previous generation produces $Qk_{eff}k_{eff}$, and so on. Thus the quantity νF is given by the series:

$$\nu F = (k_{eff}Q) + \left[k_{eff}\left(k_{eff}Q\right)\right] + \left\{k_{eff}k_{eff}\left(k_{eff}Q\right)\right\} + \ldots$$

Therefore, as an assembly gets closer to a critical configuration, the neutron multiplication becomes infinite. For $k_{eff} < 1$, the series converges and the neutron multiplication M relates as

$$M = \frac{Q\left[1 + k_{eff} + k_{eff}^2 + k_{eff}^3 + \ldots\right]}{Q} = \frac{1}{1 - k_{eff}}$$

As we would expect, as k_{eff} goes to unity, M goes to infinity. Reactivity changes can similarly be found by measuring the multiplication at different subcritical configurations and utilizing the ratio

$$\frac{M_1}{M_2} = \frac{1 - k_{eff,1}}{1 - k_{eff,2}}$$

For measuring the multiplication in a reactor core, the source and detector are first placed on opposite sides of an unloaded assembly. An initial count rate C_o is measured. Fuel is added one element at a time. The count rate is measured after each fuel addition. The multiplication can then be approximated as

$$M_n = \frac{C_n}{C_o}$$

From a graph of the inverse multiplication *1/M* versus the number of elements added, one can estimate the critical fuel loading by extrapolating a line from the last two data points. When the reactor approaches critical ($k_{eff} = 1$), the multiplication factor goes to infinity. Therefore the point at which the extrapolation line of the inverse multiplication factor data points intersects the x-axis is an approximation of the expected critical loading.

For conservatism, it is common practice to only load half of the expected number of fuel elements required to go critical, and then to construct a new extrapolation line. It is also conservative to start loading fuel near the detector and proceeding toward the source.

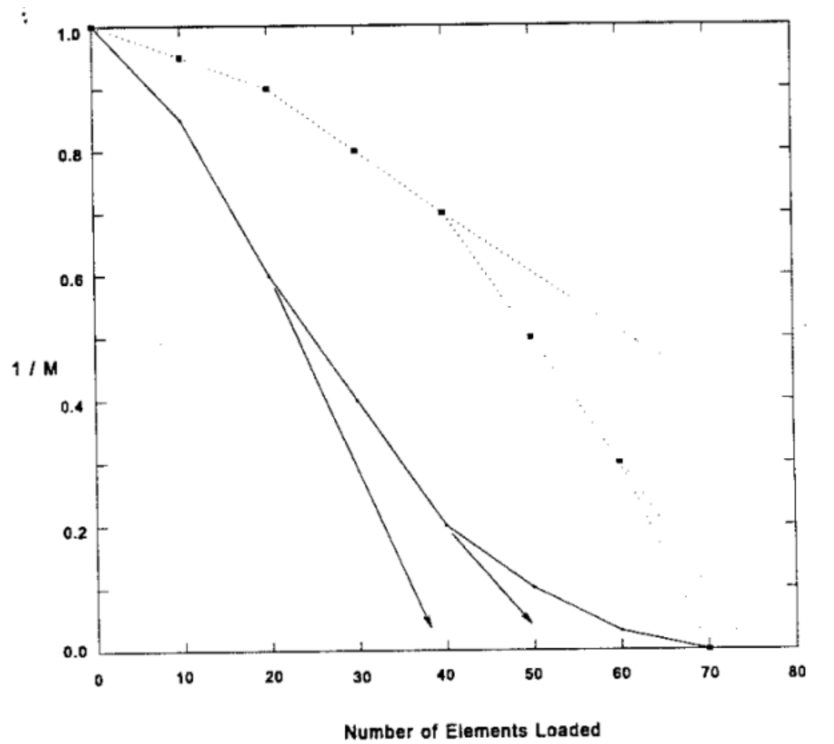

Figure 18 - Plot of inverse multiplication factor 1/M versus fuel loading for a TRIGA. Solid line - loading begins near detector, proceeds toward source. Dashed line – loading begins near source, proceeds toward detector.

Period and Reactivity

INTRODUCTION

The purpose of this discussion is to develop the theoretical support for the relationship between the stable or asymptotic period of a reactor and the reactivity responsible for that period. The relationship developed applies to the variation of the reactor power with time after a step change is made in reactivity from an initially stead state critical condition ($\rho = 0$; $k_{eff} = 1$). The period is the time required for the reactor power to increase by a factor of *e* (2.718…). As will be seen, a step increase in reactivity leads, after a short transient period, to exponentially increasing power, i.e., to a constant or stable and positive period.

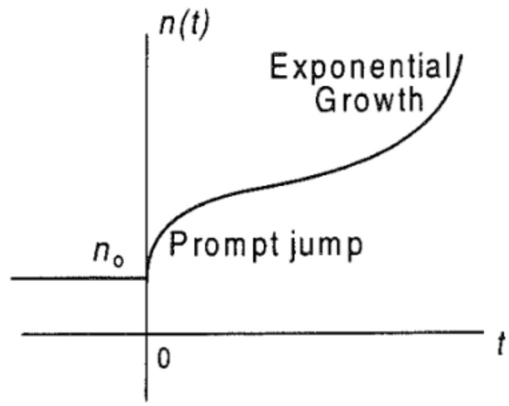

Figure 19 - Reactor power following a step increase in reactivity

The relationship is known historically as the "inhour" equation. It applies only to the reactor operating in the "zero-power" region, ie., in the absence of temperature feedback. In the case of the TRIGA reactor, the relationship is valid effectively for thermal power less than 1000 W. Furthermore, the relationship applies strictly only to reactor operation without an external source of neutrons. Practically, source effects are significant only at relatively low power. For the TRIGA, the relationship is valid for thermal power in excess of about 1 W. Therefore, measurements are useful for thermal power between 1 W and 1000 W.

The inhour equation provides an extremely important, precise, safe, and widely used method of determining reactivity through a relatively simple period measurement. Such measurements are important in control rod calibration, analysis of poison compensation, and analysis of temperature compensation.

Notation

$n(t)$	a quantity proportional to average reactor power, average neutron flux density, or average neutron density. It is very conveniently interpreted as average neutron density (cm^{-3}).
n_o	initial neutron density (cm^{-3}), the value of $n(t)$ prior to time $t = 0$ at which a step change in reactivity is made.
T	stable reactor period (s)
$C(t)$	a quantity proportional to the average concentration of delayed neutron precursors in the reactor. Here it is conveniently interpreted as the average concentration (cm^{-3}).
C_i	Concentration of the *ith* group of precursors.
β	effective fraction of neutrons delayed in production ($\beta = 0.007$ for the TRIGA).
β_i	effective fraction of neutrons produced by the decay of the *ith* group of delayed neutron precursors.
λ_i	decay constant for the *ith* group of delayed neutron precursors (s^{-1})
ℓ^*	lifetime of one generation of neutrons (s)
ℓ	$\underline{\ell^* / k_{eff}}$, the effective lifetime of one generation of neutrons (43 µs for the TRIGA, based on the 2008 SAR).
ρ	the reactivity step change made at time $t = 0$

THE POINT REACTOR KINETICS EQUATIONS (PRKE)

The term "point" refers to the fact that, in this approximate model of reactor behavior, it is assumed that the reactor power varies proportionally through its volume. Thus it is necessary to consider only an average value of the power density, etc., hence a single, or point, value.

The Reactor With No Delayed Neutrons

Consider a length of time Δt equal to the lifetime of one generation of neutrons. According to the definition of the effective multiplication factor,

$$n(t+\Delta t) - n(t) = k_{eff}\, n(t) - n(t)$$

$$\frac{dn}{dt} = \lim_{\Delta t \to 0} \frac{n(t+\Delta t) - n(t)}{\Delta t} = \frac{nk_{eff} - n}{\ell *} = \frac{n}{\ell}\rho \qquad (1)$$

$$n(t) = n_o e^{\frac{\rho t}{\ell}}$$

The stable period in the last equation is $T = \ell / \rho$. For $\rho = 0.0035$, which is not an unusual value in reactor operation, with $\ell = 43$ μs, the stable period is only 12 milliseconds – far too short for successful control. Fortunately, not all neutrons are prompt. The effective fraction of delayed neutrons is $\beta = 0.007$. These delayed neutrons play a key role in reactor control and stability.

The Reactor With a Single Group of Delayed Neutrons

Here the approximation is made that all delayed neutrons are released from a single precursor, with a representative decay constant of $\lambda = 0.08\ s^{-1}$. This approximation can be useful in illustrating the differential equations characterizing reactor behavior. Equation 1 needs to be modified as follows:

$$\frac{dn}{dt} = \frac{nk_{eff}(1-\beta) - n}{\ell *} = \frac{n}{\ell}(\rho - \beta) + \lambda C \qquad (2)$$

That is, the source of prompt neutrons is reduced by the factor of (1-β). Delayed neutrons appear with the source term λC. There is a corresponding differential equation describing the behavior of the delayed neutron precursors, namely,

$$\frac{dC}{dt} = \frac{\beta n}{\ell} - \lambda C \qquad (3)$$

Note that at steady state both derivatives are zero. This requires that $\rho = 0$, and that

$$C_o = \frac{\beta n_o}{\ell} \qquad (4)$$

The Reactor With Six Groups of Delayed Neutrons

It has been found experimentally that reactor behavior is well described in the framework of six groups of delayed neutrons. For a ^{235}U-fueled reactor such as the TRIGA, data in Table 1 apply. Note that, for the longest-lived group, $1/\lambda_1 = 80s$. It will be seen that this is the limiting stable period for reactor shutdown.

In other words, no matter how subcritical a reactor is, it cannot be shut down with a stable period shorter than about 80 s.

Table 1 - Group constants for thermal neutron induced fission of ^{235}U. Source: Duderstadt and Hamilton (1976).

Group	$T_{1/2}$ (seconds)	β_i / β
1	54.51	0.038
2	21.84	0.213
3	6.00	0.188
4	2.23	0.407
5	0.496	0.128
6	0.179	0.026

The reactor kinetics equations now take the form:

$$\frac{dn}{dt} = \frac{n}{\ell}(\rho - \beta) + \sum_{i=1}^{6} \lambda_i C_i \tag{5}$$

$$\frac{dC_i}{dt} = \frac{\beta_i n}{\ell} - \lambda_i C_i \,,\ i = 1 \ldots 6 \tag{6}$$

Note that at steady state all derivatives are zero. This requires that $\rho = 0$ and that

$$C_{i,0} = \frac{\beta_i n_o}{\ell} \tag{7}$$

SOLUTION OF THE PRKE FOR A STEP CHANGE IN REACTIVITY

Initial Conditions

Until time $t = 0$, the reactor is at steady state. Then a step change in reactivity from zero to $\pm\rho$ is made. Initially, $n(0) = n_o$ and initial precursor concentrations are given by Eq. 4 or 7.

Formal Solution

The PRKE are initial value problems, described by linear differential equations with constant coefficients. As such, the solution for *n(t)* is, formally,

$$n(t) = n_o \sum_j A_j e^{s_j t} \tag{8}$$

in which the constants A_j and s_j depend on ρ as well as the nuclear parameters λ_i, β_i, and ℓ. It can be shown that, for one group of delayed neutrons, there are two terms in the summation of Eq. 8. For six groups of delayed neutrons, there are seven terms. The exponential nature of the solution for *n(t)* is evidenced in Figs. 20 and 21.

Note how the same change in reactivity, when positive, has a far greater effect on the reactor power than when negative. Note too that, for positive changes in reactivity, as ρ increases, the period becomes shorter and shorter. For negative changes in reactivity, the ρ increases, the period approaches a constant value (the 80 second period mentioned above).

One could proceed with the solution of Eq. 8 and evaluate all the constants. This is not necessary, however, if one is interested only in the stable period. In that case, it is necessary to determine the controlling value of s_j, i.e., the most positive of all the values. This procedure is described in the next section, and leads to the so-called "inhour equation."

STABLE PERIOD VS. REACTIVITY

In principle, the values of s_j may be positive or negative real numbers. For this problem, all s_j are unique.

THE LAPLACE TRANSFORM OF THE PRKE SOLUTION

It is left as an exercise for the student to show that, for one group of delayed neutrons, the Laplace transform $\overline{n}(s)$ of $n(t)$ is given by

$$\overline{n}(s) = \frac{n_o\left[\ell + \dfrac{\beta}{s+\lambda}\right]}{s\ell - \rho + \dfrac{\beta s}{s+\lambda}} \tag{9}$$

Note that, when fractions are cleared in the denominator, there results a quadratic equation with two roots, s_1 and s_2. At this point it is not necessary to evaluate these roots. For six groups of delayed neutrons,

$$\overline{n}(s) = \frac{n_o\left[\ell + \sum_{i=1}^{6}\dfrac{\beta_i}{s+\lambda_i}\right]}{s\ell - \rho + \sum_{i=1}^{6}\dfrac{\beta_i s}{s+\lambda_i}} \tag{10}$$

Note that, when fractions are cleared in the denominator, there results a seventh order polynomial with roots s_1 to s_2. At this point it is not necessary to evaluate these roots.

The Heaviside Expansion Theorem for Laplace Transforms

If $\bar{f}(s)$, the Laplace transform of $f(t)$, is in the form of $p(s)/q(s)$, and $q(s)$ is an mth order polynomial $q(s) = (s-s_1)(s-s_2)\ldots(s-s_m)$ with no roots repeated, then

$$f(t) = \sum_{j=1}^{m}\frac{p(s_j)e^{s_j t}}{q'(s_j)} \tag{11}$$

in which s_j are the roots of $q(s)$, i.e., the poles of $\bar{f}(s)$, and

$$q'(s_j) = \left|\frac{dq}{ds}\right|_{s=s_j} \tag{12}$$

Thus the determination of the asymptotic time behavior of *n(t)* reduces to examination of the poles of the function $\bar{f}(s)$, or the roots of the denominator of Eq. 10, or, for one group of delayed neutrons, Eq. 9.

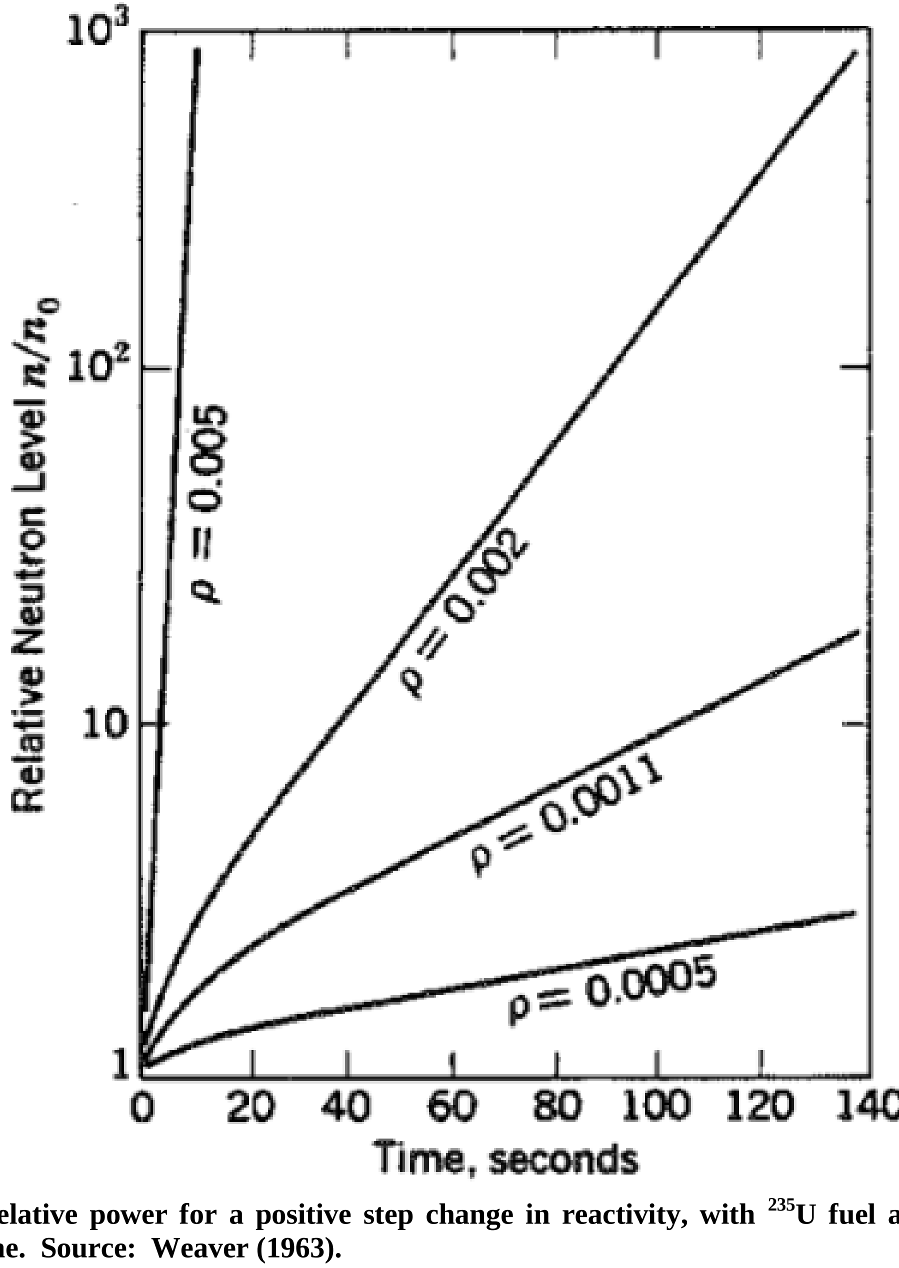

Figure 20 - Relative power for a positive step change in reactivity, with ^{235}U fuel and a 100 ms neutron lifetime. Source: Weaver (1963).

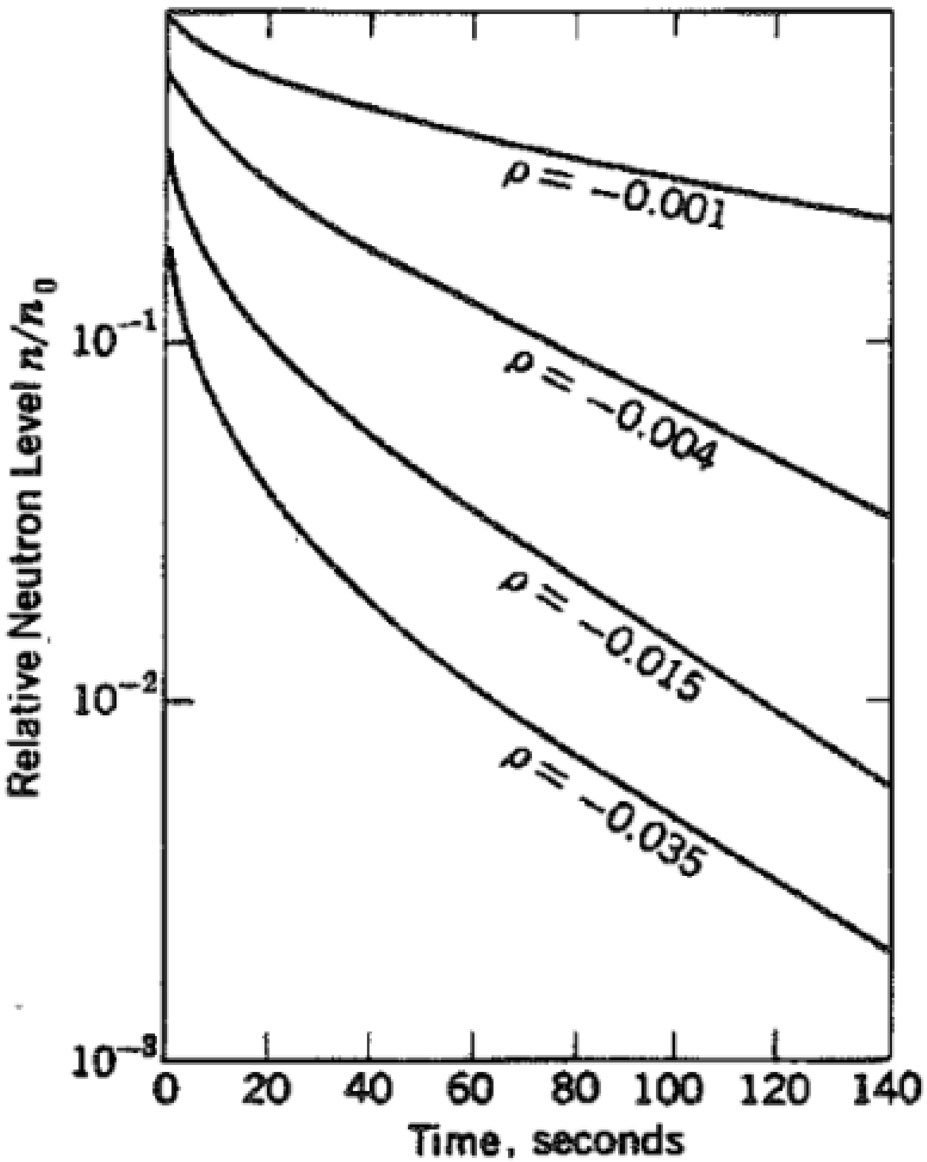

Figure 21 - Relative power for a negative step change in reactivity, with ^{235}U fuel and a 100 ms neutron lifetime. Source: Weaver (1963).

Graphical Solution for One Group of Delayed Neutrons

The poles of $\bar{f}(s)$ may be found by setting the denominator of Eq. 9 to zero, *i.e.*, setting

$$\frac{\rho}{\beta} - \frac{s\ell}{\beta} = \frac{s}{s+\lambda}, \text{ or} \qquad (13)$$
$$\text{LHS} = \text{RHS}$$

The graphical solution of this equation is illustrated in Figure 22. The RHS has two branches, depending on whether $s \geq -\lambda$ or $s < -\lambda$. The figure illustrates two cases for the LHS, one for positive reactivity, and one for negative reactivity. The intersection of the LHS with the two branches of the RHS marks two poles. The pole greatest in magnitude, i.e., most positive, and marked with s^*, is the pole that determines the asymptotic behavior of $n(t)$, i.e., the stable period T, namely,

$$\lim_{t\to\infty} n(t) = e^{s^*t} = e^{t/T} \tag{14}$$

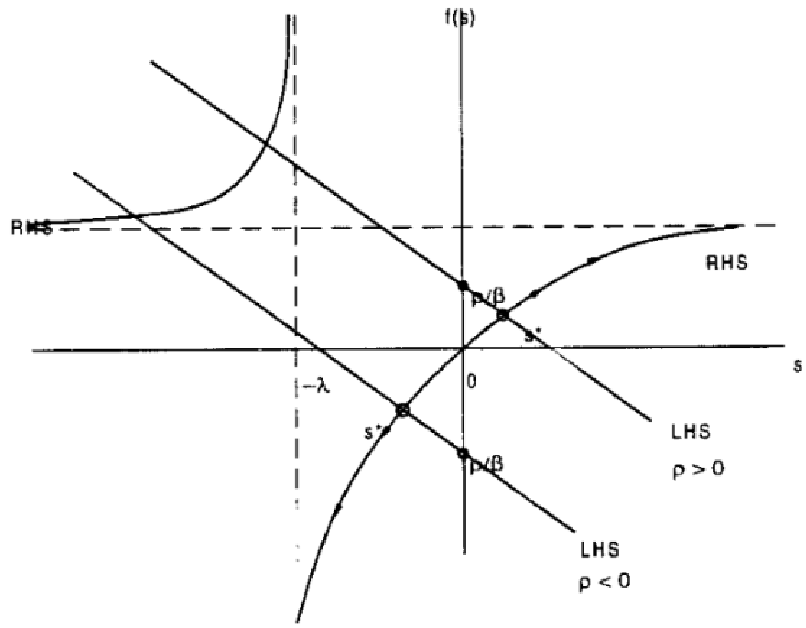

Figure 22 - Graphical determination of stable period for one group of delayed neutrons

If ρ is positive, there is one positive pole at $s = s^*$ and one negative pole. As $\rho \rightarrow +\infty$, $s^* \rightarrow +\infty$ and $T \rightarrow 0$. If ρ is negative, there are two negative poles, with the more positive at $s = s^*$. As $\rho \rightarrow -\infty$, $s^* \rightarrow -\lambda$ and $T \rightarrow -1/\lambda$.

THE INHOUR EQUATION

Since $s^* = 1/T$ must satisfy Eq. 13, it follows that

$$\frac{\rho}{\beta} = \frac{\ell}{\beta T} + \frac{1}{1+\lambda T} \tag{15}$$

Similarly, for six groups of delayed neutrons,

$$\frac{\rho}{\beta} = \frac{\ell}{\beta T} + \sum_{i=1}^{6} \frac{\beta_i / \beta}{1+\lambda_i T} \tag{16}$$

This equation is a key equation in reactor operations, reactor stability analysis and control theory, and reactor safety analysis. It relates a measured period to a causative reactivity. It allows prediction of time behavior for a given change in reactivity. The period-reactivity relationship given by this equation is displayed in Figure 23, which has been computed for the delayed neutron parameters given in Table 1 and an effective prompt neutron lifetime of 43 μs, which the TRIGA prompt neutron lifetime tabulated in the 2008 KSU Reactor Safety Analysis Report.

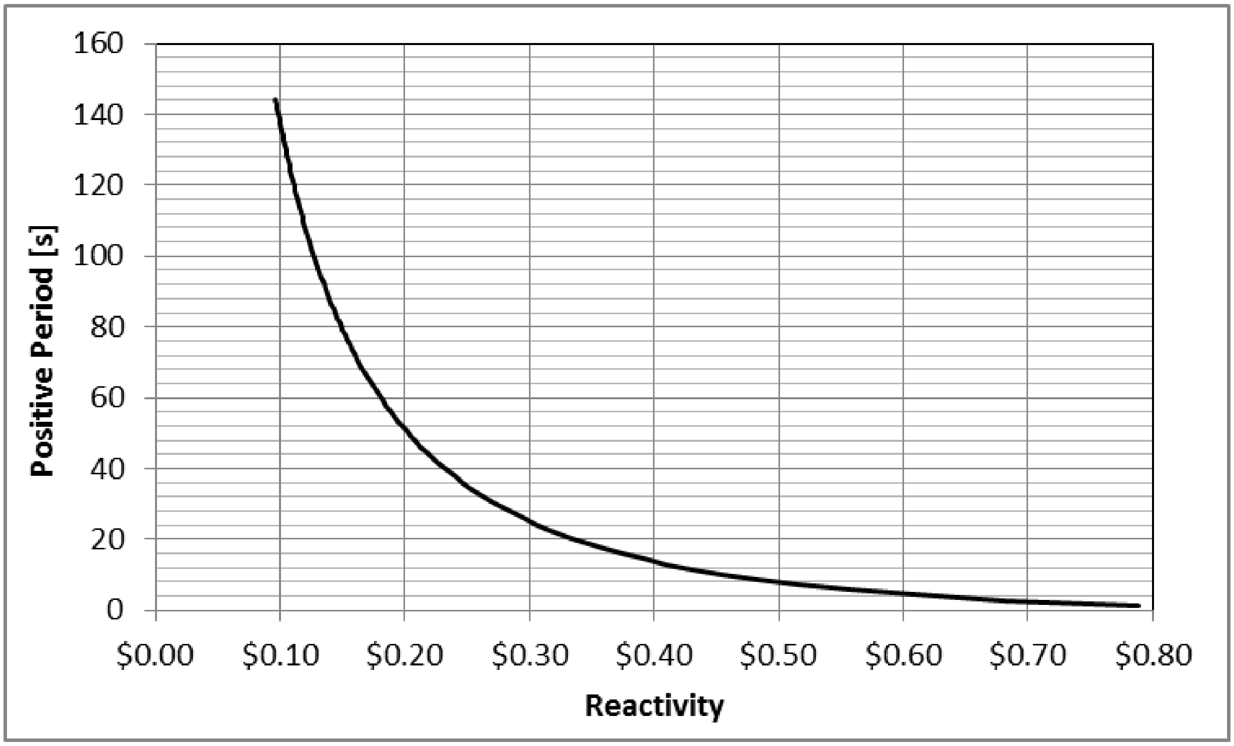

Figure 23 - Period vs. reactivity for a ^{235}U-fuelled thermal reactor

Temperature Feedback

GENERAL FORMULATION

The temperature coefficient of reactivity is defined as:

$$\alpha \equiv \frac{\partial \rho}{\partial T} \tag{1}$$

Since $\rho \equiv \dfrac{k_{eff}-1}{k_{eff}}$ and $k_{eff} \approx 1$,

$$\alpha \cong \frac{1}{k_{eff}}\frac{\partial k_{eff}}{\partial T} = \frac{1}{\eta}\frac{\partial \eta}{\partial T} + \frac{1}{\varepsilon}\frac{\partial \varepsilon}{\partial T} + \frac{1}{p}\frac{\partial p}{\partial T} + \frac{1}{f}\frac{\partial f}{\partial T} + \frac{1}{P_{Th}}\frac{\partial P_{Th}}{\partial T} + \frac{1}{P_F}\frac{\partial P_F}{\partial T} \tag{2}$$

In other words, α can be broken into six additive terms corresponding to the six terms in the effective multiplication factor, k_{eff}, namely:

$$\alpha = \alpha_\eta + \alpha_\varepsilon + \alpha_p + \alpha_f + \alpha_{P_{Th}} + \alpha_{P_F} \tag{3}$$

GENERAL EFFECTS OF TEMPERATURE ON REACTIVITY

1. Thermal expansion affects core dimensions and thus the geometric buckling.
2. Density changes affect macroscopic cross sections. Differential changes from region to region affect moderator / fuel ratios.
3. Temperature affects thermal motion and resonance absorption.
4. Temperature affects the neutron energy spectrum and thus microscopic cross sections. Effects may be different for capture and fission.

It is possible to take all of these effects into account by evaluating the partial derivatives of Eq. 2.

THERMAL PROPERTIES OF THE ZRH TRIGA FUEL MATRIX

1. H atoms are bound to Zr atoms in a tetrahedral crystal lattice.
2. Vibrational energy levels of the H atom exist at fixed intervals of ΔE = 0.013 eV above the ground state.
3. At all temperatures, most H atoms are in the ground vibrational level. However, as temperature increases, greater proportions of the H atoms exist in vibrational levels of higher energy, e.g., 0.13 eV, 0.26 eV, and so on.
4. At any temperature, interactions with H atoms in the fuel cannot reduce neutron energies below 0.13 eV. Thermalization of neutrons in the TRIGA takes place in water or graphite. Because the fission cross section decreases with increasing neutron energy, for a neutron born in the fuel to become effective at inducing fission, it must diffuse out of the fuel, become thermalized, and then diffuse into the fuel as a thermal neutron.

5. As temperature increases, thermal neutrons in the fuel are more likely to interact with H atoms in excited vibrational states. Such interactions transfer energy from the H atom to the neutron, in effect causing spectrum hardening (i.e., up-scattering of neutrons with respect to energy). The up-scattered neutrons are relatively ineffective at inducing fission. To be effective, they must again diffuse from the fuel, thermalize in the moderator or reflector, and diffuse back into the fuel, all with increasing likelihood of capture or leakage.
6. In summary, the TRIGA has a prompt and very negative temperature coefficient of reactivity. In large part, this is due to the hardening of the neutron spectrum by interaction with excited hydrogen atoms in the fuel. It results in significant hardening of the neutron energy spectrum as the fuel temperature increases. This hardening is much greater than would be expected on the basis of Maxwell-Boltzmann statistics.

REPRESENTATIVE DATA FOR THE TRIGA

The following data are for a TRIGA reactor with ℓ = 40 μs and β = 0.007 and are the results of General Atomics' calculations. Terms in k_{eff} were calculated for two conditions:

1. Fuel and moderator temperatures T_F and T_M = 23°C;
2. T_M = 23°C, T_F = 400°C, k_{eff} = 1.0000.

Individual coefficients of reactivity were calculated, for example, as:

$$\alpha_\eta \equiv \frac{1}{\eta}\frac{\partial \eta}{\partial T} \cong \frac{1}{\eta}\frac{\Delta \eta}{\Delta T} \cong \frac{1}{\eta}\frac{(\eta_2 - \eta_1)}{(T_{f2} - T_{f1})} \tag{4}$$

Non-leakage probabilities were based on age-diffusion theory, namely

$$P_f = e^{-B^2 \tau_{Th}} \text{ and } P_{Th} = \frac{1}{1 + L^2 B^2} \tag{5}$$

Fuel Temp. (°C)	23	400	
H_2O Temp. (°C)	23	23	$\alpha \cdot 10^5$
η	2.0730	2.0638	–1.18
f	0.7241	0.7115	–4.65
p	0.8768	0.8690	–2.37
ϵ	1.0520	1.0550	+0.76
k_∞	1.3845	1.3462	–7.44
τ_{th} (cm^2)	22.23	22.00	
L^2 (cm^2)	2.48	3.04	
B^2 (cm^{-2})	0.0119	0.0119	
P_f	0.7676	0.7697	+0.73
P_t	0.9713	0.9651	–1.70
k_{eff}	1.0320	1.0000	–8.41

Expressed in units of dollars, $\alpha = -8.41 \times 10^{-5}/0.007 = -\$0.012/°C$. For the KSU TRIGA, α is more nearly -$0.017/°C.

POWER COEFFICIENT OF REACTIVITY

With the TRIGA, there is a unique relation between power and average core temperature (see attached figures). For the KSU TRIGA, the power coefficient of reactivity is about -0.008 to -0.01 $ / kW.

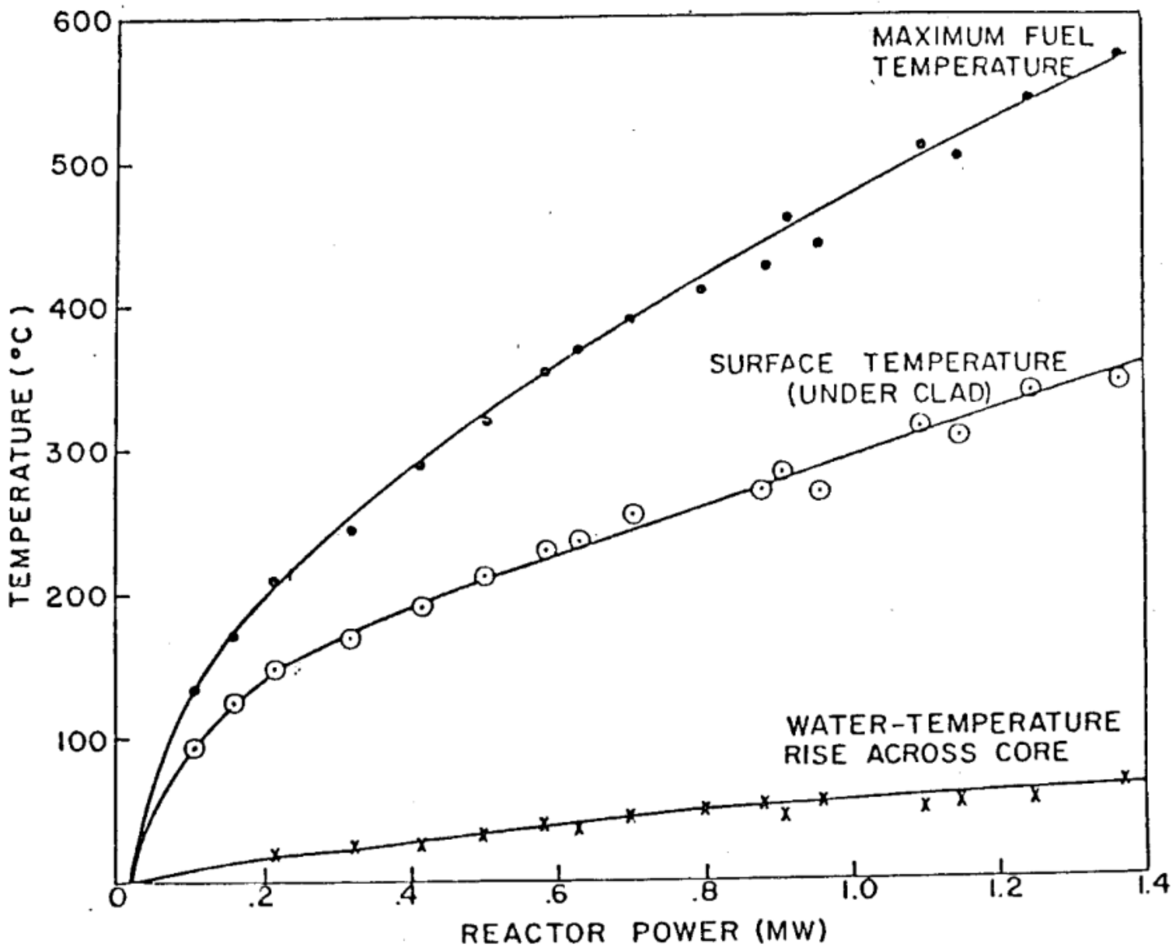

Figure 24 - Typical component temperatures in high-power quasi-equilibrium experiments [Torrey Pines TRIGA]

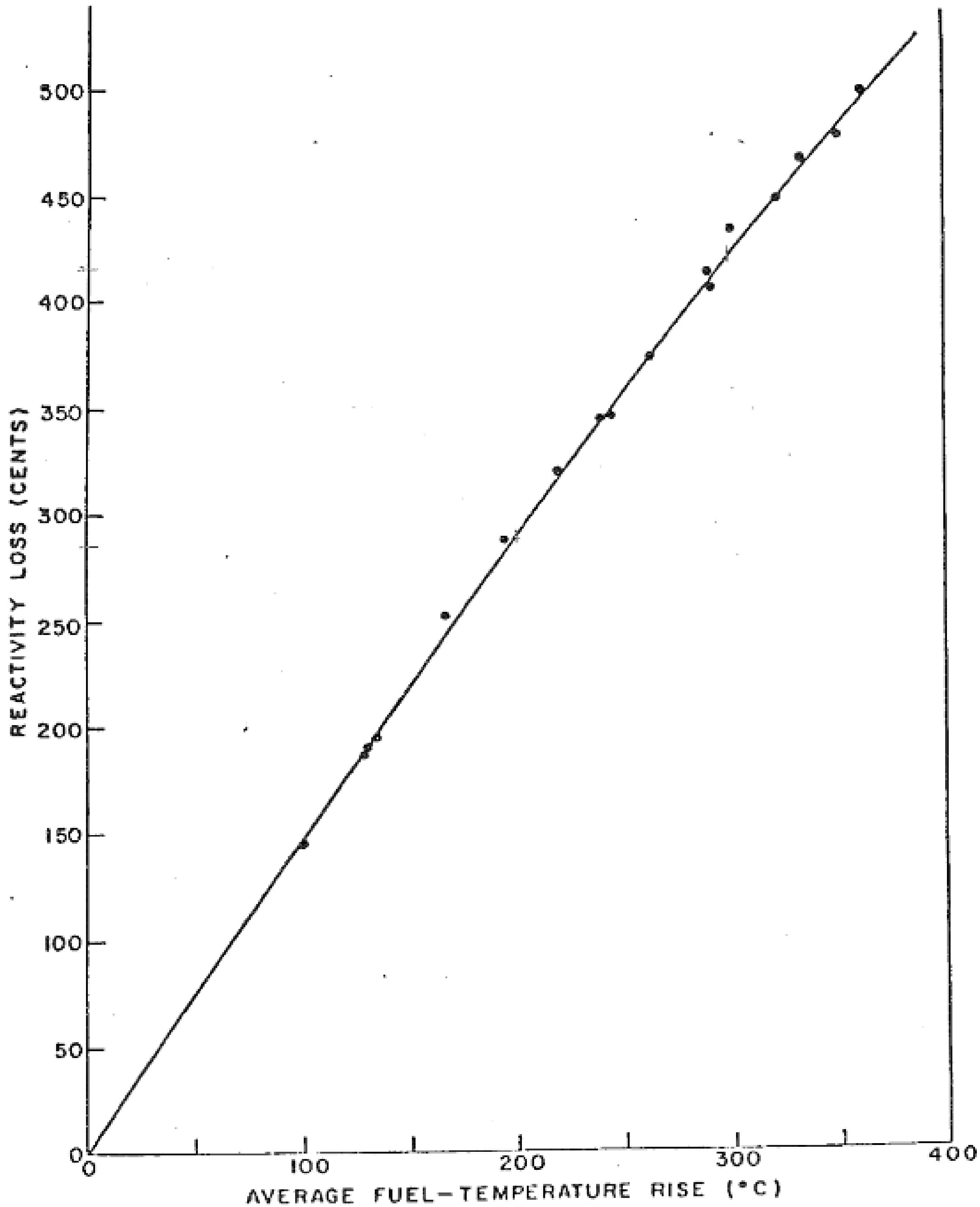

Figure 25 - Reactivity loss as a function of average fuel temperature rise with constant average core water temperature [Torrey Pines TRIGA]

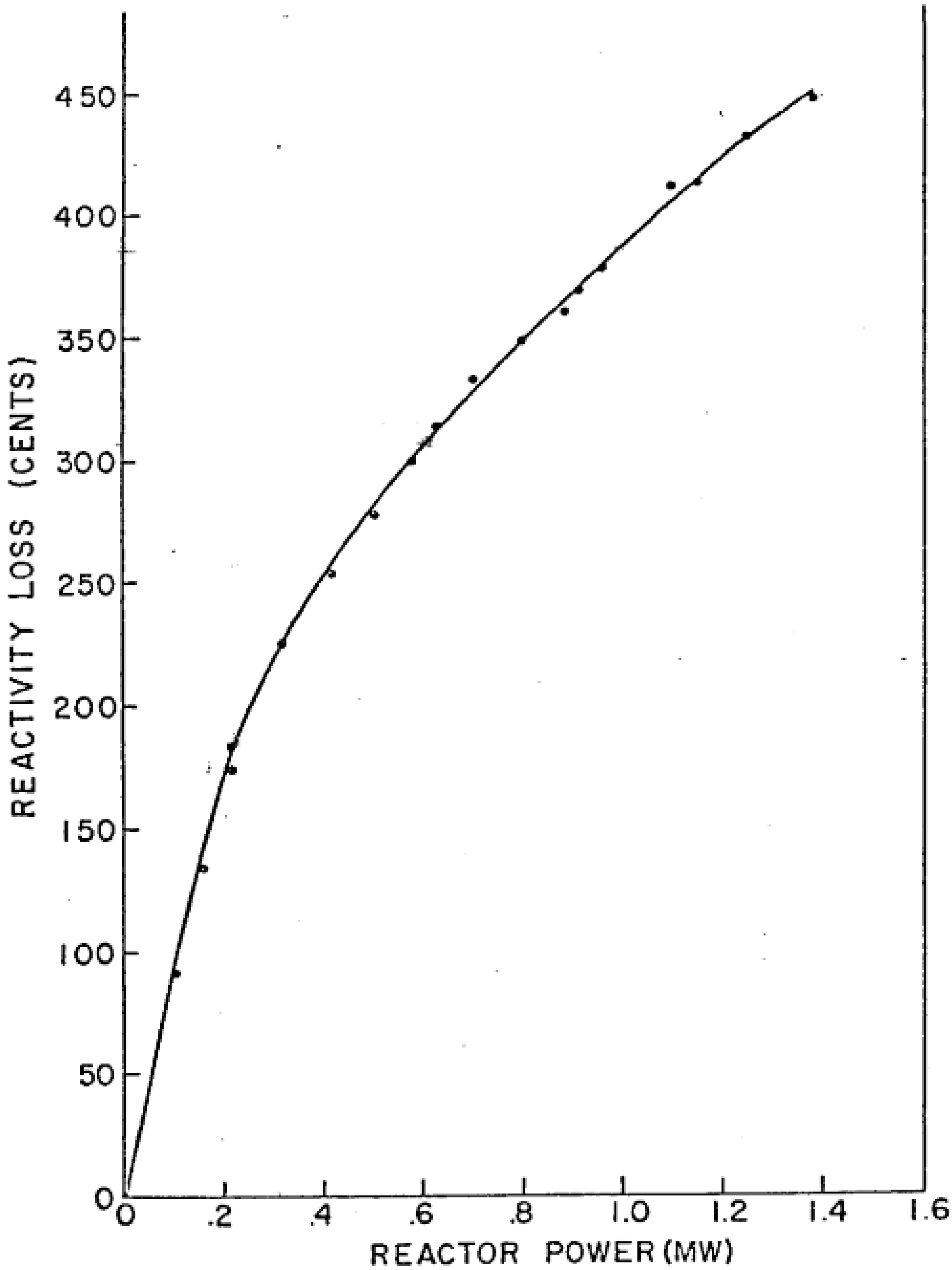

Figure 26 - Reactivity loss versus power level in high-power quasi-equilibrium experiments [Torrey Pines TRIGA]

Power Excursions

POWER EXCURSION CALCULATIONS

The analysis is based on the following conditions:

1. The reactor is initially at steady state power n_o (W) and the point reactor kinetics equations (PRKE) apply.
2. The temperature coefficient of reactivity α is negative and constant (about 8.4 x 10^{-5} for the TRIGA).
3. The system heat capacity K (J/°C) is constant (about 200 kg x 300 J / (kg °C) = 6 x 10^4 J / °C for the TRIGA). The energy coefficient of reactivity is A = α / K (about 1.4 x 10^{-9} J^{-1} for the TRIGA).
4. Time dependent, externally imposed reactivity is denoted as $\rho_o(t)$, feedback reactivity as $\rho_f(t)$, and the total as ρ(t).
5. Except as noted below, the reactor system is adiabatic and all energy released by fission leads to temperature increase, not phase change.

Equations are written for one group of delayed neutrons. Additional conditions are imposed at various stages of the development.

GOVERNING EQUATIONS

$$\frac{dn}{dt} = \left[\rho(t) - \beta\right]\frac{n(t)}{\ell} + \lambda C(t) \tag{1}$$

$$\frac{dC}{dt} = \beta\frac{n(t)}{\ell} - \lambda C(t) \tag{2}$$

$$\frac{d\rho}{dt} = \frac{d\rho_o}{dt} - An(t) \tag{3}$$

Initially, ρ = 0, $n = n_o$, and $C = C_o = \beta n_o/\lambda\ell$. The last equation arises from condition 5, and may be stated as:

$$\rho(t) = \rho_o(t) + \rho_f(t) = \rho_o(t) - A\int_0^t dt'n(t') \tag{4}$$

If there were heat loss from the system during a power excursion and if the rate of heat loss were simply proportional to the temperature rise of the system, then the above equation would be modified to read:

$$\rho(t) = \rho_o(t) + \rho_f(t) = \rho_o(t) - A\int_0^t dt'n(t')e^{-R(t-t')} \tag{5}$$

in which R is the heat loss rate (W) per net joule of energy gained during the excursion. The attached figures illustrate solutions to the above equations. Notation in the figures is as follows: $\dot{E}$ is the power density (W / cm^3); proportional to n; δk is the reactivity ρ; B_ϵ, proportional to A, has units cm^3/J. The figures are from Keepin, Physics of Nuclear Kinetics, Addison-Wesley, 1965.

THE NORDHEIM-FUCHS PROMPT BURST MODEL FOR A STEP CHANGE IN REACTIVITY

The basis for this power excursion approximation is that the excursion is so rapid that delayed neutrons may be neglected and that the system is adiabatic. The model is also for a positive step change in reactivity, ρ_o. Other conditions and definitions are as follows:

1. The initial power n_o is so small that it may be taken as 0.
2. The initial reactivity $\rho_o \gg \beta$.
3. The cumulative energy release (J) is $E(t) = \int_o^t dt' n(t')$.
4. The system average temperature rise (K) is $\theta(t) = \frac{A}{\alpha} E(t)$.
5. Peak values are denoted as $\hat{n}, \hat{E}$, etc.

Thus, Eqs. 1 – 3 reduce to:

$$\frac{dn}{dt} = [\rho(t) - \beta] \frac{n(t)}{\ell} \tag{6}$$

$$\frac{d\rho}{dt} = -An(t) \tag{7}$$

with initial conditions $\rho(0)=\rho_o$ and $n(0) = n_o \approx 0$.

PEAK POWER

Equations 6 and 7 yield the following relationship between ρ and n:

$$\frac{dn}{d\rho} = \frac{dn/dt}{d\rho/dt} = -\frac{(\rho - \beta)}{A\ell} \tag{8}$$

with the solution

$$n(\rho) - n_o = -\frac{1}{A\ell} \int_{\rho_o}^{\rho} d\rho(\rho - \beta) = \frac{1}{2A\ell} \left[(\rho_o - \beta)^2 - (\rho - \beta)^2 \right]. \tag{9}$$

Since we take $n_o = 0$ and since it is apparent from Eq. 6 that $n = \hat{n}$ when $\rho = \beta$,

$$\hat{n} = \frac{\beta^2}{2A\ell} \left[\frac{\rho_o}{\beta} - 1 \right]^2. \tag{10}$$

INITIAL PERIOD T

At the start of a power excursion, $\rho = \rho_o$. The power rise is proportional to $e^{t/T}$ where the initial period T is given by

$$\frac{1}{T} = \frac{1}{n}\frac{dn}{dt} = \frac{\beta}{\ell}\left[\frac{\rho_o}{\beta} - 1\right]. \tag{11}$$

TIME DEPENDENCE OF REACTIVITY

From Eqs. (7) and (9),

$$\frac{d\rho}{dt} = -\frac{1}{2\ell}\left[(\rho_o - \beta)^2 - (\rho - \beta)^2\right]. \tag{12}$$

Now let $\hat{t}$ represent the time at which $n = \hat{n}$, i.e., the time at which ρ = β. Then the solution of Eq. 12 can be written as:

$$-\frac{1}{2\ell}\int_{\hat{t}}^{t} dt = \int_{\beta}^{\rho(t)} d\rho\left[(\rho_o - \beta)^2 - (\rho - \beta)^2\right]^{-1}, \tag{13}$$

or

$$\left[\rho(t) - \beta\right] = -\left[\rho_o - \beta\right]\tanh\left[\frac{t - \hat{t}}{2T}\right]. \tag{14}$$

TIME DEPENDENCE OF POWER

Since ρ(t) is now known, Eq. 14 may be substituted into Eq. 9 and Eq. 10 may be invoked to yield

$$n(t) = \hat{n}\cdot\mathrm{sech}^2\left[\frac{t - \hat{t}}{2T}\right]. \tag{15}$$

PULSE DURATION

The duration τ of the power excursion is defined as the FWHM of *n(t)*. In Eq. 15, setting $n/\hat{n} = \frac{1}{2}$ and $t - \hat{t} = \tau/2$ yields

$$\tau = 4T\cosh^{-1}\sqrt{2} \approx 3.5T = 3.5\frac{\ell/\beta}{\rho_o/\beta - 1}. \tag{16}$$

ENERGY RELEASE

From the definition of energy release *E(t)* and from Eq. 15,

$$E(t) = \int_0^t dt'\, n(t') = \hat{n}\int_0^t dt'\mathrm{sech}^2\left[\frac{t - \hat{t}}{2T}\right] \tag{17}$$

$$E(t) = 2T\hat{n}\left\{1 + \tanh\left[\frac{t-\hat{t}}{2T}\right]\right\} = \frac{\beta}{A}\left(\frac{\rho_o}{\beta} - 1\right)\left\{1 + \tanh\left[\frac{t-\hat{t}}{2T}\right]\right\}.$$

The total energy release $\hat{E} \equiv E_\infty$ or

$$\hat{E} = 2\frac{\beta}{A}\left(\frac{\rho_o}{\beta} - 1\right). \tag{18}$$

TEMPERATURE RISE

Since the temperature rise $\theta(t) = \dfrac{A}{\alpha}E(t)$,

$$\theta(t) = \frac{\beta}{\alpha}\left(\frac{\rho_o}{\beta} - 1\right)\left\{1 + \tanh\left[\frac{t-\hat{t}}{2T}\right]\right\}. \tag{19}$$

The maximum temperature rise $\hat{\theta} = \theta_\infty$, or

$$\hat{\theta} = 2\frac{\beta}{\alpha}\left(\frac{\rho_o}{\beta} - 1\right). \tag{20}$$

Note that $\theta(t)/\hat{\theta} = E(t)/\hat{E} = 0.5\left\{1 + \tanh\left[\dfrac{t-\hat{t}}{2T}\right]\right\}$.

RULES OF THUMB FOR POWER EXCURSIONS

The following equations, in which $\$ = \rho_o/\beta$, are useful in extrapolation and interpolation of power excursion data.

$$\hat{n} \propto (\$ - 1)^2$$
$$T \propto (\$ - 1)^{-1}$$
$$\tau \propto (\$ - 1)^{-1}$$
$$\hat{E} \propto (\$ - 1)$$
$$\hat{\theta} \propto (\$ - 1)$$

TYPICAL RESULTS FOR THE TRIGA

The following are representative (but certainly not exact) values for the TRIGA, with $\beta = 0.007$, $\alpha = 8.4 \times 10^{-5}\ K^{-1}$, $A = 1.4 \times 10^{-9}\ J^{-1}$, and $\ell = 7 \times 10^{-5}$ s. This leads to:

$$\hat{n} \propto 250(\$-1)^2 \text{ MW}$$
$$T \propto 10(\$-1)^{-1} \text{ ms}$$
$$\tau \propto 35(\$-1)^{-1} \text{ ms}$$
$$\hat{E} \propto 10(\$-1) \text{ MJ}$$
$$\hat{\theta} \propto 170(\$-1) \text{ K (core average)}$$
$$\hat{\theta} \propto 170(\$-1) \text{ K (B-ring)}$$

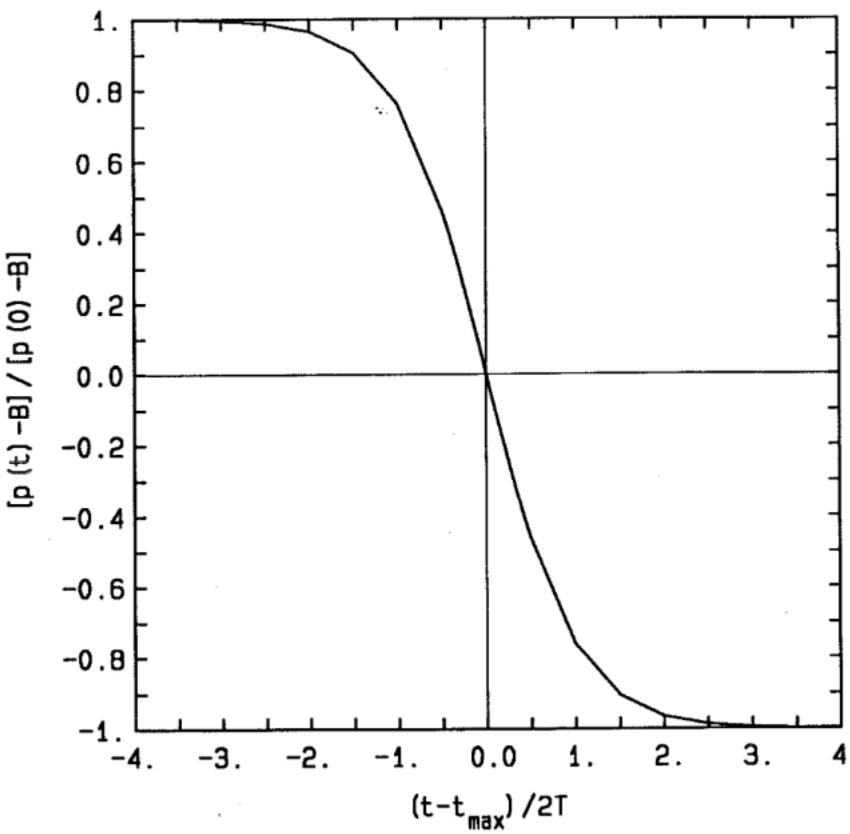

Figure 27 - Temperature dependence of reactivity during a pulse

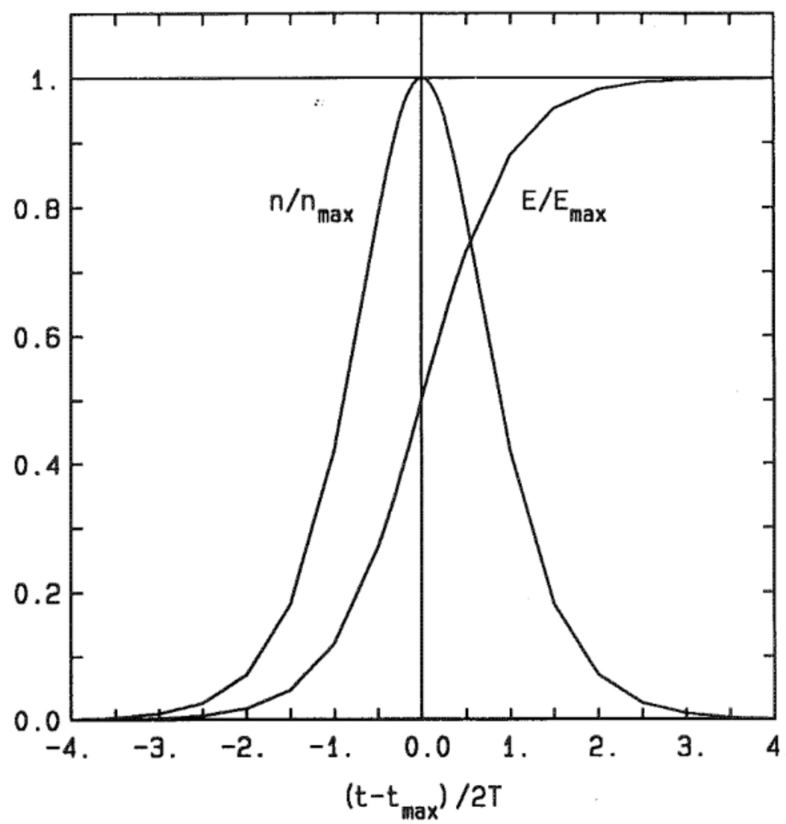

Figure 28 - Time dependence of power *n* and energy *E* during a pulse

Fission Product Poisoning

LOSS IN REACTIVITY VIA THERMAL UTILIZATION

During reactor operation, thermal utilization is reduced because of both the buildup of fission product poisons, in particular ^{149}Sm and ^{135}Xe. Both poisons increase subsequent to operation; however, because ^{135}Xe is radioactive, its effect ultimately dissipates. Since $\rho \equiv 1 - 1/k_{eff}$, it can easily be shown that, uncompensated, the effect of a poison on reactivity is given by

$$-\rho = \frac{\Sigma_{a,P}}{\Sigma_{a,F}\eta\left(\varepsilon p P_{Th} P_F\right)}, \tag{1}$$

in which $\Sigma_{a,P}$ and $\Sigma_{a,F}$ are respectively the macroscopic absorption cross sections of the poison and fuel. Since $\eta\Sigma_{a,F} = \nu\Sigma_f$, in which Σ_f is the fission cross section, and since the product of the terms in parentheses is roughly unity, we will use the approximation that

$$-\rho \approx \frac{\Sigma_{a,P}}{\nu\Sigma_f} \tag{2}$$

XENON POISONING

^{135}Xe appears in the chain

$$^{135}\mathrm{Te} \xrightarrow[\beta^-]{\text{short}} {}^{135}\mathrm{I} \xrightarrow[\beta^-]{6.7\text{–h}} {}^{135}\mathrm{Xe} \xrightarrow[\beta^-]{9.2\text{–h}} {}^{135}\mathrm{Cs} \xrightarrow[\beta^-]{2\text{x}10^6\text{–y}} {}^{135}\mathrm{Ba}$$

Approximations for fission yields and other factors are listed in the following table. Because the half life of ^{135}Te is so short, its yield is combined with that of ^{135}I. In subsequent equations, subscripts “x” and “i” refer to xenon and iodine factors.

	^{135}**I**	^{135}**Xe**
Fission yield, ϒ	0.061	0.003
Absorption cross section, σ (b)	0	$3\text{x}10^6$
Decay constant, λ (s^{-1})	$2.87\text{x}10^{-5}$	$2.09\text{x}10^{-5}$

Governing Equations

If ϕ is the thermal neutron flux density ($cm^{-2}s^{-1}$) and if I and X are the atomic densities of ^{135}I and ^{135}Xe, then

$$\frac{dI}{dt} = \Sigma_f \phi\gamma_i - \lambda_i I \tag{3}$$

$$\frac{dX}{dt} = \Sigma_f \phi\gamma_x + \lambda_i I - \lambda_x X - \sigma_x X\phi . \tag{4}$$

These equations may be solved in principle for arbitrary time dependent ϕ and for a variety of initial conditions. A few examples are as follows.

Equilibrium Poisoning

If ϕ is constant, the rates of formation and loss of iodine and xenon ultimately balance. By setting the derivatives in Eqs. 3 and 4 to zero and invoking Eq. 2, it follows that:

$$-\rho_{eq} = \frac{\sigma_x X_{eq}}{\upsilon \Sigma_f} = \frac{\nu^{-1}(\gamma_i + \gamma_x)\phi}{\lambda_x / \sigma_x + \phi} \tag{5}$$

For very high flux, ρ_{eq} is clearly independent of ϕ. For very low flux, ρ_{eq} varies in proportion to ϕ. Numerical results are as follows, with the values for full-power TRIGA operation italicized:

ϕ	$-\rho_{eq}$
10^{10}	3.7×10^{-5}
10^{11}	3.6×10^{-4}
10^{12}	3.2×10^{-3}
5×10^{12}	*1.1×10^{-2} ($1.5)*
10^{13}	1.5×10^{-2}
10^{14}	2.4×10^{-2}
∞	2.6×10^{-2} ($3.7)

Approach to Equilibrium

In this case, Eqs. 3 and 4 may be solved with ϕ constant and with conditions I = X = 0. Results are illustrated in Figure 29.

Buildup of Xenon After Reactor Shutdown

In this case, Eqs. 3 and 4 may be solved with ϕ set equal to zero, but with initial conditions for *I* and *X* dependent on the reactor state at shutdown. Results are illustrated in Figure 30, which is based on full-power TRIGA operation at $\phi = 5 \times 10^{12}$ n / cm^2 / s and equilibrium xenon prior to shutdown.

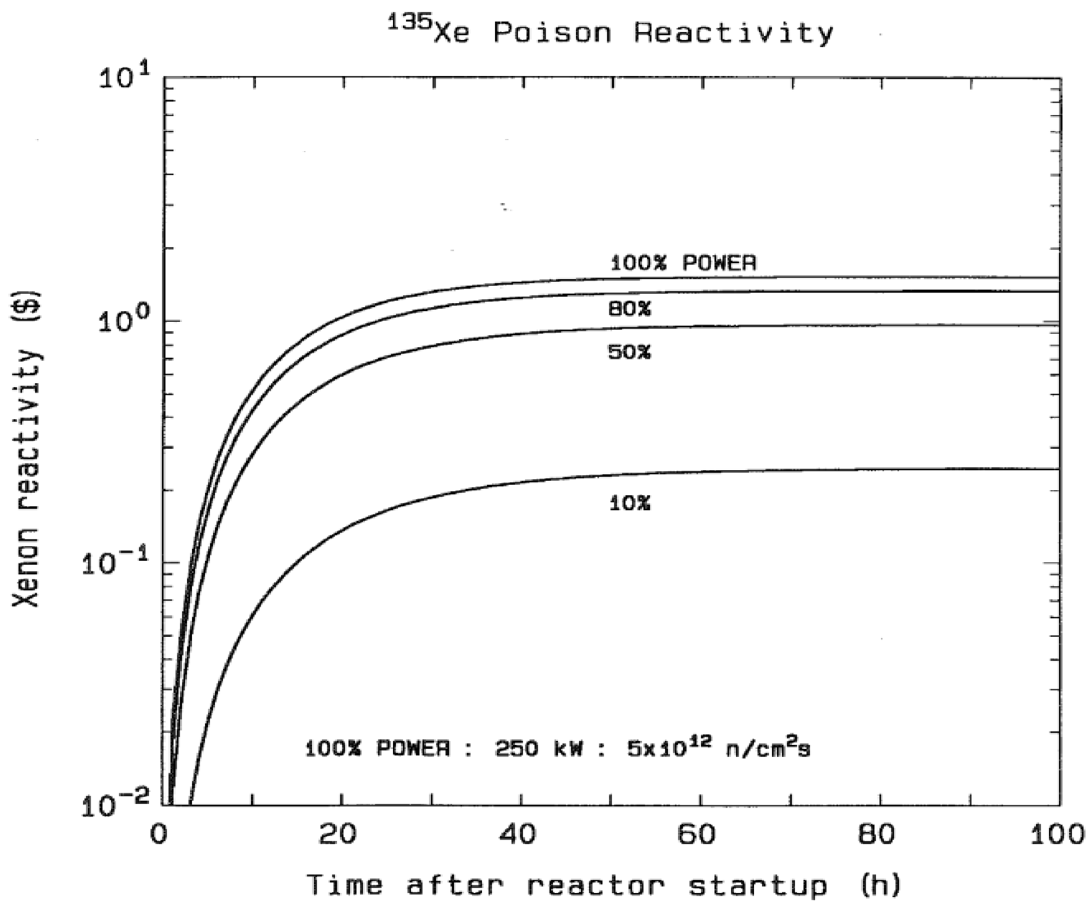

Figure 29 - Xenon reactivity during approach to equilibrium

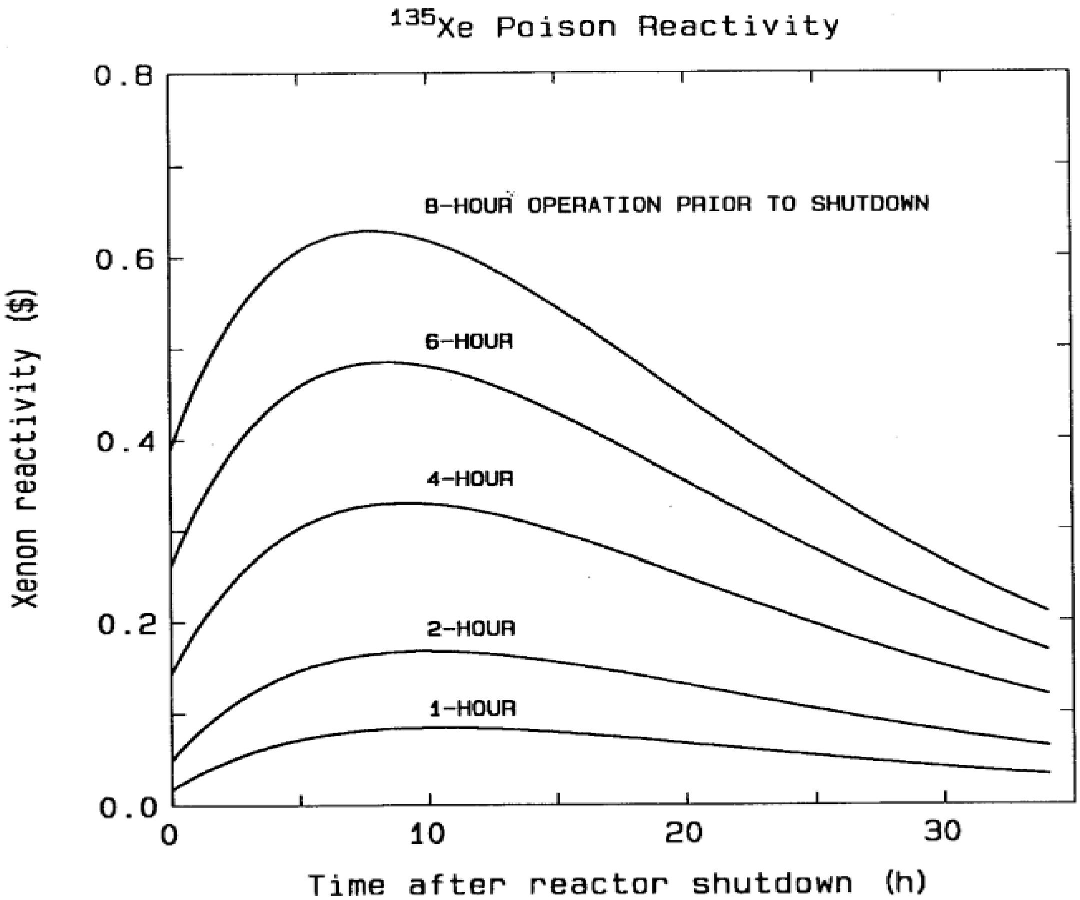

Figure 30 - Xenon reactivity after shutdown

SAMARIUM POISONING

^{149}Sm appears in the decay chain

$$^{149}\text{Nd} \xrightarrow[\beta^-]{\text{short}} {}^{149}\text{Pm} \xrightarrow[\beta^-]{\text{54-h}} {}^{149}\text{Sm (stable)}$$

Approximations for fission yields and other factors are listed in the following table. Because the half-life of ^{149}Nd is so short, its yield is combined with that of ^{149}Pm.

	^{149}Pm	**^{149}Sm**
Fission yield, γ	0.01	0
Absorption cross section, σ (b)	0	5×10^4
Decay constant, λ (s^{-1})	3.6×10^{-6}	--

Governing Equations

If ϕ is the thermal neutron flux density (cm^{-2}s^{-1}) and if *P* and *S* are the atomic densities (cm^{-3}) of ^{149}Pm and ^{149}Sm, then

$$\frac{dP}{dt} = \Sigma_f \phi \gamma - \lambda P \tag{6}$$

$$\frac{dS}{dt} = \lambda P - \sigma S \phi \tag{7}$$

Subscripts are not required because γ, λ, and σ are uniquely determined.

Equilibrium Poisoning

If ϕ is constant, the rates of formation and loss of Pm and Sm ultimately balance. By setting the derivatives in Eqs. 6 and 7 equal to zero, and invoking Eq. 2, it follows that:

$$-\rho_{eq} = \frac{\sigma S_{eq}}{\nu \Sigma_f} = \frac{\gamma}{\nu} = 0.004\,(\$0.57) \tag{8}$$

At equilibrium, ρ_{eq} is constant, independent of ϕ.

Approach to Equilibrium

In this case, Eqs. 6 and 7 may be solved with ϕ constant and with initial conditions $P = S = 0$:

$$\frac{\rho(t)}{\rho_{eq}} = \left[1 - \frac{\lambda e^{-\sigma\phi t} - \sigma\phi e^{-\lambda t}}{\lambda - \sigma\phi}\right] \tag{9}$$

For $\sigma\phi \ll \lambda$ (a common case),

$$\frac{\rho(t)}{\rho_{eq}} = \left[1 - e^{-\sigma\phi t}\right] \tag{10}$$

Results are illustrated in Figure 31 as the ratio ρ / ρ_{eq}.

Buildup of Samarium after Reactor Shutdown

In this case, Eqs. 6 and 7 may be solved with ϕ set equal to 0, but with initial conditions for *P* and *S* dependent on the reactor state at shutdown. For cases in which equilibrium has been reached prior to shutdown, as illustrated in Figure 32,

$$\frac{\rho(t)}{\rho_{eq}} = 1 + \frac{\sigma\phi}{\lambda}\left(1 - e^{-\lambda t}\right) \tag{11}$$

The buildup is very slow because of the 54-hour half-life of ^{149}Pm. For the case of a TRIGA operating at full power, $\phi = 5 \times 10^{12}$,

$$\frac{\rho_{\max}}{\rho_{eq}} = 1 + \frac{\sigma\phi}{\lambda} = 1.07 \tag{12}$$

Burnup of Samarium After Reactor Restart

Suppose a reactor is restarted after all of the ^{149}Pm has decayed to ^{149}Sm, i.e., with $\rho = \rho_{max}$. After restart, the initial ^{149}Sm is consumed, according to

$$\rho_{old} = \rho_{\max} e^{-\sigma\phi t} \tag{13}$$

However, new ^{149}Sm is being formed, as described in Eq. 9. Thus, the time dependence of the poison effect on reactivity, as shown in Figure 32, is given by

$$\frac{\rho(t)}{\rho_{eq}} = \left[1 + \frac{\sigma\phi}{\lambda}\right] e^{-\sigma\phi t} + \left[1 - \frac{\lambda e^{-\sigma\phi t} - \sigma\phi e^{-\lambda t}}{\lambda - \sigma\phi}\right] \tag{14}$$

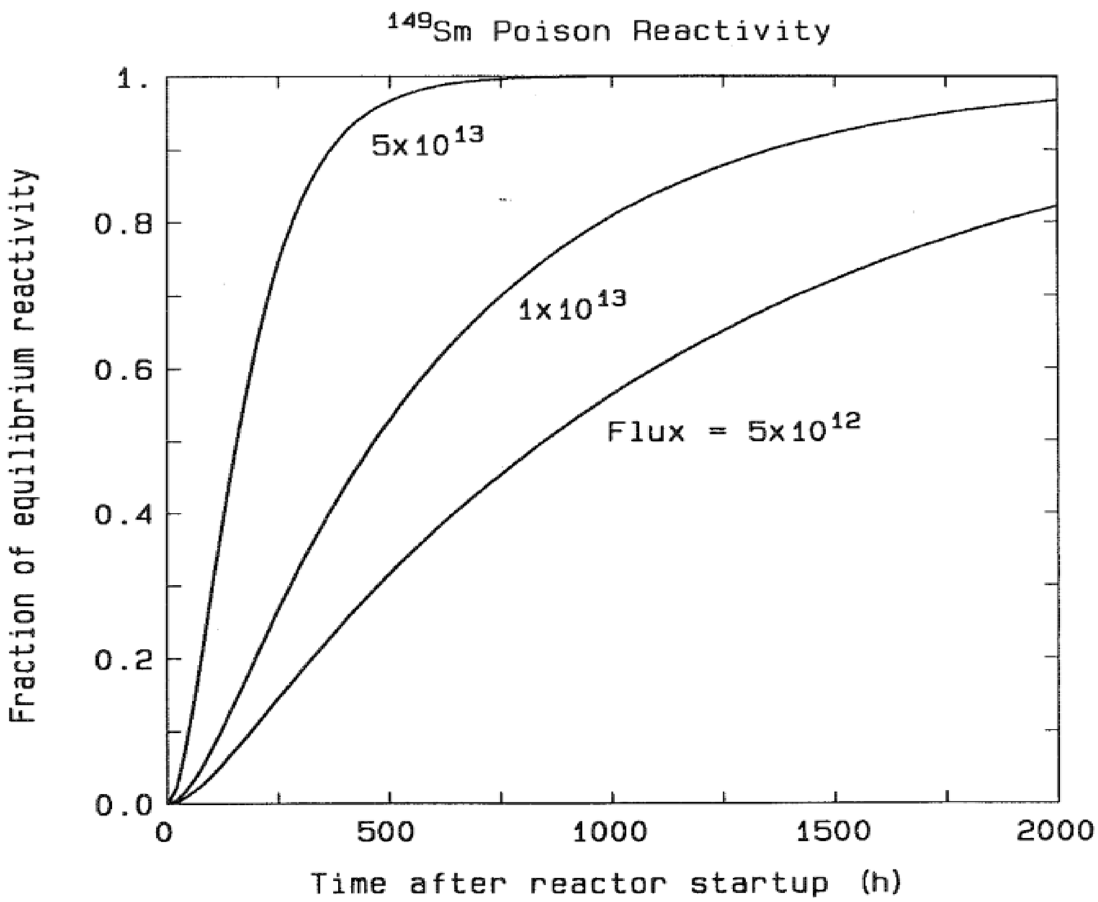

Figure 31 - Samarium reactivity during approach to equilibrium

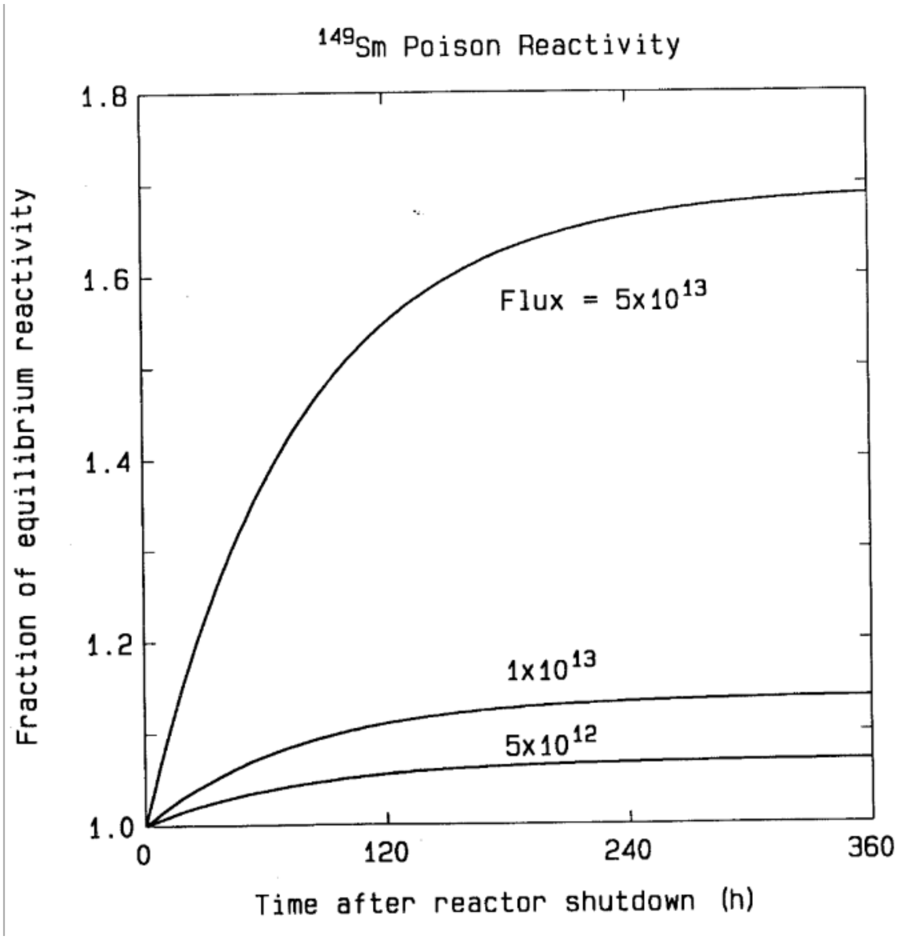

Figure 32 - Samarium reactivity after shutdown

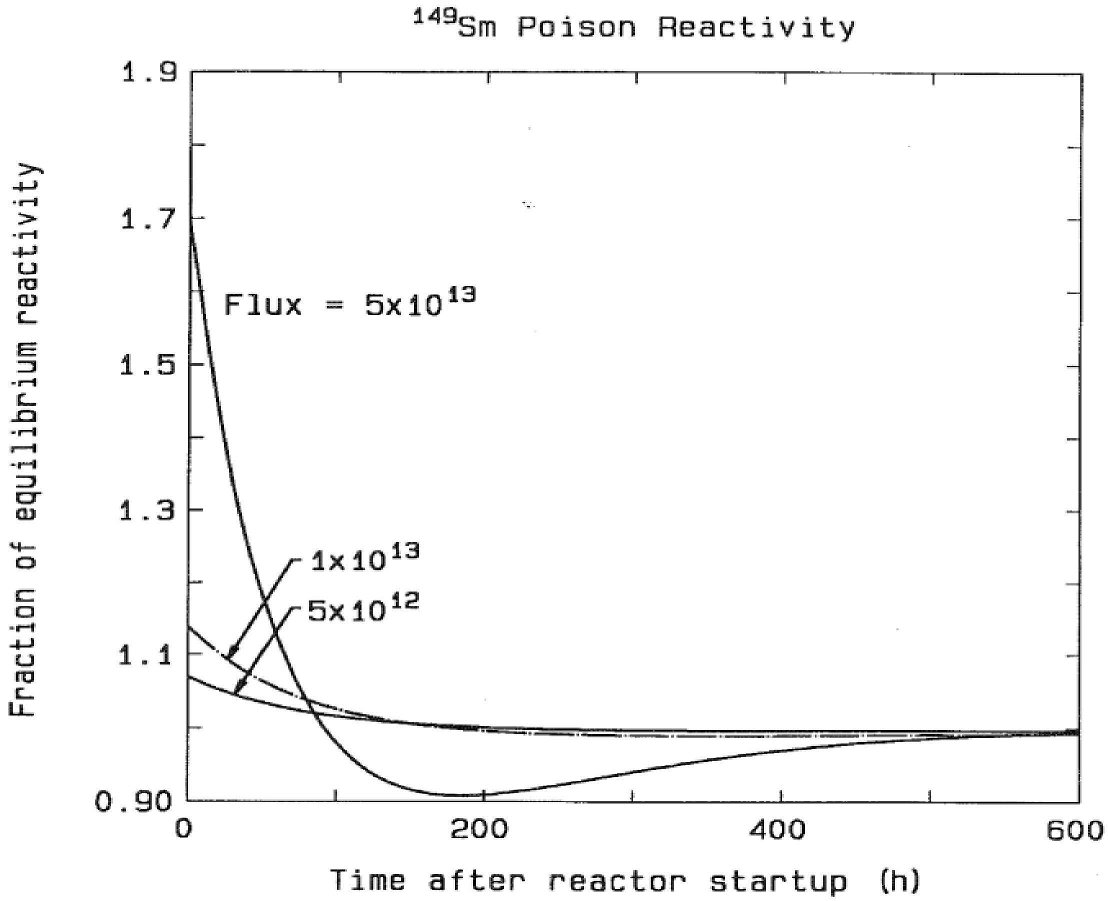

Figure 33 - Samarium reactivity after reactor restart

Principles of Neutron Activation Analysis

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GENERAL PRINCIPLES

When a material is irradiated by neutrons, individual isotopes of each element present may in principle be transmuted to radioactive isotopes (radionuclides). Associated with the decay of each radionuclide may be the emission of gamma rays with many possible discrete energies. The decay of any one radionuclide is characterized by a unique half-life $t_{1/2}$ or decay constant λ given by

$$\lambda = \frac{\ln 2}{t_{1/2}}$$

Let the mass of the element irradiated be *m*, the fractional natural abundance of the isotope transmuted be *g*, and the fractional emission frequency of a gamma ray of interest be *f*. This frequency is the probability that any one gamma ray emitted will carry the energy of interest. During irradiation, the intensity of the neutrons is characterized by a flux density ϕ equal to the neutron path length traveled per unit volume per unit time. The probability that an atom is transmuted is proportional not only to the flux density of neutrons and atomic density of target atoms, but also to the "cross section" σ. Suppose that the mass *m* is irradiated for time t_1 and allowed to "cool" for time t_2. It can be shown that the number of gamma rays emitted per unit time with the energy of interest is given by

$$\text{emission rate} = \frac{mgN_A}{\mathrm{A}} f\sigma\phi\left(1-e^{-\lambda t_1}\right)e^{-\lambda t_2}$$

in which N_A is Avogadro's number. As one would expect, the emission rate is directly proportional to element mass, isotopic abundance, emission frequency, cross section, and neutron flux density.[5] The exponential term $\left(1-e^{-\lambda t_1}\right)$ accounts for the approach to equilibrium, i.e., to a sufficiently large quantity of the radioactive isotope that the rate of radioactive decay is just balanced by the rate of production of the isotope in the reactor. The exponential term $e^{-\lambda t_2}$ accounts for the decay of the radioisotope during the cooling stage between irradiation and spectroscopic measurements.

[5] The produce σϕ is shorthand for the integral $\int dE\sigma(E)\phi(E)$ which accounts for the energy dependence of the activation cross section and the energy spectrum of the neutron flux density. The integral is frequently expressed as the sum of the products of (1) the total flux density of thermal neutrons and the average thermal neutron cross section, and (2) the total fast (non-thermal) neutron flux density and the so-called resonance cross section. In the nuclear reactor used in this work, the thermal and fast neutron flux densities are nearly equal, thus we simply multiply the thermal neutron flux density by the sum of the thermal neutron activation cross section and the resonance integral.

Table 2 lists natural abundances and cross sections for activation of certain isotopes of heavy elements. The yield term in the table needs to be applied when there are two possible activation products. For example, ^{69m}Zn, a *metastable* state of ^{69}Zn, is produced in only 7% of the activations of ^{68}Zn.

Table 2 - Abundances, cross sections, and activation yields of heavy elements.

Element	Z	A	Abundance (%)	Cross section (b) thermal	Cross section (b) resonance	Product	Half life	Yield (%)
Cr	24	50	4.32	15.9	7.6	^{51}Cr	27.70 d	100
Zn	30	64	48.6	1.4		^{65}Zn	244.4 d	100
		68	18.8	1.072	3.3	^{69m}Zn	13.76 h	7
As	33	75	100	4.3	60	^{76}As	26.32 h	100
Se	34	74	0.87	51.8	565	^{75}Se	119.8 d	100
		80	49.82	0.61	1.7	^{81}Se	18.6 m	85
						^{81m}Se	56.8 m	15
		82	0.045	0	13	^{83}Se	25 m	87
						^{83m}Se	70 s	13
Ag	47	107	51.35	37.2	94	^{108}Ag	2.37 m	92
		109	48.65	91	1450	^{110}Ag	24.57 s	94
						^{110m}Ag	249.9 d	6
Cd	48	114	28.86	0.336	20	^{115}Cd	53.5 h	89
						^{115m}Cd	44.6 d	11
Ba	56	130	0.101	13.5	150	^{131}Ba	11.8 d	100
		138	71.66	0.35	0.2	^{139}Ba	83.1 m	100
Hg	80	196	0.146	3200	472	^{197}Hg	64.14 h	96
						^{197m}Hg	23.8 h	4
		202	29.8	4.9	4.9	^{203}Hg	46.6 d	100

Source: Ryman [1] and Kocher [2].

Gamma Ray Emission Properties of Activation Products

Table 3 lists the various radionuclides produced by the activation of select heavy elements. Half-lives of the radionuclides are listed as are gamma ray energies and emission frequencies. Also listed in the table are energies of gamma rays emitted by other radionuclides potentially interfering in the activation analysis. This table is useful in planning neutron activation analysis and in interpretation of gamma ray energy spectra. Helmke [3] gives a similar table and in addition recommends that comprehensive activation analysis of soils may be accomplished optimally by measuring gamma ray spectra immediately after irradiation, after three days, after ten days, and after forty days. The first measurement, of course, permits determination of short-lived activation products. By the time of the last measurement, all but the longer-lived radionuclides will have decayed to negligible activities.

USE OF STANDARDS AND FLUX MONITORS

As described in the Appendix, if the gamma rays from the element are measured for a time t_3 using a detector with a photopeak[6] efficiency ϵ, the number of counts registered in the photopeak is given by

[6] Of the various ways gamma rays may interact in a detector, the one of importance in gamma ray spectroscopy is the photoelectric effect. The photoelectron carries nearly all the original energy of the

$$P = \frac{mgfN_A\sigma\phi\varepsilon}{A\lambda} \cdot C$$

where the time factor C is given by

$$C(t_1, t_2, t_3) = \left(1 - e^{-\lambda t_1}\right)\left(e^{-\lambda t_2} - e^{-\lambda(t_2 + t_3)}\right)$$

Because the cross section σ and the flux density ϕ are usually not known with precision, standards and flux monitors are commonly used in neutron activation analysis. In ratios of measured photopeak counts for an unknown and a standard containing a known quantity of an element, the cross section, natural abundance, photopeak efficiency, and gamma ray emission efficiency cancel. While the flux densities during irradiation do not cancel in the ratio, flux monitors may be used to account for differences between flux densities during irradiations. A flux monitor might be a short section of pure iron wire fastened to a specimen or standard container. Emission of gamma rays from the activated flux monitors can be measured, and, from the measurements, variations in flux density can be accounted for. Details of the analysis procedure are described in the Appendix.

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4. Keller, J.M., *TPASGAM: Radiation Decay Library of Gamma-Ray Energies, Branching Ratios, and Cross Sections*, issued as Data Library DLC88 by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, TN.

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gamma ray and events all appear as a "photopeak" in the measured energy spectrum of the gamma rays.

Table 3 - Characteristics of activation products and potentially interfering radionuclides

				Potential Interference		
Nuclide	$T_{1/2}$	Energy (keV)	Emission percent	Nuclide	Energy (keV)	$T_{1/2}$
^{24}Na	15.0 h	1369	100			
		2754	100			
^{42}K	12.4 h	1524	18			
^{46}Sc	83.8 d	889	100	^{110m}Ag	884	250. d
		1121	100	^{214}Bi	1121	Nat'l
				^{182}Ta	1121	115. d
^{47}Ca	4.54 d	1297	75	^{152}Eu	1299	12.7 y
^{51}Cr	27.7 d	320	10	^{105}Rh	319	1.48 d
^{59}Fe	44.5 d	1099	57			
		1292	43			
^{60}Co	5.28 y	1173	100			
		1333	100			
^{65}Ni	2.52 h	1115	15	^{65}Zn	1115	244. d
		1482	24	^{82}Br	1476	1.47 d
^{65}Zn	244. d	1115	51	^{152}Eu	1112	13.6 y
^{69m}Zn	13.8 h	439	95			
^{75}Se	120. d	121	17	^{152}Eu	122	13.6 y
				^{131}Ba	124	11.6 d
		136	60	^{181}Hf	133	42.4 d
				^{197m}Hg	134	24.0 h
		265	60			
		280	25	^{197m}Hg	279	24.0 h
		401	11			
^{76}As	1.10 d	560	45	^{134}Cs	563	2.07 y
		660	6			
^{82}Br	1.47 d	554	71			
		619	43			
		698	28			
		777	83	^{152}Eu	779	13.6 y
		828	24			
		1327	17	^{60}Co	1333	5.28 y
		1476	17			

Table 3, continued.

Nuclide	$T_{1/2}$	Energy (keV)	Emission percent	Potential Interference Nuclide	Potential Interference Energy (keV)	Potential Interference $T_{1/2}$
^{105}Rh	1.48 d	319	19	^{51}Cr	320	27.7 d
^{110m}Ag	250. d	658	94	^{76}As	660	1.10 d
		677	11			
		707	17			
		763	23			
		884	73	^{46}Sc	889	83.8 d
		937	34			
		1384	24			
		1505	13			
^{115}Cd	2.22 d	336	50			
		528	29			
^{122}Sb	2.70 d	564	71	^{134}Cs	570	2.07 y
^{124}Sb	60.2 d	603	98	^{134}Cs	605	2.07 y
		722	11			
		1690	48			
^{131}Ba	11.6 d	124	29	^{152}Eu	122	13.6 y
				^{75}Se	121	120. d
		216	20			
		373	15			
		496	47			
^{134}Cs	2.07 y	563	9	^{76}As	560	1.10 d
		570	16			
		605	98	^{124}Sb	603	60.2 d
		802	9			
		896	88			
^{140}La	1.68 d	329	21			
		487	46	^{47}Ca	489	4.54 d
		752	4			
		816	24			
		868	6			
		925	7			
		1597	95			
^{141}Ce	32.5 d	145	48			

Table 3, continued.

				Potential Interference		
Nuclide	$T_{1/2}$	Energy (keV)	Emission percent	Nuclide	Energy (keV)	$T_{1/2}$
^{152}Eu	13.6 y	122	28	^{75}Se	121	120. d
				^{131}Ba	124	11.6 d
		344	26	^{181}Hf	346	42.4 d
		779	13	^{82}Br	777	1.47 d
		964	15	^{160}Tb	966	72.0 d
				^{232}Th	969	Nat'l
		1112	14	^{65}Zn	1115	244. d
		1408	21			
^{153}Sm	1.95 d	103	28	^{182}Ta	100	115. d
^{160}Tb	72.0 d	299	27			
		879	29			
		966	24	^{232}Th	969	Nat'l
		1178	14			
^{177}Lu	6.74 d	113	7			
		208	11			
^{181}Hf	42.4 d	133	41	^{197m}Hg	134	24.0 h
				^{75}Se	136	120. d
		346	12	^{152}Eu	344	13.6 y
		482	83	^{187}W	480	23.6 h
^{182}Ta	115. d	100	14	^{153}Sm	103	1.95 d
		1121	35	^{46}Sc	1121	83.8 d
				^{214}Bi	1121	Nat'l.
		1189	16			
		1221	27			
^{187}W	23.6 h	480	21	^{181}Hf	482	42.4 d
		686	26			
^{197m}Hg	24.0 h	134	43	^{181}Hf	133	42.4 d
				^{75}Se	136	120. d
		279	6			
^{197}Hg	2.67 d	279	71			
		279	71			
^{203}Hg	46.6 d	279	82			

Source: Keller [4].

APPENDIX – DATA ANALYSIS PROCEDURE

Assumptions

A sample specimen containing an unknown mass of some reference element is irradiated independently of a standard specimen containing a known mass of the element. Both the unknown and the standard are irradiated with flux monitors such as pure iron wires of known mass. The following assumptions and approximations apply to the neutron activation analysis procedure.

a. The unknown and the standard are of such small masses that neutron flux depression within the samples is negligible.

b. Specimen holders such as polyethylene vials are uncontaminated by the reference element or else provisions have been made for correction of results to account for such contamination.

c. The neutron angular and energy distribution is the same during the irradiation of the unknown and the standard.

d. The masses and volumes of the standard and the unknown are about the same. The specimen containers are identical.

e. The flux monitors for the unknown and the standard are about the same weight and are secured to the containers in identical geometries.

f. The standard and unknown are analyzed for gamma ray intensities in the same geometry and data are corrected for background. The same is true for the flux monitors, although a different spectrometer and different geometry may be used for the monitors.

Notation

The following notation applies to unknowns, standards, and flux monitors. Subscripts “s” and “u” apply respectively to standards and unknowns. Superscript “w” denotes flux monitor.

Normally the flux density and cross section apply to thermal neutrons. However, if there is a substantial flux density of fast neutrons and/or a large resonance integral, it is necessary, at least conceptually, to account for activation by fast neutrons. As a reasonable approximation for the TRIGA reactor, for which the thermal neutron and fast neutron flux densities are about the same, one may treat ϕ as the thermal neutron flux density but treat σ as the sum of the thermal neutron cross section and the resonance integral.

Symbol	Designation
A	atomic mass (amu)
E	gamma ray energy (MeV)
S	emission rate (s^{-1}) for a specific gamma ray
f	frequency of emission of a gamma ray
g	natural abundance of isotope examined
m	element mass (g) in specimen or flux monitor
M	measured specimen or flux monitor mass (g)
$\mathbf{N_A}$	Avogadro's number
P	measured counts in a photopeak
$\mathbf{t_1}$	irradiation time (s)
$\mathbf{t_2}$	cooling time (s)
$\mathbf{t_3}$	counting real time (s)
$\mathbf{t_4}$	counting live time (s)
X	mass concentration of element (e.g., ppm)
ϵ	spectrometer photopeak efficiency
λ	decay constant of isotope examined (s^{-1})
ϕ	neutron flux density ($cm^{-2}\,s^{-1}$)
σ	neutron cross section (cm^2)

Working Equations

The following equation is useful for predicting the exposure rate to be expected from a specimen or flux monitor at various times after irradiation. It gives the source strength in terms of gamma rays per second at time t_2 after irradiation for any one gamma ray associated with any one isotope of the reference element or the flux monitor.

$$S = \frac{mgfN_A\sigma\phi}{A}\left(1-e^{-\lambda t_1}\right)e^{-\lambda t_2} \tag{1}$$

From that same gamma ray, the exposure rate $\dot{X}$ (R/h) at a distance of one foot from the specimen is given approximately by

$$\dot{X} = \left(\frac{6SE}{3.7\times10^{10}}\right) \tag{2}$$

The following equation gives the number of photopeak counts for a given gamma ray emitted from a specimen or flux monitor.

$$P = \frac{mgfN_A\sigma\phi\varepsilon}{A\lambda}\cdot C \tag{3}$$

where the time factor C is given by

$$C(t_1,t_2,t_3) = \left(1-e^{-\lambda t_1}\right)\left(e^{-\lambda t_2} - e^{-\lambda(t_2+t_3)}\right) \tag{4}$$

Measured photopeak counts from the unknown are in the following ratio, as required by Eq. 3:

$$\frac{m_u}{m_s} = \frac{P_u}{P_s}\frac{C_s}{C_u}\frac{\phi_s}{\phi_u} \tag{5}$$

independent of *f*, *g*, ϵ, and σ for the particular unknown element and gamma ray of interest. The ratio of the flux densities under which standard and unknown are irradiated may also be obtained from Eq. 3 evaluated for flux-monitor spectroscopy under the two conditions of irradiation, namely,

$$\frac{\phi_s}{\phi_u} = \frac{M_u^w}{M_s^w}\frac{C_u^w}{C_s^w}\frac{P_s^w}{P_u^w} \tag{6}$$

in which m_u^w / m_s^w has been replaced by its equal M_u^w / M_s^w. This ratio is independent of *f*, *g*, ϵ, and σ for the flux monitor and the associated gamma ray of interest. It follows that the ratio of the element masses in the unknown and standard are given by

$$\frac{m_u}{m_s} = \frac{P_u}{P_s}\frac{P_s^w}{P_u^w}\frac{M_u^w}{M_s^w}\frac{C_s}{C_u}\frac{C_u^w}{C_s^w} \tag{7}$$

It follows that the ratio of element concentrations in the unknown and standard is given by

$$\frac{X_u}{X_s} = \frac{C_u^w}{C_u}\frac{M_u^w}{M_u}\frac{P_u}{P_u^w}\cdot \mathcal{F} \tag{8}$$

in which $\mathcal{F}$ is a calibration factor determined exclusively from data related to the standard specimen. The function is given by

$$\mathcal{F} = \frac{C_s}{C_s^w}\frac{M_s}{M_s^w}\frac{P_s^w}{P_s} \tag{9}$$

This calibration factor need be determined only once, provided that the specimen-container and counting geometries remain unchanged.

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TRIGA Reactor Kinetic Data

Core II-16

$k_{eff} = \eta \epsilon p f P_f P_{th}$

Factor	Designation	TRIGA Value (ARO)
η	Reproduction factor	2.073
ϵ	Fast fission factor	1.052
p	Resonance escape probability	0.877
f	Thermal utilization factor	0.7136
P_f	Fast non-leakage probability	0.767
P_{th}	Thermal non-leakage probability	0.971
K_{eff}	Effective multiplication factor	1.0164

REACTIVITY DESCRIPTORS (ALL RODS OUT)

$$\Delta k = k_{eff} - 1 = 0.0164$$

$$\rho = \frac{\Delta k}{k_{eff}} = 0.0162$$

$$\%\Delta k = \rho \times 10^2 = 1.62$$

$$PCM = \rho \times 10^5 = 1620$$

$$\$ = \rho / \beta_{eff} = \$2.31$$

INHOUR EQUATION: PERIOD *T* VS. REACTIVITY

$$\rho = \frac{\ell}{T} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T}$$

in which ℓ is the prompt neutron lifetime (about 43 μs for the TRIGA), β_i are fission yields for groups of delayed neutron precursors, and λ_i are decay constants for groups of delayed neutron precursors

NOTE: For the calculation of doubling time and period in the following tables, a prompt neutron lifetime of 43 μs (based on the 2008 SAR) and group constants from Duderstadt and Hamilton were used.

Doubling Time - Reactivity Relationships for the KSU TRIGA Reactor

Doubling Time (s)	Period (s)	ρ/β ($)	ρ	PCM	k_{eff}
1	1.44	$0.734	0.00514	514	1.00516
2	2.89	$0.617	0.00432	432	1.00434
3	4.33	$0.543	0.00380	380	1.00382
4	5.77	$0.490	0.00343	343	1.00344
5	7.21	$0.450	0.00315	315	1.00316
6	8.66	$0.417	0.00292	292	1.00293
7	10.10	$0.390	0.00273	273	1.00274
8	11.54	$0.367	0.00257	257	1.00257
9	12.98	$0.347	0.00243	243	1.00244
10	14.43	$0.330	0.00231	231	1.00231
11	15.87	$0.315	0.00220	220	1.00221
12	17.31	$0.301	0.00211	211	1.00211
13	18.76	$0.289	0.00202	202	1.00202
14	20.20	$0.277	0.00194	194	1.00195
15	21.64	$0.267	0.00187	187	1.00187
16	23.08	$0.258	0.00180	180	1.00181
17	24.53	$0.249	0.00174	174	1.00175
18	25.97	$0.241	0.00169	169	1.00169
19	27.41	$0.234	0.00164	164	1.00164
20	28.85	$0.227	0.00159	159	1.00159
21	30.30	$0.220	0.00154	154	1.00154
22	31.74	$0.214	0.00150	150	1.00150
23	33.18	$0.208	0.00146	146	1.00146
24	34.62	$0.203	0.00142	142	1.00142
25	36.07	$0.198	0.00139	139	1.00139
26	37.51	$0.193	0.00135	135	1.00135
27	38.95	$0.189	0.00132	132	1.00132
28	40.40	$0.184	0.00129	129	1.00129
29	41.84	$0.180	0.00126	126	1.00126
30	43.28	$0.176	0.00123	123	1.00123
31	44.72	$0.172	0.00121	121	1.00121
32	46.17	$0.169	0.00118	118	1.00118
33	47.61	$0.165	0.00116	116	1.00116
34	49.05	$0.162	0.00113	113	1.00114
35	50.49	$0.159	0.00111	111	1.00111
36	51.94	$0.156	0.00109	109	1.00109
37	53.38	$0.153	0.00107	107	1.00107
38	54.82	$0.150	0.00105	105	1.00105
39	56.27	$0.148	0.00103	103	1.00103
40	57.71	$0.145	0.00101	101	1.00102
41	59.15	$0.142	0.00100	100	1.00100
42	60.59	$0.140	0.00098	98	1.00098
43	62.04	$0.138	0.00096	96	1.00097
44	63.48	$0.136	0.00095	95	1.00095
45	64.92	$0.133	0.00093	93	1.00093
46	66.36	$0.131	0.00092	92	1.00092
47	67.81	$0.129	0.00090	90	1.00091
48	69.25	$0.127	0.00089	89	1.00089
49	70.69	$0.125	0.00088	88	1.00088
50	72.13	$0.124	0.00086	86	1.00087

Doubling Time - Reactivity Relationships for the KSU TRIGA Reactor

Doubling Time (s)	Period (s)	ρ/β ($)	ρ	PCM	k_{eff}
51	73.58	$0.122	0.00085	85	1.00085
52	75.02	$0.120	0.00084	84	1.00084
53	76.46	$0.118	0.00083	83	1.00083
54	77.91	$0.117	0.00082	82	1.00082
55	79.35	$0.115	0.00081	81	1.00081
56	80.79	$0.114	0.00080	80	1.00080
57	82.23	$0.112	0.00078	78	1.00079
58	83.68	$0.111	0.00077	77	1.00077
59	85.12	$0.109	0.00076	76	1.00076
60	86.56	$0.108	0.00075	75	1.00076
61	88.00	$0.106	0.00075	75	1.00075
62	89.45	$0.105	0.00074	74	1.00074
63	90.89	$0.104	0.00073	73	1.00073
64	92.33	$0.103	0.00072	72	1.00072
65	93.78	$0.101	0.00071	71	1.00071
66	95.22	$0.100	0.00070	70	1.00070
67	96.66	$0.099	0.00069	69	1.00069
68	98.10	$0.098	0.00069	69	1.00069
69	99.55	$0.097	0.00068	68	1.00068
70	100.99	$0.096	0.00067	67	1.00067
71	102.43	$0.095	0.00066	66	1.00066
72	103.87	$0.094	0.00066	66	1.00066
73	105.32	$0.093	0.00065	65	1.00065
74	106.76	$0.092	0.00064	64	1.00064
75	108.20	$0.091	0.00063	63	1.00064
76	109.64	$0.090	0.00063	63	1.00063
77	111.09	$0.089	0.00062	62	1.00062
78	112.53	$0.088	0.00062	62	1.00062
79	113.97	$0.087	0.00061	61	1.00061
80	115.42	$0.086	0.00060	60	1.00060
81	116.86	$0.085	0.00060	60	1.00060
82	118.30	$0.084	0.00059	59	1.00059
83	119.74	$0.084	0.00059	59	1.00059
84	121.19	$0.083	0.00058	58	1.00058
85	122.63	$0.082	0.00057	57	1.00057
86	124.07	$0.081	0.00057	57	1.00057
87	125.51	$0.081	0.00056	56	1.00056
88	126.96	$0.080	0.00056	56	1.00056
89	128.40	$0.079	0.00055	55	1.00055
90	129.84	$0.078	0.00055	55	1.00055
91	131.29	$0.078	0.00054	54	1.00054
92	132.73	$0.077	0.00054	54	1.00054
93	134.17	$0.076	0.00053	53	1.00053
94	135.61	$0.076	0.00053	53	1.00053
95	137.06	$0.075	0.00052	52	1.00052
96	138.50	$0.074	0.00052	52	1.00052
97	139.94	$0.074	0.00052	52	1.00052
98	141.38	$0.073	0.00051	51	1.00051
99	142.83	$0.072	0.00051	51	1.00051
100	144.27	$0.072	0.00050	50	1.00050

Period - Reactivity Relationships for the KSU TRIGA Reactor

Period (s)	Doubling Time (s)	ρ/β ($)	ρ	PCM	k_{eff}
1	0.69	$0.788	0.00552	552	1.00555
2	1.39	$0.681	0.00477	477	1.00479
3	2.08	$0.610	0.00427	427	1.00429
4	2.77	$0.558	0.00390	390	1.00392
5	3.47	$0.517	0.00362	362	1.00363
6	4.16	$0.483	0.00338	338	1.00339
7	4.85	$0.455	0.00318	318	1.00319
8	5.55	$0.431	0.00302	302	1.00303
9	6.24	$0.410	0.00287	287	1.00288
10	6.93	$0.392	0.00274	274	1.00275
11	7.62	$0.375	0.00263	263	1.00263
12	8.32	$0.360	0.00252	252	1.00253
13	9.01	$0.347	0.00243	243	1.00243
14	9.70	$0.335	0.00234	234	1.00235
15	10.40	$0.324	0.00227	227	1.00227
16	11.09	$0.313	0.00219	219	1.00220
17	11.78	$0.304	0.00213	213	1.00213
18	12.48	$0.295	0.00206	206	1.00207
19	13.17	$0.287	0.00201	201	1.00201
20	13.86	$0.279	0.00195	195	1.00196
21	14.56	$0.272	0.00190	190	1.00190
22	15.25	$0.265	0.00185	185	1.00186
23	15.94	$0.258	0.00181	181	1.00181
24	16.64	$0.252	0.00177	177	1.00177
25	17.33	$0.246	0.00173	173	1.00173
26	18.02	$0.241	0.00169	169	1.00169
27	18.71	$0.236	0.00165	165	1.00165
28	19.41	$0.231	0.00162	162	1.00162
29	20.10	$0.226	0.00158	158	1.00158
30	20.79	$0.222	0.00155	155	1.00155
31	21.49	$0.217	0.00152	152	1.00152
32	22.18	$0.213	0.00149	149	1.00149
33	22.87	$0.209	0.00146	146	1.00147
34	23.57	$0.205	0.00144	144	1.00144
35	24.26	$0.202	0.00141	141	1.00141
36	24.95	$0.198	0.00139	139	1.00139
37	25.65	$0.195	0.00136	136	1.00137
38	26.34	$0.192	0.00134	134	1.00134
39	27.03	$0.188	0.00132	132	1.00132
40	27.73	$0.185	0.00130	130	1.00130
41	28.42	$0.182	0.00128	128	1.00128
42	29.11	$0.180	0.00126	126	1.00126
43	29.81	$0.177	0.00124	124	1.00124
44	30.50	$0.174	0.00122	122	1.00122
45	31.19	$0.172	0.00120	120	1.00120
46	31.88	$0.169	0.00118	118	1.00119
47	32.58	$0.167	0.00117	117	1.00117
48	33.27	$0.164	0.00115	115	1.00115
49	33.96	$0.162	0.00114	114	1.00114
50	34.66	$0.160	0.00112	112	1.00112

Period - Reactivity Relationships for the KSU TRIGA Reactor

Period (s)	Doubling Time (s)	ρ/β ($)	ρ	PCM	k_{eff}
51	35.35	$0.158	0.00110	110	1.00111
52	36.04	$0.156	0.00109	109	1.00109
53	36.74	$0.154	0.00108	108	1.00108
54	37.43	$0.152	0.00106	106	1.00106
55	38.12	$0.150	0.00105	105	1.00105
56	38.82	$0.148	0.00104	104	1.00104
57	39.51	$0.146	0.00102	102	1.00102
58	40.20	$0.144	0.00101	101	1.00101
59	40.90	$0.143	0.00100	100	1.00100
60	41.59	$0.141	0.00099	99	1.00099
61	42.28	$0.139	0.00098	98	1.00098
62	42.98	$0.138	0.00096	96	1.00097
63	43.67	$0.136	0.00095	95	1.00095
64	44.36	$0.135	0.00094	94	1.00094
65	45.05	$0.133	0.00093	93	1.00093
66	45.75	$0.132	0.00092	92	1.00092
67	46.44	$0.130	0.00091	91	1.00091
68	47.13	$0.129	0.00090	90	1.00090
69	47.83	$0.128	0.00089	89	1.00089
70	48.52	$0.126	0.00088	88	1.00088
71	49.21	$0.125	0.00087	87	1.00088
72	49.91	$0.124	0.00087	87	1.00087
73	50.60	$0.122	0.00086	86	1.00086
74	51.29	$0.121	0.00085	85	1.00085
75	51.99	$0.120	0.00084	84	1.00084
76	52.68	$0.119	0.00083	83	1.00083
77	53.37	$0.118	0.00082	82	1.00082
78	54.07	$0.117	0.00082	82	1.00082
79	54.76	$0.115	0.00081	81	1.00081
80	55.45	$0.114	0.00080	80	1.00080
81	56.14	$0.113	0.00079	79	1.00079
82	56.84	$0.112	0.00079	79	1.00079
83	57.53	$0.111	0.00078	78	1.00078
84	58.22	$0.110	0.00077	77	1.00077
85	58.92	$0.109	0.00077	77	1.00077
86	59.61	$0.108	0.00076	76	1.00076
87	60.30	$0.107	0.00075	75	1.00075
88	61.00	$0.106	0.00075	75	1.00075
89	61.69	$0.106	0.00074	74	1.00074
90	62.38	$0.105	0.00073	73	1.00073
91	63.08	$0.104	0.00073	73	1.00073
92	63.77	$0.103	0.00072	72	1.00072
93	64.46	$0.102	0.00071	71	1.00071
94	65.16	$0.101	0.00071	71	1.00071
95	65.85	$0.100	0.00070	70	1.00070
96	66.54	$0.100	0.00070	70	1.00070
97	67.24	$0.099	0.00069	69	1.00069
98	67.93	$0.098	0.00069	69	1.00069
99	68.62	$0.097	0.00068	68	1.00068
100	69.31	$0.096	0.00068	68	1.00068

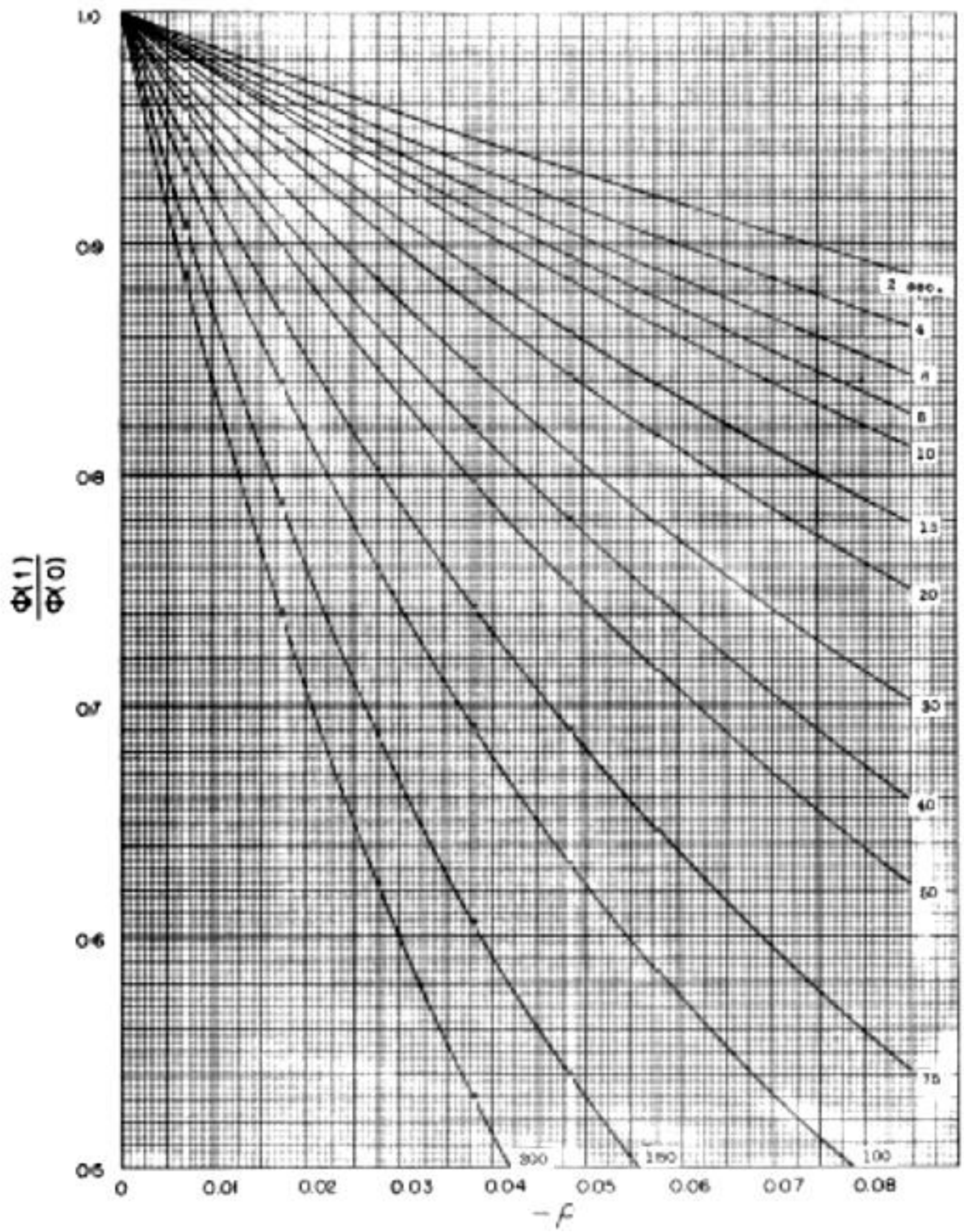

Figure 34 – Change in reactor flux following a step insertion of reactivity (from KSU TRIGA Hazards Summary Report).

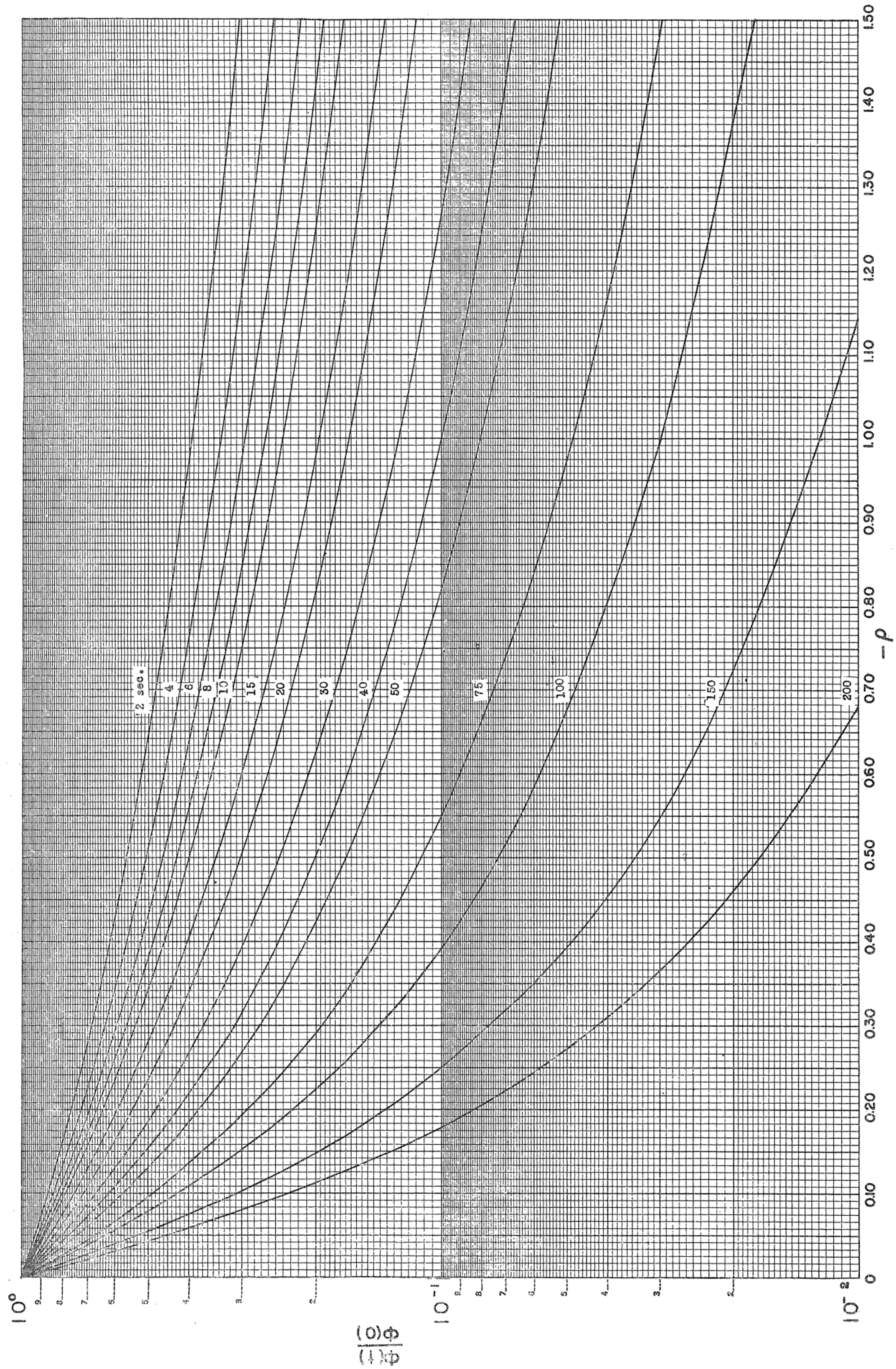

Figure 35 - Change in reactor flux following a step insertion of reactivity (from KSU TRIGA Hazards Summary Report).

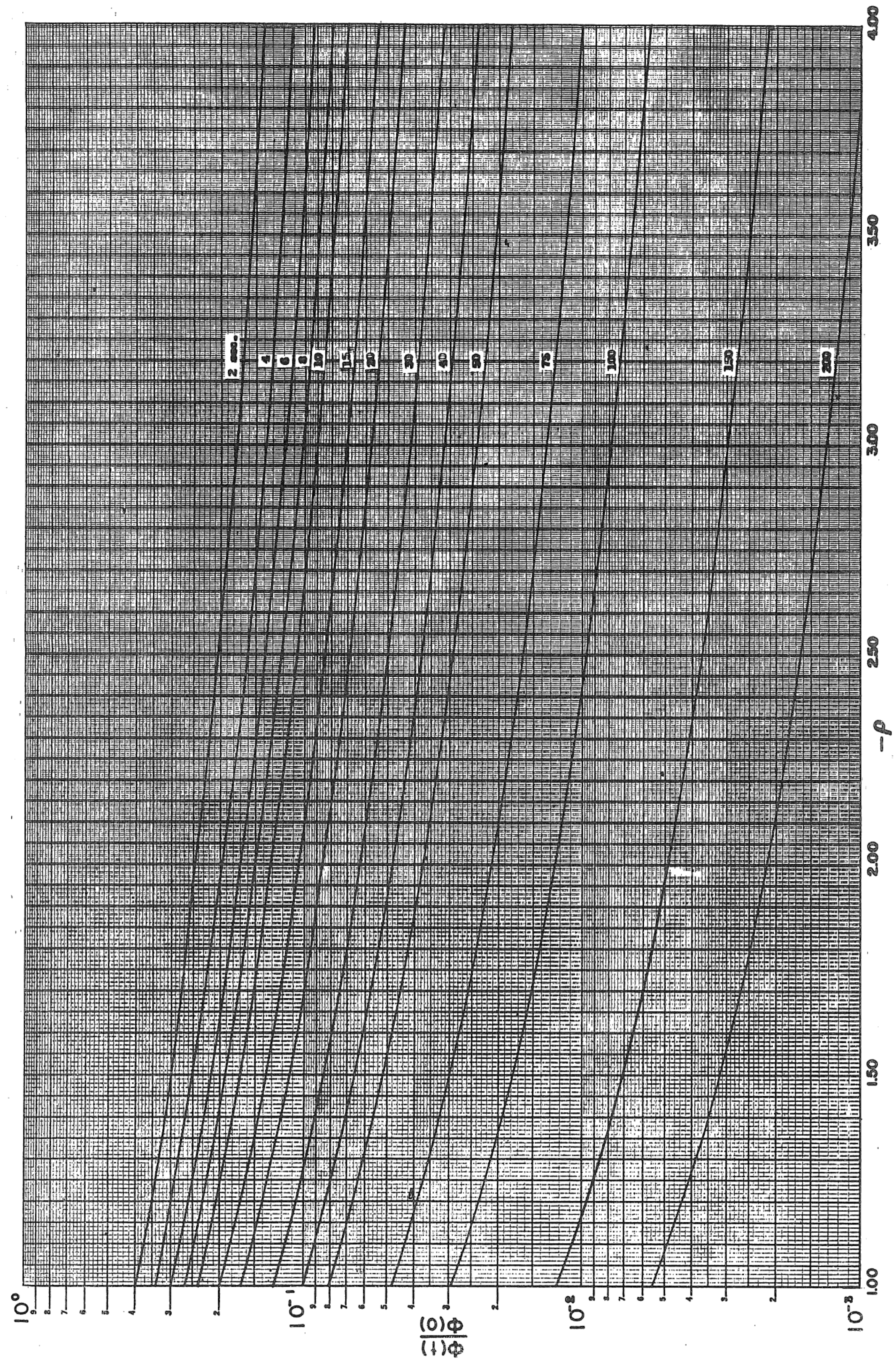

Figure 36 - Change in reactor flux following a step insertion of reactivity (from KSU TRIGA Hazards Summary Report).