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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

DATE: 12 September 2007

SUBJECT: **Response to September 11, 2007 Request for Additional Information (TAC NO. MC9031)**

Update to Proposed Kansas State University Research Reactor (Docket 50-188) Safety Analysis Report Chapter 7 and Technical Specifications

Dear Mr. Hughes:

The response to the September 07 Request for Additional Information follows, with the revised proposal for the related to the proposed Chapter 7 of the Safety Analysis report and Technical Specifications attached.

ITEM 1: TS 2.2.3 - Your safety analyses has shown that under the proposed Technical Specifications (TSs) the request of a licensed power of 1250 kW thermal (kW(t)) is appropriate. In addition, the accident analyses would allow a Limiting Safety System Setting (LSSS) of 1250 kW(t) and it would meet the requirements of 10 CFR 50.36. It is appropriate that the TSs provide the limits of operation and allow the licensee to administratively determine the actual setpoint and operation power, based on the accuracy and precision of the instrumentation, to assure that the limits are not exceeded. TS 2.2.3, as written, establishes a "no operation zone" between the licensed power and the LSSS and causes unnecessary confusion. Please propose a change in this TS to remove this confusion.

RESPONSE: Corrected to reflect 1250 kW (thermal) with a statement in the bases section that the actual set point will be established to ensure the LSSS is met.

ITEM 2: TS 5.1.3 (3) - In a previous RAI most TSs related to the use of aluminum clad TRIGA fuel were eliminated. TS 5.1.3 (3) is not adequately supported in the SAR and TSs. Please remove this TS if that was the intention, otherwise provide proper bases in the SAR and supporting TSs.

RESPONSE: Removed 5.1.3(3)

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ITEM 3: TS 3.4.3 - The description of the (standard) control rod interlock in the basis does not match the function as described in Table 2. Clarify the TS.

RESPONSE: *NOTE[1] The interlock descriptions in Chapter 7 were rewritten to indicate that two pulsing interlocks have a safety basis and are therefore included in Technical Specifications (preventing movement of standard rods in the pulse mode and preventing pulsing in steady state mode). These requirements are placed in 3.4 and 4.4 (Safety Channel and Control Rod Operability) with applicable requirements and bases statements. The interlocks that do not have a safety basis are removed from the Technical Specifications.*

C. Interlocks

Several interlocks are built into the control system of the reactor to prevent improper operation. These interlocks are hard-wired into the control rod drive circuitry. Two of the interlocks are required by Technical Specifications as they (1) ensure power level monitoring is adequate to detect power level during pulsing, and (2) ensure pulsing operations remain within assumptions of the safety analysis. One of the interlocks (3) provides a method of assuring that a power level detector is available to monitor startup, although the interlock may be bypassed under certain conditions. Two of the interlocks (4 and 5) are legacies of previous licensing bases and maintained as a good operating practice. The interlocks are described below:

1. *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.*

   Pulse operations are likely to exceed the maximum range of the power level instruments used in the steady state mode; therefore this interlock ensures that power level monitoring is configured for pulse operations.

2. *Only the pulse rod can be withdrawn when the reactor is in the PULSE mode to limit control rod reactivity addition during a pulse (i.e., to the pulse rod only). This interlock does not prevent the scramming of any control rod. The interlock function is provided manually engaging the source interlock with a pushbutton switch on the NLW-1000 instrument, to configure the channel after the reactor mode selector switch is placed in the pulse mode.*

   The amount of reactivity added during a pulse is controlled by the position of the pulse rod. Analysis assumes reactivity addition to the maximum nominal value of the pulse rod, and this interlock prevents movement of additional control rods during pulsing operations that could add to the pulse rod reactivity addition.

3. *An interlock prevents control rod withdrawal (shim, regulating and safety rods only) unless the count rate neutron channel is indicating > 2 cps, above the minimum sensitivity of the NLW-1000 power level monitoring channel. This interlock ensures that a neutron detector is functioning during startup, and provides the mechanism for implementing the pulse interlock as described below.*

   The interlock may be bypassed (a) during fuel loading operations when core inventory is not high enough to multiply the source above 2 cp, or (b) if channel operability can be verified by a source check.

4. *An interlock prevents simultaneous withdrawal of two or more control rods when the reactor is in the steady state mode.*

   The Technical Specifications for the original 100 kW facility operating license included a maximum reactivity addition rate; this interlock ensured that the maximum reactivity addition rate was not challenged by inserting reactivity form multiple control rods. The facility operating license was subsequently revised (as indicated in Chapter 1) to include pulsing operations, where up to $2.50 can be added in a fraction of a second, effectively removing the safety basis for the reactivity addition rate. The interlock remains in effect as a good operating practice.

5. *An interlock prevents reactor pulses from being fired if the reactor power is above 10 kW (normally set at 1 kW).*

   Previous safety bases did not address operations with elevated fuel temperature, while current accident analysis (Chapter 13) provides analysis with pulsing from power operations. The interlock remains in effect as a good operating practice.

ITEM 4: TS 3.8.4 - Action C uses "ASAP" as a completion time. This is not defined. Define the term or change it to a defined term.

RESPONSE: *Changed to IMMEDIATE*

ITEM 5: TS 3.5.4 - Operable is misspelled in Required Action A.2.d. Please correct.

RESPONSE: *Corrected*

ITEM 6: TS 6.2 - This TS contains 2 subsections listed as "e." Please correct.

RESPONSE: *Corrected*

ITEM 7: TS 6.8 - This TS contains the title "Director, Division of Reactor Licensing, NRC." The use of "NRC" would be more generic and appropriate.

RESPONSE: *Revised*

ITEM 8: TS 6.11.a) - Add Region IV to this 24 hr notification TS to meet the requirements of 10 CFR 50.36(c)(7)(ii).

RESPONSE: *Revised*

ITEM 9: TS 6.11.b) - Remove "to the NRC Operations Center." The first sentence in the TS 6.11 already provides the appropriate addressee for written reports.

RESPONSE: *Revised*

ITME 10: TS 6.11.c) - Remove "to the USNRC, Region IV, 611 Ryan Drive, Suite 400, Arlington, TX 76011-4005." The first sentence in the TS 6.11 already provides the appropriate addressee for written reports.

RESPONSE: *Revised*

ITEM 11: SAR Section 7.3.1 - First Channel In a previous version of the SAR the power level setting of the described interlock was 1 kW and not 10 kW? Discuss the reason for this change. Propose a TS for this interlock, otherwise provide justification for not making this interlock a TS.

RESPONSE: *See NOTE[1] above; in particular, the original Technical Specification required 1 kW, while the actual setpoint was established at 10 kW; neither the revised hazard analysis nor the previous version of the SAR did adequately reflected this situation.*

5. *An interlock prevents reactor pulses from being fired if the reactor power is above 10 kW (normally set at 1 kW).*

   Previous safety bases did not address operations with elevated fuel temperature, while current accident analysis (Chapter 13) provides analysis with pulsing from power operations. The interlock remains in effect as a good operating practice.

ITEM 12: SAR Section 7.3.1 - Third Channel The description in this section of the scrams that are bypassed or not when in pulse mode is not clear. In particular if this instrument is being used during pulse mode, is the high voltage scram bypassed? Please discuss.

RESPONSE: *The Chapter was revised in several places of 7.3.1 to better describe pulsing operations.*

ITEM 13: Propose a safety channel specification in section 3 of the TSs and an appropriate surveillance specification in section 4 of the TSs for the high voltage scrams on the power level instruments, or provide justification for not doing so.

RESPONSE: *Loss of high voltage is related to operability of the power level instruments; therefore an ACTION statement was placed in 3.3 for a loss of high voltage and the basis statement* *indicates instrument operability of the channel is the basis for the value.*

(3.3.4 Actions)

A.2 High voltage to reactor power level detector less than 90% of required operating value	*Establish REACTOR SHUTDOWN condition* *AND* *Enter REACTOR SECURED mode*	*A.2. IMMEDIATE*

(3.3.5 Basis)

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.

(4.3.2 Specification)

CHANNEL CHECK high voltage to required power level instruments	*DAILY*

ITEM 14: SAR Section 7.3.4.b - This section describes the pulse preset timer scram. Justify why including this scram as a TS is not necessary, otherwise propose a TS.

RESPONSE: *Current accepted value for the 250 kW TRIGA core prompt temperature coefficient is -\$0.17 per °C; General Atomics Report GA-4339 indicates the 1 MW TRIGA Mk III prompt temperature coefficient is is -\$0.18 per °C. Based on Chapter 4 analysis, a steady state operation at full power requires more that \$4.3 reactivity. With a maximum excess reactivity of \$3.00, the reactor cannot operate steady state at 1.25 MW and therefore pulsed insertion of maximum available excess reactivity will not create a condition where the reactor can operate in excess of the LSSS. The original Hazards Report for the Oregon State University 250 kW TRIGA Mark II Reactor, GA-6499, Figure 4.1 (Estimated reactivity loss versus power) shows that steady state operation at approximately 1.1 MW requires \$4.00, with \$3.00 corresponding to approximately 0.65 MW. The Safety Analysis Report for the Illinois Advanced TRIGA (August 1967) Figure III-13 shows at 1.25 MW a reactivity loss ranging from approximately \$4.2 to \$4.9, with a \$3.00 reactivity loss resulting from operation at power levels ranging from approximately 0.55 MW to 0.75 MW.*

Since a pulse cannot add enough reactivity to cause steady state power to exceed the steady state LSSS, there is no safety basis for requiring a reactor trip following a pulse to the maximum excess reactivity.

ITEM 15: SAR Section 7.3.4.c - The interlocks described in this section are in the surveillance section TS 4.4.2 but they are not specified in section 3 of the TSs. Specify the interlocks in section 3 of the TSs so it is clear what functionality is surveilled in TS 4.4.2.

RESPONSE: *The interlock description in Chapter 7 were rewritten to indicate that two pulsing interlocks have a safety basis and are therefore included in Technical Specifications (preventing movement of standard rods in the pulse mode and preventing pulsing in steady state mode). These requirements are placed in 3.4 and 4.4 (Safety Channel and Control Rod Operability) with applicable requirements and bases statements. Descriptions of the remaining interlocks indicate the historical reason/source of the interlock and the reason that the non-safety related interlocks are being maintained. The interlocks that do not have a safety basis are removed from the Technical Specifications*

(Table 2: Required Safety System Channels)

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

(3.4.5 Basis)

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 surveillance was revised to require a functional test.

(4.4.2 Specifications)

CONTROL ROD (STANDARD) position interlock functional test
Pulse rod interlock functional test

The basis was revised to explicitly identify the interlock functions.

(4.4.3 Basis)

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.

ITEM 16: SAR Section 12.5.3 and TS 6.11.c) - This section of the SAR states that a 30 day notification shall be made of any permanent changes in Facility Manager or head of the Department of Mechanical and Nuclear Engineering. This is consistent with ANS/ANSI 15.1. Propose an addition to TS 6.11.c) to require this report, or justify not doing so.

RESPONSE: *Revised*

If you have any questions or comments concerning this matter, you may contact me at 785-532-6657 or whaley@ksu.edu.

I verify under penalty of perjury that the foregoing is true and correct,

Executed on 12 September 2007,

P. M. Whaley

MARCIA E. CHACON
NOTARY PUBLIC
STATE OF KANSAS
My Appt. Exp. 5-12-2011

Marcia E Chacon 9-12-2007

Docket No. 50-188
Enclosures:

Proposed Safety Analysis Report, Chapter 11
Proposed Technical Specifications

Table of Contents

1. 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	Acticns 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 firmly 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 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.

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 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 1,150°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.

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 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

CONDITION	REQUIRED ACTION	COMPLETION TIME
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
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 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.

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
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.

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
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.

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

(1) 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.

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

(1) 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 130°F 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.

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.

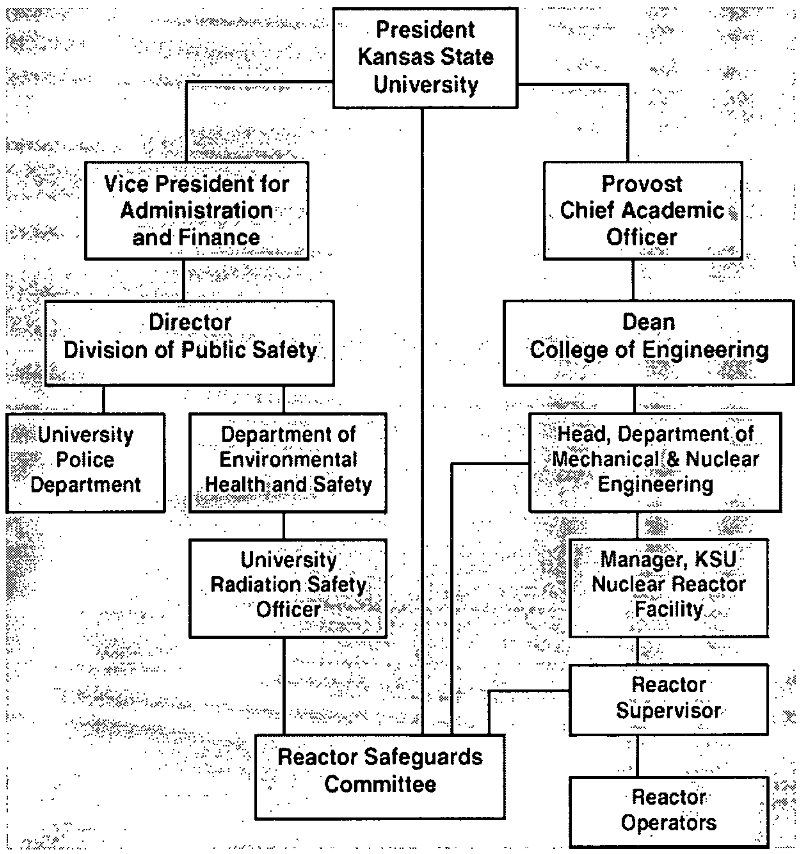


Figure TS.1: Organization and Management Structure for the K-State Reactor

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.

6.7 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.

7. INSTRUMENTATION AND CONTROL SYSTEMS

Much of the reactor's original instrumentation and control (I&C) systems were replaced during the control room modifications in 1993 and 1994. The original console and vacuum-tube instruments were replaced by a surplus solid-state console obtained from U.S. Geological Survey's TRIGA Mark I reactor. This console was then outfitted with new N-1000 series neutronic channels from General Atomics. These channels have optically isolated outputs, allowing other devices to utilize the neutronic data.

7.1 Summary Description

The bulk of the reactor I&C systems are hard-wired analog systems primarily manufactured by General Atomics and widely used at various NRC-licensed facilities. The general layout of these systems is shown in Figure 7.1.

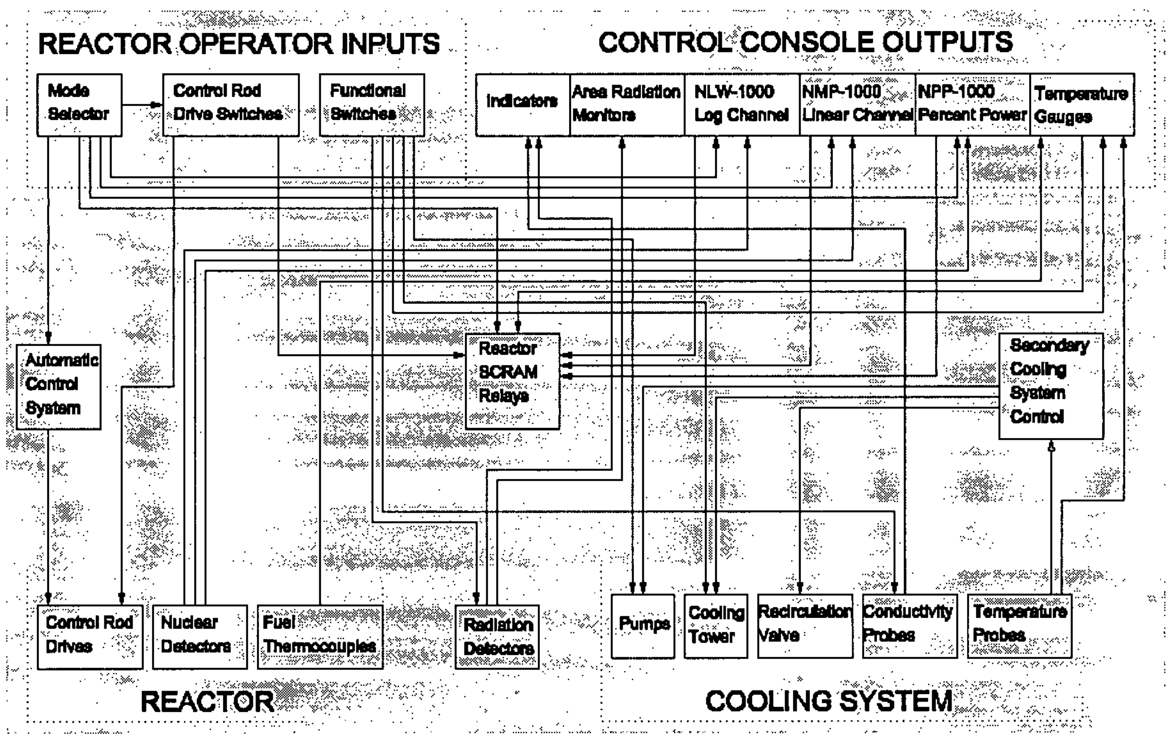


Figure 7.1, Inter-connectivity Diagram.

The reactor control system (RCS) consists of the instrumentation channels, the control rod drive circuitry and interlocks, and an automatic flux controller. The RCS measures several key reactor parameters including power, fuel temperature, water temperature, and water conductivity. Three neutronic instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. These provide at least two indications of reactor power from source range to power range during steady state operations. During pulsing operations (with the reactor mode selector switch in "Hi Pulse") the linear channel is disabled and the power channel switches to sub-channel of the percent power channel (capable of monitoring maximum pulse power). In additions, a "pulse interlock" is manually initiated on the wide wide-range logarithmic channel (which disables the channel). During pulsing operations a another neutron sensitive measuring channel is added to the central thimble to record pulse data. Fuel temperatures can be monitored on both the console and on an auxiliary panel. Primary water temperature is displayed on the console and measured by an RTD in the water box. Titanium electrodes at the entrance and exit of the cleanup loop measure water conductivity.

The control rod drives and their associated circuitry are simple in design. A rotary switch configures the primary mode of operation, namely automatic, steady-state, or pulse mode. Numerical indicators give drive position, with illuminated switches to manipulate the rods and to indicate rod and drive status. Several interlocks are incorporated to prevent unintentional rapid insertions of reactivity, except in pulse mode. An automatic control system links the RCS with the neutronic channels providing regulation of the power level.

The reactor protection system (RPS) is a component of the RCS instruments. The RPS will initiate a reactor scram if any of several measured parameters in the RCS are outside of their limited safety system settings. The reactor scram effectively places the reactor in a subcritical configuration by releasing the control rods from their respective drives. Since the rods are no longer physically attached to the drive, they fall into the reactor core by gravity. High reactor power, high fuel temperature, loss of detector high voltage, loss of building power, and short reactor period will automatically cause all of the control rods to be dropped into the reactor core. A bar above the control rod drive switches allows this system to be actuated manually. Since the core is cooled by natural convection, no other engineered safety features are necessary for safe reactor shutdown.

The control console and display instruments are primarily housed in a control console, with auxiliary instruments located in a rack next to the console. At the console, the reactor operator has direct control over mode of operation, control rod drive positions, cooling system operation, opening of reactor bay doors, and manual scram of the reactor. Display instruments located in the control console provide measurements of reactor power, control rod positions, primary water temperature, and fuel temperature. Indicators in the console display scram information, low air pressure, low primary water level, high reactor sump water level, sump high water level, sump overflow water level, secondary surge tank level low, source interlock status, reactor bay upper door open, reactor bay lower door open, thermal column door open, person on stairway, and rod drive status. Secondary surge tank makeup is controlled with a backlit pushbutton that indicates surge tank level and surge tank makeup valve status. Console instruments monitor temperatures in the primary cooling system, and instrumented fuel elements. An intercom system in the console provides communication to numerous locations around the reactor bay and staff offices. In the auxiliary rack, the operator can control pneumatic transfer system operation and actuate timers for testing the control rod drop time. Display instruments located in this rack include, primary water conductivity, water activity, remote area radiation monitors, and a strip-chart output of reactor power. Several audible alarms indicate high radiation levels in the primary coolant and at various locations throughout the reactor bay. A breaker-box in the control room provides control over electrical devices in the reactor facility, including ventilation systems.

Radiation protection instruments are distributed throughout the reactor bay. All instruments have visual indication of radiation level, visible alarm conditions, and audible alarms. Radiation area monitors (RAM) strategically cover potential radiation areas throughout the reactor bay. A combined pool surface and primary water monitor indicated water activity. A 5 $R \cdot h^{-1}$ evacuation alarm is located on the 22-foot level. A continuous air monitor (CAM) is energized during reactor operation.

The human-machine interface principles incorporated into control room design allow the reactor to be operated by a single individual. All monitoring instruments are visible to the reactor operator at the console. The instruments and controls necessary for reactor operation are within reach of the operator, including an intercom and telephone. Surveillance instruments are located next to the console, with visual and audible alarms to signal the operator to abnormal conditions.

7.2 Design of Instrumentation and Control System

7.2.1 Design Criteria

Reliability of essential equipment is ensured through redundancy. Multiple instruments and safety systems perform similar functions for all modes of reactor operation. The construction and installation of instruments was performed according to applicable regulations at the time of introduction. However, all crucial instruments are checked daily for calibration and operability. Testing and calibration procedures exist for repair and general service. The majority of these I&C systems were manufactured by General Atomics, or other industrial manufacturers of nuclear equipment. Crucial systems to be considered include neutronic instruments, control rod drives, radiation monitors, and control systems.

Redundancy is designed into each of these systems. During steady state operation, a minimum of two neutronic channels provide reactor power level indication, two of which provide high power level RPS actuation (scram). These neutronic instruments are tested prior to reactor operation for demonstration of scram capability. Two are also tested for operability by internal calibration tests. There are two fuel temperature indications. The control rod drives drop their rods into the reactor core upon loss of power or RPS actuation, providing sufficient shutdown margin with even the most reactive rod stuck out. Multiple remote area radiation monitors cover important areas, including two directly above the reactor core and two monitoring primary coolant activity.

The maximum steady state power level for KSU TRIGA Mark II reactor is proposed to be 1,250 kW. Similar reactors operate up to 2 MW with 2 GW pulses. Therefore, the limited safety systems settings associated with reactor are extremely conservative when compared to the safety limits of the reactor. Thus the reactor has a considerable safety margin.

7.2.2 Design-Basis Requirements

The primary function of the RCS is to govern the manner in which reactivity is varied in the reactor core. The RCS system should prevent the reactor operator from unintentionally inserting large amounts of reactivity, through various interlock systems. The operator should only be able to remove one rod at a time from the reactor core, preventing large insertion rates. The pulse rod must not be able to be rapidly ejected from the core while in steady-state operation. Furthermore, the pulse rod should be the only rod that can be withdrawn in pulse mode, preventing supercritical pulses. There should also be an interlock to prevent startup without a power level signal above the minimum instrument sensitivity, preventing unmonitored or unanticipated criticality. Rod position indicators should show the rod position to 0.2% or total travel for accurate reactivity calculations.

Another primary function of the RCS is to provide the reactor operator with reactor status information. Reactor power, a crucial parameter, requires at least two instruments to provide confirmation of reactor power from shutdown to operating levels. Three instruments are used to cover this range: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. Accuracy of measurement at full rated power increases accordingly with the refinement of scale. The log channel provides gross reactor power indication and is accurate to 20% of scale, the linear channel is accurate to 5% of scale, and the percent power channel is accurate to 3% of scale. The percent power channel will also display pulse parameters for large pulses. These instruments are calibrated annually and checked for operability at the start of each operating day. An additional channel is installed and calibrated in the central thimble to record pulse data. Fuel temperature must be monitored during pulsing operation.

The primary function of the RPS is to automatically insert the control rods into the reactor core when certain parameters deviate from limited safety system settings. Several scrams involve the neutronic channels in the RCS. If 110% rated power level is exceeded in steady state mode, one of two trip-points will scram the reactor. Failure of the high voltage power supplies for operating neutronic channels will also cause a scram. For pulsing operations, a scram will be actuated when the fuel temperature is in excess of 450°C. Manual scram will be available in all modes of reactor operation. Rod drop times for the standard rods will be measured regularly to ensure proper RPS function. No other ESF features are required in this design.

The primary function of the radiation monitoring instruments is for personnel protection measures and emergency assessment actions. The area monitors provide the reactor operator with information regarding the actual radiation environment inside the reactor bay. With this knowledge, reactor users can be informed of possible hazards. A 5 $R \cdot h^{-1}$ monitor on the 22-foot level signals personnel to evacuate the reactor bay. A number of survey instruments (ion chambers, rem balls, G-M counters) are also available to personnel. Other instruments such as the constant air monitor, pool surface monitor, and water box monitor indicate the presence of dispersible radioactive materials, an indication of possible fuel cladding failures.

The control room is designed so a single operator can manipulate all significant controls without leaving the room. The reactor operator should be able to de-energize all equipment and experiments in the reactor bay. The control room should provide sufficient ventilation to provide cooling of the reactor instruments.

7.2.3 System Description

The overall system layout is depicted in Figure 7.2. The majority of the RCS is housed in a General Atomics (GA) console originally manufactured for the USGS reactor, which is shown with modifications in Figure 7.2. A detailed description of this figure is provided in Table 7.1. Figure 7.3 shows a representative layout of the auxiliary instrumentation rack. Since the instrument racks are general use equipment, configuration may be changed to allow better utilization of space, installation of new equipment, support specific equipment modifications, etc. without affecting function. The functions of each piece of equipment in this configuration are discussed in following sections with additional figures showing location and layout.

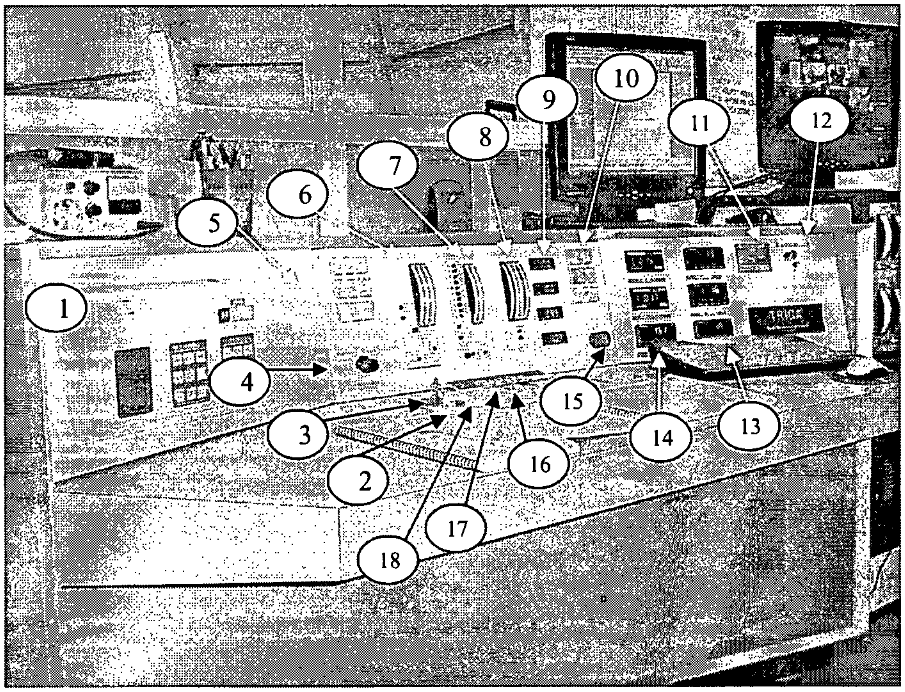


Figure 7.2, USGS TRIGA Console with Modifications.

7.2.4 System Performance Analysis

The system performance of the current I&C systems surpasses the original equipment. Reliability has been high, with few unanticipated reactor shutdowns. Since daily checkouts are performed, any discrepancies would be observed and corrected in a prompt manner. The opto-isolated outputs of the neutronic channels allow the data to be utilized by other devices without concern over those devices affecting the channels. A line conditioner provides regulated power to the instruments, protecting the equipment from electrical disruptions.

7.2.5 Conclusion

The current I&C systems outperform the original equipment supplied with the reactor, while meeting all of the necessary design bases for the facility. The human design factors used in control room development allow the reactor to be operated by a single individual. Checkout and testing procedures ensure that all equipment is maintained in operational status.

7.3 Reactor Control System

The bulk of the reactor control system (RCS) is housed in the USGS console shown in Figure 7.2. The remainder is contained in the auxiliary rack-mount panel next to the console, shown in Figure 7.3. The RCS consists of the instrumentation channels, the control rod drive circuitry and interlocks, and an automatic flux controller. These are shown in Figure 7.4. The RCS measures several key reactor parameters including power, fuel temperature, water temperature, and water conductivity.

Table 7.1, Description of Figure 7.2.

Number	Function	Description
1	Facility intercom	Control panel
2	Console Power	Push Button Switch
3	Magnet Power / Scram Reset	Key Switch
4	Mode selector switch	Rotary switch
5	SCRAM and interlock indicators	Annuciators
6	NLW-1000, wide range log power level indicator	
7	NMP-1000, multi-range linear power level indicator	
8	NPP-1000, Power & Pulse power level indicator	
9	Rod position indicators	
10	Process controls and annunciators	Low Air Pressure, Hi & Hi-Hi sump level, surge tank level & makeup, Upper & Lower Doors, and Cooling System Power
11	Period scram bypass	Key Switch
12	Source range interlock override	Key Switch
13	Fuel temperature instruments	
14	Cooling system temp instruments	
15	%-Demand potentiometer	Auto-flux control pot.
16	Rod controls	Backlit pushbuttons for UP, DOWN, CONT, and ON/AIR
17	Manual SCRAM bar	
18	Pulse/transient rod air solenoid control	Backlit pushbutton

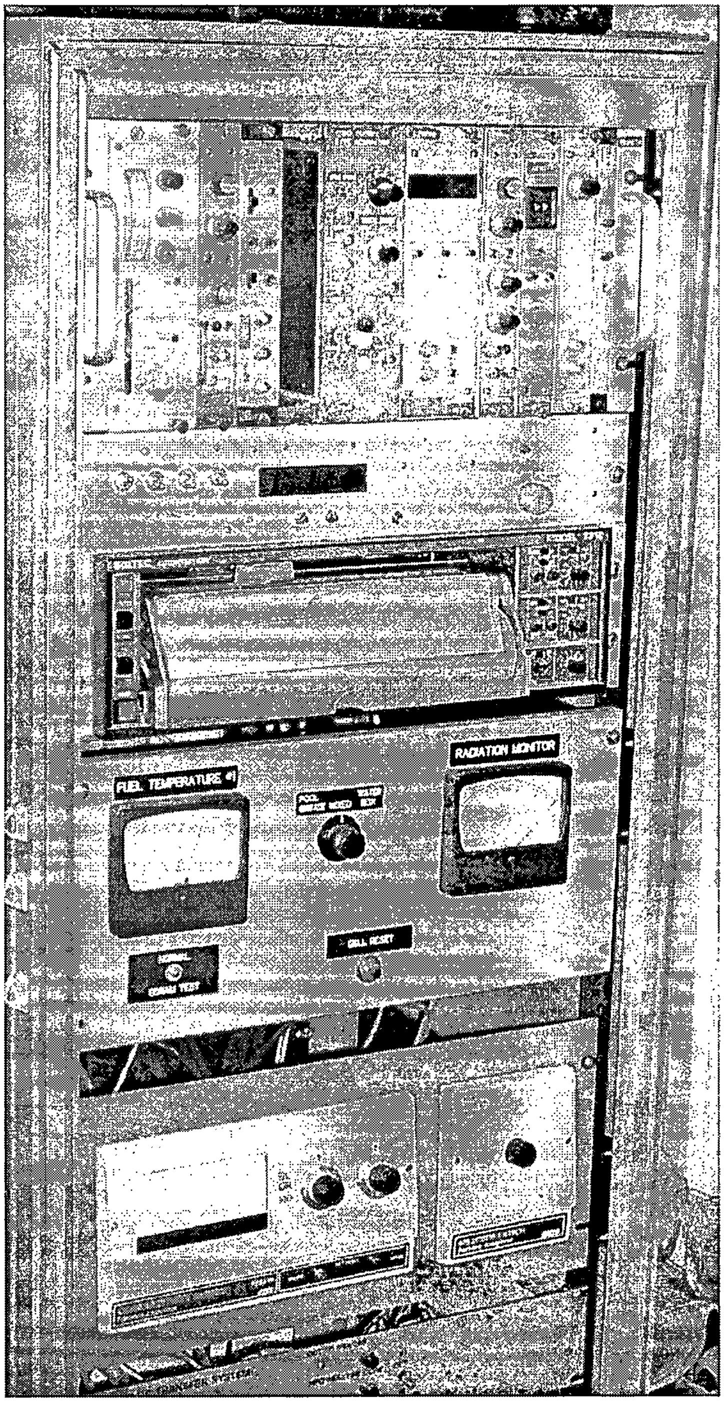


Figure 7.3, Instrumentation Rack.

7.3.1 Neutronic Instruments (Reactor Power)

Three neutronic instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel, as shown in Figure 7.5. Wiring diagrams and calibration procedures are found in the instrument maintenance manuals listed in the bibliography.

The wide-range log channel uses a fission counter for detecting thermal neutrons in the range of 1.4 to 1.4 x 10^5 nv, and provides approximately 0.7 counts·nv^{-1}. The detector has an aluminum case, an aluminum electrode, a U_3O_8 (>90% enriched in ^{235}U) coating 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 it is located on the 12-foot level.

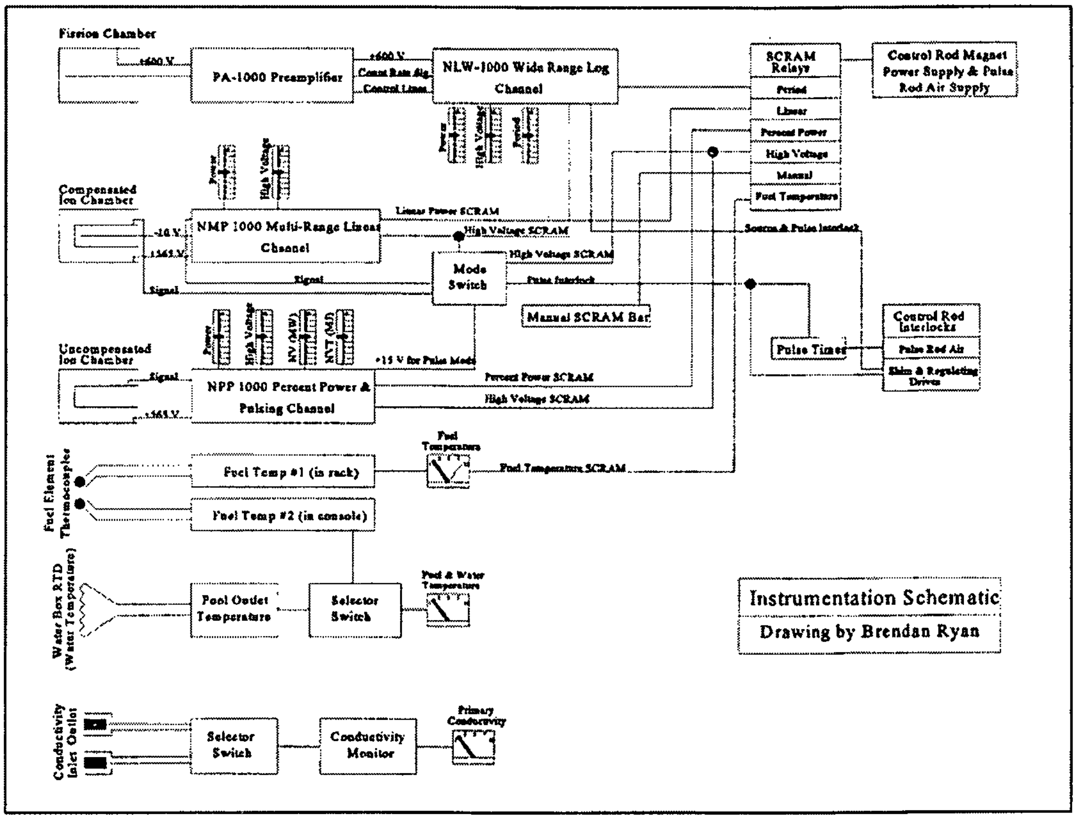


Figure 7.4, Instrumentation Diagram.

The remainder of the channel circuitry is located in the NLW-1000 unit in the central console. The NLW-1000 unit supplies the high voltage for the detector and power for the preamplifier. 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 power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. The period meter has a scram at 3 sec and there is a high voltage scram, both of which are bypassed in pulse mode. This channel also provides a protective interlock which prevents rod withdrawal when indicated neutron flux is < 2 cps, which is also activated in pulse mode to prevent removal of the shim, safety and regulating rods. This interlock acts by disabling the channel, effectively protecting the channel from over range condition and preventing a period trip. Another interlock prevents pulsing when reactor power is above 10 kW (normally set at 1 kW). The unit has two calibration checks in pulse mode, two in current mode, and checks for the period and high voltage scrams.

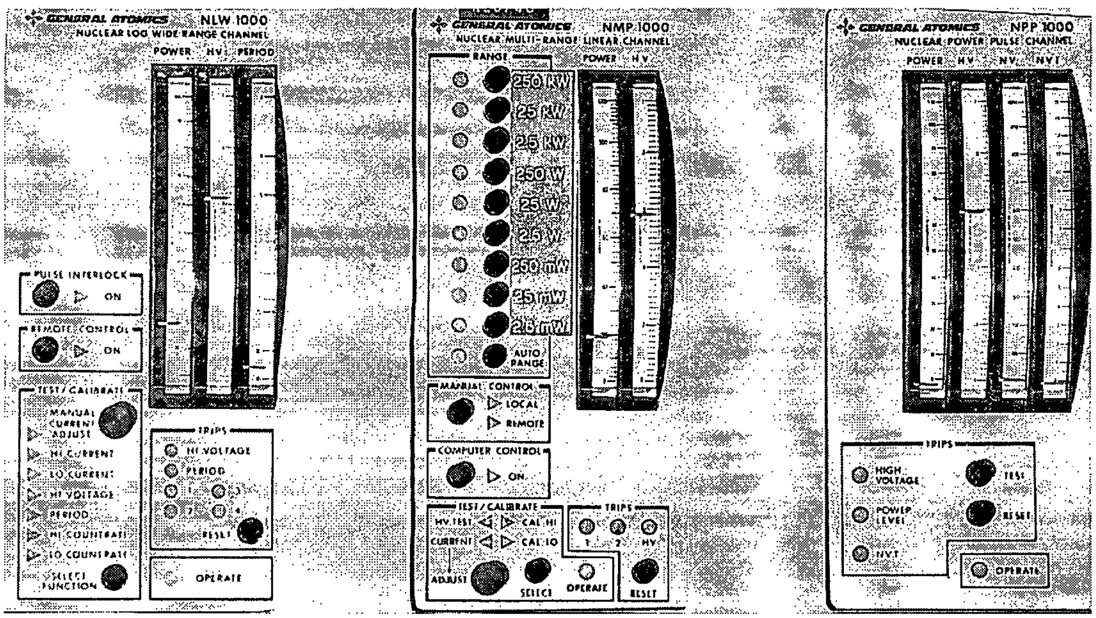


Figure 7.5, N-1000 Series Instruments.

The second channel provides 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 NMP-1000 unit in the center console, which in turn supplies high and compensation voltages. This unit features automatic or manual ranging to select the appropriate decade of power displayed. The instrument provides two indicators, power and high voltage. The power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. In addition there is also a high power level scram (normally set for 104% nominal rated power) and a high voltage scram. The signal from the detector and the high voltage scram are bypassed in the pulsing mode. The unit has two calibration checks, an auto-ranging test feature, and checks for high power level and high voltage scrams. Placing the reactor mode selector switch in "Hi Pulse" disables the channel, thereby preventing an over range condition and preventing a reactor trip at the nominal steady state power level trip point.

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 detector sends its signal to the NPP-1000 instrument in the center console, which provides a visual indication of reactor power, high voltage, nv, and nvt measurements. The NPP-1000 supplies the high voltage for the detector. There is a high power scram (normally set for 104% of full power) and a high voltage scram. In pulse mode, the channel is designed to monitor maximum power and integral output of a reactor pulse, disabling the normal steady state power channel portion of the instrument. This action consequently disables the nominal steady state power level trip point. Pulse output readings are measured in reference to the 250 MW maximum. Hence an additional channel is added to the central thimble to permit recovery of data from pulses of various magnitudes. The unit has checks for high power and high voltage scrams.

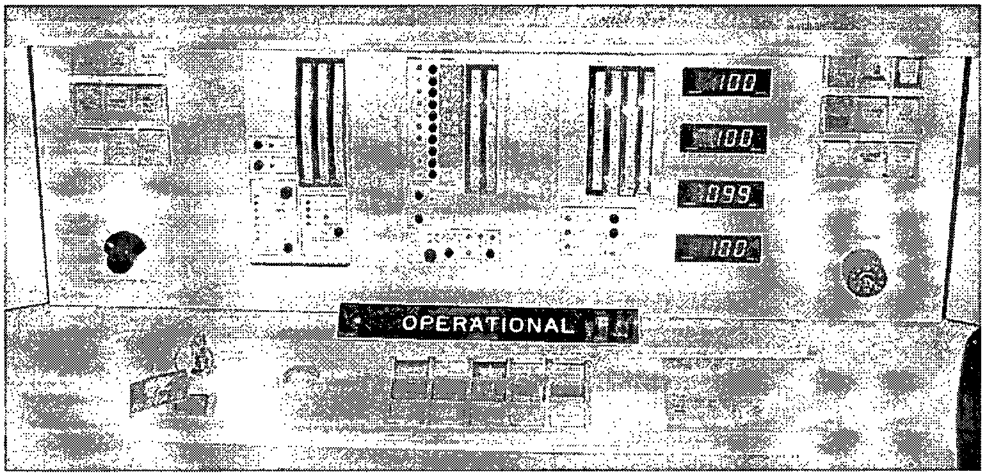


Figure 7.6, N-1000 Series Instrument Installation.

An added pulsing channel consists of a small BF_3 chamber, which can be inserted into the central thimble of the reactor core. A separate high voltage supply powers the instrument and a multi-range picoammeter reads the detector current. A reference voltage output of the picoammeter is sent to a computer in the control room, which collects the pulse data. This channel is calibrated prior to pulsing operations and range selected in advance based upon the anticipated peak power.

The original instruments that the N-1000 series units replaced are still housed in the control console for backup use. These older analog devices have all of the same measurement and RPS features, except that they lack opto-isolated outputs for computer acquisition of reactor data. They also require more manual input as the linear channel does not possess auto-ranging features. These instruments were used for many years at USGS and for one year at K-State until the N-1000 units arrived, and provide adequate backup for an interim time while the N-1000 series units are serviced. Wiring diagrams and calibration procedures for these instruments are located in the maintenance manual for the USGS console.

7.3.2 Temperature

Temperature indications for the primary water and specific B-Ring fuel elements are provided on the front section of the control panel. The instrumented fuel elements have three chromel-alumel thermocouples in the fuel element that are used for temperature indication on the console or in the instrumentation rack. The thermocouples are located 0.76 cm (0.3-in) below the fuel surface, spaced at the midpoint of the element and at ±, 2.5 cm from the midpoint; an averaged value from all three thermocouples is typically used for instrument readingsThe temperature in the primary cleanup loop is a nickel alloy thermistor, and is displayed on a console meter, which is shared with fuel temperature via a rotary switch. Another indication of fuel temperature is located in the instrumentation rack with the capability of initiating reactor scram if the measured fuel temperature exceeds a preset value (normally 400 °C).

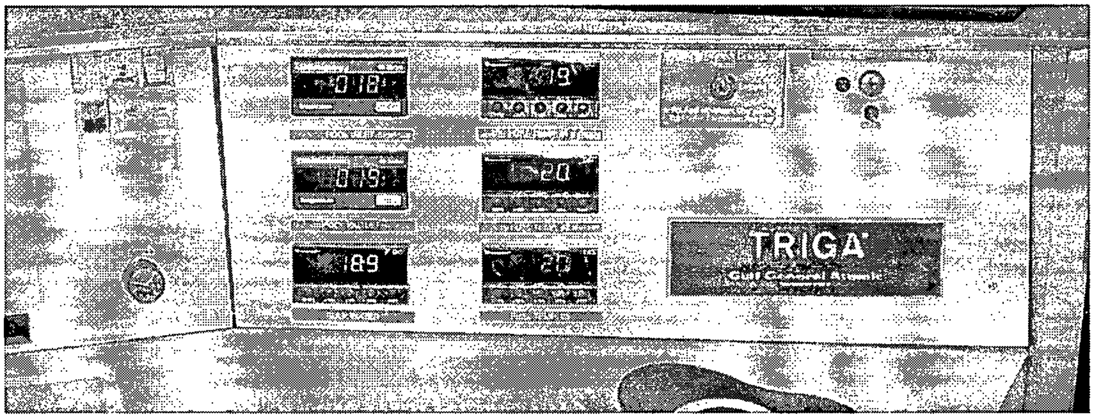


Figure 7.7, Console Temperature Instrumentation.

Several other temperature measurements can be obtained from the computer in the control room. The computer can read two additional fuel thermocouples from other fuel elements in various positions in the reactor core. Additionally AD590 temperature transducers are located on the inlet and exit of both the primary and secondary sides of the heat exchanger to evaluate performance. Two other transducers are located in the reactor tank for bulk pool temperature measurement and high temperature alarm.

7.3.3 Water Conductivity

Primary water conductivity is measured at the inlet and outlet of the purification loop by titanium electrode cells that send signals to a bridge circuit in the instrumentation rack. The bridge circuit is automatically temperature compensated and nulled to provide good conductivity measurements over all reactor conditions. The inlet and outlet conductivities provide a good indication of the overall purity of the primary water and the effectiveness of the ion exchanger.

7.3.4 Control Rod Drives

Four control rods are required for reactor operations at 1,250 kW to meet reactivity requirements: a shim rod, a regulating rod, a transient rod, and a safety rod. The shim, regulating and safety rods share identical control circuitry (Figure 7.9) and provide coarse and fine power control. The standard rod drives are analog-motor and control systems. Drive position is determined by voltage drop across a potentiometer that is adjusted as the control rod drive is moved. The position indicator is coupled to a shaft coupled to the drive motor shaft with a setscrew. The pulse rod is designed so that it can be rapidly ejected from the core to a preset height to initiate a reactor pulse. However, it still functions as a normal control rod in steady state mode. All rods can be individually scrammed without shutting down the reactor.

a. Standard Control Rod Drives

The standard rod drive mechanism (see Figure 7.8) is an electric motor actuated linear drive, equipped with a magnetic coupler. Its purpose is to adjust the reactor control rod position. In the analog drive motor, a 110-V, 60-cps, two phase motor drives a pinion gear and a 10-turn potentiometer. The potentiometer provides rod position indication. 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 draw tube. If the magnet is energized, the armature and connecting rod will follow the draw tube 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 (5 cm). of the bottom of its travel, the piston is restrained by the dashpot action of the restricted ports in the lower end of the barrel. This restraint cushions bottoming impact. Control rod drop times are measured semi-annually and must be less than one second.

The analog 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 (see Figure 7.7), electrically locking the motor. Clockwise (up) or counter-clockwise (down) rotation is enabled by shifting the phase between the windings with a 1-μF capacitor; motor control switches allow the appropriate phase shift. The stepper motor operates using phase switched direct current power. The motor shaft advances 200 steps per revolution (1.8 degrees per step). Since current is maintained on the motor winding when the motor is not being stepped, high holding torque is maintained. A translator module drives the stepping motor.

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, which 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 draw tube down, then the rod down microswitch must be open before the magnet down microswitch during coupled withdrawal of the control rod.

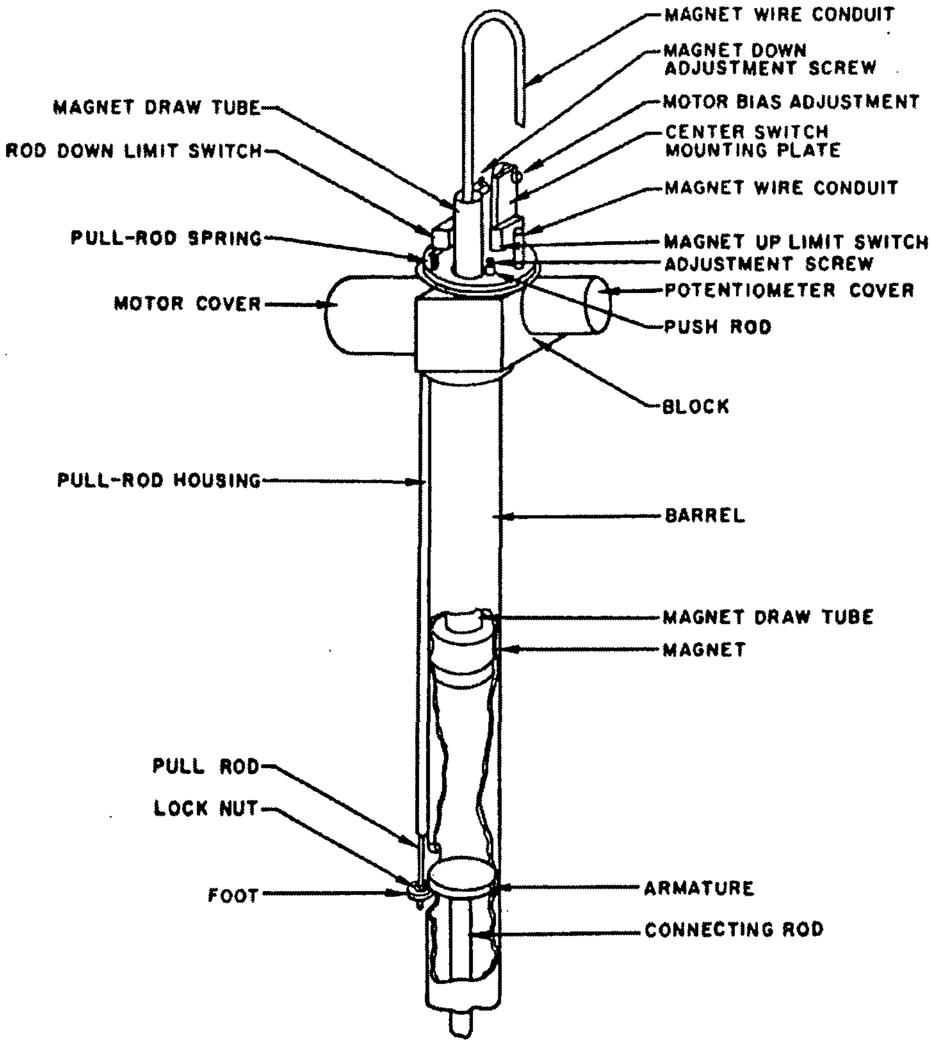
Three lights indicate that, 1) the magnet drawtube is full up, 2) the magnet drawtube is full down, and 3) the armature and magnet are coupled. When the magnet drawtube is full up, microswitch (S901) is actuated opening the short across the magnet up light (DS321). When the magnet draw tube is full down, microswitch (S902) is actuated opening the short across the magnet down light (DS324). When the control rod drops, the non-actuated 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, 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 indicating 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 (with the exception that one unit uses a stepper motor) and they are, therefore, interchangeable.

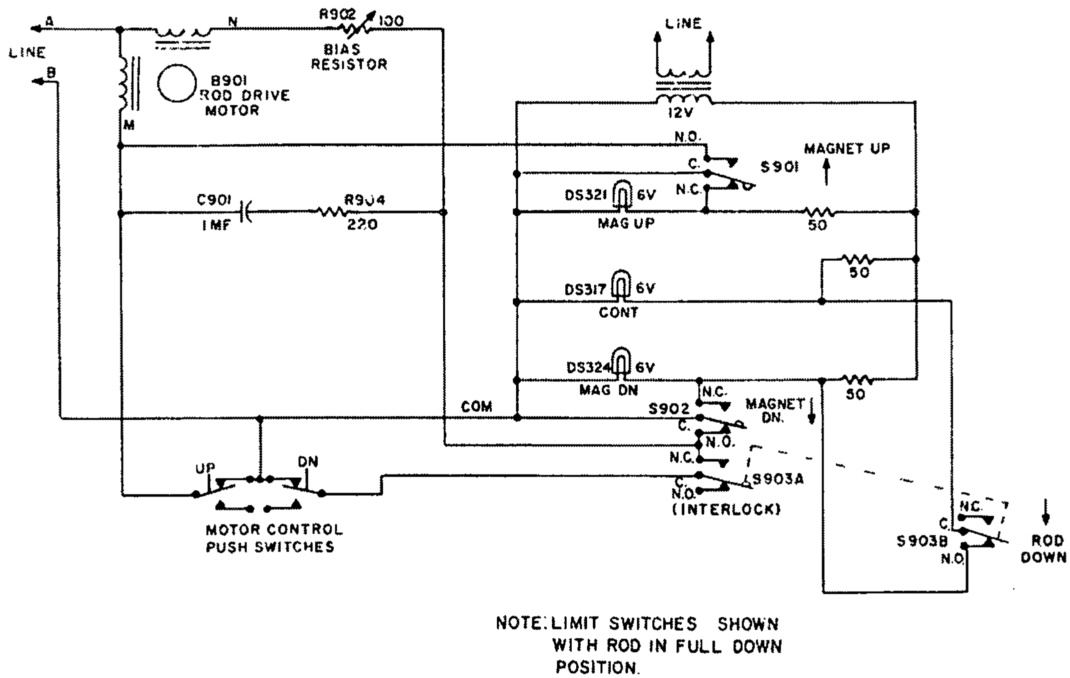
The rod position indicators are three digit, LED display indicators that receive a variable DC voltage input from 10-turn potentiometers that are driven by the respective rod drive motors. The digital display is simply a voltmeter, since the voltage across the potentiometer is directly related to the 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 on the drive and the associated wiring.

Normal rod motion speed is about 12-in. per minute. Using rod speed, rod position indication at UP and DOWN limit switch positions, and respective rod worth curves, the operator can determine the reactivity insertion rate for a given interval of rod motion.



Standard rod drive mechanism

Figure 7.8, Regulating, Safety and Shim Rod Drive.



Rod drive, motor control, and indicator lamp circuit

Figure 7.9, Control Rod Drive Circuit.

b. Transient Rod Drive

The transient rod-drive (Figure 7.10) is an electrically controlled, pneumatically operated, mechanically limited system. The transient rod and aluminum extension rod are 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. A three-way solenoid valve mounted below the support controls air to the piston. 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, safety and regulating control rods. The drive circuitry is identical to the shim, safety and regulating rods (Figure 7.9), except that the motor is not continuously energized. Relays in the drive unit allow it act similarly to the electrically locked motors. Since air is used to support the rod, there is no compensation for rod weight. The remaining differences involve the pneumatic relay controls.

The solenoid valve is actuated by means of the console mounted transient rod fire and air release (scram) 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 release 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. Micro-switches are used to indicate the extreme positions, up or down, of the shock absorber. In steady state mode, an interlock prevents actuation of the rod fire switch if the drive is not in its fully down position.

In the pulse mode, a variable timer (usually six seconds) de-energizes the solenoid valve 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 the steady state mode of operation, the adjustable (normally six second) timer is disengaged and the cylinder remains pressurized. If the air supply for the pulse rod drive should drop below approximately 45 psig, an amber low air pressure warning light will be actuated on the control console. Loss of air pressure will cause the rod to fall into the core.

c. Interlocks

Several interlocks are built into the control system of the reactor to prevent improper operation. These interlocks are hard-wired into the control rod drive circuitry. Two of the interlocks are required by Technical Specifications as they (1) ensure power level monitoring is adequate to detect power level during pulsing, and (2) ensure pulsing operations remain within assumptions of the safety analysis. One of the interlocks (3) provides a method of assuring that a power level detector is available to monitor startup, although the interlock may be bypassed under certain conditions. Two of the interlocks (4 and 5) are legacies of previous licensing bases and maintained as a good operating practice. The interlocks are described below:

1. 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.

   Pulse operations are likely to exceed the maximum range of the power level instruments used in the steady state mode; therefore this interlock ensures that power level monitoring is configured for pulse operations.

2. Only the pulse rod can be withdrawn when the reactor is in the PULSE mode to limit control rod reactivity addition during a pulse (i.e., to the pulse rod only). This interlock does not prevent the scramming of any control rod. The interlock function is provided manually engaging the source interlock with a pushbutton switch on the NLW-1000 instrument, to configure the channel after the reactor mode selector switch is placed in the pulse mode.

   The amount of reactivity added during a pulse is controlled by the position of the pulse rod. Analysis assumes reactivity addition to the maximum nominal