UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FACILITY OPERATING LICENSE
DOCKET NO. 50-607
DEPARTMENT OF THE AIR FORCE AT McCLELLAN AIR FORCE BASE

License No. R-130

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

   A. The application for license, filed by the Department of the Air Force at McClellan Air Force Base, on October 23, 1996, as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. Construction of the facility was completed in substantial conformity with the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   F. The licensee is a Federal agency and will use the facility for defense programs and research. The licensee, in accordance with 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," is not required to furnish proof of financial protection. The licensee has executed an indemnity agreement that satisfies the requirements of 10 CFR Part 140 of the Commission's regulations;

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

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

I. The receipt, possession, and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23, and 70.31.

2. Facility Operating License No. R-130 is hereby issued to the Department of the Air Force at McClellan Air Force Base as follows:

A. The license applies to the training reactor and isotopes production, General Atomics (TRIGA) nuclear reactor (the facility) owned by the Department of the Air Force at McClellan Air Force Base (the licensee). The facility is located on the licensee's site at McClellan Air Force Base and is described in the licensee's application for license of October 23, 1996, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Department of the Air Force at McClellan Air Force Base:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location at McClellan Air Force Base, in accordance with the procedures and limitations set forth in this license.

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use up to 21.0 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 4 grams of contained uranium-235 of any enrichment in the form of fission chambers; up to 16.1 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of plates; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess, and use a 4-curie sealed americium-beryllium neutron source in connection with operation of the facility; a 55-millicurie sealed cesium-137 source for instrument calibrations; small instrument calibration and check sources of less than 0.1 millicurie each; and to possess, use, but not separate, except for byproduct material produced in reactor experiments, such byproduct material as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not in excess of 2300 kilowatts (thermal) and in the pulse mode with reactivity insertions not to exceed $1.75 (1.23 $\%\Delta k/k$).

(2) Technical Specifications

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

(3) Physical Security Plan

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security plan, including all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The approved plan, which is exempt from public disclosure pursuant to the provisions of 10 CFR 2.790, is entitled "Physical Security Plan for the McClellan Nuclear Radiation Center (MNRC) TRIGA Reactor Facility," Revision 3, dated August 1996.

D. This license is effective as of the date of issuance and shall expire twenty (20) years from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: August 13, 1998

TECHNICAL SPECIFICATIONS

FOR THE

MCCLELLAN NUCLEAR RADIATION CENTER (MNRC)

REACTOR FACILITY

DOCUMENT NUMBER: MNRC-0004-DOC

TECHNICAL SPECIFICATION APPROVAL

This "Technical Specification for the McClellan Nuclear Radiation Center (MNRC) Reactor" has undergone the following coordination.

Reviewed by: ______________________________
Health Physics Supervisor (Date)

Reviewed by: ______________________________
Operations Supervisor (Date)

Approved by: ______________________________
Chief, Nuclear Licensing and Operations (Date)

Approved by: ______________________________
Chairman, SM-ALC Nuclear Safety Committee (Date)

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS
FOR THE
MCCLELLAN NUCLEAR RADIATION CENTER (MNRC)
REACTOR FACILITY

General

The McClellan Nuclear Radiation Center (MNRC) reactor is operated by the United States Air Force at McClellan Air Force Base, Sacramento CA. The MNRC research reactor is a TRIGA type reactor. The MNRC provides a high sensitivity inspection capability for detection of early stage corrosion in aluminum aircraft components, thereby reducing aircraft crash risk and reducing repair cost. In addition, the MNRC provides a wide range of irradiation services for both military and non military tasks. The reactor operates at a nominal steady state power level up to and including 2 MW. The MNRC reactor is also capable of square wave and pulse operational modes. The MNRC reactor fuel is less than 20% enriched in uranium-235.

1.0 Definitions

1.1 As Low As Reasonably Achievable (ALARA). The term "as low as reasonably achievable" when applied to radiation exposures and releases of radioactive materials in effluents means as low as reasonably achievable taking into account the state of technology, and the economics of improvements in relation to the benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of nuclear energy in the public interest.

1.2 Licensed Operators. A MNRC reactor operator is an individual licensed by the Nuclear Regulatory Commission (e.g., senior reactor operator or reactor operator) to carry out the duties and responsibilities associated with the position requiring the license.

1.2.1 Senior Reactor Operator. An individual who is licensed to direct the activities of reactor operators and to manipulate the controls of the facility.

1.2.2 Reactor Operator. An individual who is licensed to manipulate the controls of the facility and perform reactor- related maintenance.

1.3 Channel. A channel is the combination of sensor, line, amplifier, processor, and output devices which are connected for the purpose of measuring the value of a parameter.

1.3.1 Channel Test. A channel test is the introduction of a signal into the channel for verification that it is operable.

1.3.2 Channel Calibration. A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include a channel test.

1.3.3 Channel Check. A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.4 Confinement. Confinement means isolation of the reactor room air volume such that the movement of air into and out of the reactor room is through a controlled path.

1.5 Experiment. Any operation, hardware, or target (excluding devices such as detectors, fission chambers, foils, etc), which is designed to investigate specific reactor characteristics or which is intended for irradiation within the reactor tank, or in a beamport or experiment facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.5.1 Experiment, Moveable. A moveable experiment is one where it is intended that the entire experiment may be moved in or near the reactor core or into and out of the reactor while the reactor is operating.

1.5.2 Experiment, Secured. A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible conditions.

1.5.3 Experiment Facilities. Experiment facilities shall mean the pneumatic transfer tube, central thimble, beamtubes, irradiation facilities in the core or in the reactor tank, and radiography bays.

1.5.4 Experiment Safety System. Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.6 Fuel Element, Standard. A fuel element is a single TRIGA element. The fuel is U-ZrH clad in stainless steel. The zirconium to hydrogen ratio is nominally 1.65 +/- 0.05. The weight percent (wt %) of uranium can be either 8.5, 20 or 30 wt %, with an enrichment of less than 20% U-235. A standard fuel element may contain a burnable poison.

1.7 Fuel Element, Instrumented. An instrumented fuel element is a standard fuel element fabricated with thermocouples for temperature measurements. An instrumented element shall have at least one operable thermocouple embedded in the fuel near the axial and radial midpoints.

1.8 Measured Value. The measured value is the value of a parameter as it appears on the output of a channel.

1.9 Mode, Steady-State. Steady-state mode operation shall mean operation of the MNRC reactor with the selector switch in the automatic or manual mode position.

1.10 Mode, Square-Wave. Square-wave mode operation shall mean operation of the MNRC reactor with the selector switch in the square-wave mode position.

1.11 Mode, Pulse. Pulse mode operation shall mean operation of the MNRC reactor with the selector switch in the pulse mode position.

1.12 Operable. Operable means a component or system is capable of performing its intended function.

1.13 Operating. Operating means a component or system is performing its intended function.

1.14 Operating Cycle. The period of time starting with reactor startup and ending with reactor shutdown.

1.15 Protective Action. Protective action is the initiation of a signal or the operation of equipment within the MNRC reactor safety system in response to a variable or condition of the MNRC reactor facility having reached a specified limit.

1.15.1 Channel Level. At the protective instrument channel level, protective action is the generation and transmission of a scram signal indicating that a reactor variable has reached the specified limit.

1.15.2 Subsystem Level. At the protective instrument subsystem level, protective action is the generation and transmission of a scram signal indicating that a specified limit has been reached.

NOTE: Protective action at this level would lead to the operation of the safety shutdown equipment.

1.15.3 Instrument System Level. At the protective instrument system level, protective action is the generation and transmission of the command signal for the safety shutdown equipment to operate.

1.15.4 Safety System Level. At the reactor safety system level, protective action is the operation of sufficient equipment to immediately shut down the reactor.

1.16 Pulse Operational Core. A pulse operational core is a reactor state operational core for which the maximum allowable pulse reactivity insertion has been determined.

1.17 Reactivity, Excess. Excess reactivity is that amount of reactivity that would exist if all control rods (control, regulating, etc.) were moved to the maximum reactive condition from the point where the reactor is at ambient temperature and the reactor is critical. (k_{eff}=1)

1.18 Reactivity Limits. The reactivity limits are those limits imposed on the reactivity conditions of the reactor core.

1.19 Reactivity Worth of an Experiment. The reactivity worth of an experiment is the maximum value of the reactivity change that could occur as a result of changes that alter experiment position or configuration.

1.20 Reactor Controls. Reactor controls are apparatus and/or mechanisms the manipulation of which directly affect the reactivity or power level of the reactor.

1.21 Reactor Core, Operational. The MNRC reactor operational core is a core for which the parameters of excess reactivity, shutdown margin, fuel temperature, power calibration and reactivity worths of control rods and experiments have been determined to satisfy the requirements set forth in these Technical Specifications.

1.22 Reactor Operating. The MNRC reactor is operating whenever it is not secured or not shut down.

1.23 Reactor Safety Systems. Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.24 Reactor Secured. The MNRC reactor is secured when the console key switch is in the off position and the key is removed from the lock and under the control of a licensed operator, and the conditions of 'a' or 'b' exist:

a. (1) The minimum number of control rods are fully inserted to ensure the reactor is shutdown, as required by technical specifications; and

a. (2) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives, unless the control rod drives are physically decoupled from the control rods; and

a. (3) No experiments in or near the reactor are being moved or serviced that have on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or $1.00, whichever is smaller, or

b. The reactor contains insufficient fissile materials in the reactor core, adjacent experiments or control rods to attain criticality under optimum available conditions of moderation and reflection.

1.25 Reactor Shutdown. The MNRC reactor is shutdown if it is subcritical by at least one dollar ($1.00) both in the Reference Core Condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

1.26 Reference Core Condition. The condition of the core when it is at ambient temperature (cold T<28°C), the reactivity worth of xenon is negligible (< $0.30) (i.e., cold, clean, and critical), and the central irradiation facility contains the graphite thimble plug and the aluminum thimble plug.

1.27 Research Reactor. A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research development, education, and training, or experimental purposes, and which may have provisions for the production of radioisotopes.

1.28 Rod, Control. A control rod is a device fabricated from neutron absorbing material, with or without a fuel or air follower, which is used to establish neutron flux changes and to compensate for routine reactivity losses. The follower may be a stainless steel section. A control rod shall be coupled to its drive unit to allow it to perform its control function, and its safety function when the coupling is disengaged. This safety function is commonly termed a scram.

1.28.1 Regulating Rod. A regulating rod is a control rod used to maintain an intended power level and may be varied manually or by a servo-controller. A regulating rod shall have scram capability.

1.28.2 Standard Rod. The regulating and shim rods are standard control rods.

1.28.3 Transient Rod. The transient rod is a control rod that is capable of providing rapid reactivity insertion to produce a pulse or square wave.

1.29 Safety Channel. A safety channel is a measuring channel in the reactor safety system.

1.30 Safety Limit. Safety limits are limits on important process variables, which are found to be necessary to reasonably protect the integrity of the principal barriers which guard against the uncontrolled release of radioactivity.

1.31 Scram Time. Scram time is the elapsed time between reaching a limiting safety system set point and the control rods being fully inserted.

1.32 Scram, External. The external scrams consist of those shutdown signals that do not originate from the reactor control system.

1.33 Shall, Should and May. The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; the word "may" to denote permission, neither a requirement nor a recommendation.

1.34 Shutdown Margin. Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety system starting from any permissible operating condition with the most reactive rod assumed to be in the most reactive position, and once this action has been initiated, the reactor will remain subcritical without further operator action.

1.35 Shutdown, Unscheduled. An unscheduled shutdown is any unplanned shutdown of the MNRC reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

1.36 Surveillance Activities. In general, two types of surveillance activities are specified: operability checks and tests, and calibrations. Operability checks and tests are generally specified as daily, weekly or quarterly. Calibration times are generally specified as quarterly, semi-annually, annually, or biennially.

1.37 Surveillance Intervals. Maximum intervals are established to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term. The allowable surveillance interval is the interval between a check, test, or calibration, whichever is appropriate to the item being subjected to the surveillance, and is measured from the date of the last surveillance. Allowable surveillance intervals shall not exceed the following:

1.37.1 Annual - interval not to exceed fifteen (15) months.

1.37.2 Semiannual - interval not to exceed seven and a half (7.5) months.

1.37.3 Quarterly - interval not to exceed four (4) months.

1.37.4 Monthly - interval not to exceed six (6) weeks.

1.37.5 Weekly - interval not to exceed ten (10) days.

1.38 Unreviewed Safety Questions. A proposed change, test or experiment shall be deemed to involve an unreviewed safety question:

a. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c. If the margin of safety as defined in the basis for any technical specification is reduced.

1.39 Value, Measured. The measured value is the value of a parameter as it appears on the output of a channel.

1.40 Value, True. The true value is the actual value of a parameter.

1.41 Watchdog Circuit. The watchdog circuit is a surveillance circuit provided by the Data Acquisition Computer (DAC) and the Control System Computer (CSC) to ensure proper operation of the reactor computerized control system.

1.42 Loss of Coolant Accident (LOCA). The loss of coolant accident (LOCA) assumes the complete loss of coolant to the reactor core. The LOCA assumes (SAR Chapter 13) that the reactor core becomes uncovered instantaneously after extended operation at a power level of two (2) Megawatts.

1.43 Emergency Core Cooling System (ECCS). The emergency core cooling system (ECCS) provides a domestic source of water to cool the reactor core in the event of a loss of coolant accident (LOCA). The emergency core cooling system assures that the fuel temperature safety limit will not be exceeded during a LOCA.

2.0 Safety Limits and Limiting Safety System Setting.

2.1 Safety Limits.

Applicability - This specification applies to the temperature of the reactor fuel in a standard TRIGA fuel element.

Objective - The objective is to define the maximum temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specification -

a. The maximum fuel temperature in a standard TRIGA fuel element shall not exceed 930°C during steady-state operation.

b. The maximum temperature in a standard TRIGA fuel element shall not exceed 1100°C during pulse operation.

Basis -

a. This fuel safety limit applies for conditions in which the cladding temperature is above 500°C (Safety Analysis Report (SAR), Section 4.5.4.1.3). The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as it can be measured directly. A loss in the integrity of the fuel element cladding could arise if the cladding stress exceeds the ultimate strength of the cladding material. The fuel element cladding stress is a function of the element's internal pressure while the ultimate strength of the cladding material is a function of its temperature. The cladding stress is a result of the internal pressure due to the presence of air, fission product gases and hydrogen from the disassociation of hydrogen and zirconium in the fuel moderator. Hydrogen pressure is the most significant. The magnitude of the pressure is determined by the fuel moderator temperature and the ratio of hydrogen to zirconium in the alloy. At a fuel temperature of 930°C for $ZrH_{1.7}$ fuel, the cladding stress due to the internal pressure is equal to the ultimate strength of the cladding material at the same temperature (SAR fig 4.18). This is a conservative limit since the temperature of the cladding material is always lower than the fuel temperature. (See Chapter 4 of the Safety Analysis Report.)

b. This fuel safety limit applies for conditions in which the cladding temperature is less than 500°C (SAR Chapter 4, Section 4.5.4.1.3). This analysis shows that a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C. SAR, figure 4.17 shows that the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. Therefore, if the stress produced by the hydrogen overpressure inside the fuel element is less than 59,000 psi, the fuel element will not undergo loss of cladding integrity. Referring to Figure 4.18 and considering $UZrH_{1.7}$ fuel with a peak temperature of 1000°C, the stress on the clad is found to be 24,000 psi. Therefore, a fuel temperature limit of 1100°C is a limit where no loss of cladding integrity will occur. Operational data shows that GA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad. (SAR, ref 4.21.)

2.2 Limiting Safety System Setting.

2.2.1 Fuel Temperature.

Applicability - This specification applies to the protective action for the reactor fuel element temperature.

Objective - The objective is to prevent the fuel element temperature safety limit from being reached.

Specification - The limiting safety system setting shall be 750°C (operationally this may be set more conservatively) as measured in an instrumented fuel element. One instrumented element shall be located in the analyzed peak power location of the reactor operational core.

Basis - For steady-state operation of the reactor, the limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 750°C provides a safety margin at the point of measurement of at least 137°C for standard TRIGA fuel elements in any condition of operation. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is near the center and the mid-plane of the fuel element.

For pulse operation of the reactor, the same limiting safety system setting applies. However, the temperature channel will have no effect on limiting the peak power generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to limit the energy release after the pulse if the transient rod should not reinsert and the fuel temperature continues to increase.

3.0 Limiting Conditions For Operation

3.1 Reactor Core Parameters

3.1.1 Steady-State Operation

Applicability - This specification applies to the maximum reactor power attained during steady-state operation.

Objective - The objective is to assure that the reactor safety limit (fuel temperature) is not exceeded, and to provide for a setpoint for the high flux limiting safety systems, so that automatic protective action will prevent the safety limit from being reached during steady-state operations.

Specification - The nominal reactor steady-state power shall not exceed 2.0 MW. The automatic scram setpoints for the reactor power level safety channels shall be set at 2.2 MW or less. For the purpose of testing the reactor steady-state power level scram, the power shall not exceed 2.3 MW.

Basis - Operational experience and thermal-hydraulic calculations demonstrate that MNRC TRIGA fuel elements may be safely operated at power levels up to 2.3 MW with natural convection cooling. (Ref SAR Section 4.6.2.)

3.1.2 Pulse or Square Wave Operation

Applicability - This specification applies to the peak temperature generated in the fuel as the result of a step insertion of reactivity.

Objective - The objective is to assure that the fuel temperature safety limit will not be exceeded.

Specification - (a) For the pulse mode of operation, the maximum insertion of reactivity shall be 1.23% $\Delta k/k$ ($1.75); (b) For the square wave mode of operation the maximum step insertion of reactivity shall be 0.63% $\Delta k/k$ ($0.90).

Basis - Standard TRIGA Fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 to 1.7. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures in excess of 100°C. However, after extensive steady state operation at two (2) MW the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the radial edge of the fuel. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected. If the pulse insertion is such that the temperature of the fuel exceeds about 875°C, then the pressure may be sufficient to cause expansion of microscopic holes in the fuel that grow with each pulse. The analysis in SAR, section 13.2.2.2.1, shows that the limiting pulse, for the worst case conditions, is 1.34% Δk/k ($1.92). Therefore, the 1.23% Δk/k ($1.75) limit is below the worst case reactivity insertion accident limit. The $0.90 square wave step insertion limit is also well below the worst case reactivity insertion accident limit.

3.1.3 Reactivity Limitations

Applicability - These specifications apply to the reactivity conditions of the reactor core and the reactivity worths of the control rods and apply to all modes of reactor operation.

Objective - The objective is to assure that the reactor can be placed in a shutdown condition at all times and to assure that the safety limit shall not be exceeded.

Specifications -

a. Shutdown Margin - The reactor shall not be operated unless the shutdown margin provided by the control rods is greater than 0.35% Δk/k ($0.50) with:

(1) The reactor in any core condition,

(2) The most reactive control rod assumed fully withdrawn,

(3) Absolute value of all movable experiments analyzed in their most reactive condition or $1.00 whichever is greater.

b. Excess Reactivity - The maximum available excess reactivity (reference core condition) shall not exceed 6.65% Δk/k ($9.50).

Basis -

a. This specification assures that the reactor can be placed in a shutdown condition from any operating condition and remain shutdown, even if the maximum worth control rod should stick in the fully withdrawn position (SAR Section 4.5.5).

b. This specification sets an overall reactivity limit which provides adequate excess reactivity to override the xenon buildup, to overcome the temperature change in going from zero power to 2 MW, to permit pulsing at the $1.75 level, to permit irradiation of negative worth experiments and account for fuel burn up over time. An adequate shutdown margin exists with an excess of $9.50 for the two analyzed cores: (SAR Section 4.5.5).

3.2 Reactor Control and Safety Systems

3.2.1 Control Rods

Applicability - This specification applies to the function of the control rods.

Objective - The objective is to determine that the control rods are operable.

Specifications - The reactor shall not be operated unless the control rods are operable and,

a. Control rods shall not be considered operable if damage is apparent to the rod or drive assemblies.

b. The scram time measured from the instant a signal reaches the value of a limiting safety system setting to the instant that the slowest control rod reaches its fully inserted position shall not exceed one (1) second.

Basis -

a. The apparent condition of the control rod assemblies shall provide assurance that the rods shall continue to perform reliably as designed.

b. This assures that the reactor shall shutdown promptly when a scram signal is initiated (see SAR Chapter 13).

3.2.2 Reactor Instrumentation

Applicability - This specification applies to the information which shall be available to the reactor operator during reactor operations.

Objective - The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification(s) - The reactor shall not be operated unless the channels described in Table 3.2.2 are operable and the information is displayed on the reactor console.

Table 3.2.2
Required Reactor Instrumentation
(Minimum Number Operable)

Measuring Channel	Steady State	Pulse	Square Wave	Channel Function	Surveillance Required*
a. Reactor Power Level Safety Channel	2	0	2	Scram at 2.2 MW or less	D1,M,A1
b. Linear Power Channel	1	0	1	Automatic Power Control	D1,M,A1
c. Log Power Channel	1	0	1	Startup Control	D1,M,A1
d. Fuel Temperature Channel	2	2	2	Fuel Temperature	D1,M,A1

e. Pulse Channel	0	1	0	Measures Pulse NV & NVT	M,A1

(*) Where: D1 - Channel check during each day's operation
M1 - Channel test monthly
A1 - Channel calibration annually

Basis -

Specification (a) Table 3.2.2. The two reactor power level safety channels assure that the reactor power level is properly monitored and indicated in the reactor control room (SAR Sections 7.1.2 & 7.1.2.2).

Specification (b, c, & e) Table 3.2.2. The linear channel, log channel, and pulse channel assures that the reactor power level and energy are adequately monitored (SAR Sections 7.1.2 & 7.1.2.2).

Specification (d) Table 3.2.2. The fuel temperature channels assure that the fuel temperature is properly monitored and indicated in the reactor control room (SAR Section 4.5.4.1).

3.2.3 Reactor Scrams and Interlocks

Applicability - This specification applies to the scrams and interlocks.

Objective - The objective is to assure that the reactor is placed in the shutdown condition promptly and that the scrams and interlocks are operable for safe operation of the reactor.

Specifications - The reactor shall not be operated unless the scrams and interlocks described in Table 3.2.3 are operable:

Table 3.2.3
Required Scrams and Interlocks

Scram	Steady State	Pulse	Square Wave	Channel Function	Surveillance Requirements*
a. Console Manual Scram	1	1	1	Manual Scram and Automatic Scram Alarm	M
b. Reactor Room Manual Scram	1	1	1	Manual Scram and Automatic Scram Alarm	M
c. Radiography Bay Manual Scrams	4	4	4	Manual Scram and Automatic Scram Alarm	M
d. Reactor Power Level Safety Scram	2	0	2	Automatic Scram Alarms & Scram at 2.2 MW or less	M

e. High Voltage Power Supplies Scram	2	1	2	Automatic Scram Alarm & Scram on Loss of High Voltage to the Reactor Power Level Safety Channels	M
f. Fuel Temperature Scram	2	2	2	Automatic Scram Alarms & Scrams on indicated fuel temperature of 750°C or less	M
g. Watchdog Circuits Scrams	2	2	2	Automatic Scram Alarms	M
h. External Scram	2	2	2	Automatic Scram and Alarm if experimental or radiography scram interlocks are activated	M
i. One Kilowatt Pulse & Square Wave Interlock	0	1	1	Prevents initiation of a a step reactivity insertion above a reactor power level of 1 KW	M
j. Low Source Level Rod Withdrawal Prohibit Interlock	1	1	1	Prevents withdrawal of any control rod if the log channel reads less than 1.5 times the indicated log channel current level with the neutron source removed from the core	M
k. Control Rod Withdrawal Interlock	1	1	1	Prevents simultaneous withdrawal of two or more rods in manual mode	M
l. Magnet Power Key Switch Scram	1	1	1	Deenergizes the control rod magnets	M

(*) Where: M - channel test monthly

Basis -

Specification (a) Table 3.2.3. The console manual scram allows rapid shutdown of the reactor from the control room (SAR Section 7.1.2.5).

Specification (b) Table 3.2.3. The reactor room manual scram allows rapid shutdown of the reactor from the reactor room.

Specification (c) Table 3.2.3. The radiography bay manual scrams allow rapid shutdown of the reactor from any of the radiography bays (SAR Section 9.6.3).

Specification (d) Table 3.2.3. The automatic power level safety scram assures the reactor will be shutdown if the power level exceeds 2.2 MW, therefore not exceeding the safety limit (SAR Section 4.7.2).

Specification (e) Table 3.2.3. The loss-of-high voltage scram assures that the reactor power level safety channels operate within their intended range as required for proper functioning of the power level scrams (SAR Sections 7.1.2.1 & 7.1.2.2).

Specification (f) Table 3.2.3. The fuel temperature scrams assure that the reactor will be shutdown if the fuel temperature exceeds 750°C, therefore ensuring the safety limit will not be exceeded (SAR Sections 4.5.4.1 & 4.7.2).

Specification (g) Table 3.2.3. The watchdog circuits assure that the control system computer and the data acquisition computer are functioning properly (SAR Section 7.2).

Specification (h) Table 3.2.3. The external scrams assure that the reactor will be shutdown if the radiography bay doors and reactor concrete shutters are not in the proper position for personnel entry into the bays (SAR Section 9.6). Experimental scrams, a subset of the external scrams, also assures the integrity of the reactor system, the experiment, the facility, and the safety of the facility personnel and the public.

Specification (i) Table 3.2.3. The interlock preventing the initiation of a step reactivity insertion at a level above one (1) kilowatt assures that the pulse magnitude will not allow the fuel element temperature to exceed the safety limit (SAR Section 7.1.2.5).

Specification (j) Table 3.2.3. The low source level rod withdrawal prohibit interlock ensures an adequate source of neutrons is present for safe startup of the reactor (SAR Section 7.1.2.5).

Specification (k) Table 3.2.3. The control-rod withdrawal interlock prevents the simultaneous withdrawal of two or more control rods thus limiting the reactivity-insertion rate from the control rods in manual mode (SAR Section 7.1.2.5).

Specification (l) Table 3.2.3. The magnet current key switch prevents the control rods from being energized without inserting the key. Turning off the magnet current key switch de-energizes the control rod magnets and results in a scram (SAR Section 7.1.2.5).

3.2.4 Reactor Fuel Elements

Applicability - This specification applies to the physical dimensions of the fuel elements as measured on the last surveillance test.

Objective - The objective is to verify the integrity of the fuel-element cladding.

Specification - The reactor shall not be used for normal operation with damaged fuel. All fuel elements shall be inspected visually for damage or deterioration as per Technical Specifications Section 4.2.4. A fuel element shall be considered damaged and must be removed from the core if:

a. In measuring the transverse bend, the bend exceeds 0.125 inch (3.175 mm) over the full length 23 inches (584 mm) of the cladding, or,

b. In measuring the elongation, its length exceeds its initial length by 0.125 inch (3.175 mm), or,

c. A cladding failure exists as indicated by measurable release of fission products.

d. Visual inspection identifies bulges, grcss pitting, or corrosion.

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. The above limits on the allowable distortion of a fuel element correspond to strains that are considerably lower than the strain expected to cause rupturing of a fuel element. Limited operation in the steady state or pulsed mode may be necessary to identify a leaking fuel element especially if the leak is small.

3.3 Reactor Coolant Systems

Applicability - These specifications apply to the operation of the reactor water measuring systems.

Objective - The objective is to assure that adequate cooling is provided to maintain fuel temperatures below the safety limit, and that the water quality remains high to prevent damage to the reactor fuel.

Specifications - The reactor shall not be operated unless the systems and instrumentation channels described in Table 3.3 are operable, and the information is displayed locally or in the control room.

Table 3.3
Required Water Systems and Instrumentation

Measuring Channel/System	Minimum Number Operable	Function: Channel/System	Surveillance Requirements*
a. Tank Core Inlet Temperature Monitor	1	For operation of the reactor at 1.5 MW or higher, alarms on high heat exchanger outlet temperature of 35°C (95°F)	D1
b. Tank Low Water Level Monitor	1	Alarms if water level drops below a depth of 23 feet	M
c. Purification** Inlet Conductivity Monitor	1	Alarms if the water conductivity is greater than 5 micromhos/cm	D1,M,S
d. Emergency Core Cooling System	1	Provides domestic supply of water to cool fuel in the event of a LOCA for a minimum of 3.7 hours at 20 gpm from an appropriate nozzle	D1,S

(*) Where: D1 - channel check during each day's operation
A - channel calibration annually
Q .nnel test quarterly
S - channel calibration semiannually
M - channel test monthly

(**) The purification inlet conductivity monitor can be out-of-service for no more than 3 hours before the reactor shall be shutdown.

Basis -

Specification (a) Table 3.3. The core inlet temperature alarm assures that large power fluctuations will not occur (SAR Section 4.7.2).

Specification (b) Table 3.3. The minimum height of 23 feet of water above the tank bottom guarantees that there is sufficient water for effective cooling of the fuel and that the radiation levels at the top of the reactor are within acceptable limits. The tank water level monitor alarms if the water level drops below 23 feet, (7.01 meter) (SAR Section 11.1.5.1).

Specification (c) Table 3.3. The water conductivity remaining below 5 micromhos/cm averaged over a week, results in a minimization of activation of water impurities and contamination of reactor fuel and minimization of reactor structure corrosion.

Specification (d) Table 3.3. This system will mitigate the LOCA event analyzed in the SAR Chapter 13, Section 13.2.

3.4 Reactor Room Exhaust System

Applicability - These specifications apply to the operation of the reactor room exhaust system.

Objective - The objectives of this specification are as follows:

a. Reduce concentrations of airborne radioactive material in the reactor room.

b. Maintain the reactor room pressure negative with respect to surrounding areas.

c. Assure continuous air flow through the reactor room in the event of a LOCA.

Specification -

a. The reactor shall not be operated unless the reactor room exhaust system is in operation and the pressure in the reactor room is negative relative to surrounding areas.

b. The reactor room exhaust system shall be operable within one half hour of the onset of a LOCA.

Basis - Operation of the reactor room exhaust system assures that:

a. Concentrations of airborne radioactive material in the reactor room and in air leaving the reactor room will be reduced due to mixing with exhaust system air (SAR Section 9.5.1).

b. There will be a timely, adequate and continuous air flow through the reactor room to keep the fuel temperature below the safety limit in the event of a LOCA.

c. Pressure in the reactor room will be negative relative to surrounding areas due to air flow patterns created by the reactor room exhaust system (SAR Section 9.5.1).

3.5 This section intentionally left blank.

3.6 This section intentionally left blank.

3.7 Reactor Radiation Monitoring Systems

3.7.1 Monitoring Systems

Applicability - This specification applies to the information which shall be available to the reactor operator during reactor operation.

Objective - The objective is to require that sufficient information regarding radiation levels and radioactive effluents is available to the reactor operator to assure safe operation of the reactor.

Specifications - The reactor shall not be operated unless the channels described in Table 3.7.1 are operable, the readings are below the alarm setpoints, and the information is displayed in the control room. The stack and reactor room CAMS shall not be shutdown at the same time during reactor operation.

Table 3.7.1
Required Radiation Monitoring Instrumentation

Measuring Equipment	Minimum Number Operable	Channel Function	Surveillance Requirements*
a. Facility Stack Monitor	1	Monitors Argon-41 and radioactive particulates and alarms	D1,W,S
b. Reactor Room Radiation Monitor	1	Monitors the radiation level in the reactor room and alarms	D1,W,S
c. Purification System Radiation Monitor	1	Monitors radiation level at the demineralizer station and alarms	D1,W,S
d. Reactor Room Continuous Air Monitor	1	Monitors air from the reactor room for particulate and gaseous radioactivity and alarms	D1,W,S

(*) Where: D1 - channel check during each days operation
S - channel calibration semiannually
W - channel test

* monitors may be placed out-of-service for up to 2 hours for calibration and maintenance. During this out-of-service time, no experiment or maintenance activities shall be conducted which could

result in alarm conditions (e.g., airborne releases or high radiation levels)

Basis -

Specification (a) Table 3.7.1. The facility stack monitor provides information to operating personnel regarding the release of radioactive material to the environment (SAR Section 11.1.1.1.4). The alarm setpoint on the facility stack monitor is set to limit Argon-41 concentrations to less than 10 CFR Part 20, Appendix B, Table 2, Column 1 values (averaged over one year) for unrestricted locations outside the operations area.

Specification (b) Table 3.7.1. The reactor room radiation monitor provides information regarding radiation levels in the reactor room during reactor operation (SAR Section 11.1.5.1).

Specification (c) Table 3.7.1. The radiation monitor located next to the purification system resin canisters provides information regarding radioactivity in the primary system cooling water (SAR Section 11.1.5.4.2).

Specification (d) Table 3.7.1. The reactor room continuous air monitor provides information regarding airborne radioactivity from the reactor room (SAR Sections 11.1.1.1.2 & 11.1.1.1.5).

3.7.2 Effluents - Argon-41 Discharge Limit

Applicability. This specification applies to the concentration of Argon-41 that may be discharged from the MNRC reactor facility.

Objective. The objective is to ensure that the health and safety of the public is not endangered by the discharge of Argon-41 from the MNRC reactor facility.

Specification. The annual average unrestricted area concentration of Argon-41 due to releases of this radionuclide from the MNRC, and the corresponding annual radiation dose from Argon-41 in the unrestricted area (at the operations boundary), shall not exceed the applicable levels in 10 CFR Part 20.

Basis. The annual average concentration limit for Argon-41 in air in the unrestricted area is specified in Appendix B, Table 2, Column 1 of 10 CFR Part 20. 10 CFR 20.1301 specifies dose limitations in the unrestricted area (at the operations boundary.) 10 CFR 20.1101 specifies a constraint on air emissions of radioactive materials to the environment. The Safety Analysis Report, Section 11.1.1.1.4 estimates that the routine Argon-41 releases and the corresponding doses in the unrestricted area will be below these limits.

3.8 Experiments

3.8.1 Reactivity Limits

Applicability. This specification applies to the reactivity limits on experiments installed in the reactor and in-tank experiment facilities.

Objective. The objective is to assure control of the reactor during the irradiation or handling of experiments adjacent to or in the reactor core.

Specification. The reactor shall not be operated unless the following conditions governing experiments exist:

a The absolute reactivity worth of any moveable experiment shall be less than one (1) dollar (0.7% $\Delta k/k$).

b. The absolute reactivity worth of any single secured experiment shall be less than the maximum allowed pulse ($1.75) (1.23% $\Delta k/k$).

c. The absolute total reactivity worth of in-tank experiments shall not exceed an absolute value of one dollar and ninety-two cents ($1.92) (1.34% $\Delta k/k$), including the potential reactivity which might result from malfunction, flooding, voiding, or removal and insertion of the experiment.

Basis-

a. A limitation of less than $1.00 on the reactivity worth of a single moveable experiment will assure that the pulse limit of $1.75 is not exceeded (SAR, Chapter 13). In addition, limiting the worth of each moveable experiment to less than $1.00 will assure that the additional increase in transient power and temperature will be slow enough so that the fuel temperature scram will be effective (SAR, Chapter 13).

b. The absolute worst event which may be considered in conjunction with a single secured experiment is its sudden accidental or unplanned removal while the reactor is operating. This would result in a reactivity increase less than a pulse of $1.92, analyzed in Chapter 13.

c. It is conservatively assumed that simultaneous removal of all experiments in the reactor at any given time shall not exceed the maximum reactivity insertion limit. Chapter 13, Section 13.2.2.2.1, of the SAR shows that an insertion of $1.92 worth of reactivity would be needed to reach the fuel temperature safety limit.

3.8.2 Materials Limit

Applicability. This specification applies to experiments installed in the reactor and its experiment facilities.

Objective. The objective is to prevent damage to the reactor or significant releases of radioactivity by limiting material quantity and radioactive material inventory of the experiment.

Specification. The reactor shall not be operated unless the following conditions governing experiment materials exist:

a. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall be appropriately encapsulated.

b. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum strontium inventory is no greater than 5 millicuries.

c. Explosive materials in quantities greater than 25 milligrams shall not be irradiated in the reactor tank. Explosive materials in quantities of 25 milligrams or less may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

d. Explosive materials in quantities of three (3) pounds TNT equivalent or less may be irradiated in any radiography bay. The irradiation of explosives in any bay is limited to those assemblies where a safety analysis has been performed that shows that there is no damage to the reactor safety systems upon detonation (SAR Section

13.2.6.2).

Basis -

a. Appropriate encapsulation is required to lessen the experimental hazards of some types of materials.

b. The 1.5 curies limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, occupational doses and doses to members of the general public in the unrestricted areas shall be within the limits in 1O CFR 20 (SAR Section 13.2.6.2).

c. This specification is intended to prevent damage to vital equipment by restricting the quantity of explosive materials within the reactor tank (SAR Section 13.2.6.2).

d. The failure of an experiment involving the irradiation of 3 lbs TNT equivalent or less in any radiography bay external to the reactor tank will not result in damage to the reactor controls or the reactor tank. Safety Analyses have been performed (SAR Section 13.2.6.2) which show that up to six (6) lbs of TNT equivalent can be safely irradiated in any radiography bay. Therefore, the three (3) lb limit gives a safety margin of two (2).

3.8.3 Failure and Malfunctions

Applicability. This specification applies to experiments installed in the reactor and its experimental facilities.

Objective. The objective is to prevent damage to the reactor or significant releases of radioactive materials in the event of an experiment failure.

Specifications

a. All experiment materials which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the reactor building or into the unrestricted area, the quantity and type of material to be irradiated shall be limited such that the airborne concentration of radioactivity shall not exceed the applicable limits of 10 CFR Part 20 (at the operations boundary), assuming 100% of the gases or aerosols escape.

b. In calculations pursuant to (a) above, the following assumptions shall be used:

(1) If the effluent from an experiment facility exhausts through a stack which is closed on high radiation levels, at least 1O% of the gaseous activity or aerosols produced will escape.

(2) If the effluent from an experiment facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron and larger particles, at least 10% of these will escape.

(3) For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undistributed column of water above the core, at least 1O% of these vapors can escape.

c. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, an evaluation shall be made to determine the need for corrective action. Inspection and any corrective action taken shall be reviewed by the Facility Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis -

a. This specification is intended to reduce the likelihood that airborne radioactivity in excess of the limits of 10 CFR Part 20 shall be released into the reactor building or to the unrestricted area (SAR Section 13.2.6.2).

b. These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure (SAR Section 13.2.6.2).

c. Normal operation of the reactor with damaged reactor fuel or structural damage is prohibited to avoid release of fission products. Potential damage to reactor fuel or structure must be brought to the attention of the Facility Director or his designated alternate for review to assure safe operation of the reactor (SAR Section 13.2.6.2).

4.0 Surveillance Requirements

General. The surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which can not be performed with the reactor operating may be deferred to the end of the operating cycle. If the reactor is not operated for a reasonable time, a reactor system or measuring channel surveillance requirement may be waived during the associated time period. Prior to reactor system or measuring channel operation, the surveillance shall be performed for each reactor system or measuring channel for which surveillance was waived. A reactor system or measuring channel shall not be considered operable until it is successfully tested.

4.1 Reactor Core Parameters

4.1.1 Steady State Operation

Applicability. This specification applies to the surveillance requirement for the power level monitoring channels.

Objective. The objective is to verify that the maximum power level of the reactor does not exceed the authorized limit.

Specification. An annual channel calibration shall be made of the power level monitoring channel. If a channel is removed, replaced, or unscheduled maintenance is performed, or a significant change in core configuration occurs, a channel calibration shall be required. Discovery of noncompliance with specification 4.1.1 shall limit reactor operations to that required to perform the surveillance.

Basis. The annual power level channel calibration will assure that the indicated reactor power level is correct.

4.1.2 Shutdown Margin and Excess Reactivity

Applicability. These specifications apply to the surveillance requirements for reactivity control of the reactor core.

Objective. The objective is to measure and verify the reactivity worth, performance, and operability of those systems affecting the reactivity of the reactor.

Specifications.

a. The total reactivity worth of each control rod and the shutdown margin shall be determined annually or following any significant change in core or control rod configuration. The shutdown margin shall be verified by meeting the requirements of Section 3.1.3(a).

b. The core excess reactivity shall be verified:

(1) Prior to each startup operation and,

(2) Following any change in core loading or configuration.

Discovery of noncompliance with Specifications 4.1.2(a-b) shall limit reactor operations to that required to perform the surveillance.

Basis -

a. The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the excess reactivity of the core. Past experience with similar reactors gives assurance that measurements of the control rod reactivity worth on an annual basis is adequate to assure no significant changes in the shutdown margin, provided no core loading or configuration changes have been made.

b. Determining the core excess reactivity prior to each reactor startup shall assure that specification 3.1.3(b) shall be met, and that the critical rod positions do not change unexpectedly.

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

Applicability. This specification applies to the surveillance of the control rods.

Objective. The objective is to inspect the physical condition of the reactor control rods and establish the operable condition of the rods.

Specification(s). Control rod worths shall be determined annually or after physical removal or any significant change in core or control rod configuration.

a. Each control rod shall be inspected at annual intervals by visual observation of the fueled sections and absorber sections plus examination of the linkages and drives.

b. The scram time of each control rod shall be measured semiannually.

Discovery of noncompliance with specification 4.2.1 (a-b) shall limit reactor operations to that required to perform the surveillance.

Basis (Specification 4.2.1.a-b). Annual determination of control rod worths or measurements after any physical removal or significant change in core loading or control rod configuration provides information about changes in reactor total reactivity and individual rod worths. The frequency of inspection for the control rods shall provide periodic verification of the condition of the control rod assemblies. The specification intervals for scram time assure operable performance of the control rods.

4.2.2 Reactor Instrumentation

Applicability. These specifications apply to the surveillance requirements for measurements, tests, calibration and acceptability of the reactor instrumentation.

Objective. The objective is to ensure that the power level and fuel temperature instrumentation are operable.

Specifications.

a. The reactor power level safety channels shall have the following:

(1) A channel test monthly or after any maintenance which could affect their operation.

(2) A channel check during each day's operation.

(3) A channel calibration annually.

b. The Linear Power Channel shall have the following:

(1) A channel test monthly or after any maintenance which could affect its operation.

(2) A channel check during each day's operation.

(3) A channel calibration annually.

c. The Log Power Channel shall have the following:

(1) A channel test monthly or after any maintenance which could affect operations.

(2) A channel check during each day's operation.

(3) A channel calibration annually.

d. The fuel temperature measuring channels shall have the following:

(1) A channel test monthly or after any maintenance which could affect operations.

(2) A channel check during each day's operation.

(3) A channel calibration annually.

e. The Pulse Energy Integrating Channel shall have the following:

(1) A channel test monthly.

(2) A channel calibration annually.

Discovery of noncompliance with specifications 4.2.2(a-e) shall limit reactor operation to that required to perform the surveillance.

Basis-

a. A daily channel check and monthly test, plus the annual calibration, will assure that the reactor power level safety channels operate properly.

b. A channel test monthly of the reactor power level multi-range channel will assure that the channel is operable and responds correctly. The channel check will assure that the reactor power level multi-range linear channel is operable on a daily basis. The channel calibration annually of the multi-range linear channel will assure that the reactor power will be accurately measured so the authorized power levels are not exceeded.

c. A channel test monthly will assure that the reactor power level wide range log channel is operable and responds correctly. A channel check of the reactor power level wide range log channel will assure that the channel is operable on a daily basis. A channel calibration will assure that the channel will indicate properly at the corresponding power levels.

d. A channel test monthly and check during each day's operation, plus the annual calibration, will assure that the fuel temperature measuring channels operate properly.

e. A channel test monthly plus the annual channel calibration will assure the pulse energy integrating channel operates properly.

4.2.3 Reactor Scrams and Interlocks

Applicability. These specifications apply to the surveillance requirements for measurements, test, calibration, and acceptability of the reactor scrams and interlocks.

Objective. The objective is to ensure that the reactor scrams and interlocks are operable.

Specification.

a. Console Manual Scram. A channel test shall be performed monthly.

b. Reactor Room Manual Scram. A channel test shall be performed monthly.

c. Radiography Bay Manual Scrams. A channel test shall be performed monthly.

d. Reactor Power Level Safety Scram. A channel test shall be performed monthly.

e. High-Voltage-Power Supply Scrams. A channel test shall be performed monthly.

f. Fuel Temperature Scram. A channel test shall be performed monthly.

g. Watchdog Circuits Scrams. A channel test shall be performed monthly.

h. External Scrams. A channel test shall be performed monthly.

i. The One Kilowatt Pulse Interlock. A channel test of the one kilowatt interlock shall be performed monthly.

j. Low Source Level Rod Withdrawal Prohibit Interlock. A channel test shall be performed monthly.

k. Control Rod Withdrawal Interlocks. A channel test shall be performed monthly.

l. Magnet Power Key Switch Scram. A channel test shall be performed monthly.

Discovery of noncompliance with Specifications 4.2.3(a-l) shall limit reactor operation to that required to perform the surveillance.

Basis -

a. A channel test monthly of the Console Manual Scram will assure that the scram is operable.

b. A channel test monthly of the Reactor Room Manual Scram will assure that the scram is operable.

c. A channel test monthly of the Radiography Bay Manual Scrams will assure that the scrams are operable.

d. A channel test monthly of the Reactor Power Level Safety Scrams will assure that the scrams are operable.

e. A channel test monthly of the Loss-of-High-Voltage Scram will assure that the high voltage power supplies are operable and respond correctly.

f. A channel test monthly of the Fuel Temperature Scrams will assure that the scrams are operable.

g. A channel test monthly of the Watchdog Circuits Scrams weekly will assure that the scram circuits are operable.

h. A channel test monthly of the External Scrams will assure that the scrams are operable and respond correctly.

i. A channel test monthly will assure that the One Kilowatt Pulse Interlock works properly.

j. A channel test monthly of the Low Source Level Rod Withdrawal Prohibit Interlock will assure that the interlock is operable.

k. A channel test monthly of the Control Rod Withdrawal Interlock will assure that the interlock is operable.

l. A channel test monthly of the Magnet Current Key Switch will assure that the scram is operable.

4.2.4 Reactor Fuel Elements

Applicability. This specification applies to the surveillance requirements for the fuel elements.

Objective. The objective is to verify the continuing integrity of the fuel-element cladding.

Specification. To assure the measurement limitations in Section 3.2.4 are met, the following shall be done:

a. The lead elements (i.e., all elements adjacent to the transient rod, with the exception of the instrumented fuel elements), and all elements adjacent to the central irradiation facility shall be inspected annually.

b. The instrumented fuel elements shall be inspected if the elements adjacent to it fail to pass the visual and/or physical measurement requirements of Section 3.2.4

Discovery of noncompliance with specification 4.2.4 shall limit operations to that required to perform the surveillance.

Basis (Specification 4.2.4 a-b). The above specification assures that the lead fuel elements shall be inspected regularly and the integrity of the lead fuel elements shall be maintained. These are the fuel elements with the highest power density as analyzed in the SAR. The instrumented fuel element is excluded to reduce the risk of damage to the thermocouples.

4.3 Reactor Coolant Systems

Applicability. This specification applies to the surveillance requirements for the reactor water measuring and the emergency core cooling systems.

Objective. The objective is to assure that the reactor tank water temperature monitoring system, the tank water level alarm, the water conductivity cells and the ECCS are operable.

Specification.

a The reactor tank core inlet temperature monitor shall have the following:

(1) A channel check during each day's operation.

(2) A channel test quarterly.

(3) A channel calibration annually.

b. The reactor tank low water level monitoring system shall have the following:

(1) A channel test monthly.

c. The purification inlet conductivity monitors shall have the following:

(1) A channel check during each day's operation.

(2) A channel test monthly.

(3) A channel calibration semiannually.

d. The Emergency Core Cooling System (ECCS) shall have the following:

(1) A channel check prior to operation.

(2) A channel calibration semiannually.

Discovery of noncompliance with specifications 4.3(a-d) shall limit operations to that required to perform the surveillance.

Basis -

a. A channel test quarterly assures the core inlet temperature monitor responds correctly to an input signal.

A channel check during each day's operation assures the channel is operable. A channel calibration annually assures the monitoring system reads properly.

b. A channel test monthly assures that the low water level monitoring system responds correctly to an input signal.

c. A channel test monthly assures that the purification inlet conductivity monitors respond correctly to an input signal. A channel check during each day's operation assures that the channel is operable. A channel calibration semiannually assures the conductivity monitoring system reads properly.

d. A channel check prior to operation assures the ECCS is operable. A channel calibration semiannually assures the ECCS performs as required.

4.4 Reactor Room Exhaust System

Applicability. This specification applies to the surveillance requirements for the reactor room exhaust system.

Objective. The objective is to assure that the reactor room exhaust system is operating properly.

Specification. The reactor room exhaust system shall have a channel check during each day's operation.

Discovery of noncompliance with this specification shall limit operations to that required to perform the surveillance.

Basis - A channel check during each day's operation of the reactor room exhaust system shall verify that the exhaust system is maintaining a negative pressure in the reactor room relative to the surrounding facility areas.

4.5 This section intentionally left blank.

4.6 This section intentionally left blank.

4.7 Reactor Radiation Monitoring Systems

Applicability. This specification applies to the surveillance requirements for the reactor radiation monitoring systems.

Objective. The objective is to assure that the radiation monitoring equipment is operating properly.

Specification.

a. The facility stack monitor shall have the following:

(1) A channel check during each day's operation.

(2) A channel test weekly.

(3) A channel calibration semiannually.

b. The reactor room radiation monitor shall have the following:

(1) A channel check during each day's operation.

(2) A channel test weekly

(3) A channel calibration semiannually.

c. The purification system radiation monitor shall have the following:

(1) A channel check during each day's operation.

(2) A channel test weekly.

(3) A channel calibration semiannually.

d. The reactor room Continuous Air Monitor (CAM) shall have the following:

(1) A channel check during each day's operation.

(2) A channel test weekly.

(3) A channel calibration semiannually.

Discovery of noncompliance with specifications 4.7(a-d) shall limit operations to that required to perform the surveillance.

Basis.

a. A channel check of the facility stack monitor system during each day's operation will assure the monitor is operable. A channel test weekly will assure that the system responds correctly to a known source. A channel calibration semiannually will assure that the monitor reads correctly.

b. A channel check of the reactor room radiation monitor during each day's operation will assure that the monitor is operable. A channel test weekly will ensure that the system responds to a known source. A channel calibration of the monitor semiannually will assure that the monitor reads correctly.

c. A channel check of the purification system radiation monitor during each day's operation assures that the monitor is operable. A channel test weekly will ensure that the system responds to a known source. A channel calibration of the monitor semiannually will assure that the monitor reads correctly.

d. A channel check of the reactor room Continuous Air Monitor (CAM) during each day's operation will assure that the CAM is operable. A channel test weekly will assure that the CAM responds correctly to a known source. A channel calibration semiannually will assure that the CAM reads correctly.

4.8 Experiments

Applicability. This specification applies to the surveillance requirements for experiments installed in the reactor or experiment facilities.

Objective. The objective is to prevent the conduct of experiments or irradiations which may damage the reactor or release excessive amounts of radioactive materials as a result of experimental failure.

Specifications.

a. A new experiment shall not be installed in any MNRC reactor irradiation facility until a safety analysis has been performed and reviewed for compliance with the Limitations on Experiments, (Technical Specifications Section 3.8) and 10 CFR 50.59 by the Facility Director or his designee.

b. All experiments performed at the MNRC shall meet the conditions of an approved Facility Use Authorization. Facility use authorizations and experiments carried out under these authorizations shall be reviewed and approved in accordance with the Utilization of Technology and Industrial Support Nuclear Facilities Document (MNRC-0027-DOC). An experiment classified as an approved experiment shall not be placed in any MNRC reactor irradiation facility until it has been reviewed for compliance with the approved experiment and facility use authorization by the Operations Supervisor and the Health Physics Supervisor, or their designated alternates.

c. The reactivity worth of any in-tank experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment. Whenever a measurement is done it shall be done at ambient conditions. Experiments shall be identified and a log or other record maintained while the experiment is in the reactor or in one of the in-tank experiment facilities.

<u>Basis</u>.

a & b. Experience at most TRIGA reactor facilities verifies the importance of reactor staff and safety committee reviews of proposed experiments.

c. Measurement of the reactivity worth of an experiment or estimation of the reactivity worth based on previous or similar measurements shall verify that the experiment is within authorized reactivity limits.

5.0 <u>Design Features</u>

5.1 <u>Site and Facility Description</u>.

5.1.1 <u>Site</u>

<u>Applicability</u>. This specification applies to the McClellan Nuclear Radiation Center site location and specific facility design features.

<u>Objective</u>. The objective is to specify those features related to the Safety Analysis evaluation.

<u>Specification</u>.

a. The site location is si approximately 8 miles (13 km) north-by-northeast of downtown Sacramento, California. The site of the MNRC facility is about 3000 ft (0.6 mi or 0.9 km) west of Watt Avenue, and 4500 ft (0.9 mi or 1.4 km) south of E Street.

b. The restricted area is that area inside the fence surrounding the reactor building. The unrestricted area is that area outside the fence surrounding the reactor building.

c. The TRIGA reactor is located in Building 258, Room 201 of the McClellan Nuclear Radiation Center (MNRC). This building has been designed with special safety features.

d. The core is below ground level in a water filled tank and surrounded by a concrete shield.

e. Restricted access areas include the reactor control room, reactor room, equipment room, and the radiography bays during normal duty hours.

Basis.

a. Information on the surrounding population, the hydrology, seismology, and climatography of the site have been presented in Chapter 2 of the Safety Analysis Report.

b. The room enclosing the reactor has been designed with systems related to the safe operation of the facility.

c. The below grade core design is to negate the consequences of an aircraft hitting the reactor building. This accident was analyzed in Chapter 13 and found to be beyond a credible accident scenario.

d. The restricted access to specific facility areas assure that proper controls are established for the safety of the public and for the security of the special nuclear materials.

5.1.2 Facility Exhaust

Applicability. This specification applies to the facility which houses the reactor.

Objective. The objective is to assure that provisions are made to restrict the amount of radioactivity released into the environment, or during a LOCA, the system is to assure proper removal of heat from the reactor room.

Specifications.

a. The MNRC reactor facility shall be equipped with a system designed to filter and exhaust air from the MNRC facility. The system shall have an exhaust stack height of a minimum of 18.2m (60 feet) above ground level.

b. Manually activated shutdown controls for the exhaust system shall be located in the reactor control room.

Basis. The MNRC facility exhaust system is designed such that the reactor room and radiography bays are maintained at a negative pressure with respect to the surrounding areas. The free air volume within the MNRC facility is confined to the facility when there is a shutdown of the exhaust system. Controls for startup, filtering, and normal operation of the exhaust system are located in the reactor control room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reactor control room with a minimum of exposure to operating personnel.

5.2 Reactor Coolant System

Applicability. This specification applies to the reactor coolant system.

Objective. The objective is to assure that adequate water is available for cooling and shielding during normal reactor operation or during a LOCA.

Specification.

a. During normal reactor operation the reactor core shall be cooled by a natural convection flow of water.

b. The reactor tank water level alarm shall activate if the water level in the reactor tank drops below a depth of 23 ft.

c. During a LOCA the reactor core shall be cooled for a minimum of 3.7 hours at 20 gpm by a source of water from the ECCS.

Basis.

a. SAR, Chapter 4, Section 4.6, Table 4-19, shows that fuel temperature limit of 930°C will not be exceeded under natural convection flow conditions.

b. A reactor tank water low level alarm sounds when the water level drops significantly. This alarm annunciates in the reactor control room and at the command post during off duty hours so that appropriate corrective action can be taken to restore water for cooling and shielding.

c. SAR, Chapter 13, Section 13.2, analyzes the requirements for cooling of the reactor fuel and shows that the fuel safety limit is not exceeded under LOCA conditions during this water cooling.

5.3 Reactor Core and Fuel

5.3.1 Reactor Core

Applicability. This specification applies to the configuration of the fuel.

Objective. The objective is to assure that provisions are made to restrict the arrangement of fuel elements so as to provide assurance that excessive power densities will not be produced.

Specification. For operation at 0.5 MW or greater, the core shall be an arrangement of 100 or more fuel elements. Below 0.5 MW there is no minimum required number of fuel elements. In a mixed 20/20 and 8.5/20 fuel loading:

Mixed J Core

a. No fuel shall be loaded into Hex Rings A or B.

b. A fuel followed control rod located in an 8.5 wt% environment shall contain 8.5 wt% fuel.

20E Core

a. No fuel shall be loaded into Hex Rings A or B.

b. Fuel followed control rods may contain either 8.5 wt% or 20 wt% fuel.

c. Variations to 20E having 20 wt% fuel in Hex Ring C requires the 20 wt% fuel to be loaded into corner positions only and graphite dummy elements in the flat positions. The performance of fuel temperature measurements shall apply to variations to the as-analyzed 20E configurations.

Basis. In order to meet the power density requirements discussed in Chapter 4 of the Safety Analysis Report, no less than 100 fuel elements and the above loading restrictions will be allowed in an operational 0.5 MW or greater core. Specification 20E(3) allows for variations of the as-analyzed core with the condition that temperature limits are being maintained.

5.3.2 Reactor Fuel

Applicability. These specifications apply to the fuel elements used in the reactor core.

Objective. The objective is to assure that the fuel elements are of such design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specification. The individual unirradiated TRIGA fuel elements shall have the following characteristics

a. Uranium content: 8.5, 20 or 30 wt%, uranium, enriched to a nominal less than 20% U-235.

b. Hydrogen to zirconium atom ratio (in the ZrH_x): 1.60 to 1.70 (1.65+/- 0.05).

c. Cladding: stainless steel, nominal 0.5mm (0.020 inch) thick.

Basis -

a. The design basis of a TRIGA core loaded with TRIGA fuel demonstrates that limiting operation to 2.3 megawatts steady state or to a 36 megawatt-sec pulse assures an ample margin of safety between the maximum temperature generated in the fuel and the safety limit for fuel temperature. The fuel temperatures are not expected to exceed 630°C during any condition of normal operation.

b. Analysis shows that the stress in a TRIGA fuel element, H/Zr ratios between 1.6 and 1.7, is equal to the clad yield strength when both fuel and cladding temperature are at the safety limit 930°C. Since the fuel temperatures are not expected to exceed 630°C during any condition of normal operation, there is a margin between the fuel element clad stress and its ultimate strength.

c. Safety margins in the fuel element design and fabrication allow for normal mill tolerances of purchased materials.

5.3.3 Control Rods and Control-Rod Drives

Applicability. This specification applies to the control rods and control-rod drives used in the reactor core.

Objective. The objective is to assure the control rods and control-rod drives are of such a design as to permit their use with a high degree of reliability with respect to their physical, nuclear, and mechanical characteristics.

Specification.

a. All control rods shall have scram capability and contain stainless steel, borated graphite, B_4C powder, or boron and its compounds in solid form as a neutron poison. The shim and regulating rods shall have fuel followers sealed in stainless steel. The transient rod shall have an air filled follower and is sealed in an aluminum tube.

b. The control-rod drives shall be the standard GA rack and pinion type with an electromagnet and armature attached.

Basis.

a. The neutron poison requirements for the control rods are satisfied by using neutron absorbing borated graphite B_4C powder, or boron and its compounds. These materials shall be contained in a suitable clad material such as stainless steel or aluminum to assure mechanical stability during movement and to isolate the neutron poison from the tank-water environment. Scram capabilities are provided for rapid insertion of the control rods.

b. The standard GA TRIGA control-rod drive meets the requirements for driving the control rods at the proper speeds, and the electromagnet and armature provide the requirements for rapid insertion capability. These drives have been tested and proven in many TRIGA reactors.

5.4 Fissionable Material Storage

Applicability. This specification applies to the storage of reactor fuel at a time when it is not in the reactor core.

Objective. The objective is to assure that the fuel which is being stored will not becor ... critical and will not reach an unsafe temperature.

Specification.

a. All fuel elements not in the reactor core shall be stored (wet or dry) in a geometrical array where the k_{eff} is less than 0.9 for all conditions of moderation.

b. Irradiated fuel elements shall be stored in an array which shall permit sufficient natural convection cooling by water or air such that the fuel-element temperature shall not exceed the safety limit.

Basis. a&b. The limits imposed by Technical Specifications 5.4.a and 5.4.b assure safe storage.

6.0 Administrative Controls

6.1 Organization. The MNRC facility shall be under the direct control of the MNRC Director or a licensed senior reactor operator (SRO) designated by him to be in direct control. The MNRC Director shall be accountable to the Responsible Commander for the safe operation and maintenance of the reactor and its associated equipment.

6.1.1 Structure. The management for operation of the MNRC facility shall consist of the organizational structure as shown in Figures 6.1 through 6.3.

6.1.2 Responsibilities. The MNRC Director shall be accountable to the Responsible Commander for the safe operation and maintenance of the reactor and its associated equipment. The MNRC Director shall review and approve all experiments and experimental procedures prior to their use in the reactor. Individuals in the management organization (i.e., reactor operations supervisor, health physics supervisor, etc.) shall be responsible for implementing the policies and operation of the facility, and shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to the operating license and technical specifications.

6.1.3 Staffing.

6.1.3.1 The minimum staffing when the reactor is not shutdown shall be:

a. A reactor operator in the control room.

b. A second person in the facility area that can perform prescribed instructions.

c. A senior reactor operator shall be present whenever a reactor startup is performed, fuel is being moved, or experiments are being placed in the reactor tank.

d. A senior reactor operator readily available. The available senior reactor operator should be within thirty (30) minutes of the facility and reachable by telephone.

6.1.3.2 A list of reactor facility personnel by name and telephone number shall be available to the reactor operator in the control room. The list shall include:

a. Management personnel.

b. Health Physics personnel.

c. Reactor Operations personnel.

6.1.4 Selection and Training of Personnel. The selection, training and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors (ANS 15.4). Qualification and requalification of licensed operators shall be subject to an approved Nuclear Regulatory Commission (NRC) program.

6.2 Review and Inspection

General Policy. Nuclear facilities shall be designed, constructed, operated, and maintained in such a manner that facility personnel, the general public, and both government and nongovernment property are not exposed to undue risk. These activities shall be conducted in accordance with applicable governmental regulatory requirements.

The Responsible Commander shall institute this policy as the facility license holder. The Nuclear Safety Committee (NSC) has been chartered to assist in meeting this responsibility by providing objective and independent reviews, evaluations, advice and recommendations on matters affecting nuclear safety. The following describes the composition and conduct of the NSC.

6.2.1 NSC Composition and Qualifications. The Responsible Commander shall appoint the chairman of the NSC. The chairman shall appoint a Nuclear Safety Committee of at least five (5) members knowledgeable in fields which relate to nuclear safety. The NSC shall evaluate and review safety standards associated with the operation and use of the nuclear facilities.

6.2.2 NSC Charter and Rules. The NSC shall conduct its review and audit functions in accordance with a written charter. This charter shall include provisions for:

a. Meeting frequency (full committee shall meet at least semiannually).

b. Voting rules.

c. Quorums.

d. Method of submission and content of presentations to the committee.

e. Use of subcommittees.

f. Review, approval and dissemination of meeting minutes.

6.2.3 Review Function. The responsibilities of the NSC, or designated subcommittee thereof, shall include but are not limited to the following:

a. Review of approved experiments utilizing the reactor facilities.

b. Review of all proposed changes to the facility Technical Specifications or SAR.

c. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question, in accordance with 10 CFR 50.59, or require a change to the Technical Specifications. This determination may be in the form of verifying a decision already made by the Facility Director.

d. Review of the operation and operational records for both reactor operations and health physics.

e. Review of abnormal performance of facility equipment and operating anomalies.

f. Review of all reportable events.

6.2.4 Inspection Function. The NSC, or a subcommittee thereof, shall inspect reactor operations and health physics annually. The annual inspection shall include, but not be limited to the following:

a. Inspection of the reactor operating and health physics records.

b. Inspection of the reactor facility.

c. Examination of reportable events.

d. Determination of the adequacy of standard operating procedures.

e. Verification of the effectiveness of the training program.

f. Verification of conformance of operations with the operating license and Technical Specifications and applicable regulations.

6.3 Radiation Safety. The Health Physics Supervisor shall be responsible for implementation of the MNRC Radiation Safety Program. The program should use the guidelines of the American National Standard for Radiation Protection at Research Reactor Facilities (ANSI/ANS 15.11). The Health Physics Supervisor shall report to the MNRC Director.

6.4 Procedures. Written procedures shall be prepared and approved prior to initiating any of the activities listed in this section. The procedures shall be approved by the MNRC Director. A periodic review of procedures shall be performed and documented in a timely manner by the MNRC staff to assure they are current. Procedures shall be adequate to assure the safe operation of the reactor but shall not preclude the use of independent judgement and action should the situation require. Procedures shall be in effect for the following items:

6.4.1 Reactor Operations Procedures.

a. Startup, operation, and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Control rod removal or replacement.

d. Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety.

e. Testing and calibration of reactor instrumentation and controls, control rods and control rod drives.

f. Administrative controls for operations, maintenance, and conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of required plans such as emergency or security plans.

h. Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes.

6.4.2 Health Physics Procedures.

a. Testing and calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, and portable radiation monitoring instrumentation.

b. Working in laboratories and other areas where radioactive materials are used.

c. Facility radiation monitoring program including routine and special surveys, personnel monitoring, monitoring and handling of radioactive waste, and sampling and analysis of solid and liquid waste, and gaseous effluents released from the facility. The program shall include a management commitment to maintain exposures and releases as low as reasonably achievable (ALARA).

d. Monitoring radioactivity in the environment surrounding the facility.

e. Administrative guidelines for the facility health physics program to include personnel orientation and training.

f. Receipt of radioactive materials at the facility, and unrestricted release of materials and items from the facility which may contain induced radioactivity or radioactive contamination.

g. Leak testing of sealed sources containing radioactive materials.

h. Special nuclear material acco· ·tability.

i. Transportation of radioactive materials.

Changes to the above procedures shall require approval of the MNRC Director. All substantive changes shall be documented.

6.5 Experiment Review and Approval. All new classes of experiments (designated Facility Use Authorization) shall be approved by the Nuclear Safety Committee (NSC) and the MNRC Director. All specific experiments to be performed under the provision of an approved Facility Use Authorization shall be approved by the MNRC Director.

a. Approved experiments shall be carried out in accordance with established approval procedures.

b. Substantive change to previously approved experiments shall require the same review and approval as a new experiment.

c. Minor changes to an experiment that do not significantly alter the experiment may be approved by a senior reactor operator.

6.6 Required Actions

6.6.1 Actions to be taken in case of a safety limit violation.
In the event of a safety limit violation (fuel temperature), the following actions shall be taken:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The safety limit violation shall be promptly reported to the MNRC Director.

c. The safety limit violation shall be reported to the chairman of the NSC and to the Nuclear Regulatory Commission by the MNRC Director.

d. A safety limit violation report shall be prepared. The report shall describe the following:

(1) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

(2) Effect of violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

(3) Corrective action to be taken to prevent recurrence.

e. The safety limit violation report shall be reviewed by the NSC and then be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation of the reactor.

6.6.2 Actions to be taken for a reportable occurrence.

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

a. Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by the MNRC Director or his designated alternate.

b. The occurrence shall be reported to the MNRC Director or his designated alternate. The MNRC Director shall report the occurrence to the Nuclear Regulatory Commission as required by these Technical Specifications or any applicable regulations.

c. All occurrences shall be reported to the NSC at the same time the NRC is notified. All occurrence reports should be reviewed by the NSC before being sent to the NRC.

6.7 Reports. All written reports shall be sent within the prescribed interval to the Nuclear Regulatory Commission, Washington DC 20555, Attn: Document Control Desk.

6.7.1 Operating Reports. An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted within six months following the end of each calendar year. Each annual report shall include the following information:

a. A brief summary of operating experiences 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.

b. A tabulation showing the energy generated by the reactor (in megawatt hours), hours the reactor was critical, and the total cumulative energy output since initial criticality.

c. The number of emergency shutdowns and inadvertent scrams, including reasons therefor.

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required.

e. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures, and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50.

f. A summary of the nature and amount of radioactive effluents released or discharged to the environment beyond the effective control of the licensee as measured at or prior to the point of such release or discharge, including the following:

(1) Liquid Effluents (summarized on a monthly basis)

(a) Liquid radioactivity discharged during the reporting period tabulated as follows:

(i) The total estimated quantity of radioactivity released (in curies).

(ii) An estimation of the specific activity for each detectable radionuclide present if the specific activity of the released material after dilution is greater than 1×10^{-7} microcuries/ml.

(iii) A summary of the total release in curies of each radionuclide determined in (ii) above for the reporting period based on representative isotopic analysis.

(iv) An estimated average concentration of the released radioactive material at the point of release for each month in which a release occurs, in terms of microcuries/ml and the fraction of the applicable limit in 10 CFR 20.

(b) The total volume (in gallons) of effluent water (including diluent) released during each period of liquid effluent release.

(2). Airborne Effluents (summarized on a monthly basis)

(a) Airborne radioactivity discharged during the reporting period (in curies) tabulated as follows:

(i) The total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method.

(ii) The total estimated quantity (in curies) of Argon-41 released during the reporting period based on data from an appropriate monitoring system.

(iii) The estimated maximum annual average concentration of Argon-41 in the unrestricted area (in microcuries/ml), the estimated corresponding annual radiation dose at this location (in millirem), and the fraction of the applicable 10 CFR 20 limits for these values.

(iv) The total estimated quantity of radioactivity in particulate form with half-lives greater than eight days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system.

(v) The average concentration of radioactive particulates with half-lives greater than eight days released (in microcuries/ml) during the reporting period.

(3) Solid Waste (summarized on an annual basis)

(a) The total amount of solid waste packaged (in cubic feet).

(b) The total activity in solid waste (in curies).

(c) The dates of shipment and disposition (if shipped offsite).

g. An annual summary of the radiation exposure received by facility operations personnel, by facility users, and by visitors in terms of the average radiation exposure per individual and the greatest exposure per individual in each group.

h. An annual summary of the radiation levels and levels of contamination observed during routine surveys performed at the facility in terms of average and highest levels.

i. An annual summary of any environmental surveys performed outside the facility.

6.7.2 Special Reports. Special reports are used to report unplanned events as well as planned major facility and administrative changes. The following classifications shall be used to determine the appropriate reporting schedule.

a. A report within 24 hours by telephone or similar conveyance to the NRC Operations Center of:

(1) Any accidental release of radioactivity into unrestricted areas above applicable unrestricted area concentration limits, whether or not the release resulted in property damage, personal injury or exposure;

(2) Any violation of a safety limit;

(3) Operation with a limited safety system setting less conservative than specified in Section 2.0, Limiting Safety System Settings;

(4) Operation in violation of a Limiting Condition for Operation;

(5) Failure of a required reactor or experiment safety system component which could render the system incapable of performing its intended safety function unless the failure is discovered during maintenance tests or a period of reactor shutdown;

(6) Any unanticipated or uncontrolled change in reactivity greater than $1.00;

(7) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could have resulted in operation of the reactor outside the specified safety limits, and;

(8) A measurable release of fission products from a fuel element

b. A report within 14 days in writing to the NRC, Document Control Desk, Washington DC 20555.

(1) Those events reported as required by Sections 6.7.2.a.(1) through 6.7.2.a.(8).

(2) The written report (and, to the extent possible, the preliminary telephone report or report by similar conveyance) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

c. A report within 30 days in writing to the NRC, Document Control Desk, Washington DC 20555.

(1) Any significant variation of measured values 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 (SAR);

(3) Any changes in facility organization or personnel;

(4) Any observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused an existence or development of an unsafe condition with regard to reactor operations.

6.8 Records. Records may be in the form of logs, data sheets, or other suitable forms The required information may be contained in single or multiple records, or a combination thereof. Records and logs shall be prepared for the following items and retained for a period of at least five years for items a. through f., and indefinitely for items g. through k. (Note: Annual reports, if they contain all of the required information, may be used as records for items g. through k.)

a. Normal reactor operation.

b. Principal maintenance activities.

c. Those events reported as required by Sections 6.7.1 and 6.7.2.

d. Equipment and component surveillance activities required by the Technical Specifications.

e. Experiments performed with the reactor.

f. Airborne and liquid radioactive effluents released to the environments and solid radioactive waste shipped off-site.

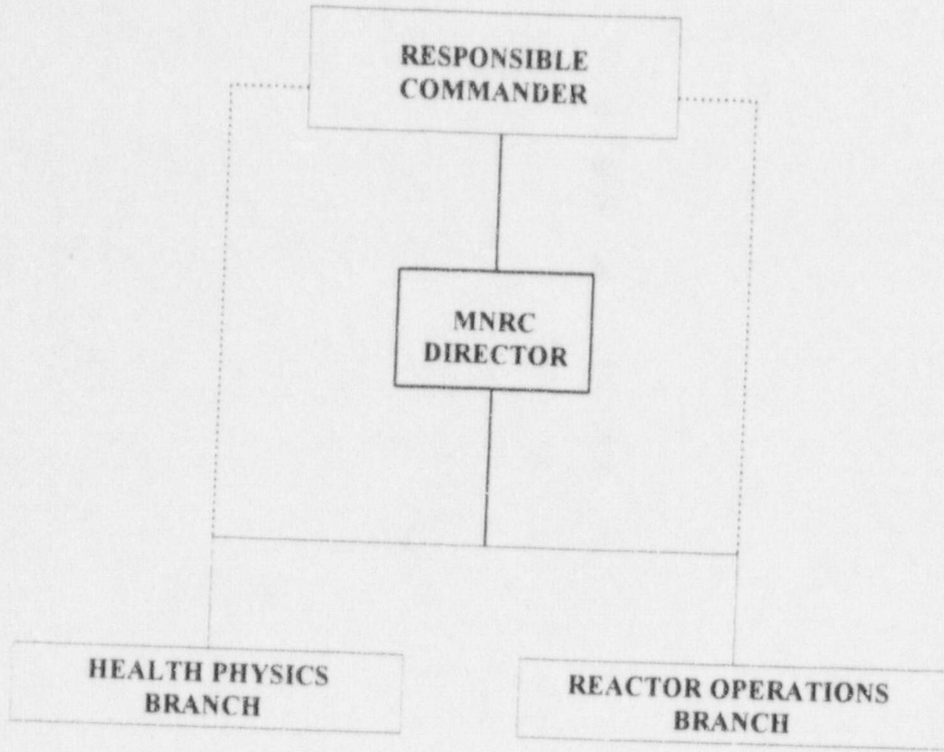
g. Offsite environmental monitoring surveys.

h. Fuel inventories and transfers.

i. Facility radiation and contamination surveys.

j. Radiation exposures for all personnel.

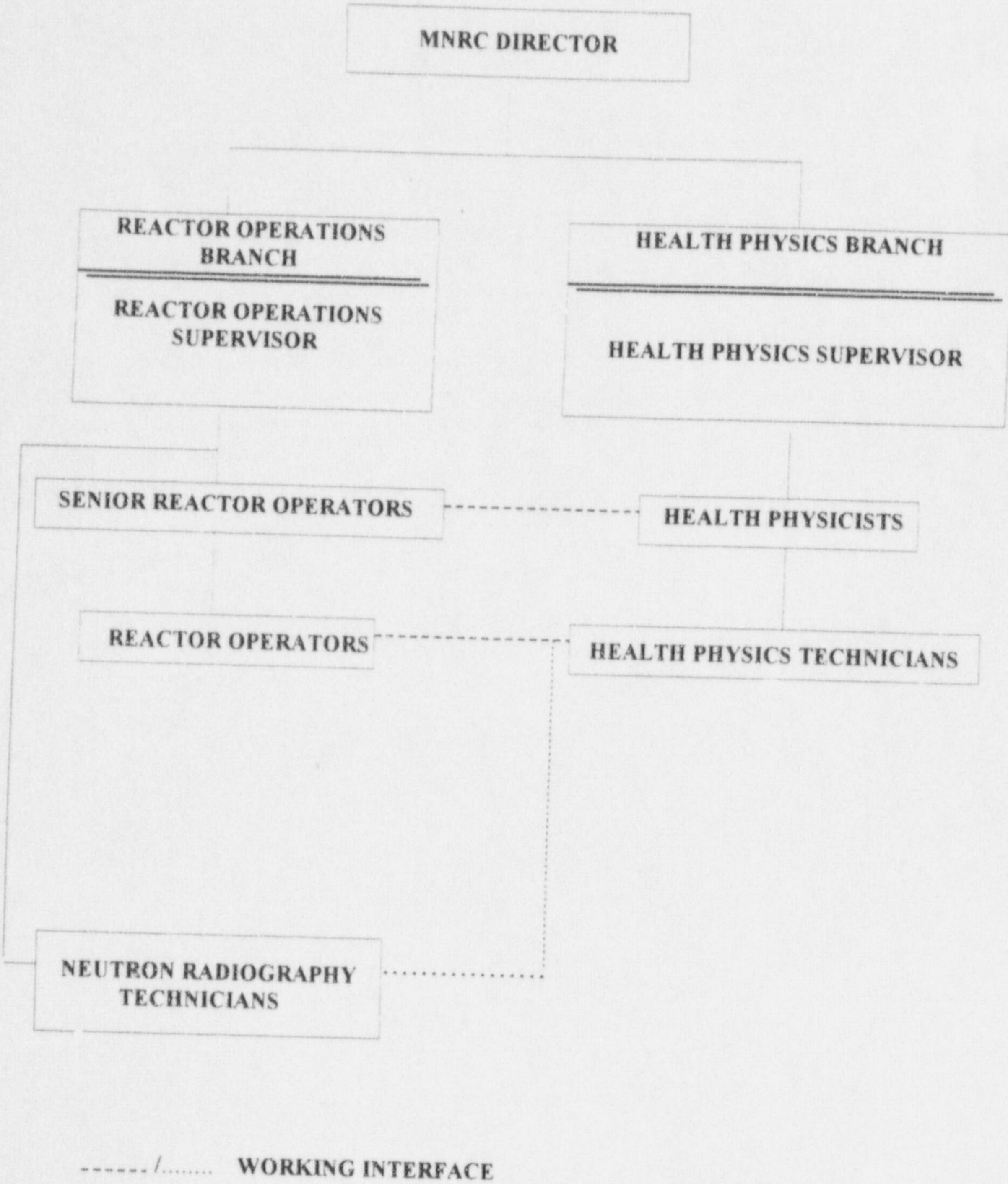
k. Updated, corrected, and as-built drawings of the facility.



Path for issues unresolved at MNRC Director Level

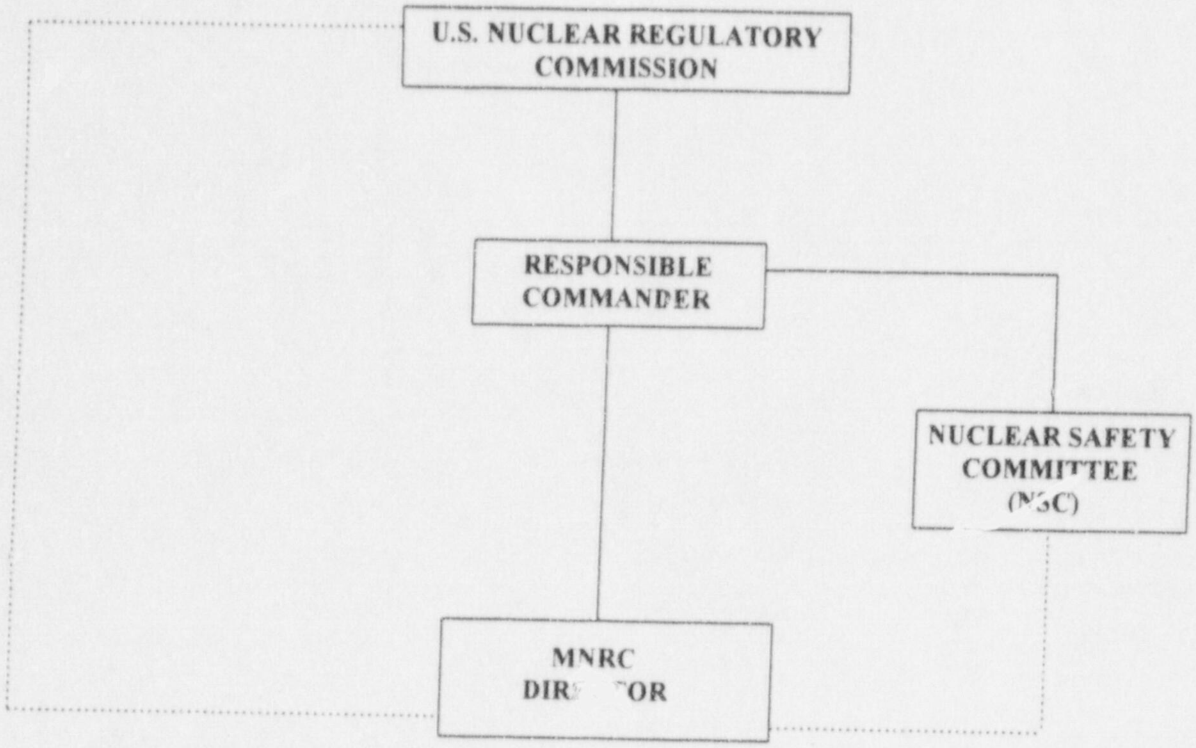
McClellan Air Force Base Nuclear Operations Organization

Figure 6.1



McClellan Nuclear Radiation Center (MNRC) Organization

Figure 6.2



INDICATES NUCLEAR SAFETY AND LICENSING ROUTE

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McClellan Air Force Base Nuclear Safety and Licensing Organization

Figure 6.3