August 9, 2001

Dr. Kevin Smith, Vice Chancellor
Office of the Chancellor
University of California, Davis
One Shields Avenue
Davis, CA 95616-8558

SUBJECT: ISSUANCE OF AMENDMENT NO. 4 TO AMENDED FACILITY OPERATING LICENSE NO. R-130 - REGENTS OF THE UNIVERSITY OF CALIFORNIA (TAC NO. 8391)

Dear Dr. Smith:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. R-130 for the McClellan Nuclear Radiation Center (MNRC) TRIGA Research Reactor. The amendment consists of changes to the Technical Specifications (TSs) in response to your submittal of May 11, 2001.

The amendment reflects the administrative changes to the TSs as a result of the transfer of the license from the Department of the Air Force to the Regents of the University of California. There are other, non-administrative changes, which are also reflected in this amendment and which are discussed in the enclosed safety evaluation report.

Sincerely,

/RA/

Warren J. Eresian, Project Manager
Operational Experience
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-607

Enclosures: 1. Amendment No. 4
2. Safety Evaluation

cc w/enclosures:
Please see next page

cc:

Dr. Wade J. Richards
5335 Price Avenue, Bldg. 258
McClellan AFB, CA 95652-2504

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

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REGENTS OF THE UNIVERSITY OF CALIFORNIA AT

McCLELLAN NUCLEAR RADIATION CENTER

DOCKET NO. 50-607

AMENDMENT TO AMENDED FACILITY OPERATING LICENSE

Amendment No. 4
License No. R-130

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

   A. The application for an amendment to Amended Facility Operating License No. R-130 filed by the Regents of the University of California at McClellan Nuclear Radiation Center (the licensee) on May 11, 2001, conforms to the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission as stated in Chapter I of Title 10 of the *Code of Federal Regulations* (10 CFR);

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

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

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

   E. This amendment is issued in accordance with the regulations of the Commission as stated in 10 CFR Part 51, and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105, and publication of notice for this amendment is not required by 10 CFR 2.106.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(ii) of Amended Facility Operating License No. R-130 is hereby amended to read as follows:

   2.C.(ii) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Warren J. Eresian, Project Manager
Operational Experience
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A, Technical
Specification Changes

Date of Issuance: August 9, 2001

ENCLOSURE TO LICENSE AMENDMENT NO. 4

AMENDED FACILITY OPERATING LICENSE NO. R-130

DOCKET NO. 50-607

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
i	i
ii	ii
iii	iii
iv	iv
v	v
vi	vi
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3	3
4	4
6	6
7	7
9	9
13	13
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TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF CALIFORNIA - DAVIS MCCLELLAN NUCLEAR RADIATION CENTER

(UCD/MNRC)

DOCUMENT NUMBER: MNRC-0004-DOC-11

Rev 11, 12/10/99

TECHNICAL SPECIFICATIONS APPROVAL

These "Technical Specifications" for the University of California at Davis/McClellan Nuclear Radiation Center (UCD/MNRC) Reactor have undergone the following coordination:

Reviewed by:______________________________
Health Physics Supervisor (Date)

Reviewed by:______________________________
Reactor Operations Supervisor (Date)

Approved by:______________________________
UCD/MNRC Director (Date)

Approved by:______________________________
Chairman, UCD/MNRC (Date)
Nuclear Safety Committee

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS
FOR THE
UNIVERSITY OF CALIFORNIA - DAVIS/MCCLELLAN NUCLEAR RADIATION CENTER (UCD/MNRC)

General

The University of California - Davis/McClellan Nuclear Radiation Center (UCD/MNRC) reactor is operated by the University of California, Davis, California (UCD). The UCD/MNRC research reactor is a TRIGA-type reactor. The UCD/MNRC provides state-of-the-art neutron radiography capabilities. In addition, the UCD/MNRC provides a wide range of irradiation services for both research and industrial needs. The reactor operates at a nominal steady state power level up to and including 2 MW. The UCD/MNRC reactor is also capable of square wave and pulse operational modes. The UCD/MNRC reactor fuel is less than 20% enriched in uranium-235.

1.0 Definitions

1.1 As Low As Reasonably Achievable (ALARA). As defined in 10 CFR, Part 20.

1.2 Licensed Operators. A UCD/MNRC licensed 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 an 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 reactor experiment facilities 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, beamtubes, irradiation facilities in the reactor 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 fuel 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 UCD/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 UCD/MNRC reactor with the selector switch in the square-wave mode position.

1.11 Mode, Pulse. Pulse mode operation shall mean operation of the UCD/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 UCD/MNRC reactor safety system in response to a variable or condition of the UCD/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 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 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 position 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 UCD/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 UCD/MNRC reactor is operating whenever it is not shutdown or secured.

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 UCD/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

(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

(3) No experiments in any reactor experiment facility, or in any other way near the reactor, are being moved or serviced if the experiments 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 UCD/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 and clean), and the central irradiation facility contains the graphite thimble plug and the aluminum thimble plug (CIF-1).

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. External scrams may arise from the radiography bay doors, radiography bay ripcords, bay shutter interlocks, and any scrams from an experiment.

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 UCD/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 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 gasses 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 SAR Chapter 4, Section 4.5.4.)

b. This fuel safety limit applies for conditions in which the cladding temperature is less than 500°C. Analysis (SAR Chapter 4, Section 4.5.4.1.1), shows that a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is estimated to be 470°C. Further analysis (SAR Section 4.5.4.1.2), shows that the internal pressure for both $Zr_{1.65}$ (at 1150°C) and $Zr_{1.7}$ (at 1100°C) increases to a peak value at about 0.3 sec, at which time the pressure is about one-fifth of the equilibrium value or about 400 psi (a stress of 14,700 psi). The yield strength of the cladding at 500°C is about 59,000 psi.

Calculations for step increases in power to peak $ZrH_{1.65}$ fuel temperature greater than 1150°C, over a 200°C range, show that the time to reach the peak pressure and the fraction of equilibrium pressure value achieved were approximately the same as for the 1150°C case. Similar results were found for fuel with ZrH1.7. Measurements of hydrogen pressure in TRIGA fuel elements during transient operations have been made and compared with the results of analysis similar to that used to make the above prediction. These measurements indicate that in a pulse where the maximum temperature in the fuel was greater than 1000°C, the pressure ($ZrH_{1.65}$) was only about 6% of the equilibrium value evaluated at the peak temperature. Calculations of the pressure gave values about three times greater than the measured values. The analysis gives strong indications that the cladding will not rupture if fuel temperatures are never greater than 1200°C to 1250°C, providing the cladding temperature is less than 500° C. For fuel with $ZrH_{1.7}$,a conservative safety limit is 1100°C. As a result, at this safety limit temperature, the class pressure is a factor of 4 lower than would be necessary for cladding failure.

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 the 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 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 operation.

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 UCD/MNRC TRIGA fuel elements may be safely operated at power levels up to 2.3 MW with natural convection cooling. (SAR Chapter 4, 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 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. Analysis (SAR Chapter 13, Section 13.2.2.2.1), shows that the limiting pulse, for the worst case conditions, is 1.34% $\Delta k/k$ ($1.92). Therefore, the 1.23% $\Delta k/k$ ($1.75) limit is below the worse case reactivity insertion accident limit. The $0.90 square wave step insertion limit is also well below the worse case reactivity insertion accident limit.

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 shut down promptly when a scram signal is initiated (SAR Chapter 13, Section 13.2.2.2.2).

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 - 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 Requirements*
a. Reactor Power Level Safety Channel	2	0	2	Scram at 2.2 MW or less	D,M,A
b. Linear Power Channel	1	0	1	Automatic Power Control	D,M,A
c. Log Power Channel	1	0	1	Startup Control	D,M,A
d. Fuel Temperature Channel	2	2	2	Fuel Temperature	D,M,A
e. Pulse Channel	0	1	0	Measures Pulse NV & NVT	P,A

(*) Where: D - Channel check during each day's operation
M - Channel test monthly
A - Channel calibration annually
P - Channel test prior to pulsing operation

Basis -

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 Chapter 7, Sections 7.1.2 & 7.1.2.2).

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.

Specification - 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. Primary Coolant Core Inlet Temperature Monitor	1	For operation of the reactor at 1.5 MW or higher, alarms on high heat exchanger outlet temperature of 45°C (113°F)	D,Q,A
b. Reactor Tank Low Water Monitor	1	Alarms if water level drops below a depth of 23 feet in the reactor tank	M
c. Purification** Inlet Conductivity Monitor	1	Alarms if the primary coolant water conductivity is greater than 5 micromhos/cm	D,M,S
d. Emergency Core Cooling System	1	For operation of the reactor at 1.5MW or higher, provides water to cool fuel in the event of a Loss of Coolant Accident for a minimum of 3.7 hours at 20 gpm from an appropriate nozzle	D,S

(*) Where:
D - channel check during each day's operation
A - channel calibration annually
Q - channel 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 -

a. Table 3.3. The primary coolant core inlet temperature alarm assures that large power fluctuations will not occur (SAR Chapter 4, Section 4.6.2).

b. Table 3.3. The minimum height of 23 ft. of water above the reactor 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 tank are within acceptable limits. The reactor tank water level monitor alarms if the water level drops below a height of 23 ft. (7.01m) above the tank bottom (SAR Chapter 11, Section 11.1.5.1).

c. Table 3.3. Maintaining the primary coolant water conductivity below 5 micromhos/cm averaged over a week will minimize the activation of water impurities and also the corrosion of the reactor structure.

d. Table 3.3. This system will mitigate the Loss of Coolant Accident 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. To reduce concentrations of airborne radioactive material in the reactor room, and maintain the reactor room pressure negative with respect to surrounding areas.

b. To assure continuous air flow through the reactor room in the event of a Loss of Coolant Accident.

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 Loss of Coolant Accident.

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 Chapter 9, Section 9.5.1). 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 Chapter 9, Section 6.5.1).

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 Loss of Coolant Accident.

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.

Specification - 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	D,W,A
b. Reactor Room Radiation Monitor	1	Monitors the radiation level in the reactor room and alarms	D,W,A
c. Purification System Radiation Monitor	1	Monitors radiation level at the demineralizer station and alarms	D,W,A
d. Reactor Room Continuous Air Monitor	1	Monitors air from the reactor room for particulate and gaseous radioactivity and alarms	D,W,A

(*) Where: D - channel check during each day's operation
A - channel calibration annually
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 -

a. Table 3.7.1. The facility stack monitor provides information to operating personnel regarding the release of radioactive material to the environment (SAR Chapter 11, 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.

b. Table 3.7.1. The reactor room radiation monitor provides information regarding radiation levels in the reactor room during reactor operation (SAR Chapter 11, Section 11.1.5.1), to limit occupational radiation exposure to less than 10 CFR 20 limits.

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 Chapter 11, Section 11.1.5.4.2) and allows assessment of radiation levels in the area to ensure that personnel radiation doses will be below 10 CFR Part 20 limits.

d. Table 3.7.1. The reactor room continuous air monitor provides information regarding airborne radioactivity in the reactor room, (SAR Chapter 11, Sections 11.1.1.1.2 & 11.1.1.1.5), to ensure that occupational exposure to airborne radioactivity will remain below the 10 CFR Part 20 limits.

3.7.2 Effluents - Argon-41 Discharge Limit

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

Specification - The annual average unrestricted area concentration of Argon-41 due to releases of this radionuclide from the UCD/MNRC, and the corresponding annual radiation dose from Argon-41 in the unrestricted area shall not exceed the applicable levels in 1O 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. 10 CFR 20.1101 specifies a constraint on air emissions of radioactive material to the environment. The SAR Chapter 11, 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 specific reactor experiment facilities.

Objective - The objective is to assure control of the reactor during the irradiation or handling of experiments in the specifically designated reactor experiment facilities.

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

a. The absolute reactivity worth of any single moveable experiment in the pneumatic transfer tube, the central irradiation facility, the central irradiation fixture 1 (CIF-1), or any other in-core or in-tank irradiation facility, shall be less than \$1.00 (0.7% Δk/k), except for the automated central irradiation facility (ACIF) (See 3.8.1.c below).

b. The absolute reactivity worth of any single secured experiment positioned in a reactor in-core or in-tank irradiation facility shall be less than the maximum allowed pulse (\$1.75) (1.23% Δk/k).

c. The absolute total reactivity worth of any single experiment or of all experiments collectively positioned in the ACIF shall be less than the maximum allowed pulse (\$1.75) (1.23% Δk/k).

d. The absolute total reactivity of all experiments positioned in the pneumatic transfer tube, and in any other reactor in-core and in-tank irradiation facilities at any given time shall be less than one dollar and ninety-two cents (\$1.92) (1.34% Δk/k), including the potential reactivity which might result from malfunction, flooding, voiding, or removal and insertion of the experiments.

Basis -

a. A limitation of less than one dollar ($1.00) (0.7% $\Delta k/k$) on the reactivity worth of a single movable experiment positioned in the pneumatic transfer tube, the central irradiation facility (SAR, Chapter 10, Section 10.4.1), the central irradiation fixture-1 (CIF-1) (SAR Chapter 10, Section 10.4.1), or any other in-core or in-tank irradiation facility, will assure that the pulse limit of $1.75 is not exceeded (SAR Chapter 13, Section 13.2.2.2.1). In addition, limiting the worth of each movable 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, Section 13.2.2.2.1).

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. For such an event, the reactivity limit for fixed experiments ($1.75) would result in a reactivity increase less than the $1.92 pulse reactivity insertion needed to reach the fuel temperature safety limit (SAR Chapter 13, Section 13.2.2.2.1).

c. A reactivity limit of less than $1.75 for any single experiment or for all experiments collectively positioned in the sample can of the automated central irradiation facility (ACIF) (SAR Chapter 10, Section 10.4.2) is based on the pulsing reactivity insertion limit (Technical Specification 3.1.2) (SAR Chapter 13, Section 13.2.2.2.1) and on the design of the ACIF, which allows control over the positioning of samples into and out of the central core region in a manner identical in form, fit, and function to a control rod.

d. It is conservatively assumed that simultaneous removal of all experiments positioned in the pneumatic transfer tube, and in any other reactor in-core and in-tank irradiation facilities at any given time shall be less than the maximum reactivity insertion limit of $1.92. The SAR Chapter 13, Section 13.2.2.2.1 indicates that a pulse reactivity insertion of $1.92 would be needed to reach the fuel temperature safety limit.

3.8.2 Materials Limit

Applicability - This specification applies to experiments installed in reactor experiment facilities.

Objective - The objective is to prevent damage to the reactor or significant releases of radioactivity by limiting material quantity and the 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. Each experiment in the I-125 production facility shall be controlled such that the total inventory of I-125 dispensed or stored in the reactor room glove box shall not exceed 20 curies.

d. Each experiment in the I-125 production facility shall be controlled such that the total inventory of I-125 being processed at any one time in the reactor room fume hood shall not exceed 200 millicuries. An additional 800 millicuries of I-125 in sealed storage containers may also be present in the reactor room fume hood.

e. Explosive materials in quantities greater than 25 milligrams of TNT equivalent shall not be irradiated in the reactor tank. Explosive materials in quantities of 25 milligrams of TNT equivalent 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.

f. Explosive materials in quantities of three (3) pounds of 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 Chapter 13, 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 10 CFR 20 (SAR Chapter 13, Section 13.2.6.2).

c&d. Limiting the total I-125 inventory to twenty (20.0) curies in the reactor room glove box and to one (1.0) curie in the reactor room fume hood assures that, if these inventories of I-125 are totally released into their respective containments, occupational doses and doses to members of the general public in the unrestricted areas shall be within the limits of 10 CFR 20 (SAR Chapter 13, Section 13.2.6.2).

e. This specification is intended to prevent damage to vital equipment by restricting the quantity of explosive materials within the reactor tank (SAR Chapter 13, Section 13.2.6.2).

f. 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 Chapter 13, Section 13.2.6.2) which show that up to six (6) pounds of TNT equivalent can be safely irradiated in any radiogaphy bay. Therefore, the three (3) pound limit gives a safety margin of two (2).

3.8.3 Failure and Malfunctions

Applicability - This specification applies to experiments installed in reactor experiment facilities.

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

Specification -

a. All experiment materials which could off-gas, sublime, volatilize, or produce aerosols under:

(1) normal operating conditions of the experiment or the 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 in the experiment shall be limited such that the airborne radioactivity in the reactor room will not result in exceeding the applicable dose limits in 10 CFR Part 20 in the unrestricted area, assuming 100% of the gases or aerosols escapes.

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 10% 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°C 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 UCD/MNRC 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 the reactor room or the unrestricted area will result in exceeding the applicable dose limits in 10 CFR Part 20.

b. These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure (SAR Chapter 13, 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 shall be brought to the attention of the UCD/MNRC Director or his designated alternate for review to assure safe operation of the reactor (SAR Chapter 13, 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.

Discovery of noncompliance with Technical Specifications 4.3.a-c shall limit operations to that required to perform the surveillance. Noncompliance with Technical Specification 4.3.d shall limit operations to less than 1.5 MW.

Basis -

a. A channel test quarterly assures the water temperature monitoring system 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 that the emergency core cooling system is operable for power levels above 1.5 MW. A channel calibration semiannually assures that the Emergency Core Cooling System performs as required for power levels above 1.5 MW.

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 annually.

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 annually.

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 annually.

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 annually.

Discovery of noncompliance with Technical 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 annually 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 annually 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 annually 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 annually will assure that the CAM reads correctly.

4.8 Experiments

Applicability - This specification applies to the surveillance requirements for experiments installed in any UCD/MNRC reactor experiment facility.

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 experiment~~al~~ failure.

Specification -

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

b. All experiments performed at the UCD/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 the (UCD) McClellan Nuclear Radiation Center Research Reactor Facility Document (MNRC-0027-DOC). An experiment classified as an approved experiment shall not be placed in any UCD/MNRC experiment facility until it has been reviewed for compliance with the approved experiment and Facility Use Authorization by the Reactor Manager and the Health Physics Manager, or their designated alternates.

c. The reactivity worth of any experiment installed in the pneumatic transfer tube, or in any other UCD/MNRC reactor in-core or in-tank irradiation facility 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.

d. Experiments shall be identified and a log or other record maintained while experiments are in any UCD/MNRC reactor experiment facility.

Basis -

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.

d. Maintaining a log of experiments while in UCD/MNRC reactor experiment facilities will facilitate maintaining surveillance over such experiments.

5.0 Design Features

5.1 Site and Facility Description.

5.1.1 Site

Applicability - This specification applies to the UCD/MNRC site location and specific facility design features.

Objective - The objective is to specify those features related to the Safety Analysis evaluation.

Specification -

a. The site location is situated approximately 8 miles (13 km) north-by-northeast of downtown Sacramento, California. The site of the UCD/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 UCD/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.

Basis -

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

b. The restricted area is controlled by the UCD/MNRC Director.

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

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

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 Loss of Coolant Accident, the system is to assure proper removal of heat from the reactor room.

Specification -

a. The UCD/MNRC reactor facility shall be equipped with a system designed to filter and exhaust air from the UCD/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 UCD/MNRC facility exhaust system is designed such that the reactor room shall be maintained at a negative pressure with respect to the surrounding areas. The free air volume within the UCD/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 Loss of Coolant Accident.

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. For operations at 1.5 MW or higher during a Loss of Coolant Accident the reactor core shall be cooled for a minimum of 3.7 hours at 20 gpm by a source of water from the Emergency Core Cooling System.

Basis -

a. The 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 a 24 hour monitored location so that appropriate corrective action can be taken to restore water for cooling and shielding.

c. The 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 Loss of Coolant Accident 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 reactor core shall be an arrangement of 96 or more fuel elements to include fuel followed control rods. Below 0.5 MW there is no minimum required number of fuel elements. In a mixed 20/20, 30/20 and 8.5/20 fuel loading (SAR Chapter 4, Section 4.5.5.6):

Mix J Core and Other Variations

(1) No fuel shall be loaded into Hex Rings A or B.

(2) A fuel followed control rod located in an 8.5 wt% environment shall contain 8.5 wt% fuel.

20E Core and Other Variations

(1) No fuel shall be loaded into Hex Rings A or B.

(2) Fuel followed control rods may contain either 8.5 wt% or 20 wt% fuel.

(3) Variations to the 20E core 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 core configurations.

30B Core and Other Variations

(1) No fuel shall be loaded into Hex Rings A or B.

(2) The only fuel types allowed are 20/20 and 30/20.

(3) 20/20 fuel may be used in any position in Hex Rings C through G.

(4) 30/20 fuel may be used in any position in Hex Rings D through G but not in Hex Ring C.

(5) An analysis of any irradiation facility installed in the central cavity of this core shall be done before it is used with this core.

Basis - In order to meet the power density requirements discussed in the SAR Chapter 4, Section 4.5.5.6, no less than 96 fuel elements including fuel followed control rods and the above loading restrictions will be allowed in an operational 0.5 MW or greater core. Specifications for the 20E core and for the 30B core allow for variations of the as-analyzed core with the condition that temperature limits are being maintained (SAR Chapter 4, Section 4.5.5.6 and Argonne National Laboratory Report ANL/ED 97-54).

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 nominally to less than 20% U-235.

b. Hydrogen to zirconium atom ratio (in the ZrH_x): 1.60 to l.70 (l.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 a neutron poison such as stainless steel, borated graphite, B_4C powder, or boron and its compounds in solid form. The shim and regulating rods shall have fuel followers sealed in stainless steel. The transient rod shall have an air filled follower and be 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 stainless steel, 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 become 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 - The limits imposed by Technical Specifications 5.4.a and 5.4.b assure safe storage.

6.0 Administrative Controls

6.1 Organization. The Vice Chancellor for Research shall be the licensee for the UCD/MNRC. The UCD/MNRC facility shall be under the direct control of the UCD/MNRC Director or a licensed senior reactor operator (SRO) designated by the UCD/MNRC Director to be in direct control. The UCD/MNRC Director shall be accountable to the Vice Chancellor of the Office of Research for the safe operation and maintenance of the reactor and its associated equipment.

6.1.1 Structure. The management for operation of the UCD/MNRC facility shall consist of the organizational structure as shown in Figure 6.1.

6.1.2 Responsibilities. The UCD/MNRC Director shall be accountable to the Vice Chancellor of the Office of Research for the safe operation and maintenance of the reactor and its associated equipment. The UCD/MNRC Director, or his designated alternate, shall review and approve all experiments and experiment~~al~~ procedures prior to their use in the reactor. Individuals in the management organization (e.g., Reactor Manager, Health Physics Manager, etc.) shall be responsible for implementing UCD/MNRC policies and for 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. The Reactor Manager and Health Physics Manager report directly to the UCD/MNRC Director.

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 who can perform prescribed instructions;

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

d. 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.

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, Audit, Recommendation and Approval

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

The UCD Vice Chancellor of the Office of Research shall institute the above stated policy as the facility license holder. The Nuclear Safety Committee (NSC) has been chartered to assist in meeting this responsibility by providing timely, objective, and independent reviews, audits, recommendations and approvals on matters affecting nuclear safety. The following describes the composition and conduct of the NSC.

6.2.1 NSC Composition and Qualifications. The UCD/MNRC Director shall appoint the Chairperson of the NSC. The NSC Chairperson shall appoint a Nuclear Safety Committee (NSC) of at least five (5) members knowledgeable in fields which relate to nuclear safety. The NSC shall evaluate and review nuclear safety associated with the operation and use of the UCD/MNRC.

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

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

b. Voting rules.

c. Quorums (For the full committee, a quorum will be at least five (5) members.

d. A committee review function and an audit/inspection function.

e. Use of subcommittees.

f. Review, approval and dissemination of meeting minutes.

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

a. Review approved experiments utilizing UCD/MNRC nuclear facilities.

b. Review and approve all proposed changes to the facility license, the Technical Specifications and the Safety Analysis Report, and any new or changed Facility Use Authorizations and proposed Class I modifications, prior to implementing (Class I) modifications, prior to taking action under the preceding documents or prior to forwarding any of these documents to the Nuclear Regulatory Commission.

c. Review and determine whether a proposed change, test, or experiment would constitute an unreviewed safety question or require a change to the license, to a Facility Use Authorization, or to the Technical Specifications. This determination may be in the form of verifying a decision already made by the UCD/MNRC Director.

d. Review reactor operations and operational maintenance, Class I modification records, and the health physics program and associated records for all UCD/MNRC nuclear facilities.

e. Review the periodic updates of the Emergency Plan and Physical Security Plan for UCD/MNRC nuclear facilities.

f. Review and update the NSC Charter every two (2) years.

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

h. Review all reportable occurrences and all written reports of such occurrences prior to forwarding the final written report to the Nuclear Regulatory Commission.

i. Review the NSC annual audit/inspection of the UCD/MNRC nuclear facilities and any other inspections of these facilities conducted by other agencies.

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

a. Inspection of the reactor operations and operational maintenance, Class I modification records, and the health physics program and associated records, including the ALARA program, for all UCD/MNRC nuclear facilities.

b. Inspection of the physical facilities at the UCD/MNRC.

c. Examination of reportable events at the UCD/MNRC.

d. Determination of the adequacy of UCD/MNRC standard operating procedures.

e. Assessment of the effectiveness of the training and retraining programs at the UCD/MNRC.

f. Determination of the conformance of operations at the UCD/MNRC with the facility's license and Technical Specifications, and applicable regulations.

g. Assessment of the results of actions taken to correct deficiencies that have occurred in nuclear safety related equipment, structures, systems, or methods of operations.

h. Inspection of the currently active Facility Use Authorizations and associated experiments.

i. Inspection of future plans for facility modifications or facility utilization.

j. Assessment of operating abnormalities.

k. Determination of the status of previous NSC recommendations.

6.3 Radiation Safety. The Health Physics Manager shall be responsible for implementation of the UCD/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 Manager shall report to the UCD/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 UCD/MNRC Director. A periodic review of procedures shall be performed and documented in a timely manner by the UCD/MNRC staff to assure that procedures are current. Procedures shall be adequate to assure the safe operation of the reactor, but shall not preclude the use of independent judgment 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 and security plans.

h. Actions to be taken to correct 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 radiation protection 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 accountability.

i. Transportation of radioactive materials.

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

6.5 Experiment Review and Approval. Experiments having similar characteristics are grouped together for review and approval under specific Facility Use Authorizations. All specific experiments to be performed under the provisions of an approved Facility Use Authorization shall be approved by the UCD/MNRC Director, or his designated alternate.

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

b. Substantive change to a previously approved experiment 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 Action to be taken in case of a safety limit violation. In the event of a safety limit violation (fuel temperature), the following action shall be taken:

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

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

c. The safety limit violation shall be reported to the chairman of the NSC and to the NRC by the UCD/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 the 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 reoccurrence.

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

6.6.2 <u>Actions to be taken for reportable occurrences</u>. In the event of reportable occurrences, 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 shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the UCD/MNRC Director or his designated alternate.

b. The occurrence shall be reported to the UCD/MNRC Director or the designated alternate. The UCD/MNRC Director shall report the occurrence to the NRC as required by these Technical Specifications or any applicable regulations.

c. Reportable occurrences should be verbally reported to the Chairman of the NSC and the NRC Operations Center within 24 hours of the occurrence. A written preliminary report shall be sent to the NRC, Attn: Document Control Desk, 1 White Flint North, 11555 Rockville Pike, Rockville MD 20852, within 14 days of the occurrence. A final written report shall be sent to the above address within 30 days of the occurrence.

d. Reportable occurrences should be reviewed by the NSC prior to forwarding any written report to the Vice Chancellor of the Office of Research or to the Nuclear Regulatory Commission.

6.7 <u>Reports</u>. All written reports shall be sent within the prescribed interval to the NRC, Attn: Document Control Desk, 1 White Flint North, 11555 Rockville Pike, Rockville MD 20852.

6.7.1 <u>Operating Reports</u>. 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 cumulative total energy output since initial criticality.

(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 reoccurrence of the event.

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

(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) A personnel change involving the positions of UCD/MNRC Director or UCD Vice Chancellor for Research; and

(4) Any observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the 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, to the extent they contain all of the required information, may be used as records for items g. through j.)

a. Normal reactor operation.

b. Principal maintenance activities.

c. Those events reported as required by Technical Specifications 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.

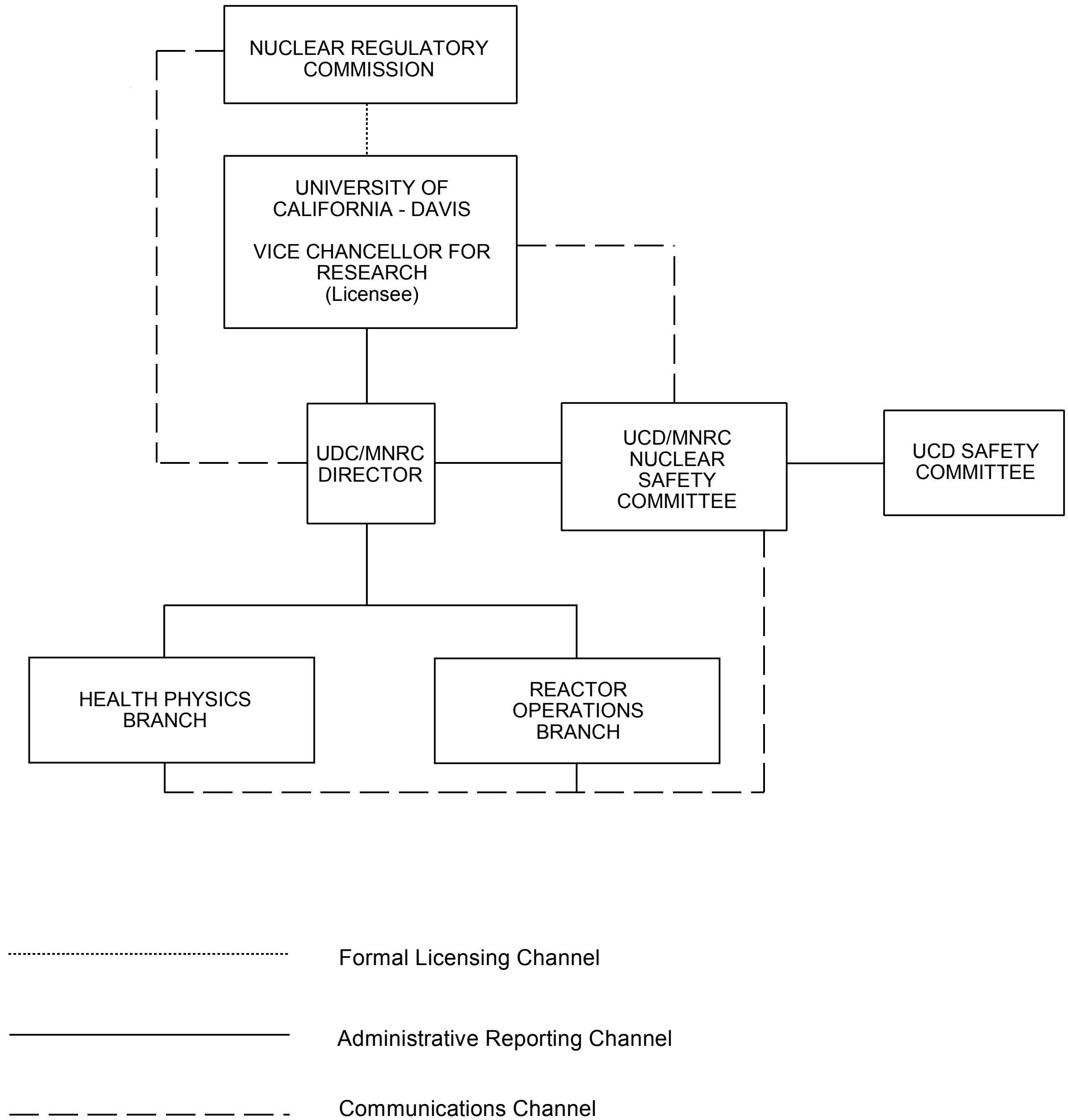
g. Offsite environmental monitoring surveys.

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.



UCD/MNRC ORGANIZATION FOR LICENSING AND OPERATION
FIGURE 6.1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO

AMENDED FACILITY OPERATING LICENSE NO. R-130

REGENTS OF THE UNIVERSITY OF CALIFORNIA AT

McCLELLAN NUCLEAR RADIATION CENTER

DOCKET NO. 50-607

1.0 INTRODUCTION

By letter dated May 11, 2001, the Regents of the University of California (the licensee) submitted a request for amendment of the Technical Specifications (TSs), Appendix A, to Facility Operating License No. R-130 for the McClellan Nuclear Radiation Center (MNRC) TRIGA research reactor. (On July 9, 2001, the licensee resubmitted the amendment request under oath. The resubmittal contained no new information.) The request provides for the following changes, which if implemented, will result in Revision 11 of the TSs:

1. On February 1, 2000, the operating license for MNRC was transferred from the Department of the Air Force to the Regents of the University of California. As a result of this transfer, a number of administrative changes simply involving name changes (e.g., changing references from “Responsible Commander” to “Vice Chancellor of the Office of Research” and “Air Force” to “University of California-Davis,” etc.) is necessary

2. Section 2.1, Basis b. This section has been expanded to include more detail regarding cladding integrity during pulsing operation.

3. Section 3.3, Table 3.3. A request to increase the alarm setpoint for the heat exchanger outlet temperature from 35 degrees Centigrade to 45 degrees Centigrade.

4. Section 4.7, Specification 4.7.a(3), 4.7.b(3) and 4.7.d(3). A request to allow channel calibrations to be performed annually rather than semiannually.

5. Section 5.3.1. A request to add the use of 30/20 TRIGA fuel and a new core fuel loading termed a 30B core.

6. Section 6.0. A request to revise the organization and duties of the Nuclear Safety Committee and to clarify the Committee’s review and audit functions to reflect the new licensee.

7. A request for approval of a new Iodine-125 production loop.

8. Section 3.8.2. Clarifies reactivity limits for experiments, and adds a new paragraph pertaining to the Iodine-125 production facility.

2.0 EVALUATION

The staff has considered each of the items 1-8 above. Each item is discussed below.

2.1 Administrative changes.

As a result of the February 1, 2000, transfer of the Operating License from the Department of the Air Force to the Regents of the University of California, the TSs must be modified to take account of administrative changes. These changes will occur in a number of places, and consist of the substitution of Department of the Air Force organizational and position titles with corresponding University of California titles. The substitutions are made on a one-for-one basis. These changes are also reflected in Figure 6.1, “UCD/MNRC Organization for Licensing and Operation.” The staff concludes that there has been no diminishment of licensee oversight (i.e., the lines of authority and responsibility have not been weakened) and that these changes are acceptable.

2.2 Section 2.1, Basis b.

The previous version of the Technical Specifications addressed the issue of the effect of pulsing on fuel clad integrity and concluded that TRIGA fuel of the type used in the McClellan reactor could be pulsed up to temperatures of 1150 degrees Centigrade without damage to the clad, provided that the clad temperature was less than 500 degrees Centigrade. The present analysis expands the discussion to include more recent measurements of hydrogen pressure resulting from pulses and concludes that the cladding will not rupture if fuel temperatures are never greater than 1200 to 1250 degrees Centigrade, providing the cladding temperature is less than 500 degrees Centigrade. Since the pulse reactivity limit remains at $1.75, the staff concludes that the bases for Section 2.1 are more conservative and this is acceptable.

2.3 Section 3.3, Table 3.3.

A re-evaluation of the thermal and hydraulic analyses and operating limits was performed by Research Reactor Safety Analysis Services (RRSAS-99-6-1, December 1999) to determine if the conservative maximum core inlet temperature (heat exchanger outlet temperature) as set by the U.S. Air Force in the original design could be raised from 35 degrees Centigrade to 45 degrees Centigrade. The effect of the lower limit is that the reactor power is required to be reduced below the license limit of 2 MW whenever ambient local weather conditions prevent the system from maintaining the heat exchanger outlet temperature at or below the lower limit.

Evaluation of data during 2 MW startup tests as well as data from subsequent steady state operations, when compared with previous calculations by Argonne National Laboratory, General Atomics published reports, and results from power upgrades at the Sandia Annular Core

Research Reactor facility shows that the maximum core inlet temperature can be raised to 45 degrees Centigrade with only a small reduction in Critical Heat Flux ratio (from 2.53 to 2.40). These numbers have been also confirmed by RELAP5 thermal hydraulic calculations. The calculations also show that there is no increase in the maximum fuel temperature or the maximum fuel clad surface temperature, two of the most important parameters which measure fuel integrity. Accordingly, the staff concludes that safety limits will not be reduced and that there is no reduction in safety margin.

2.4 Section 4.7, Specification 4.7.a(3), 4.7.b(3) and 4.7.d(3).

This section of the Technical Specifications addresses channel calibration frequencies for the stack monitor system, the reactor room radiation monitor and the reactor room continuous air monitor. These systems are presently required to be calibrated semiannually. The licensee has requested that they be calibrated annually.

The requirement for semiannual calibrations stems from the original Department of the Air Force licensing organization, but has no operational safety basis. Research reactors of similar power levels currently licensed by the NRC (National Institute of Standards and Technology, Rhode Island AEC) are permitted to calibrate similar instruments on an annual basis, since there are no operating experience data to suggest that this practice would compromise safety. In addition, the American National Standard ANSI/ANS 15.11 “Radiation Protection at Research Reactor Facilities,” states that “Instruments shall be tested at least annually in a performance quality assurance program [i.e., calibration], or more frequently if subject to extreme conditions.” The facility is not subject to extreme conditions, and the staff concludes that annual calibrations are acceptable.

2.5 Section 5.3.1.

When the McClellan reactor was originally licensed by the NRC (August 1998), the reactor was operating with a mixed core of 8.5/20 and 20/20 fuel loading (referred to as the MixJ core in the original SAR). At that time it was understood that the reactor would eventually transition to a core consisting of 20/20 and 30/20 fuel, termed a 30B core. The 30B core was analyzed in the original SAR and found to be acceptable by the NRC staff in the SER. In addition, the NRC staff had previously approved the generic use of TRIGA fuels with uranium loadings of up to 30 wt% in licensed TRIGA reactors (NUREG-1282.) The staff concludes that the introduction of 30/20 fuel is consistent with previous analyses and does not create any additional hazards.

2.6 Section 6.0.

Section 6.0 of the Technical Specifications describes the administrative controls governing the operation and maintenance of the reactor and associated equipment. There are a number of minor changes with respect to titles and some changes with respect to the composition and duties of the Nuclear Safety Committee (NSC). The review and inspection functions of the NSC have been expanded to provide additional oversight. These expanded functions include review of the Emergency Plan and Physical Security Plan, review and update of the NSC Charter every two years, review of inspections conducted by other agencies, assessment of actions taken to correct deficiencies, inspection of currently active experiments, and inspection of future plans for facility modifications or facility utilization. Since these changes increase oversight of facility operations, the staff concludes that they are acceptable.

2.7 A request for approval of a new Iodine-125 production loop.

The licensee has requested amendment of the Safety Analysis Report to provide for the installation of an Iodine-125 production loop. The purpose of the loop is to produce from ten to twenty curies of Iodine-125 for use as a medical radioisotope.

The production of Iodine-125 occurs in five steps:

1. Xenon-124 is transferred from a storage tank into an irradiation chamber located in the reactor core.

2. The Xenon-124 is irradiated over an eight to sixteen hour time span and by neutron activation results in the production of Xenon-125. The activated Xenon-124 gas contains up to 4,000 curies of Xenon-125.

3. The Xenon-125 is transferred to a tank, referred to as decay storage 1, where it decays with a 17-hour half-life to Iodine-125. After a few days, most of the Xenon-125 has decayed and the Iodine-125 plates out in the tank.

4. The Xenon-125 remaining in decay storage 1 is transferred to another tank, referred to as decay storage 2.

5. The Iodine-125 in decay storage 1 is recovered by washing the tank with a NaOH solution, resulting in a NaI solution which is packaged as a liquid and sent to an off-site user in an appropriate DOT container.

All equipment used in the production loop is located within a primary containment and a secondary containment. The primary containment houses the irradiation chamber, tubing, pneumatically operated valves, transfer vessel, decay storage 1 and decay storage 2. The secondary containment is placed around the primary containment to the irradiation chamber and allows for recovering the xenon gas if a leak occurs within the primary containment. Shielding around the secondary containment reduces radiation levels to below 10 mrem/hr. Both of these containments are within the reactor room, which has a ventilation system with isolation/recirculation capability.

There are two other structures within the reactor room which are confinement barriers designed for the safety of personnel working with the production loop. The first is a glove box which contains controls for operation of the Iodine-125 recovery system. The glove box has its own ventilation and filtration system which exhausts into the reactor room ventilation system. The second is a fume hood in which quality assurance of the Iodine-125 is performed. The fume hood also contains its own ventilation and filtration system which exhausts into the reactor room ventilation system.

The licensee has analyzed the situation (worst-case) whereby all of the Xenon-125 from the primary containment leaks into the secondary containment and subsequently leaks into the reactor room at the design leak rate of the secondary containment. Their analysis shows that exposures to personnel in the reactor room would result in a deep dose equivalent (DDE) of 17 millirem after one hour, about 1.4 millirem for a five-minute occupancy, and about 0.6 millirem for a two-minute occupancy, all well within 10 CFR 20 limits. Exposures to personnel located at the boundary of the unrestricted area for a full year would be approximately 7 millirem.

The Maximum Hypothetical Accident analyzed in the Safety Analysis Report (SAR) is a cladding rupture of one highly irradiated fuel element with no decay followed by instantaneous release of fission products into the air. At the closest distance to the site boundary (10 meters), the maximum dose to a member of the general public is 66 millirem, received over an approximately 10-minute period. The dose received at the same location due to a failure of the Iodine-125 production loop is approximately 7 millirem over a period of one year.

The staff concludes that the installation of the Iodine-125 production loop does not reduce the margin of safety with respect to 10 CFR 20 limits and that the installation of the production loop is acceptable.

2.8 Section 3.8.2.

This section of the Technical Specifications has been expanded to take account of the Iodine-125 production loop. Sections 3.8.2.c and 3.8.2.d have been added to limit the amount of Iodine-125 present in the reactor room glove box and the reactor room fume hood. Limiting the amount of Iodine-125 in these areas will reduce the occupational dose and dose to personnel in the unrestricted areas to less than 10 CFR 20 limits if the inventories of Iodine-125 are totally released within the glove box and fume hood. The staff concludes that this is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in inspection and surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, on the basis of the considerations discussed above, that (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes; and (3) such changes are in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: Warren J. Eresian

Date: August 9, 2001