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MNRC
McCLELLAN NUCLEAR RADIATION CENTER

February 29, 2000

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
1 White Flint North
11555 Rockville Pike
Rockville MD 20852

Reference: University of California, Davis, McClellan Nuclear Radiation Center (UCD/MNRC), Docket No. 50-607, Facility Operating License No. R-130

Gentlemen

As a result of the recent transfer of Facility Operating License No. R-130 for the McClellan Nuclear Radiation Center (MNRC) from the Department of the Air Force to the Regents of the University of California, we are submitting proposed new Technical specifications, an updated Final Safety Analysis Report (FSAR) and a revised Selection and Training Plan for Reactor Personnel, in order to update essential documents associated with this license transfer. In submitting these documents, we also wish to confirm that the present physical security plan and emergency response plan describe functional programs and actual response capabilities. However, because of the license transfer, these two plans are under review and will be revised in a timely manner to reflect anticipated changes in both programs.

The following descriptions provide more information regarding the documents being submitted:

A. Proposed New Technical Specifications

The proposed new Technical Specifications (Revision 11) have been revised throughout and incorporate a number of changes. The vast majority of these changes are simply administrative in nature and designed to support the recent license transfer. However, there are also changes to eliminate previous formatting and typographical errors, changes to assure consistency in terminology throughout the Technical Specifications, changes to assure correct references to related documents (e.g., the Safety Analysis Report), and minor wording changes to enhance clarity. All changes are designed by a vertical bar along the right margin.

One section that is shown as a change in Revision 11 was actually approved previously by the Commission as part of Amendment No. 1 to the Facility Operating License (dated

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December 9, 1998). However, this change to Technical Specification section 3.8.3.a, did not get included into Revision 10 of the Technical Specifications and is therefore shown as a change in Revision 11.

Another Technical Specifications change to section 3.8.1 was approved by Amendment No. 2 to the Facility Operating License (dated March 1, 1999), and involved the maximum reactivity worth of a moveable experiment. However, the new Revision 11 of the Technical Specifications also incorporates a change to section 3.8.1, which involves the rewording of subparagraphs a through c and the addition of a new subparagraph d. The Revision 11 changes *do not* introduce any new Technical Specifications, but do use the new terminology adopted for the irradiation facilities in the center region of the reactor core (FSAR Chapter 10, Section 10.4), and do clarify the reactivity limits already approved for experiments placed in specific in-core and in-tank irradiation facilities.

Other changes to Revision 11 of the Technical Specifications which should be identified are as follows:

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

2. Section 3.3, Table 3.3, Primary Coolant Core Inlet Temperature Monitor: The alarm point temperature (heat exchanger outlet temperature) has been increased from 35 degrees Centigrade to 45 degree Centigrade (FSAR Chapter 4, Table 4-21, and reference 4.40).

3. Section 4.7, Specification 4.7.a(3), 4.7.b(3) and 4.7.d(3): These specifications have all had the channel calibration frequency changed from semiannual to annual. The basis for each of these specifications addresses this change and concludes that the instrumentation will read correctly with an annual calibration.

4. Section 5.3.1: This specification is changed to add the use of 30/20 TRIGA fuel and a new core fuel loading termed a 30B core. The basis for this specification references the applicable sections in Chapter 4 of the FSAR, and FSAR Section 4.5.3 notes the limitations on this core that are subsequently included in the Technical Specifications.

5. Section 6.0: This section has numerous text changes to reflect the new licensee. In addition, there is a new Figure 6.1, Organization for Licensing and Operation, which incorporates administrative units of the new licensee. Section 6.2 has been changed slightly to reflect the organization and duties of the Nuclear Safety Committee (NSC), and to clarify the Committee's review and audit functions. Subsection 6.6.2 was changed slightly to clarify the NSC's role in the event of a reportable occurrence.

B. <u>The Updated Final Safety Analysis Report (FSAR)</u>

The FSAR also required numerous administrative changes to reflect the presence of a new licensee. In addition, there have been changes to the facility since the original FSAR was submitted to the NRC (FSAR Revision 2), and several other changes to the FSAR document have occurred based on approvals in Amendments 1 and 2 to the Facility Operating License. These latter changes (i.e., those approved in Amendments 1 and 2), were incorporated into the FSAR by the MNRC as Revision 3 prior to the recent license transfer.

Because the number of pages involved in updating the FSAR would be difficult to manage using page by page replacement, it was decided that the best approach would be to issue a completely revised and updated Revision 4 of the FSAR. Changes in Revision 4 are indicated by a vertical bar in the right margin. The change bars indicate changes to the wording shown in the previous FSAR revision, which is Revision 2 except for the very few pages changed as Revision 3. All FSAR pages have a revision number and a date, and at the beginning of each chapter there is a listing of valid pages for that chapter.

C. Revised Selections and Training Plan for Reactor Personnel

This document has been revised to reflect the presence of a new licensee and licensee organization. It also incorporates applicable requirements of 10 CFR Part 55 into the new program to be administered by the UCD/MNRC.

We hope the preceding information provides an adequate explanation of the documents we are submitting and of the changes that are involved. We will, of course, be pleased to provide any additional information you may need or to answer questions that may arise. We greatly appreciate your help with this important licensing action.

Sincerely

Wade J. Richards

WADE J. RICHARDS, Ph.D.
Director, UCD/MNRC

3 Attachments:
1. Proposed Tech Spec Rev 11
2. FSAR
3. Selection and Training Plan for Reactor Personnel

cc:
W.J. Eresian, USNRC
Region IV, USNRC
Chairperson, Nuclear Safety Committee

State of California)
) ss
County of Yolo)

Wade J. Richards, being first duly sworn on oath, deposes and says that he has affixed his signature to the letter above in his official capacity as Director, University of California, Davis, McClellan Nuclear Radiation Center; that he has signed this letter requesting Technical Specifications changes as required by 10 CFR Part 50.4b(5)(ii); that in accordance with the provisions of 10 CFR Part 50.30(b), he is attaching this affidavit; that the facts set forth in the letter within are true to his best information and belief.

Wade J. Richards

Wade J. Richards, Director

Subscribed and sworn before me, a Notary Public, in and for the County of Yolo, State of California, this 1st day of March, A.D., 2000.

Barbara Frazier

Barbara Frazier
Notary Public of California

March 12, 2002
My Commission Expires

BARBARA FRAZIER
Commission # 1176021
Notary Public - California
Yolo County
My Comm. Expires Mar 12, 2002

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF CALIFORNIA - DAVIS MCCLELLAN NUCLEAR RADIATION CENTER

(UCD/MNRC)

DOCUMENT NUMBER: MNRC-0004-DOC-11

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: ______________________________ 18 JAN 00
Health Physics Supervisor (Date)

Reviewed by: ______________________________ 01-13-00
Reactor Operations Supervisor (Date)

Approved by: ______________________________ 18 Jan 00
UCD/MNRC Director (Date)

Approved by: ______________________________ Feb 7, 00
Chairman, UCD/MNRC Nuclear Safety Committee (Date)

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 system level, protective action is the generation and transmission of the command signal for the safety shutdown equipment to operate.

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

1.16 Pulse Operational Core. A pulse operational core is a reactor 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 secured or not shut down.

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

1.24 Reactor Secured. The 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 $T<28^{o}C$), 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 being subjected to the surveillance, and is measured from the date of the last surveillance. Allowable surveillance intervals shall not exceed the following:

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

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

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

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

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

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

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

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

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

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

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

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

2.0 Safety Limit and Limiting Safety System Setting.

2.1 Safety Limit.

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

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

Specification -

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

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

Basis -

a. This fuel safety limit applies for conditions in which the cladding temperature is above 500°C (Safety Analysis Report (SAR), Chapter 4, Section 4.5.4.1.3). The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as it can be measured directly. A loss in the integrity of the fuel element cladding could arise if the cladding stress exceeds the ultimate strength of the cladding material. The fuel element cladding stress is a function of the element's internal pressure while the ultimate strength of the cladding material is a function of its temperature. The cladding stress is a result of the internal pressure due to the presence of air, fission product 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 $ZrH_{1.7}$.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 measurement of at least 137°C for standard TRIGA fuel elements in any condition of operation. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is near the center and the mid-plane of the fuel element. For pulse operation of the reactor, the same limiting safety system setting applies. However, the temperature channel will have no effect on limiting the peak power generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to limit the energy release after the pulse if the transient rod should not reinsert and the fuel temperature continues to increase.

3.0 Limiting Conditions For Operation

3.1 Reactor Core Parameters

3.1.1 Steady-State Operation

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

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

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

Basis - Operational experience and thermal-hydraulic calculations demonstrate that 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%)Δk/k ($1.75);

(b) For the square wave mode of operation the maximum step insertion of reactivity shall be 0.63%)Δk/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% Δ)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.

3.1.3 <u>Reactivity Limitations</u>

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

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

<u>Specification</u> -

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

(1) The reactor in any core condition,

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

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

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

<u>Basis</u> -

a. This specification assures that the reactor can be placed in a shutdown condition from any operating condition and remain shutdown, even if the maximum worth control rod should stick in the fully withdrawn position (SAR Chapter 4, Section 4.5.5).

b. This specification sets an overall reactivity limit which provides adequate excess reactivity to override the xenon buildup, to overcome the temperature change in going from zero power to 2 MW, to permit pulsing at the $1.75 level, to permit irradiation of negative worth experiments and account for fuel burnup over time. An adequate shutdown margin exists with an excess of $9.50 for the two analyzed cores: (SAR Chapter 4, Section 4.5.5).

3.2 Reactor Control and Safety Systems

3.2.1 Control Rods

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

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

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

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

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

Basis -

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

b. This assures that the reactor shall 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).

b, c, & e. Table 3.2.2. The linear power channel, log power channel, and pulse channel assure that the reactor power level and energy are adequately monitored (SAR Chapter 7, Sections 7.1.2 & 7.1.2.2).

d. Table 3.2.2. The fuel temperature channels assure that the fuel temperature is properly monitored and indicated in the reactor control room (SAR Chapter 4, Section 4.5.4.1).

3.2.3 Reactor Scrams and Interlocks

Applicability - This specification applies to the scrams and interlocks.

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

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

Table 3.2.3

Required Scrams and Interlocks

Scram	Steady State	Pulse	Square Wave	Channel Function	Surveillance Requirements*
a. Console Manual Scram	1	1	1	Manual Scram and Automatic Scram Alarm	M
b. Reactor Room Manual Scram	1	1	1	Manual Scram and Automatic Scram Alarm	M
c. Radiography Bay Manual Scrams	4	4	4	Manual Scrams and Automatic Scram Alarms	M
d. Reactor Power Level Safety Scrams	2	0	2	Automatic Scram Alarms & Scrams at 2.2 MW or less	M
e. High Voltage Power Supplies Scrams	2	1	2	Automatic Scram Alarms & Scrams on Loss of High Voltage to the Reactor Power Level Safety Channels	M
f. Fuel Temperature Scrams	2	2	2	Automatic Scram Alarms & Scrams on indicated fuel temperature of 750EC or less	M

g. Watchdog Circuit Scrams	2	2	2	Automatic Scram Alarms & Scrams	M
h. External Scrams	2	2	2	Automatic Scrams and Alarms if an experiment or radiography scram interlock is activated	M
i. One Kilowatt Pulse & Square Wave Interlock	0	1	1	Prevents initiation of a step reactivity insertion above a reactor power level of 1 KW	M
j. Low Source Level Rod Withdrawal Prohibit Interlock	1	1	1	Prevents withdrawal of any control rod if the log channel reads less than 1.5 times the indicated log channel current level with the neutron source removed from the core	M
k. Control Rod Withdrawal Interlock	1	1	1	Prevents simultaneous withdrawal of two or more rods in manual mode	M
l. Magnet Power Key Switch Scram	1	1	1	De-energizes the control rod magnets, scram & alarm	M

(*) Where: M - channel test monthly

<u>Basis</u> -

<u>a. Table 3.2.3</u>. The console manual scram allows rapid shutdown of the reactor from the control room (SAR Chapter 7, Section 7.1.2.5).

<u>b. Table 3.2.3</u>. The reactor room manual scram allows rapid shutdown of the reactor from the reactor room.

<u>c. Table 3.2.3</u>. The radiography bay manual scrams allow rapid shutdown of the reactor from any of the radiography bays (SAR Chapter 9, Section 9.6.3).

d. Table 3.2.3. The automatic power level safety scram assures the reactor will be shut down if the power level exceeds 2.2 MW, therefore not exceeding the safety limit (SAR Chapter 4, Section 4.7.2).

e. Table 3.2.3. The loss-of-high-voltage scram assures that the reactor power level safety channels operate within their intended range as required for proper functioning of the power level scrams (SAR Chapter 7, Sections 7.1.2.1 & 7.1.2.2).

f. Table 3.2.3. The fuel temperature scrams assure that the reactor will be shut down if the fuel temperature exceeds 750EC, therefore ensuring the safety limit will not be exceeded (SAR Chapter 4, Sections 4.5.4.1 & 4.7.2).

g. Table 3.2.3. The watchdog circuits assure that the control system computer and the data acquisition computer are functioning properly (SAR Chapter 7, Section 7.2).

h. Table 3.2.3. The external scrams assure that the reactor will be shut down if the radiography bay doors and reactor concrete shutters are not in the proper position for personnel entry into the bays (SAR Chapter 9, Section 9.6). External scrams from experiments, a subset of the external scrams, also assure the integrity of the reactor system, the experiment, the facility, and the safety of the facility personnel and the public.

i. Table 3.2.3. The interlock preventing the initiation of a step reactivity insertion at a level above one (1) kilowatt assures that the pulse magnitude will not allow the fuel element temperature to exceed the safety limit (SAR Chapter 7, Section 7.1.2.5).

j. Table 3.2.3. The low source level rod withdrawal prohibit interlock assures an adequate source of neutrons is present for safe startup of the reactor (SAR Chapter 7, Section 7.1.2.5).

k. Table 3.2.3. The control rod withdrawal interlock prevents the simultaneous withdrawal of two or more control rods thus limiting the reactivity-insertion rate from the control rods in manual mode (SAR Chapter 7, Section 7.1.2.5).

l. Table 3.2.3. The magnet current key switch prevents the control rods from being energized without inserting the key. Turning off the magnet current key switch de-energizes the control rod magnets and results in a scram (SAR Chapter 7, Section 7.1.2.5).

3.2.4 Reactor Fuel Elements

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

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

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

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

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

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

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

Basis - The most severe stresses induced in the fuel elements result from pulse operation of the reactor, during which differential expansion between the fuel and the cladding occurs and the pressure of the gases within the elements increases sharply. The above limits on the allowable distortion of a fuel element correspond to strains that are considerably lower than the strain expected to cause rupturing of a fuel element. Limited operation in the steady state or pulsed mode may be necessary to identify a leaking fuel element especially if the leak is small.

3.3 Reactor Coolant Systems

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

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

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 <u>This section intentionally left blank</u>.

3.6 <u>This section intentionally left blank</u>.

3.7 <u>Reactor Radiation Monitoring Systems</u>

3.7.1 <u>Monitoring Systems</u>

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

<u>Objective</u> - 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.

<u>Specification</u> - 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

<u>REQUIRED RADIATION MONITORING INSTRUMENTATION</u>

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 cannisters 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 10 CFR Part 20.

Basis - The annual average concentration limit for Argon-41 in air in the unrestricted area is specified in Appendix B, Table 2, Column 1 of 10 CFR Part 20. 10 CFR 20.1301 specifies dose limitations in the unrestricted area. 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% $\Delta k/k$), 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% $\Delta k/k$).

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% $\Delta k/k$).

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% $\Delta k/k$), 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. 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.

d. 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. 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).

d. The failure of an experiment involving the irradiation of 3 lbs TNT equivalent or less in any radiography bay external to the reactor tank will not result in damage to the reactor controls or the reactor tank. Safety Analyses have been performed (SAR 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 or the unrestricted area

will not result in exceeding the applicable dose limits in 10 CFR Part 20, 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 130EC and where vapors formed by boiling this material can escape only through an undistributed column of water above the core, at least 10% 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.

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

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

4.1.2 Shutdown Margin and Excess Reactivity

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

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

Specification -

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

b. The core excess reactivity shall be verified:

(1) Prior to each startup operation and,

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

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

Basis -

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

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

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

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

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

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

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

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

Discovery of noncompliance with Technical Specifications 4.2.1.a-b shall limit reactor operations to that required to perform the surveillance.

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

4.2.2 Reactor Instrumentation

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

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

Specification -

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

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

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

(3) A channel calibration annually.

b. The Linear Power Channel shall have the following:

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

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

(3) A channel calibration annually.

c. The Log Power Channel shall have the following:

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

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

(3) A channel calibration annually.

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

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

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

(3) A channel calibration annually.

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

(1) A channel test monthly.

(2) A channel calibration annually.

Discovery of noncompliance with Technical Specifications 4.2.2.a-e shall limit reactor operation to that required to perform the surveillance.

Basis -

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

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

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

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

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

4.2.3 Reactor Scrams and Interlocks

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

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

Specification -

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

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

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

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

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

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

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

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

i. The One Kilowatt Pulse Interlock. A channel test shall be performed monthly.

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

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

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

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

Basis -

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

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

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

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

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

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

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

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

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

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

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

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

4.2.4 <u>Reactor Fuel Elements</u>

<u>Applicability</u> - This specification applies to the surveillance requirements for the fuel elements.

<u>Objective</u> - The objective is to verify the continuing integrity of the fuel element cladding.

<u>Specification</u> - To assure the measurement limitations in Section 3.2.4 are met, the following shall be done:

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

b. Instrumented fuel elements shall be inspected if any of the elements adjacent to it fail to pass the visual and/or physical measurement requirements of Section 3.2.4. Discovery of noncompliance with Technical Specification 4.2.4 shall limit operations to that required to perform the surveillance.

<u>Basis (Technical Specifications 4.2.4.a-b)</u> - The above specifications assure that the lead fuel elements shall be inspected regularly and the integrity of the lead fuel elements shall be maintained. These are the fuel elements with the highest power density as analyzed in the SAR Chapter 4, Section 4.5.5.6. The instrumented fuel element is excluded to reduce the risk of damage to the thermocouples.

4.3 <u>Reactor Coolant Systems</u>

<u>Applicability</u> - This specification applies to the surveillance requirements for the reactor water measuring systems and the emergency core cooling system.

<u>Objective</u> - The objective is to assure that the reactor tank water temperature monitoring system, the tank water level alarm, the water conductivity cells and the emergency core cooling system are all operable.

<u>Specification</u> -

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

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

(2) A channel test quarterly.

(3) A channel calibration annually.

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

(1) A channel test monthly.

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

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

(2) A channel test monthly.

(3) A channel calibration semiannually.

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

(1) A channel check prior to operation.

(2) A channel calibration semiannually.

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 <u>Reactor Room Exhaust System</u>

<u>Applicability</u> - This specification applies to the surveillance requirements for the reactor room exhaust system.

<u>Objective</u> - The objective is to assure that the reactor room exhaust system is operating properly.

<u>Specification</u> - 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.

<u>Basis</u> - 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 <u>Reactor Radiation Monitoring Systems</u>

<u>Applicability</u> - This specification applies to the surveillance requirements for the reactor radiation monitoring systems.

<u>Objective</u> - The objective is to assure that the radiation monitoring equipment is operating properly.

<u>Specification</u> -

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 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 Operations Supervisor and the Health Physics Supervisor, 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/McClellan Nuclear Radiation Center. 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 4.4.5.7):

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 1.70 (1.65+/- 0.05).

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

Basis -

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

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

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

5.3.3 Control Rods and Control-Rod Drives

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

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

Specification -

a. All control rods shall have scram capability and contain 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 shall review and approve all experiments and experiment procedures prior to their use in the reactor. Individuals in the management organization (e.g., reactor operations supervisor, health physics supervisor, 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 operations supervisor and health physics supervisor 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 quorom 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 Supervisor 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 Supervisor 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.

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 Actions to be taken for reportable occurrences. 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.

c. The number of emergency shutdowns and inadvertent scrams, including reasons for the shutdowns or scrams.

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

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

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

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

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

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

2 An estimation of the specific activity for each detectable radionuclide present if the specific activity of the released material after dilution is greater than $1x10^{-7}$ microcuries/ml.

3 A summary of the total release in curies of each radionuclide determined in 2 above for the reporting period based on representative isotopic analysis.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(2) Any violation of a safety limit;

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

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

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

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

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

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

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

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

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

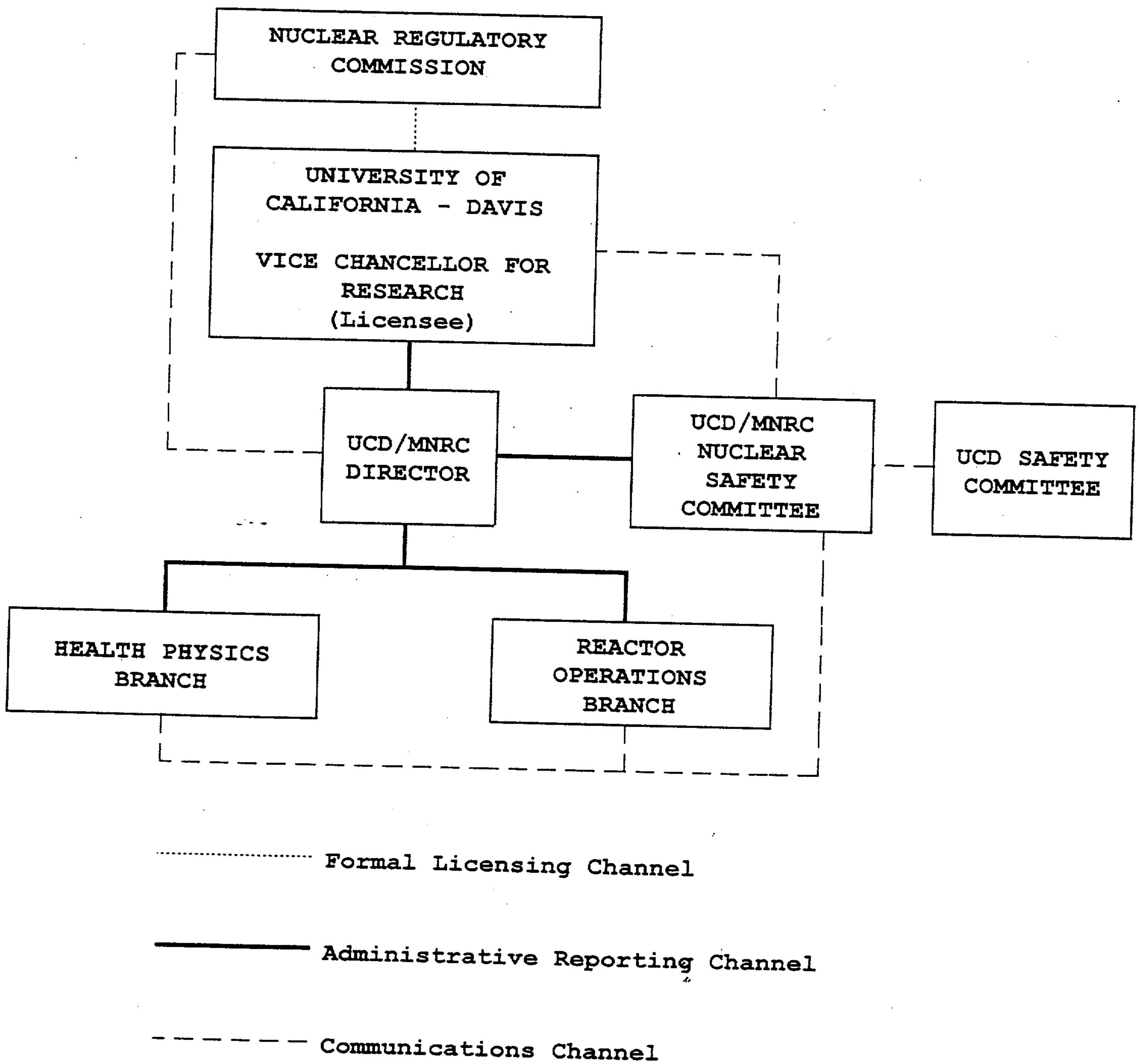
g. Offsite environmental monitoring surveys.

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