UCDAVIS
MNRC
McCLELLAN NUCLEAR RADIATION CENTER

5335 PRICE AVENUE
BUILDING 258
McCLELLAN, CA 95652
PHONE (916) 614-6200
FAX (916) 614-6250
WEB http //ovcr ucdavis edu/mnrc

October 17, 2002

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

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

Subject: Requested Amendment No. 5 to Facility Operating License No. R-130 McClellan Nuclear Radiation Center

Gentlemen

It is requested that the attached proposed Revision 12 to the UCD/MNRC Technical Specifications be approved for use. The nature of the changes are explained in Section A of this letter. Also, attached are the Safety Analysis Report (SAR) pages that fully describe the proposed changes.

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

1. The proposed new Technical Specifications (Rev. 12) have been revised to incorporate the new management position of "Site Manager." This position will report directly to the UCD Director. The Reactor Manager and Health Physics Manager will report directly to the Site Manager.

2. The most substantive change involves increasing the Iodine-125 production rate from 20 to 61 Curies. This increase in Iodine-125 production is shown (i.e., Chapter 13, Section 13.2.6, pages 45-56) to result in personnel exposures, under accident condition, lower than 10 CFR Part 20 limits.

A020

All changes (i.e., Rev 12) to the Technical Specifications are marked by vertical lines in the right hand margin.

Sincerely

Wade J. Richards

Wade J. Richards, Ph.D.
Director
UCD/MNRC

2 Attachments:
1. UCD/MNRC Tech Specs Rev. 12 (2 copies)
2. SAR Support Pages (2 copies)

cc: Dr. B. Klein, UCD Vice Chancellor for Research
Dr. Warren Eresian, USNRC

In accordance with 10 CFR 50.30(b), I declare under penalty of perjury that the foregoing is true and correct.

Executed on 10-17-02 / Wade J. Richards
Date Signature

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF CALIFORNIA - DAVIS MCCLELLAN NUCLEAR RADIATION CENTER

(UCD/MNRC)

DOCUMENT NUMBER: MNRC-0004-DOC-12

TECHNICAL SPECIFICATIONS APPROVAL

Revision 12 of the "Technical Specifications" for the University of California at Davis/McClellan Nuclear Radiation Center (UCD/MNRC) Reactor have undergone the following coordination:

Reviewed by: Health Physics Manager 10/16/02 Date

Reviewed by: Reactor Manager 10/16/02 Date

Reviewed by: Charles C. Smith Site Manager 10-17-02 Date

Approved by: Wade Richards UCD/MNRC Director 10-17-02 Date

Approved by: AG Johnson Chairman, UCD/MNRC Nuclear Safety Committee 10/14/02 Date

Technical Specifications

Rev 12 09/2002

Page		
Title Page	Rev 12	9/2002
Approval Page	Rev 12	9/2002
32	Rev 12	9/2002
Figure 6.1	Rev 12	9/2002

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

Page

Page

Page

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

General

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

1.0 Definitions

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1.22 Reactor Operating The UCD/MNRC reactor is operating whenever it is not shutdown or secured.

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

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

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

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

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

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

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

1.26 Reference Core Condition. The condition of the core when it is at ambient temperature (cold T<28° C), the reactivity worth of xenon is negligible (< $0.30) (i.e., cold and clean), and the central irradiation facility contains the graphite thimble plug and the aluminum thimble plug (CIF-1).

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

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

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

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

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

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

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

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

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

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

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

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

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

1.37 Surveillance Intervals. Maximum intervals are established to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term. The allowable surveillance interval is the interval between a check, test, or calibration, whichever is appropriate to the item 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 (LSSS)

2.1 Safety Limits

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

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

Specification -

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

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

Basis -

a. This fuel safety limit applies for conditions in which the cladding temperature is above 500 °C (Safety Analysis Report (SAR), 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 ZrH1.7. Measurements of hydrogen pressure in TRIGA fuel elements during transient operations have been made and compared with the results of analysis similar to that used to make the above prediction These measurements indicate that in a pulse where the maximum temperature in the fuel was greater than 1000 °C, the pressure ($ZrH_{1.65}$) was only about 6% of the equilibrium value evaluated at the peak temperature. Calculations of the pressure gave values about three times greater than the measured values. The analysis gives strong indications that the cladding will not rupture if fuel temperatures are never greater than 1200 °C to 1250°C, providing the cladding temperature is less than 500 ° C. For fuel with $ZrH_{1.7}$,a conservative safety limit is 1100 °C. As a result, at this safety limit temperature, the class pressure is a factor of 4 lower than would be necessary for cladding failure.

2.2 Limiting Safety System Setting

2.2.1 Fuel Temperature

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

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

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

Basis - For steady-state operation of the reactor, the limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 750 °C provides a safety margin at the point of the measurement of at least 137 °C for standard TRIGA fuel elements in any condition of operation. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is near the center and mid-plane of the fuel element. For pulse operation of the reactor, the same limiting safety system setting applies. However, the temperature channel will have no effect on limiting the peak power generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to limit the energy release after the pulse if the transient rod should not reinsert and the fuel temperature continues to increase.

3.0 Limiting Conditions For Operation

3.1 Reactor Core Parameters

3 1.1 Steady-State Operation

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

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

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

Basis - Operational experience and thermal-hydraulic calculations demonstrate that UCD/MNRC TRIGA fuel elements may be safely operated at power levels up to 2.3 MW with natural convection cooling. (SAR Chapter 4, Section 4.6.2.)

3.1.2 Pulse or Square Wave Operation

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

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

Specification -

a. For the pulse mode of operation, the maximum insertion of reactivity shall be 1.23% Δ k/k ($1.75);

b. For the square wave mode of operation, the maximum 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% Δ k/k ($1.92). Therefore, the 1.23% Δk/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 Reactivity Limitations

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

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

Specification -

a. Shutdown Margin - The reactor shall not be operated unless the shutdown margin provided by the control rods is greater than 0.35% Δ 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. Excess Reactivity - The maximum available excess reactivity (reference core condition) shall not exceed 6.65% Δ k/k ($9.50).

Basis -

a. This specification assures that the reactor can be placed in a shutdown condition from any operating condition and remain shutdown, even if the maximum worth control rod should stick in the fully withdrawn position (SAR 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 750°C or less	M
g. Watchdog Circuit	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

Basis -

a Table 3 2 3 The console manual scram allows rapid shutdown of the reactor from the control room (SAR Chapter 7, Section 7.1.2.5).

b Table 3 2 3 The reactor room manual scram allows rapid shutdown of the reactor from the reactor room.

c Table 3 2 3. 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 shutdown 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 750 ° C, 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 This section intentionally left blank

3.6 This section intentionally left blank

3 7 Reactor Radiation Monitoring Systems

3 7.1 Monitoring Systems

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

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

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

Table 3.7.1

REQUIRED RADIATION MONITORING INSTRUMENTATION

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

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

(**) monitors may be placed out-of-service for up to 2 hours for calibration and maintenance. During this out-of-service time, no experiment or maintenance activities shall be conducted which could result in alarm conditions (e.g , airborne releases or high radiation levels)

Basis -

a Table 3 7 1. The facility stack monitor provides information to operating personnel regarding the release of radioactive material to the environment (SAR Chapter 11, Section 11.1.1.1.4). The alarm setpoint on the facility stack monitor is set to limit Argon-41 concentrations to less than 10 CFR Part 20, Appendix B, Table 2, Column 1 values (averaged over one year) for unrestricted locations outside the operations area.

b Table 3 7 1 The reactor room radiation monitor provides information regarding radiation levels in the reactor room during reactor operation (SAR Chapter 11, Section 11.1.5.1), to limit occupational radiation exposure to less than 10 CFR 20 limits.

c Table 3 7 1. The radiation monitor located next to the purification system resin 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 1O CFR Part 20

Basis - The annual average concentration limit for Argon-41 in air in the unrestricted area is specified in Appendix B, Table 2, Column 1 of 10 CFR Part 20 10 CFR 20.1301 specifies dose limitations in the unrestricted area. 10 CFR 20.1101 specifies a constraint on air emissions of radioactive material to the environment. The SAR Chapter 11, Section 11.1.1.1.4 estimates that the routine Argon-41 releases and the corresponding doses in the unrestricted area will be below these limits.

3.8 Experiments

3.8 1 Reactivity Limits.

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

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

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

a. The absolute reactivity worth of any single moveable experiment in the pneumatic transfer tube, the central irradiation facility, the central irradiation fixture 1 (CIF-1), or any other in-core or in-tank irradiation facility, shall be less than $1.00 (0.7% Δ 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% Δ 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% Δ 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% Δ 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% Δ 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 thanthe maximum reactivity insertion limit of $1.92. The SAR Chapter 13, Section 13.2.2.2.1 indicates that a pulse reactivity insertion of $1.92 would be needed to reach the fuel temperature safety limit.

3.8 2 Materials Limit

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

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

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

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

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

c. Each experiment in the I-125 production facility shall be controlled such that the total inventory of I-125 in the I-125 glove box shall not exceed 40 curies.

d Each experiment in the I-125 production facility shall be controlled such that the total inventory of I-125 being handled in the I-125 fume hood at any one time in preparation for shipping shall not exceed 20 curies An Additonal 1.0 curie of I-125 (up to 400 millicuries in the form of quality assurance samples and up to 600 millicuries in sealed storage containers) may also be present in the I-125 fume hood

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

f. Explosive materials in quantities of three (3) pounds of TNT equivalent or less may be irradiated in any radiography bay. The irradiation of explosives in any bay is limited to those assemblies where a safety analysis has been performed that shows that there is no damage to the reactor safety systems upon detonation (SAR Chapter 13, Section 13 2.6.2).

Basis -

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

b. The 1.5 curies limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, occupational doses and doses to members of the general public in the unrestricted areas shall be within the limits in 10 CFR 20 (SAR Chapter 13, Section 13.2.6.2).

c&d. Limiting the total I-125 inventory to forty (40.0) curies in the I-125 glove box and to twenty-one (21.0) curies in the I-125 fume hood assures that, if either of these inventories of I-125 is totally released into its respective containment, or if both inventories are simultaneously released into their respective containments, the occupational doses and doses to members of the general public in the unrestricted areas will be within the limits of 10 CFR 20 (SAR Chapter 13, Section 13 2.6.2).

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

f. The failure of an experiment involving the irradiation of 3 lbs TNT equivalent or less in any radiography bay external to the reactor tank will not result in damage to the reactor controls or the reactor tank. Safety Analyses have been performed (SAR Chapter 13, Section 13.2.6.2) which show that up to six (6) pounds of TNT equivalent can be safely irradiated in any radiogaphy bay. Therefore, the three (3) pound limit gives a safety margin of two (2).

3.8.3 Failure and Malfunctions

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

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

Specification -

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

(1) normal operating conditions of the experiment or the reactor,

(2) credible accident conditions in the reactor, or

(3) where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the reactor building or into the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor room will not result in exceeding the applicable dose limits in 10 CFR Part 20 in the unrestricted area, assuming 100% of the gases or aerosols escapes.

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

(1) If the effluent from an experiment facility exhausts through a stack which is closed on high radiation levels, at least 10% of the gaseous activity or aerosols produced will escape.

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

(3) For materials whose boiling point is above 130 °C and where vapors formed by boiling this material can escape only through an undistributed column of water above the core, at least 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 prior to pulsing operations.

(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 prior to pulsing 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.

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

4 2.4 Reactor Fuel Elements

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

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

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

a. The lead elements (i.e., all elements adjacent to the transient rod, with the exception of 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.

Basis (Technical Specifications 4.2.4.a-b) - 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 Reactor Coolant Systems

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

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

Specification -

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

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

(2) A channel test quarterly.

(3) A channel calibration annually.

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

(1) A channel test monthly.

c The purification inlet conductivity monitors shall have the following:

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

(2) A channel test monthly.

(3) A channel calibration semiannually.

d. The Emergency Core Cooling System 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 Reactor Room Exhaust System

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

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

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

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

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

4.5 This section intentionally left blank

4.6 This section intentionally left blank.

4.7 Reactor Radiation Monitoring Systems

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

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

Specification -

a. The facility stack monitor shall have the following.

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

(2) A channel test weekly.

(3) A channel calibration annually.

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

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

(2) A channel test weekly.

(3) A channel calibration annually.

c The purification system radiation monitor shall have the following:

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

(2) A channel test weekly.

(3) A channel calibration annually.

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

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

(2) A channel test weekly.

(3) A channel calibration annually.

Discovery of noncompliance with Technical Specifications 4.7.a-d shall limit operations to that required to perform the surveillance.

Basis -

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

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

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

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

4.8 Experiments

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

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

Specification -

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

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

c. The reactivity worth of any experiment installed in the pneumatic transfer tube, or in any other UCD/MNRC reactor in-core or in-tank irradiation facility shall be estimated or measured, as appropriate, before reactor operation with said experiment. Whenever a measurement is done it shall be done at ambient conditions.

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

Basis -

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

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

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

5.0 Design Features

5.1 Site and Facility Description

5.1.1 Site

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

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

Specification -

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

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

c. The TRIGA reactor is located in Building 258, Room 201 of the UCD/MNRC. This building has been designed with special safety features.

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

Basis -

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

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

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

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

5.1.2 Facility Exhaust

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

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

Specification -

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

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

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

5.2 Reactor Coolant System

Applicability - This specification applies to the reactor coolant system.

Objective - The objective is to assure that adequate water is available for cooling and shielding during normal reactor operation or during a Loss of Coolant Accident

Specification -

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

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

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

Basis -

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

b A reactor tank water low level alarm sounds when the water level drops significantly. This alarm annunciates in the reactor control room and at a 24 hour monitored location so that appropriate corrective action can be taken to restore water for cooling and shielding.

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

5 3 Reactor Core and Fuel

5.3.1 Reactor Core

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

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

Specification - For operation at 0.5 MW or greater, the reactor core shall be an arrangement of 96 or more fuel elements to include fuel followed control rods. Below 0.5 MW there is no minimum required number of fuel elements. In a mixed 20/20, 30/20 and 8.5/20 fuel loading (SAR Chapter 4, Section 4.5.5.6):

Mix J Core and Other Variations

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

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

20E Core and Other Variations

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

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

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

30B Core and Other Variations

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

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

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

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

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

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

5.3.2 Reactor Fuel

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

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

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

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

b. Hydrogen to zirconium atom ratio (in the ZrH_x): 1.60 to I 70 (I.65+/- 0.05).

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

Basis -

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

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

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

5.3.3 Control Rods and Control Rod Drives

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

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

Specification -

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

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

Basis -

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

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

5.4 Fissionable Material Storage

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

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

Specification -

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

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

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

6.0 Administrative Controls

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

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

6.1.2 Responsibilities. The UCD/MNRC Director shall be accountable to the Vice Chancellor of the Office of Research for the safe operation and maintenance of the reactor and its associated equipment. The UCD/MNRC Director, or his designated alternate, shall review and approve all experiments and experiment procedures prior to their use in the reactor. Individuals in the management organization (e.g., Site Manager, Reactor Manager, Health Physics Manager, etc.) shall be responsible for implementing UCD/MNRC policies and for operation of the facility, and shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to the operating license and technical specifications. The Site Manager shall report directly to the UCD/MNRC Director. The Reactor Manager and Health Physics Manager report directly to the Site Manager.

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 seven (7) members knowledgeable in fields which relate to nuclear safety. The NSC shall evaluate and review nuclear safety associated with the operation and use of the UCD/MNRC.

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

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

b. Voting rules.

c. Quorums (For the full committee, a quorum will be at least seven (7) 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 for approval.

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

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

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

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

g Review abnormal performance of facility equipment and operating anomalies.

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

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

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

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

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

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

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

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

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

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

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

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

j. Assessment of operating abnormalities.

k. Determination of the status of previous NSC recommendations.

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

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

6.4.1 Reactor Operations Procedures

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

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

c. Control rod removal or replacement.

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

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

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

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

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

6.4.2 Health Physics Procedures

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

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

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

d. Monitoring radioactivity in the environment surrounding the facility.

e. Administrative guidelines for the facility radiation protection program to include personnel orientation and training.

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

g. Leak testing of sealed sources containing radioactive materials.

h. Special nuclear material accountability.

i. Transportation of radioactive materials.

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

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

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

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

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

6.6 Required Actions

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

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

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

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

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

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

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

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

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

6 6.2 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 Reports. 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 Operating Reports. An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted within six months following the end of each calendar year. Each annual report shall include the following information:

a A brief summary of operating experiences including experiments performed, changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, and results of surveillance tests and inspections.

b. A tabulation showing the energy generated by the reactor (in megawatt hours), hours the reactor was critical, and the 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.

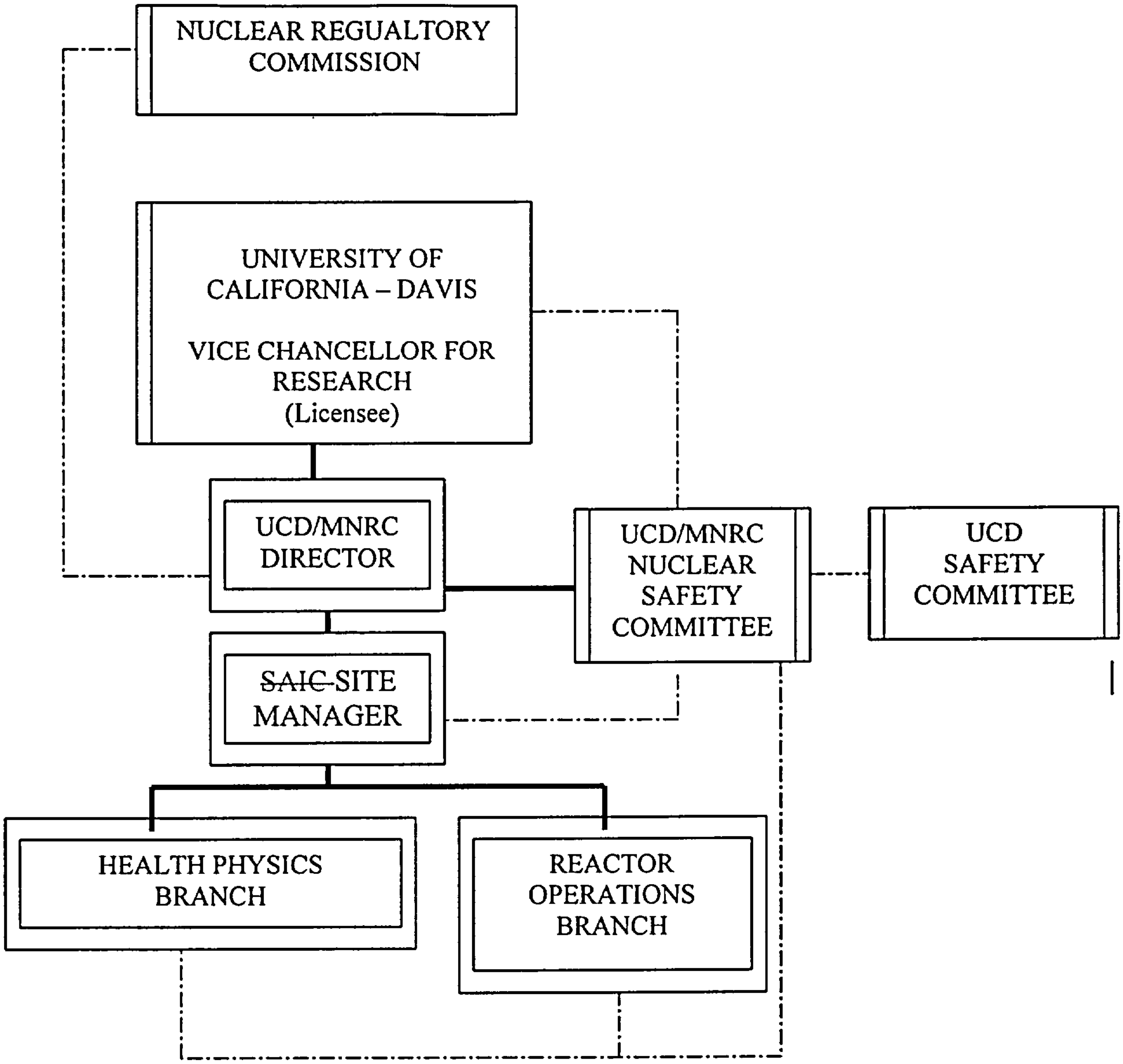
g. Offsite environmental monitoring surveys.

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.



Formal Licensing Channel

Administrative Reporting Channel

Communications Channel

UCD/MNRC ORGANIZATION FOR LICENSING AND OPERATION

FIGURE 6.1

CHAPTER 13

ACCIDENT ANALYSIS

Chapter 13 - Valid Pages
Rev.8 9/2002

i	Rev. 2	04/03/98
ii	Rev. 6	03/15/01
iii	Rev. 4	12/24/99
iv	Rev. 2	04/03/98
v	Rev. 6	03/15/01
13-1	Rev. 4	12/24/99
13-2	Rev. 4	12/24/99
13-3	Rev. 4	12/24/99
13-4	Rev. 4	12/24/99
13-5	Rev. 4	12/24/99
13-6	Rev. 2	04/03/98
13-7	Rev. 2	04/03/98
13-8	Rev. 2	04/03/98
13-9	Rev. 4	12/24/99
13-10	Rev. 2	04/03/98
13-11	Rev. 4	12/24/99
13-12	Rev. 4	12/24/99
13-13	Rev. 4	12/24/99
13-14	Rev. 4	12/24/99
13-15	Rev. 2	04/03/98
13-16	Rev. 2	04/03/98
13-17	Rev. 2	04/03/98
13-18	Rev. 2	04/03/98
13-19	Rev. 2	04/03/98
13-20	Rev. 4	12/24/99
13-21	Rev. 2	04/03/98
13-22	Rev. 2	04/03/98
13-23	Rev. 2	04/03/98
13-24	Rev. 4	12/24/99
13-25	Rev. 2	04/03/98
13-26	Rev. 2	04/03/98
13-27	Rev. 4	12/24/99
13-28	Rev. 2	04/03/98
13-29	Rev. 4	12/24/99
13-30	Rev. 4	12/24/99
13-31	Rev. 2	04/03/98
13-32	Rev. 4	12/24/99
13-33	Rev. 4	12/24/99
13-34	Rev. 4	12/24/99
13-35	Rev. 4	12/24/99
13-36	Rev. 4	12/24/99
13-37	Rev. 4	12/24/99
13-38	Rev. 6	03/15/01
13-39	Rev. 2	04/03/98
13-40	Rev. 2	04/03/98
13-41	Rev. 4	12/24/99
13-42	Rev. 4	12/24/99
13-43	Rev. 3	03/15/99
13-44	Rev. 3	03/15/99
13-45	Rev. 6	03/15/01
13-46	Rev. 6	03/15/01
13-47	Rev. 6	03/15/01
13-48	Rev. 6	03/15/01
13-49	Rev. 6	03/15/01
13-50	Rev. 6	03/15/01
13-51	Rev. 6	03/15/01
13-52	Rev. 6	03/15/01
13-53	Rev. 6	03/15/01
13-54	Rev. 6	03/15/01
13-55	Rev. 6	03/15/01
13-56	Rev. 6	03/15/01
13-57	Rev. 6	03/15/01
13-58	Rev. 6	03/15/01
13-45	Rev. 8	9/02
13-46	Rev. 8	9/02
13-47	Rev. 8	9/02
13-48	Rev. 8	9/02
13-49	Rev. 8	9/02
13-50	Rev. 8	9/02
13-51	Rev. 8	9/02
13-52	Rev. 8	9/02
13-53	Rev. 8	9/02
13-54	Rev. 8	9/02
13-55	Rev. 8	9/02
13-56	Rev. 8	9/02
13-57	Rev. 8	9/02
13-58	Rev. 8	9/02

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

REFERENCES

13.1 Credible Accident Analyses for TRIGA® and TRIGA®-Fueled Reactors. NUREG/CR- 2387, PNL-4028, April 1982.

13.2 Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content, NUREG-1537, Part 1, February 1996.

13.3 SRA (Shonka Research Associates, Inc.), Software Verification and Validation Report
for the WINCO RSAC-5 Code, Marietta, GA, 1993.

13.4 ANSI/ANS 10.4, "American National Standard Guidelines for the Verification and Validation of Scientific and Engineering Programs for the Nuclear Industry," 1987.

13.5 Research Reactor Core Conversion Guidebook, IAEA-TECDOC-643, April 1992; reprinted as UZR-27 by General Atomics.

13.6 Saphier, D., "The Simulation Language of DSNP," ANL-CT-77-20, Rev. 02, Argonne National Laboratory, 1978.

13.7 GA Report, "Low Enriched TRIGA® Fuel-Water Quench Safety Test," GA-A15413.

13.8 Larsen, G. G., "An Analysis of Heat Transfer Data for Spray-Cooled Rods," Nuclear Safety 9, No. 1 (Jan.-Feb. 1968), and Hideo, Uchida, "Experimental Investigation of Core Spray," Nuclear Safety 7, No. 3 (Spring 1966).

13.9 Jensen, R. T., "Experimental Evaluation of Emergency Core Spray Cooling Performance," USAEC Report IDO-17249, Phillips Petroleum Co., March 1968.

13.10 Sharp, R., "TRIGA® Emergency Spray Cooling Experiment," Gulf Energy & Environmental System Report GA-10521, March 31, 1971.

13.11 Razvi, J., Sorrento Electronics, Personal Communication to W. Richards, "Additional Information related to ECCS," April 4, 1997.

13.12 Boonstra, R. H., "TAC2D, A General Purpose Two-Dimensional Heat Transfer Computer Code - User's Manual," General Atomics Report GA-A 14032 (15 July 1976).

13.13 Schaefer, Bob (ANL-West) to Dan Newell (MNRC), "Power by Element in 20E with AL/C Plug," e-mail communications (Dec. 11 & 12, 1996).

13.14 Engineering Science, 1983, "Final Report Installation Restoration Program, Phase II, ...Confirmation," Vol. 1, June, Prepared for the U.S. Air Force, McClellan Air Force Base, Sacramento, CA.

13.15 Safety Analysis Report, TRIGA® Reactor Facility, University of Texas at Austin, May 1991.

13.16 10 CFR 50.59 Safety Analysis of Explosive Limits for Radiography Bays 1, 2, 3, and 4, McClellan Nuclear Radiation Center, Sacramento, CA., 1996.

13.17 Southwest Research Institute, "Safety Analysis To Determine Limiting Criteria for Explosives in Bay 3 of the McClellan Nuclear Radiation Center," September 1995.

13.18 Liu, H. Ben, "Safety Analysis for the Central Irradiation Facility (CIF) at the MNRC," Memorandum to Wade J. Richards, September 22, 1998.

13.19 Strohmayer, W.H., and M.G. Stamatelatos, "Probabilistic Assessment of the Airplane Crash Risk for the McClellan Air Force Base TRIGA® Reactor," G.A. Technologies, March 1986.

13.20 Liu, H. Ben, "Safety Analysis for I-125 Production at the UCD/MNRC," Memorandum to Dan Newell, November 14, 2000.

13.20.A Liu, H. Ben, "Safety Analysis Report for Two Irradiation Chambers in the UCD/MNRC's Reactor Core," Memorandum to Chuck Heidel, August 13, 2002.

13.21 EPA 520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," U.S. Environmental Protection Agency, September 1988.

13.0 ACCIDENT ANALYSIS

13.1 Introduction

In about 1980, the U.S. Nuclear Regulatory Commission requested an independent and fresh overview analysis of credible accidents for TRIGA® and TRIGA®-fueled reactors. Such an analysis was considered desirable since safety and licensing concepts had changed over the years. The study resulted in NUREG/CR-2387, Credible Accident Analysis for TRIGA® and TRIGA®-fueled Reactors (Reference 13.1). The information developed by the TRIGA® experience base and appropriate information from NUREG/CR-2387 serve as a basis for some of the information presented in this chapter of the UCD/MNRC Safety Analysis Report.

The reactor physics and thermal-hydraulic conditions in the UCD/MNRC TRIGA® reactor at a power level of 2 MW are established in Chapter 4. The core physics analysis demonstrates that the fundamental physical conditions in the UCD/MNRC reactor are preserved by an appropriate choice of the composition of mixed TRIGA® fueled cores containing 8.5 and 20 wt % fuel (<20 at. % enriched). A compact critical loading fueled entirely with 20 wt % fuel could have an unacceptably large peak element power with the reactor operating at the 2 MW power level. In contrast, the reference loading with all 20 wt % fuel, 20E, has acceptable power peaking because it contains a central region that does not contain fuel, which results in a larger core size (Chapter 4). This in turn results in a lowering of the maximum power generation in individual fuel elements.

The fuel temperature is a limit in both steady-state and pulse mode operation. This limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element cladding material. The strength of the cladding as a function of temperature sets the upper limit on the fuel temperature. Fuel temperature limits of 1100°C (with clad <500°C) and 930°C (with clad >500°C) for U-ZrH with a H/Zr ratio less than 1.70 have been set to preclude the loss of clad integrity (Section 4.5.4.1.3).

Nine credible accidents for research reactors were identified in NUREG-1537 (Reference 13.2) as follows:

- the maximum hypothetical accident (MHA);
- insertion of excess reactivity;
- loss of coolant accident (LOCA);
- loss of coolant flow;
- mishandling or malfunction of fuel;
- experiment malfunction;
- loss of normal electrical power;
- external events;
- mishandling or malfunction of equipment.

This chapter contains analyses of postulated accidents that have been categorized into one of the above nine groups. Some categories do not contain accidents which appeared applicable or credible for the UCD/MNRC TRIGA® reactor, but this was acknowledged in a brief discussion of the category. Some categories contain an analysis of more than one accident even though one is usually limiting in terms of impact. Any accident having significant radiological consequences was included.

For those events that do result in the release of radioactive materials from fuel, only a qualitative evaluation of the event is presented. Events leading to the release of radioactive material from a fuel element were analyzed to the point where it was possible to reach the conclusion that a particular event was, or was not, the limiting event in that accident category. The maximum hypothetical accident (MHA) for TRIGA® reactors is the cladding failure of a single irradiated element in air with no radioactive decay of contained fission products. Calculations supporting the analysis of this accident and several of the other accidents discussed in this chapter are contained in Appendix B.

13.2 Accident Initiating Events and Scenarios, Accident Analysis, and Determination of Consequences

13.2.1 Maximum Hypothetical Accident

13.2.1.1 Accident Initiating Events and Scenario

A single fuel element could fail at any time during normal reactor operation or while the reactor was shutdown, owing to a manufacturing defect, corrosion, or handling damage. This type of failure is infrequent, based on many years of operating experience with TRIGA® fuel, and such a failure would not normally incorporate all the necessary operating assumptions required to obtain a worst case fuel failure scenario.

For the UCD/MNRC TRIGA® reactor, the MHA has been defined as a cladding rupture of one highly irradiated fuel element with no decay followed by instantaneous release of fission products into the air. The failed fuel element was assumed to have been operated at the highest core power density for a continuous period of 1 year at 2 MW. This is the most severe accident for a TRIGA® and is analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in the unrestricted area.

A realistic scenario for the MHA is difficult to establish since fuel handling, the activity frequently associated with this accident, would be unlikely to occur immediately after reactor shutdown, and fuel elements would not be moved out of the reactor tank into air with no time to decay. Nevertheless, the accident has been analyzed for the UCD/MNRC TRIGA® in Appendix B and the results are summarized in this section.

13.2.1.2 Accident Analysis and Determination of Consequences

The fission product inventory used in the MHA is listed in Table B-1 of Appendix B. These data are based on compilations from Reference B.1 and have been adjusted for 2 MW operation. The data are for the volatile fission products present at shutdown in a fuel element run to saturation at the highest core power density.

A fission product release fraction of 7.7×10^{-5} is assumed for the release of noble gases and halogens from the fuel to the cladding gap. This release fraction is developed in Section 4.5.5.7, and is based on a RELAP5/3.1 calculation of fuel temperature in the hottest core element. In addition, it is assumed that 100% of the noble gases ultimately reach the unrestricted environment outside the reactor building and that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. This value for the halogens is based on historical usage and recommendations from Appendix B References B.2, B.3, B.4, B.5, and B.6, where Reference B.2 recommends a 50% release of the halogens. References B.3 and B.4 apply a natural reduction factor of 50% due to plateout in the building. This latter 50% applied to the 50% of the inventory released from the fuel element cladding gap results in 25% of the available halogen inventory reaching the outside environment. It should be noted, however, that this value appears to be quite conservative based on the 1.7% gap release fraction for halogens quoted in References B.7 and B.8.

Radiological consequence calculations were done using the Radiological Safety Analysis Computer Program (RSAC-5), Version 5.2, 02/22/94 (Reference B.9). RSAC-5 calculates the consequences of the release of radionuclides to the atmosphere and it can generate a fission product inventory; decay and ingrow the inventory with time; model the downwind dispersion of the activity; and calculate doses to downwind individuals. RSAC-5 has been subjected to extensive independent verification and validation for use in performing safety-related dose calculations to support safety analysis reports. Shonka Research Associates, Inc. (Reference 13.3) conducted this verification and validation in accordance with the guidelines presented in ANSI/ANS-10.4, "American National Standard Guidelines for the Verification and Validation of Scientific and Engineering Programs for the Nuclear Industry" (Reference 13.4).

For the MHA at the UCD/MNRC, dispersion coefficients (χ/Q values) for locations in the unrestricted area, 10 m (the UCD/MNRC perimeter fence line nearest the facility which defines the interface of the restricted and unrestricted area) out to 100 m, were input directly into the code and were calculated using Regulatory Guide 1.145 methodology (Reference B.11). Calculations were performed for Pasquill weather classifications A through F. Diffusion coefficients were taken from Reference B.12 and are presented in Table B-2. Calculations were performed assuming a ground level release at an 800 cfm reactor room release rate without any credit for stack height or building wake effects, which would only improve mixing and lower projected doses. Furthermore, it was assumed that all of the fission products were released to the unrestricted area by a single reactor room air change, which maximizes the dose rate to persons exposed to the plume during the accident and minimizes the exposure time to receive the highest estimated dose from this accident. These latter assumptions regarding release are very, very conservative since the reactor room is not at ground level and, rather than 10 meters, is approximately 30 meters from the perimeter fence. Furthermore, there are no normal or direct flow pathways to support an 800 cfm ground level flow from the reactor room to the unrestricted area.

The results of the RSAC-5 calculations for the MHA are shown in Tables B-4 and B-5. Shown are doses inside the reactor room and doses at several locations in the unrestricted area outside the UCD/MNRC (10 to 100 m from the building) as a function of weather class. Results are reported for the Committed Dose Equivalent (CDE) to the thyroid, the Committed Effective Dose Equivalent (CEDE) due to inhalation, the Deep Dose Equivalent (DDE) due to air immersion, and the Total Effective Dose Equivalent (TEDE) resulting from adding the CEDE and the DDE.

As indicated by the results in Table 13-1, the occupational dose to workers who evacuate the reactor room within 5 minutes following the MHA should be approximately 454 millirem Total Effective Dose Equivalent and 11,500 millirem Committed Dose Equivalent to the Thyroid. If evacuation were to occur within 2 minutes, as it no doubt would because the reactor room is small and easy to exit, the doses drop to 180 millirem TEDE and 4,640 millirem CDE. All of these doses are well within the NRC limits for occupational exposure as stated in 10 CFR 20.1201.

Accident: Cladding Failure in Air (MHA)				
	CDE Thyroid (millirem)	CEDE (millirem)	DDE (millirem)	TEDE (millirem)
2 minute room occupancy	4,640	140	40	180
5 minute room occupancy	11,500	360	94	454

Table 13-1 Occupational Radiation Doses in the UCD/MNRC Reactor Room Following the Maximum Hypothetical Accident.

Projected doses to the general public in the unrestricted area around the UCD/MNRC following the MHA are shown in Table 13-2. To receive the indicated dose, a person must be exposed to the airborne plume from the reactor room for the entire 9.2 minute period it is being vented. Even using this exposure requirement at the closest distance to the UCD/MNRC building (10 meters), and assuming the most unfavorable atmospheric conditions (Category F), the maximum TEDE to a member of the general public would be 66 millirem.

Although this accident and the corresponding radiation doses are never expected to occur, the maximum estimated dose of 66 millirem to the general public is still within the 100 millirem TEDE limit for the general public published in the NRC's most recent revision to 10 CFR 20 (Reference 10 CFR 20.1301). Furthermore, the above analysis clearly shows that the UCD/MNRC can be subjected to current MHA criteria and remain within dose limits established by the NRC for occupational radiation exposure and exposure of the general public. As a point of interest, should the MHA occur after 48 hours of decay, the maximum TEDE to the public drops to approximately 34 millirem.

Distance (Meters)	CDE Thyroid (millirem)	CEDE (millirem)	DDE (millirem)	TEDE (millirem)
10	1,694	53	13	66
20	1,330	42	9.9	52
40	90	2.9	6.7	0.6
80	52	1.7	3.7	5.4
100	42	1.3	3.0	4.3

Table 13-2 Radiation Doses to Members of the General Public Under the Most Conservative Atmospheric Conditions (Pasquill F) at Different Distances from the UCD/MNRC Following a Fuel Element Cladding Failure in Air with No Decay (The MHA).

- CDE - Committed Dose Equivalent
- CEDE - Committed Effective Dose Equivalent
- DDE - Deep Dose Equivalent
- TEDE - Total Effective Dose Equivalent

13.2.2 Insertion of Excess Reactivity

13.2.2.1 Accident Initiating Events and Scenarios

The most credible generic accident is the inadvertent rapid insertion of positive reactivity which could, if large enough, produce a transient resulting in fuel overheating and a possible breach of cladding integrity. Operator error or failure of the automatic power level control system could cause such an event to occur due to the uncontrolled withdrawal of a single control rod. Flooding or removal of beam tube inserts could also have a positive effect on reactivity but not as severe as removal of a control rod. In a separate scenario, a large reactivity insertion was postulated to create fuel cladding temperatures which might cause a metal-water reaction, but for many reasons this accident is not considered to be a safety risk in TRIGA® reactors.

13.2.2.2 Accident Analysis and Determination of Consequences

13.2.2.2.1 Maximum Reactivity Insertion

Raising the temperature of TRIGA® fuel has a strong, prompt negative reactivity effect, which can overcome a rapid reactivity insertion such as that produced by the firing of the transient rod. The quantity that captures this effect is the prompt negative temperature coefficient discussed in Section 4.5.4.2. There is a limit to the protection provided by this feedback, since the peak fuel temperature attained before the feedback terminates the transient increases with the magnitude of the inserted reactivity. The Nordheim-Fuchs model was used to compute the maximum reactivity pulse that can occur without exceeding the safety limit of 1100°C established in Section 4.5.4.1.3.

In the Nordheim-Fuchs model it is assumed the transient is so rapid that 1) the temperature rise is adiabatic and 2) delayed neutrons can be neglected. Thus, the model is given by the following set of coupled differential equations:

$$\frac{dn}{dt} = \frac{\rho-\beta}{l} \times n \; ;$$

$$\rho(T) = \rho_0 - \alpha(T) \times T \; ;$$

$$\frac{dT}{t} = \frac{n}{C_\rho(T)} \; ;$$

Where n is the reactor power, ρ is the time-dependent reactivity, l is the neutron lifetime, β is the effective delayed neutron fraction, T is the core-average temperature, ρ_0 is the reactivity insertion, α is the temperature feedback reactivity coefficient, and C_ρ is the whole-core heat capacity. Given values of β, l, and ρ_0, and expressions for α and C_ρ, this set of equations was solved numerically using simple finite difference techniques. The quantity of interest in the solution if ΔT, the difference between the maximum and initial values of the core-average fuel temperature. From the solution ΔT, the peak fuel temperature was found using the simple expression:

$$T_{peak} = T_0 + PF \times \Delta T \; ;$$

where T_0 is the initial temperature and PF is the total peaking factor. In the equation just described, ρ_0 is an input parameter and T_{peak} is the output, yet what is needed is the reverse; the object was to find the value of ρ_0 that yields $T_{peak} = 1100°C$. The object was attained by an iterative search. The search converged in no more than 3 iterations (estimates of ρ_0) because T_{peak} varies essentially linearly with ρ_0 over a wide range.

The following input values were used for all the results displayed here:

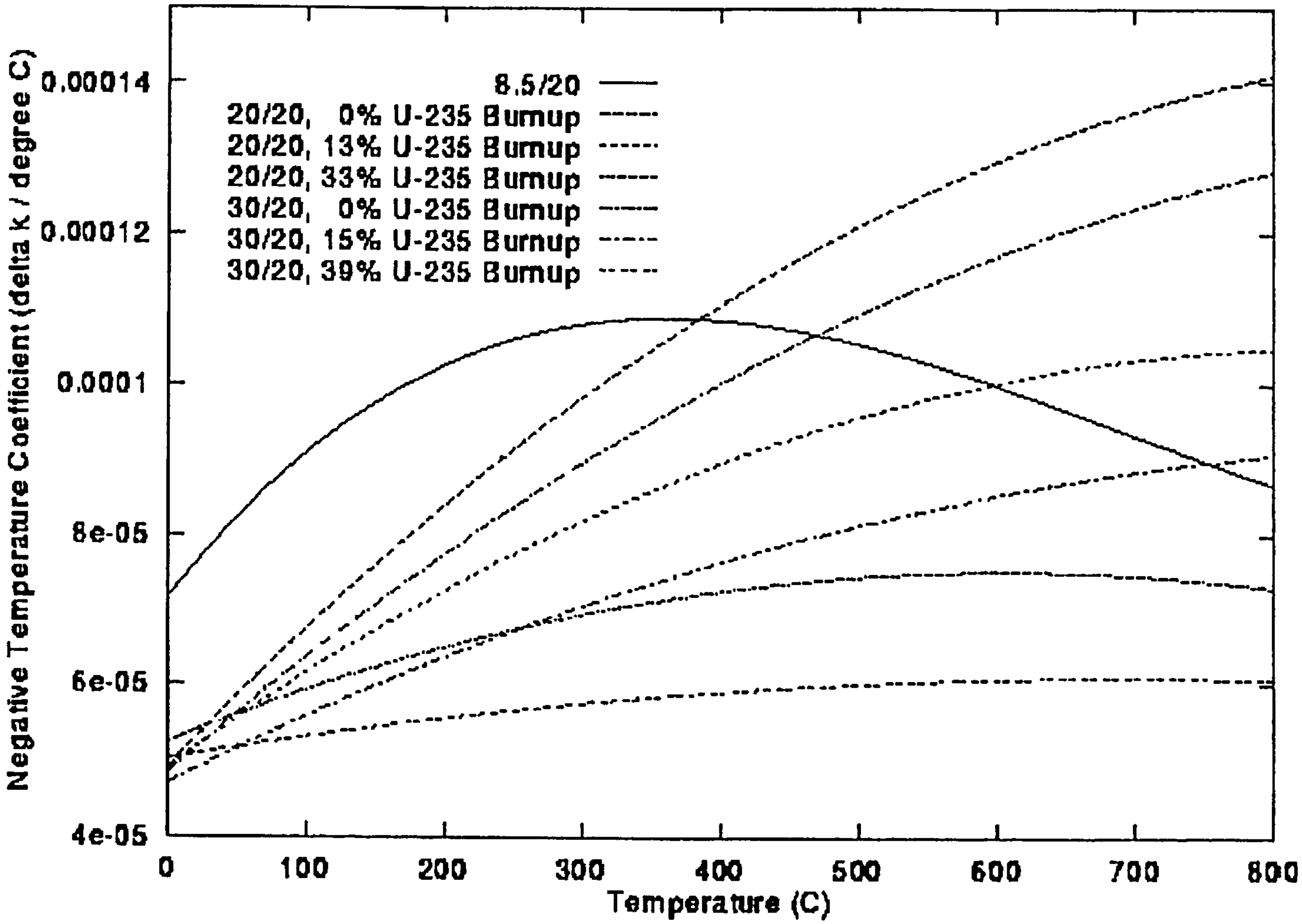
$$\beta = 0.007;$$
$$l = 32\ \mu s;$$
$$T_0 = 20°C;$$
$$PF = 4.86.$$

Although some quantities, such as the peak reactor power, depend on the value of l, T_{peak} was found not to change with reasonable variations in l. The value of β is well known. The value used for T_0 is the nominal zero-power temperature. The value of PF is the largest total peaking factor in Table 4-16 (Section 4.5.5.5). This value was determined for the 20E reference core with all fresh fuel and control rods raised 2/3 of full travel. It is a conservative value for any other permissible core loading and rod bank position.

The reactivity insertion limit is shown for seven cases in Table 13-3. Each case has a different combination of fuel type and core-average burnup. The expressions for heat capacity, Cp, as a function of temperature correspond to a minimum core size, 94 elements, and they were derived using the prescription in Reference 13.5. The insertion limit was found to be independent of Cp. The insertion limit is sensitive to the prompt negative temperature coefficient, α. The curves in Figure 13.1 show that this coefficient varies with temperature, fuel type and fuel burnup. The different expressions for α are directly responsible for the differences in the reactivity insertion limit among the seven cases in Table 13-3.

Fuel Type	Burnup (%)	(MWD/rod)	Heat Capacity C_p (watt-sec/°C)	Prompt Negative Temperature Coefficient α ($\Delta k/k°C$)	Reactivity ρ_0 ($)
8 5/30	independent		$7.31 \times 10^4 + 150T$	$7.16 \times 10^{-5} + 2.33 \times 10^{-7}T - 4.35 \times 10^{-10}T^2 + 2.09 \times 10^{-13}T^3$	2.66
20/20	0	0	$7.12 \times 10^4 + 143T$	$4.91 \times 10^{-5} + 1.93 \times 10^{-7}T - 9.73 \times 10^{-11}T^2$	2.33
20/20	13	10	$7.12 \times 10^4 + 143T$	$4.90 \times 10^{-5} + 1.32 \times 10^{-7}T - 7.82 \times 10^{-11}T^2$	2.16
20/20	33	27	$7.12 \times 10^4 + 143T$	$5.24 \times 10^{-5} + 7.45 \times 10^{-8}T - 6.13 \times 10^{-11}T^2$	2.06
30/20	0	0	$7.39 \times 10^4 + 145T$	$4.84 \times 10^{-5} + 1.59 \times 10^{-7}T - 7.34 \times 10^{-11}T^2$	2.23
30/20	15	20	$7.39 \times 10^4 + 145T$	$4.71 \times 10^{-5} + 9.13 \times 10^{-8}T - 4.63 \times 10^{-11}T^2$	2.03
30/20	39	54	$7.39 \times 10^4 + 145T$	$5.02 \times 10^{-5} + 3.10 \times 10^{-8}T - 2.24 \times 10^{-11}T^2$	1.92

Table 13-3 Maximum Reactivity Insertion and Related Quantities for Various Fuels and Burnups



PROMPT NEGATIVE TEMPERATURE COEFFICIENT FOR TRIGA® FUELS
FIGURE 13.1

The worst-case result in Table 13-3, \$1.92, was chosen as the maximum reactivity insertion allowed for the UCD/MNRC reactor. There are at least two reasons why this is a conservative bound. One is that the core-average burnup, 39% ^{235}U, is greater than is likely to be achieved, which means that there will be more prompt feedback reactivity than was used for this case. The other reason is that the peaking factor is significantly larger than would be the actual case for a highly burned 30/20 loading.

13.2.2.2.2 Uncontrolled Withdrawal of a Control Rod

Operator error or failure of the automatic power level control system could cause one of the control rods to be driven out, starting at either high or low power levels. The maximum speed of a control rod is 1.78 cm/sec (42. in./min). The maximum single rod worth for the reference loadings of Section 4.5.5 is \$2.65, but a rod worth of \$3.50 was used here to allow for reasonable variations about the reference loadings. These values were combined with a measured rod worth profile to calculate the inserted reactivity as a function of time.

The initial reactor power levels of 100 W or 2 MW were analyzed using the Dynamic Simulator for Nuclear Power Plants (DSNP) code to solve the one group point kinetics equation with a delayed neutron fraction of 0.007 and the one group decay constant equal to 0.405 sec^{-1} (Reference 13.6). The one group decay constant was chosen to match solutions near prompt critical. The feedback reactivity was assumed to be:

$$\alpha (T) = 5.018 \times 10^{-5} + 3.097 \times 10^{-8}\, T - 2.244 \times 10^{-11}\, T^2 \;\; \Delta k/k/°C.$$

This corresponds to worst-case (weakest feedback) conditions (i.e.., 30/20 fuel at end-of-life).

The heat capacity of the core was assumed to be:

$$C_P = 7.39 \times 10^4 + 145.0\, T \text{ watt-second/°C}.$$

Again, this corresponds to worst-case conditions for all 30/20 fuel loadings with 94 fueled elements. The most unfavorable initial control rod position was assumed to be 32% inserted. The insertion rate at this position is \$0.23/sec..

The amount of shutdown reactivity at the time of scram is based on the following: four control rods are capable of providing a total of \$0.50 of shutdown reactivity (conservative assumption) and the rod that is adding reactivity for the "uncontrolled withdrawal of a control rod" is available as shutdown reactivity. Thus the total shutdown reactivity is \$0.50 plus reactivity of the moving rod at the time of scram. Rod fall time of 2 seconds is assumed. The rod fall time includes the power channel delay time.

For the case with an initial reactor power at 100 watts, an average fuel temperature of 35°C, and a worst case trip level setpoint of 2.3 MW, the reactor power was calculated to reach the trip point at 4.26 seconds. Assuming it takes 0.5 seconds for the signal to cause actual release of the rods, the peak reactivity inserted would be $1.18. As shown in Section 13.2.2.2.1, this amount of reactivity could be inserted instantaneously with no adverse safety effects.

For the case with initial power at 2 MW, an average fuel temperature of 257.2°C, and a worst case trip level setpoint of 2.3 MW, the reactor power was calculated to reach the trip point in 0.54 seconds. The scram signal causing actual release of the rods occurs 0.5 seconds later. Reactivity inserted at the time all of the rods are released is $0.25. This reactivity insertion is much less than the limiting reactivity insertion derived in Section 13.2.2.2.1 for the pulse accident.

13.2.2.2.3 Uncontrolled Withdrawal of All Control Rods

This accident has been analyzed using a measured rod worth profile and a total control rod worth for five rods of $17.50. Using the DSNP model with feedback described above, initial power of 100 W, and the rods at an assumed initial position of 32% insertion, with a normal rod withdrawal rate of 1.02 cm/sec. (24 in./min.) for all five rods, an initial reactivity insertion rate of $0.66/sec. is obtained. The worst case trip level setpoint of 2.3 MW is reached at 1.73 seconds with the scram occurring at 2.23 seconds. Reactivity inserted at the time all of the rods are released is $1.52. This reactivity insertion is less than the limiting reactivity insertion derived in Section 13.2.2.2.1 for the pulse accident. For five rods to add reactivity simultaneously, there must be multiple failures in the control system. Therefore, this accident is not considered to be credible.

Since three control rods can be banked for reactor control, uncontrolled withdrawal of three control rods could be considered credible, but is bounded by the accidents analyzed.

13.2.2.2.4 Beam Tube Flooding or Removal

In the event of flooding of one or more beam tubes, air or inert gas would be substituted with water. This will constitute a positive reactivity addition. It has been estimated that the worth of one flooded beam tube is about $0.25. This amount of excess reactivity is well below the limits discussed in Section 13.2.2.2.1; therefore, it does not represent a safety significant event.

During the removal of the in-tank section of a beam tube, air and graphite will be replaced by water because a portion of the graphite reflector is removed with this section of the beam tube. Again, replacement of the air/gas with water results in a positive increase in reactivity. On the other hand, replacement of graphite with water results in a negative effect on reactivity. The net result will be a smaller reactivity addition than for beam tube flooding so this action is of even less overall consequence.

13.2.2.2.5 Metal-water Reactions

Although metal-water reactions have occurred in some reactor accidents or destructive tests, the evidence from these events and laboratory experiments shows that a dispersed liquid metal is required for a violent chemical reaction to occur (References 13.1 and 13.7). The conditions for a solid metal-water reaction are not readily achievable in a reactor system such as the UCD/MNRC.

Water quench tests on TRIGA® fuel have been conducted to fuel temperatures as high as 1200°C without significant effect. Since the operating temperatures at 2 MW do not approach this temperature, this effect does not represent a safety risk. The only credible way in which temperatures high enough to allow metal-water reactions to be created in a TRIGA® reactor is through a large reactivity excursion. The limits set on excess reactivity preclude this.

13.2.3 Loss of Coolant Accident (LOCA)

13.2.3.1 Accident Initiating Events and Scenarios

Loss of coolant from the UCD/MNRC reactor could occur primarily through one of two scenarios, pumping water from the reactor tank or reactor tank failure. These scenarios are analyzed as part of this section.

13.2.3.2 Accident Analysis and Determination of Consequences.

13.2.3.2.1 Pumping of Water from the Reactor Tank

The intake for the primary-cooling-system pump is located about 3 ft below the normal tank water level. In addition, the line is perforated from about 8 in. below the normal tank water level to the intake line entrance. The intake for the purification-system pump is through a short flexible line attached to a skimmer that floats on the surface of the tank water. However, the length of the flexible line is such as to cause loss of pump suction if the tank water level is lowered about 4 ft. Thus, the reactor tank cannot be accidentally pumped dry by either the primary pump or the purification-system pump. Also, it is not possible for other cooling system or water cleanup system components to fail and syphon water from the tank since all of the primary-water-system and purification-system piping and components are located above the normal tank water level.

The tank could be pumped out with a portable pump, but this would require deliberate action on the part of the operators and it is inconceivable that such an action would take place while the reactor was operating or at any other time without removing the fuel and taking numerous other precautions. However, if the reactor were somehow pumped dry while the reactor was shut down, the fuel temperature obtained would be considerably lower than for a loss of water while the reactor was operating, and this unlikely error would not cause damage to the fuel elements. Similarly, the dose rate from the uncovered core and the water radioactivity concentration would be less than that shown in Sections 13.2.3.2.11 and 13.2.3.2.12.

13.2.3.2.2 Reactor Tank Failure

A hole in or near the bottom of the reactor tank could cause the water level to drop below the top of the fuel elements. This event could occur either during reactor operation or while the reactor was shut down and unattended. There are no nozzles or other penetrations in the reactor tank below the normal water level, so the only mechanisms that could cause tank failure are corrosion of the tank or a mechanical failure. Leaks caused by corrosion would unquestionably be small leaks, which would be detected before the water level had lowered significantly. In such a case, makeup water could be supplied by the auxiliary make-up water system (AMUWS) until the reactor had been unloaded or the leak repaired.

Provisions to monitor for and collect tank leakage have been incorporated into the facility design. First, the tank is surrounded by corrugated metal. The corrugations provide a path to the bottom of the tank for any water leakage from the walls. Second, a drain, see Chapter 5, within the bulk shield surrounds the bottom of the tank. This drain will collect any water that may leak from the tank walls or bottom. Third, a duct leads from the drain to Radiography Bay 1 and the exit of this duct is periodically monitored for water leakage. If leakage is detected, the water could be easily collected at this point or diverted to the liquid holdup tank outside the building.

Consequences of a slow tank leak would be minimal and would require collection and containment of the water which leaked from the tank. This would be easily accomplished by using the existing liquid effluent control system described above. Small tank leaks due to corrosion are normally repairable using conventional techniques for patching aluminum, and thus it is expected that a leak could be located and fixed before there would be any significant loss of water from the tank.

An earthquake of much greater intensity than the Uniform Building Code Zone 3 earthquake appears to be the only credible mechanism for causing a large rupture in the tank, since the tank when supported by its associated biological shield structure was designed (with an importance factor of 1.5) to withstand this magnitude of earthquake. Even if such an event is assumed to cause very rapid loss of water while the reactor is operating at peak power; a reactor shutdown would be caused by voiding of water from the core, even if there were no scram.

A large rupture of the tank would obviously result in a more rapid loss of water than a leak due to corrosion or a minor mechanical failure in the tank wall. The UCD/MNRC reactor tank has no breaks in its structural integrity (i.e., there are no beam tube protrusions or other discontinuities in the reactor tank surface). In addition, the reactor core is below ground level. Thus the potential for most types of leaks is minimized.

Part of the 2 MW upgrade to the reactor included a new cavity (Bay 5) cut into the biological shield. This cut exposes the reactor tank wall below the reactor core level, and this introduces an increased possibility of draining water from the core area. While steps have been taken to minimize the probability of a tank rupture in this location, and it is believed that the likelihood of such a rupture is very low, an unplanned occurrence could nevertheless initiate such a event. Therefore, an Emergency Core Cooling System (ECCS) has been installed to cool the core until the fuel has decayed to a level where air cooling is adequate to maintain fuel temperatures below the design basis limit (see Chapter 6 for details of the ECCS design and operation).

An analysis detailing the cooling capabilities of the ECCS is described in the sections which follow. This analysis does not postulate the occurrence of a particular initiating sequence of events leading to all fuel elements in the core being uncovered. Instead, it simply assumes that the tank has ruptured and all the water is lost. Such an event has several different consequences. First there is the possibility of fuel clad rupture should the fuel temperature exceed design basis values. This event is covered in the analysis that follows, and focuses on the action of the ECCS to prevent fuel temperatures from reaching safety limits. Second, there is a possibility of personnel exposure to radiation from the uncovered reactor core due to the direct beam from the core or from radiation scattered from the reactor room walls and ceiling. Finally, there is a chance that the lost water could cause ground water contamination. Both of these latter events are also analyzed as part of the LOCA evaluation.

13.2.3.2.2.1 Description of ECCS and Assumptions

A loss-of-coolant accident (LOCA) is postulated for the UCD/MNRC in which the reactor pool is rapidly drained of water during operation at 2 MW (it is assumed that the reactor has been running at 2 MW for an infinitely long time). Because the LOCA uncovers the core quickly, the fuel clad temperature in some of the centrally located fuel elements could exceed the design basis temperature limit of 930°C after a period of at least 20 minutes.

When the reactor tank water level drops below the normal operating range (typically a loss of approximately six (6) inches of water) a tank low-level alarm sounds. This alerts the operator that action must be taken. Depending upon the rate of water loss, the suspected cause of the loss, and other considerations, several different actions may be taken by the operator in response to a reduction in the tank water level. One such action could be activation of the ECCS.

Upon activation of the ECCS, cooling water from the domestic water supply will be introduced into the reactor tank and maintained until the fuel no longer contains sufficient decay heat to present a threat to the fuel cladding or water is restored to a level above the core. If the tank water level has dropped to less than about two (2) feet above the core, water from the ECCS will be sprayed onto the top of the remaining water column above the core; however, if the tank water has dropped below or partially below core level, the ECCS water will be sprayed directly onto the core. During this time, the decay heat will be removed by the remaining tank water or by the water spray and the maximum fuel temperature will be reduced rapidly from an elevated operating temperature down to about 200°C and then gradually to 100°C with continued spray cooling.

At the end of spray cooling, natural air convection will be established in the core. During this cooling phase, the temperature of the fuel will rise slowly over several hours to a maximum and then decrease with continued air cooling. The maximum fuel and cladding temperature is controlled by the length of spray cooling and by the natural air cooling. Under the preceding conditions, no fuel cladding will be ruptured.

The detailed components of the emergency core cooling system to be used to maintain fuel temperatures below the design basis limit are described in Chapter 6. Basically the system consists of a quick connect system for coupling to the domestic water supply, sensing devices to indicate the need to initiate emergency cooling water flow, a nozzle to distribute the coolant flow over the core, a chimney mounted above the core structure to provide a sufficient channel length for maintaining sufficient air flow through the core, and a ventilation system to provide air circulation through the reactor room.

It should be noted that in a TRIGA® reactor, loss of reactor coolant water will automatically cause a complete reactor shutdown <u>even without a control rod scram</u>. Experiments with the GA subcritical assembly have indicated that the reactivity worth of the water in the core is on the order of 10% (more than 13 dollars). As a result, were the reactor to be operating during a catastrophic event in which the cooling water were completely lost, the reactor would automatically shutdown (even without insertion of control rods) once the water level dropped a few centimeters below the upper grid plate.

13.2.3.2.2.2 Spray Cooling

A considerable amount of experimental data has been gathered on the efficacy of spray cooling for a system of heated cylindrical rods in bundles. These data indicate that the amount of heat that can be removed by a water spray without the rod's wall temperature exceeding about 100°C is simply the amount of heat that would increase the enthalpy of the sprayed water from its inlet enthalpy to the saturated liquid enthalpy (Reference 13.8). The experiments were conducted for heat fluxes up to about 4 W/cm^2, which is larger than the maximum heating rate in the hottest fuel element during the loss-of-coolant accident. Even if the initial surface temperatures were very high (~900°C) before the spray is initiated, the surface temperature would be very quickly reduced to about 100°C if sufficient water is provided to remove the heat without increasing the coolant temperature to the saturation point (Reference 13.9). The spray flow rate required to cool the fuel to 100°C from 2 MW operation corresponds to 12.3 gpm through the TRIGA® core, including the consideration of peak power in the core.

Measurements have been made to determine the actual flow rate required to fulfill the 12.3 gpm flow requirement through a TRIGA® core (Reference 13.10). These tests indicated that both the nozzle type as well as its location and orientation are important in order to provide the required cooling spray. Results also showed that a total spray flow of 20 gpm from the nozzle, as specified in Reference 13.10, located approximately 2 ft. above the top grid plate will assure that adequate core spray cooling is available to meet the requirements above. Provisions have been established to ensure that sufficient spray cooling water can be supplied to the reactor core when needed from the building domestic water supply.

13.2.3.2.2.3 Air Cooling

The relatively small size (~7500 cu. ft.) of the reactor room can affect the convective air cooling of the reactor core after spray cooling ceases. In the small reactor room, hot air from the core is expected to overload the air conditioning system and raise the ambient air temperature. Since this is the air that is available for cooling the core, this situation was analyzed in detail.

The air flow in the reactor room during normal operation is the following: an exhaust flow of 800 cfm passes through absolute filters on the way to the stack, 500 cfm of which comes from the air conditioning system (1100 cfm outgoing, 1600 cfm returned) and 300 cfm comes from leaks into the reactor room from around doors or other leaks in the reactor room enclosure. Appendix D provides schematics of the reactor room and the exhaust and supply air ducts.

Although 1100 cfm is withdrawn from the room by the HVAC, and is refrigerated, and returned with an additional 500 cfm of air at ambient temperature, it will be assumed that during the LOCA event, this air flow continues but that the refrigeration fails due to an excessive heat load. (Note: If the HVAC fails, the reactor room exhaust fan will still be able to draw at least 500 cfm of ambient air in through the open HVAC damper.) Thus, 500 cfm (from the air conditioning) plus 300 cfm (from in-leakage into the reactor room) are continuously supplied to the reactor room at an ambient air temperature (~80°F) to match the 800 cfm exhaust that continues during the accident. To ensure a continuous air supply to and from the reactor room a backup power supply has been provided for the reactor room exhaust fan (EF-1).

13.2.3.2.2.4 Assumptions Made for ECCS Operation

The following assumptions are necessary to initiate and evaluate ECCS operation:

1. The ECCS will be initiated by the reactor operator if the water level drops to a level that requires the system to be turned on. Operator action and manual operation of the ECCS is considered sufficient since at least 20 minutes is available for initiation after an instantaneous loss of the tank water during operations at peak power before sufficient heat will build up in the fuel to threaten the safety limit (Reference 13.11);

2. If the reactor room continuous air monitor (CAM) actuates the recirculation mode of ventilation for the reactor room due to elevated radiation levels following tank water loss, the reactor operator will assess the situation and then switch the room ventilation from recirculation back to the manual ventilation mode (Chapter 9);

3. Based on assumption number 2, the reactor room exhaust fan will continue to extract 800 cfm from the reactor room (typically 500 cfm from the top of the reactor and 300 cfm from near the ceiling).

13.2.3.2.2.5 Performance of the ECCS

Because of the relatively small reactor room, it is necessary to consider for any air cooling portion of the loss-of-coolant accident that the initial conditions consist of an air filled reactor tank containing a hot core near its bottom and surmounted by a small reactor room (7500 cu. ft.). Hot air rises (~227 cfm) from the core in a plume, part of which is removed into the 500 cfm exhaust duct at the top of the reactor tank. The remainder of the hot air plume rises into the reactor room, mixing with the room air. Near the top of the reactor room 300 cfm of mixed air is exhausted. Ambient air at 80°F comes into the reactor room at 800 cfm.

At quasi equilibrium, the mixed air in the reactor room, including that near the top of the reactor tank, is warmer than the 80°F ambient air from the outside. This mixed air flows in a near annulus down the reactor tank adjacent to the tank wall as the hot plume from the reactor core flows upward in the center of the tank. The downflow air partially mixes with the hot air plume rising from the core and increases in temperature. This downflow air then enters the bottom of the reactor core.

An estimate of this air mixing using boundary layer analysis for the mixing region indicates that the temperature increase of the downflow air is approximately 10% of the difference between the downflow air temperature and the upflow average plume temperature.

13.2.3.2.2.6 Thermal Model for Natural Convection Air Cooling

A thermal model was constructed to assess the fuel temperatures for the LOCA event after the termination of spray cooling and with subsequent natural convection air flow through the core. The TAC2D general purpose thermal analysis code was used to calculate the maximum and average fuel temperatures for typical fueled channels representing hot, average and cold regions of the core (Reference 13.12).

Four flow channels were used to represent the natural convective flow past these three (3) fueled regions and one (1) unfueled region. No cross flow was considered between the various flow channels. One flow channel represented all the flow channels in the cooler F and G rings and one flow channel represented all the flow channels in the average powered D and E rings. Individual flow channels were modeled to represent the locally different flow channels surrounding the hottest fuel element. To complete the surface boundary conditions for these latter two flow channels, it was necessary to include in the thermal model a fifth flow channel. This channel was used to represent the temperature response of two adjacent unheated graphite elements in the C-ring and the adjacent central in-core experiment facility.

Decay heat is removed from the reactor by radial conduction to the surface of the fuel elements where it is removed by convective air currents driven by buoyant forces generated by the reactor natural convection loop. The resulting peak and average fuel temperatures were calculated for the hottest element as a function of time. The natural convection flow rate is dependent on the pressure balance in the system. The buoyancy driving head for the natural convection flow is the difference between the density head of the cooler downflow and the density head of the hot upflow. The subsequent analysis shows that a chimney two (2) feet high provides adequate buoyant driving head.

13.2.3.2.2.7 Reactor Core for LOCA

The 20E Core with the central experiment facility containing the aluminum and graphite plugs in place, all control rods fully up, and 101 fuel elements was chosen as the LOCA core configuration (Chapter 4). The axial power distribution with rearrangement required for the TAC2D code, was used for power density calculations (Reference 13.13). The results are shown in Figure 13.2.

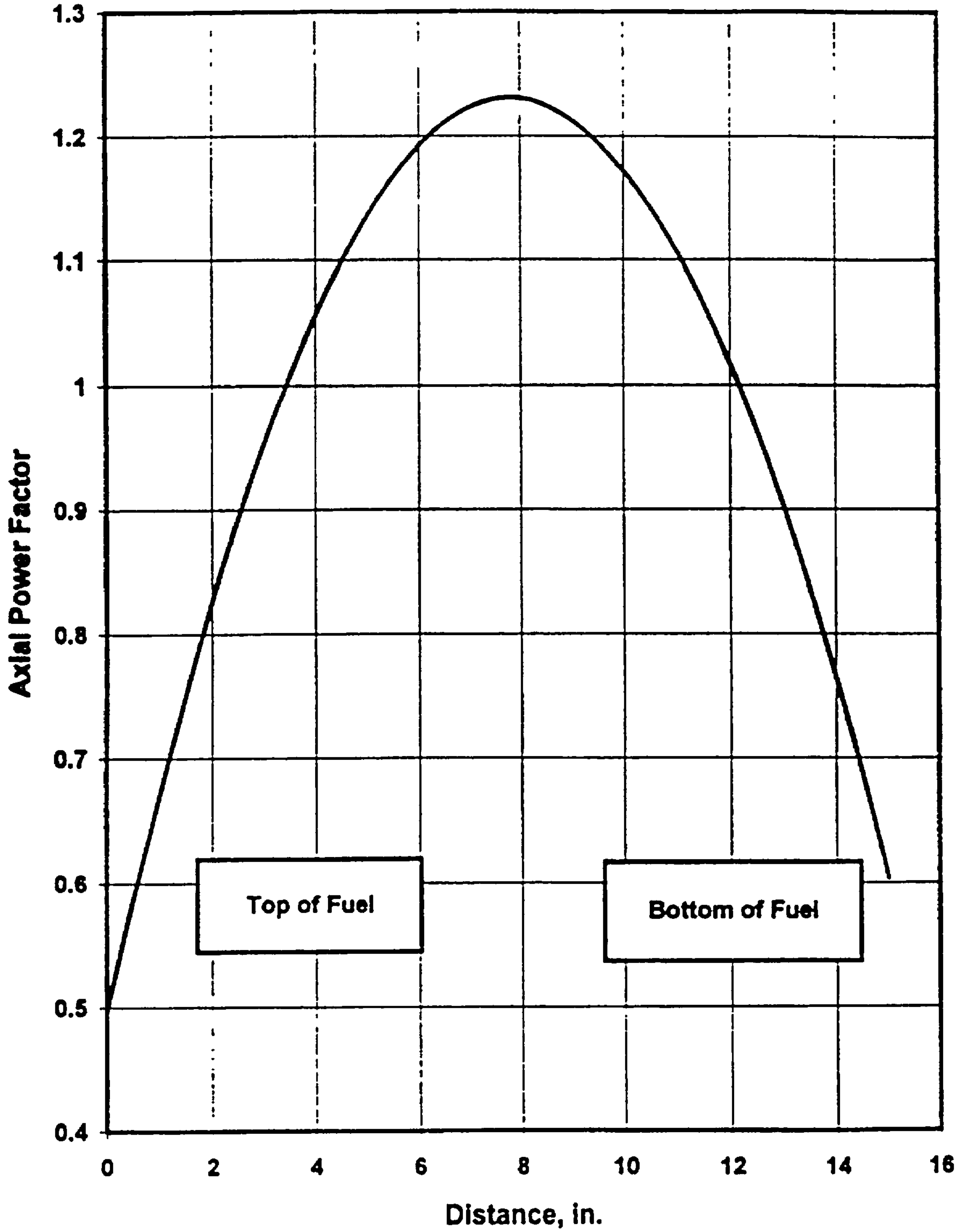
Account was taken of the five fuel followers on the control rods. For the LOCA event, the control rods are fully inserted into the core. This means that the five fuel followers are suspended below the bottom grid plate. Each of these fueled sections is located within a guide tube that has 12 openings in the surface. The 12 individual openings provide 24 in^2 of surface area and are situated symmetrically around the device to provide adequate cooling air for each element.

13.2.3.2.2.8 Mixed Air Temperature in the Reactor Room

For the design case (3.7 hours of spray cooling, 2-ft chimney) the highest average temperature in the hot air plume from the core is approximately 1360°F. At this value the plume density is very low and, consequently, the mass flow rate is low relative to the other air streams in the room. The volume of air flow in this plume is 227 cfm. It is assumed, on average, that about 100 cfm of this plume is swept into the 500 cfm duct at the top of the reactor tank and that about 127 cfm of hot air rises into the reactor room. (See Appendix D for details.) It is assumed that 800 cfm of air at 80°F is continuously supplied to the reactor room and that the duct near the top of the reactor room exhausts 300 cfm of mixed air. It is further assumed that the reactor room is small enough and the air cooling time is long enough (several hours) that a quasi steady state condition exists. That is, the temperature of the mixed air in the reactor room is simply the mixed mean temperature of the plume from the core and the incoming air streams on a mass flow basis, and further that constant specific heat and ideal gas behavior for the air streams can be assumed.

For 80°F air inlet to the room and an average plume temperature of 1364°F, the mean temperature of the mixed air is about 138°F. Even if all the hot air in the plume were to rise into the reactor room rather than a portion being drawn off at the top of the reactor tank, the mixed air temperature would only rise to 180°F, a value that does not alter significantly the cooling conditions of the reactor fuel.

In addition to the above consideration of the mixed mean air temperature in the reactor room, there is the additional consideration that the 800 cfm rate of room air exhaust will provide about 6.6 changes of room air per hour. During the two hours during which the peak fuel temperature exceeds 900°C and the average plume temperature exceeds 1290°F, the ventilation system changes the reactor room air more than 12 times while bringing into the room 80°F air at 800 cfm. This fact provides additional rationale for a quasi equilibrium condition with mixed room air at relatively low temperature.



AXIAL POWER DISTRIBUTION FOR FUEL IN CORE 20E
FIGURE 13.2

The calculation of the mixed air temperature in the reactor room is conservative. It is assumed that the hot air plume has its maximum temperature even at the start of the air cooling cycle. Actually, the plume temperature starts at 212°F, reaches 580°F in a half hour and is below 1000°F for the first hour of cooling. Under these circumstances, the reactor fuel would be cooled more efficiently since the inlet air at the bottom of the core would be lower. However, to be conservative, it is assumed that the hot air plume has its maximum temperature during the entire air cooling cycle.

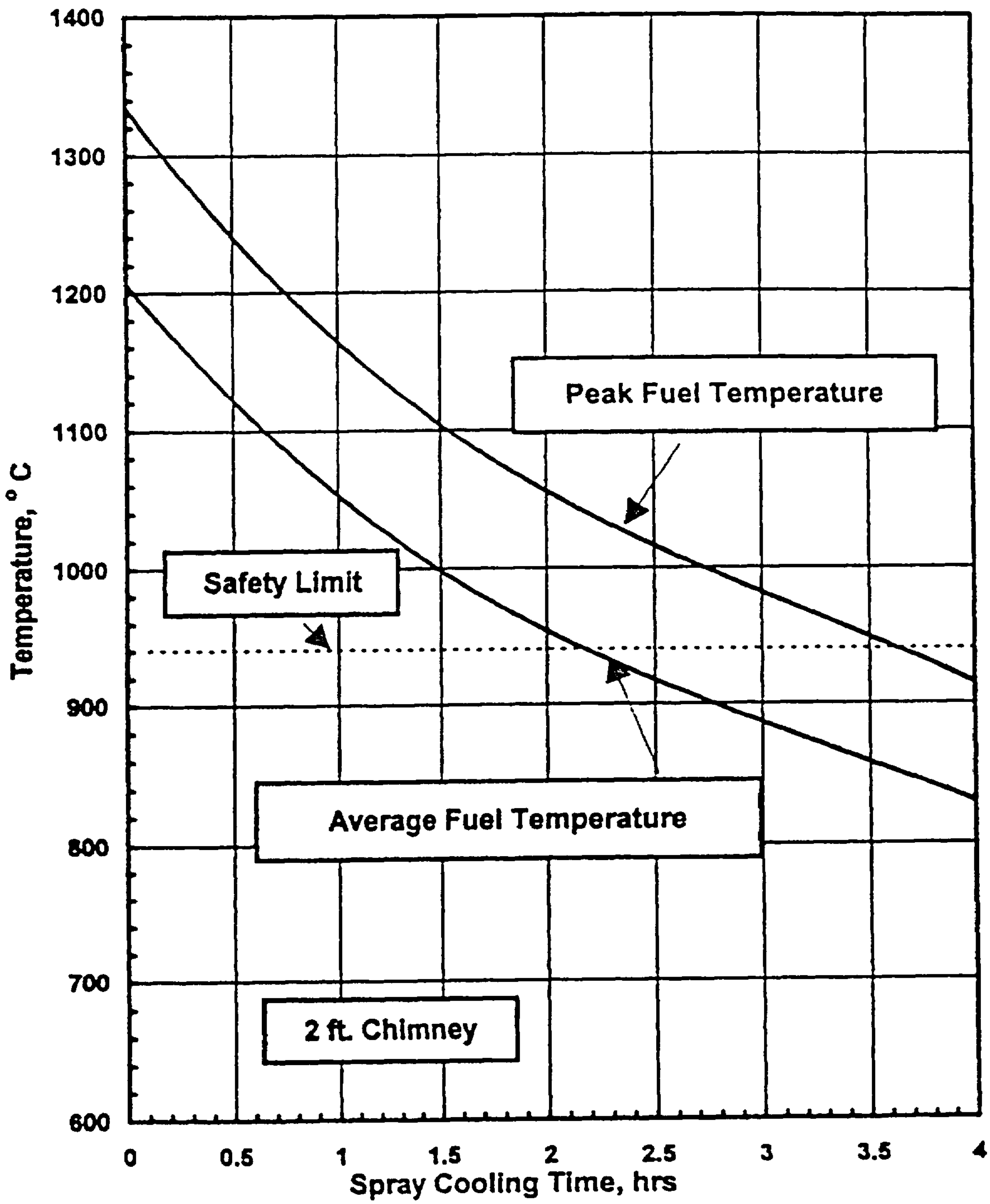
13.2.3.2.2.9 Results of ECCS Calculations

Although it is recognized that the ECCS system when hooked to the domestic water supply should be able to deliver an infinite supply of water, should the domestic water supply not be available, the ECCS function will be supplied by the backup system, the auxiliary make-up water system (AMUWS). Since this system has a limited water supply, considerations of a finite water supply with transition to air cooling were utilized in this calculation.

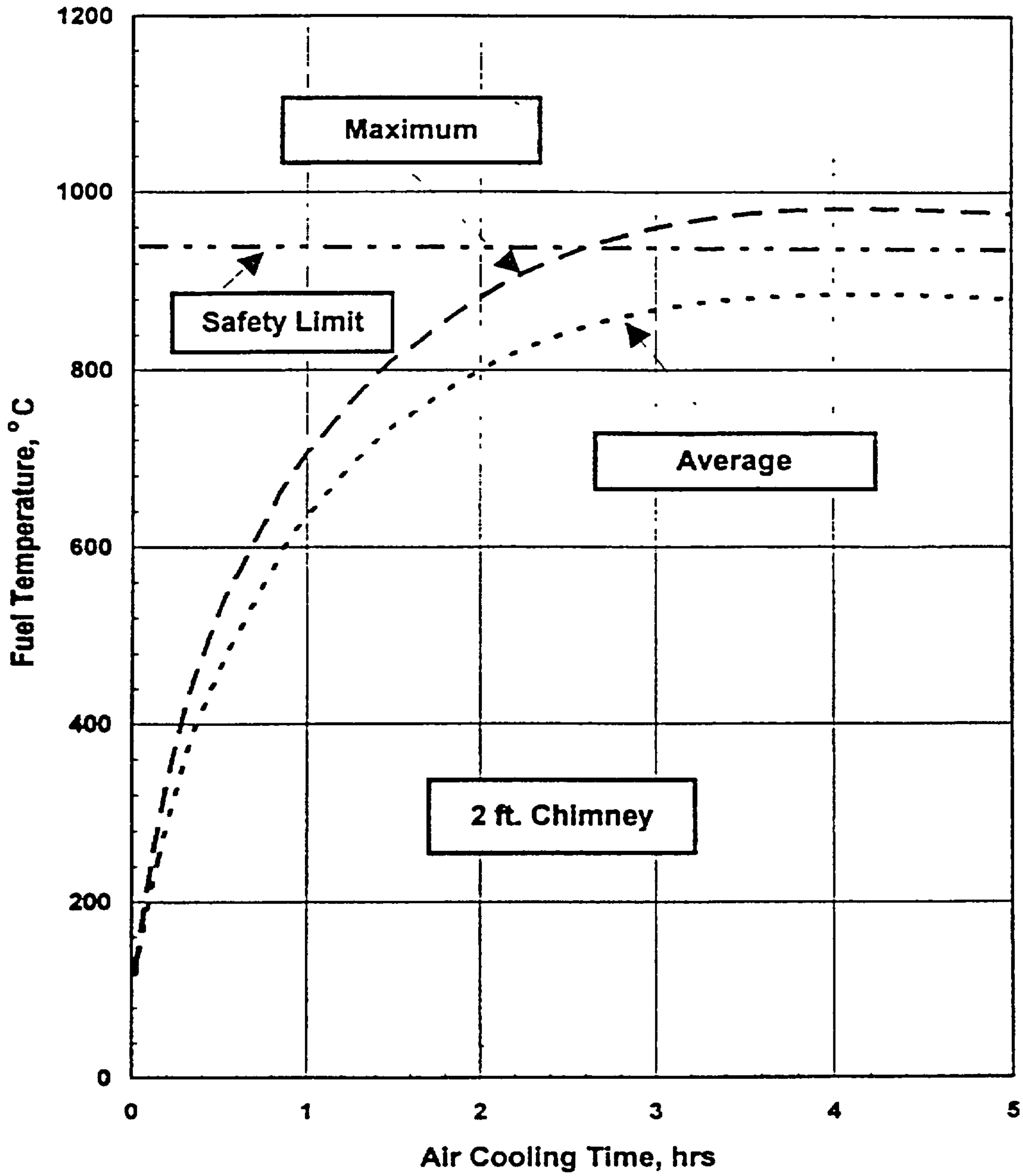
Using the preceding assumptions for the reactor core and for the temperature of the cooling air available in the reactor room, the TAC2D code was used to evaluate the cooling requirements in order to maintain fuel temperatures below safety limits. Figure 13.3 presents the peak and average fuel temperatures in the hottest fuel element during the air cooling cycle after spray cooling for time varying from zero to four hours (with a chimney height of two feet). From Figure 13.3 it may be noted that spray cooling for three hours will lower the resulting average temperature in the hottest fuel element to 886°C, well below the safety limit of 930°C. From the discussion in the following sections, it will become clear that to maintain cladding integrity it is really only necessary for the <u>average</u> temperature to be below the safety limit, since the colder sections of the fuel will act as a sink for any free hydrogen released from the hotter sections. Figure 13.3 also illustrates that with a two foot chimney and slightly more than 3.5 hours of spray cooling, the peak fuel temperature in the hottest fuel element will not exceed the safety limit of 930°C.

Figure 13.4 demonstrates the time dependent fuel temperatures (peak to average) during the air cooling cycle with a two foot chimney after spray cooling for three hours. This graph shows that the peak and average fuel temperatures reach a maximum at about 4.1 hours. While not shown in this graph, it is also clear that the six inches of the fuel in the hottest fuel element which is cooler than the average fuel temperature (886°C) has temperatures that are far below the applicable safety limit. The UCD/MNRC has elected to spray cool the fuel for at least 3.7 hours. The resulting UCD/MNRC fuel temperatures during the air cooling cycle with a 2 foot chimney are presented in Table 13-4, and the resulting cladding stresses are presented in Section 13.2.3.2.2.10.

TAC2D calculations were also made to illustrate the effect of chimney height on maximum and average fuel temperatures assuming three hours of spray cooling (or more in the absence of any chimney). These results are shown in Table 13-4. Table 13-5 shows the spray cooling requirements if the chimney height were three feet rather than two feet. As expected, the performance of a three foot chimney is better than that for a two foot chimney. However, the specifications for the location of the spray cooling nozzle suggests that the nozzle should



MAXIMUM AND AVERAGE FUEL TEMPERATURE DURING AIR COOLING CYCLE FOR VARIOUS SPRAY COOLING TIMES
FIGURE 13.3



MAXIMUM AND AVERAGE FUEL TEMPERATURES
FOR THE HOTTEST FUEL ELEMENT AS FUNCTIONS OF TIME AFTER
END OF SPRAY COOLING FOR THREE HOURS
FIGURE 13.4

T (hr) spray cooling	Chimney (ft)	T_{peak} (°C)	T_{avg} (°C)
3	3	900	813
3	2	982	886
(UCD/MNRC) 3.7	2	930*	845*
3	1	>1220**	>1112**
3	0	>1451**	>1355**
10	0	>1209**	>1122**
48	0	953***	873***

** fuel temperatures taken from Figure 13.3*
*** temperature still rising after 5 hours of air cooling*
**** temperature peaks at about 13 hours and then decreases*

Table 13-4 Maximum Fuel Temperatures with Various Chimney Heights

T(hr) spray cooling	Chimney (ft)	T_{peak} (°C)	T_{avg} (°C)
2	3	941	850
3	3	900	813
3	2	982	886

Table 13-5 Comparison of Cooling Results with 2 ft. and 3 ft. Chimneys be 26 inches above the top of the core (Reference 13.10). This height is compatible with a two foot chimney, but could be a problem with a three foot chimney.

Finally, additional elements of conservatism not mentioned earlier in this analysis provide further assurance that the cooling of the fuel will be at least as effective as described above. For instance, no account was taken of the added cooling provided by the conduction of heat from the fuel elements to the cooler portions of the fuel assembly. Similarly, the radiation of heat (especially when the fuel temperatures have reached the higher values) to the cooler parts of the system outside the core was not included in the transient heat flow considerations. Furthermore, the TAC2D calculations reported herein assumed that the temperature of the inlet air at the bottom of the core was 300°F. This is somewhat higher than would result from the considerations in Sections 13.2.3.2.2.6 and 13.2.3.2.2.8. With those earlier results, the core inlet air temperature would be about 260°F (138°F [mixed air in the reactor room] plus 122°F [ΔT from additional mixing in the tank]). If all the hot plume were to rise into the reactor room, the mixed air temperature in the room would be about 180°F. In this unlikely case, the core inlet air temperature would then be about 298°F (180°F [mixed air in the reactor room] plus 118°F [ΔT from additional mixing in the tank]). Both of these estimates of the core inlet air temperature are less than the more conservative value of 300°F used for the TAC2D calculations performed here.

13.2.3.2.2.10 Cladding Stress Analysis

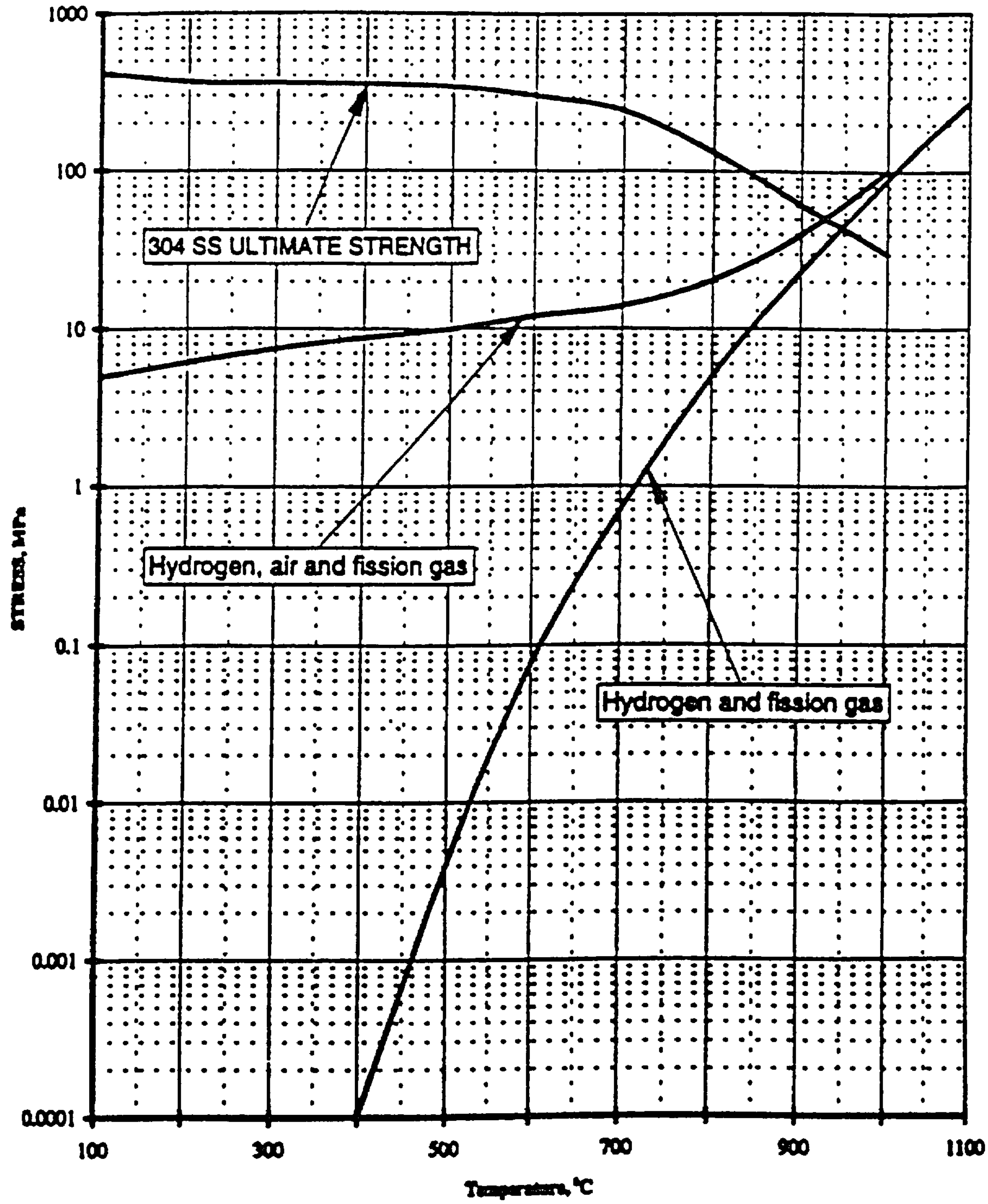
In Figures 13.3 and 13.4, it is shown that spray cooling for only three hours with a two foot chimney will assure that the average fuel temperature in the hottest fuel element will not exceed 886°C, although the corresponding peak fuel temperature will reach 982°C. Figure 13.5 presents clad strength and applied stress from equilibrium hydrogen dissociation pressure plus any other gas present within the clad as a function of fuel temperature. Early in the fuel life, there is residual air backfilling but relatively little fission gas. Both the nitrogen and oxygen form metal compounds after the fuel has been operated at full power for a period of time. Early in the effective fuel life, the air disappears as a gas leaving only hydrogen and fission gas.

For Figure 13.5 to be valid, all the fuel within the clad must have the same temperature and be at the same temperature as the clad. In this case, the Safety Limit is the crossover of the Clad Strength Curve and Gas Pressure Curve. Except for a time duration very early in the fuel element life before the air has been absorbed, the Safety Limit is about 930°C. In a real fuel element during a LOCA much of the fuel has a temperature lower than the peak temperature. For this case, the excess hydrogen gas from the hotter portions of the fuel element will disappear into the sink created by the cooler portions of the curve.

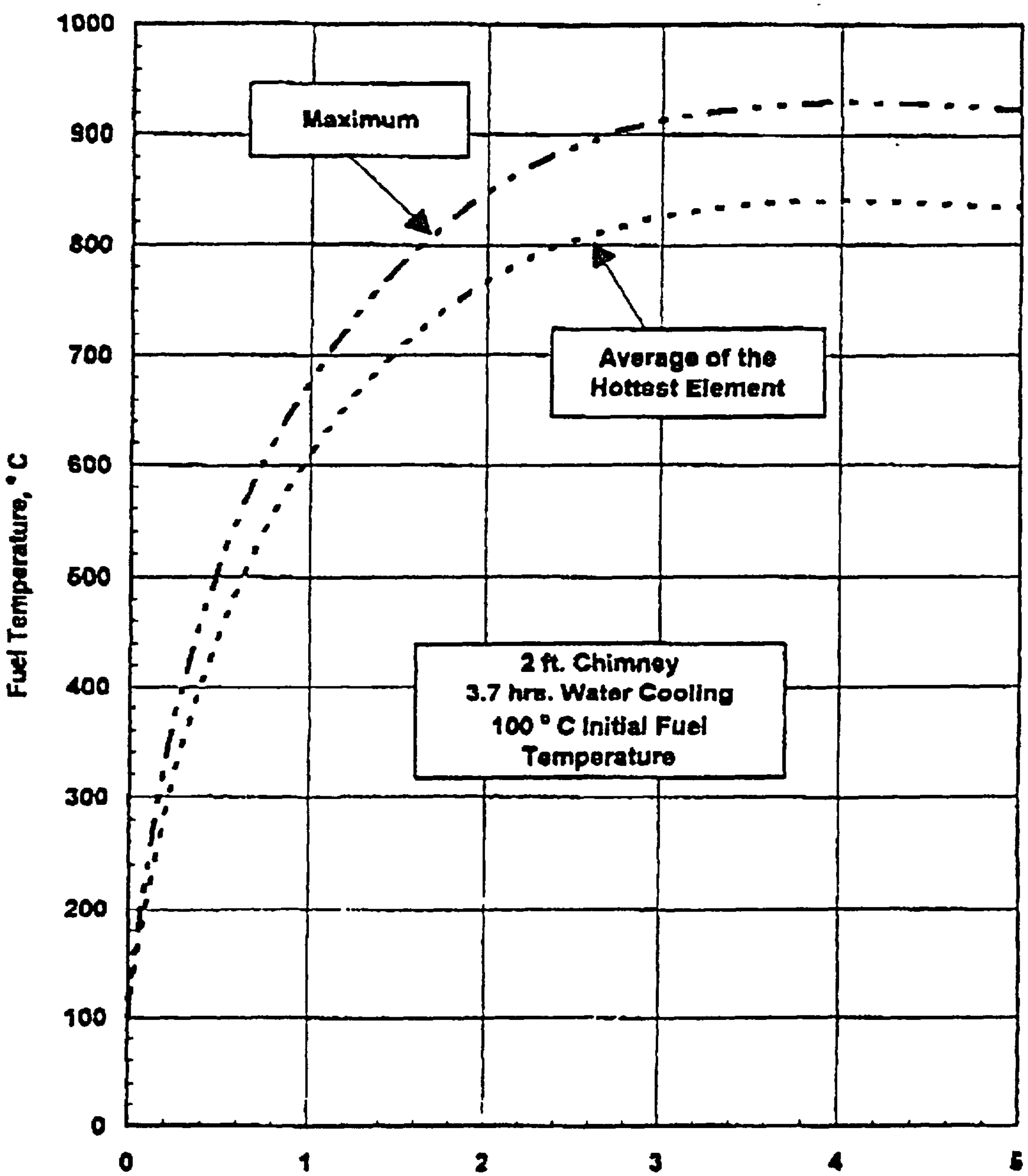
In the current example using three hours of spray cooling and a two foot chimney, the hottest fuel element ranges in temperature from 603°C at the bottom to a peak temperature of 982°C near the top. The average fuel temperature is 886°C with the bottom six (6) inches of the 15-inch fuel having fuel temperatures considerably less than the average fuel temperature. The clad temperature is a few degrees (6-8°C) cooler than the adjacent fuel temperature. There is thus a small area along the clad in which the clad temperature reaches 975°C. The curve in Figure 13.5 shows that the resulting clad strength in this region of the clad is about 35 Mpa. The peak temperature of the fuel slowly rises (over a 4-hour period) from 100°C to 980°C. Consequently, the excess hydrogen gas due to dissociation has time to be absorbed in the cooler fuel sections without raising the pressure substantially above that characteristic of the cooler section (603°C-700°C) of the fuel element. The resulting gas pressure will be less than 0.8 Mpa (700°C) which is considerably less than the 35 Mpa strength of the hot clad. There is thus no danger of clad rupture during the air cooling portion of the LOCA scenario when the fuel is previously spray cooled for only three hours. However, as a further conservatism, the UCD/MNRC will spray cool the fuel for a least 3.7 hours and will therefore experience an average fuel temperature in the hottest element of approximately 845°C and a maximum fuel temperature of 930°C during the air cooling phase (Figure 13.6). Since these temperatures are lower than those used in the preceding example, it is clear that at the UCD/MNRC there is even less danger of clad rupture during the air cooling portion of a LOCA

13.2.3.2.2.11 Ground Water Contamination

As a result of activation of impurities in the primary cooling water, the water will contain small amounts of radionuclides depending on reactor power, reactor operating time and time since reactor shutdown. To characterize the radioactivity expected to be present in the UCD/MNRC primary coolant at 2 MW, measured values for the predominant radionuclides



CLAD STRENGTH AND APPLIED STRESS RESULTING FROM EQUILIBRIUM HYDROGEN DISSOCIATION PRESSURE AS A FUNCTION OF TEMPERATURE
FIGURE 13.5



MAXIMUM AND AVERAGE FUEL TEMPERATURES
FOR THE HOTTEST FUEL ELEMENT AS FUNCTIONS OF TIME AFTER
END OF SPRAY COOLING FOR 3.7 HOURS
FIGURE 13.6

were adjusted to reflect estimated equilibrium concentrations at 2 MW (Table 11-4, Section 11.1.1.2.1). Next, a calculation was made to determine the length of time for the lost coolant to reach ground water.

The relationship to determine the time (t) for water to move from a point under the reactor tank a distance, D, to ground water is:

$$t=D/(K \times I);$$

where:

t = penetration time (sec.);
D = depth of penetration with time (ft);
I = hydraulic gradient = 1.0;
K = hydraulic conductivity = 4.57×10^{-4} ft/sec (Reference 13.14).

If it is assumed that the ground water is 80 feet below the UCD/MNRC site, it would require more than 36 hours for it to be reached if the reactor tank containment were breached. The radionuclide concentrations present in the reactor tank water upon reaching the ground water were then calculated utilizing a 36 hour delay time. These values are presented in Table 13-6. As shown, Aluminum-28, Magnesium-27, and Nitrogen-16 are gone by the time the tank water reaches the ground water, and most of the other radionuclides will have undergone some degree of decay during the first 36 hours. Decay will, of course, vary depending on the radionuclide, but Argon-41 activity would fall to about 6×10^{-12} μCi/ml during the first 36 hours. Because of its low solubility in water, argon has no limiting water concentration under 10 CFR Part 20. However, this concentration level is well below the 10 CFR Part 20 air concentration limit for the unrestricted area. Since Argon-41 is only a concern from a dose standpoint when an individual is immersed in an Argon-41 cloud, and since the concentration in this situation is well below the air or cloud limit for the unrestricted area, Argon-41 is not a problem in the ground water.

The concentration of Manganese-56 in the reactor primary water will be about 4.7×10^{-4} μCi/ml. This means that at initial release the Manganese-56 concentration is 6.7 times higher than the 7×10^{-5} μCi/ml unrestricted area concentration limit in 10 CFR Part 20. However, as shown in Table 13-6, the Manganese-56 concentration is far below the 10 CFR Part 20 limit by the time it reaches ground water.

The estimated Hydrogen-3 (tritium) level is dependent upon how long the reactor has operated since initial startup and how much non-radioactive makeup water has been added prior to the LOCA. As shown in Table 11-4, after 20 years of operation at 2 MW with no addition of clean makeup, water the tritium concentration may reach 1.3×10^{-2} μCi/ml, but this is definitely an upper limit estimate and a concentration closer to 1.0×10^{-3} μCi/ml (the 10 CFR 20 concentration limit) is expected for at least the first several years. However, the tritium concentration in the water when it is released will be largely unchanged when and if the tank water reaches the ground water. Even so, the potential tritium dose to members of the general public who might consume the ground water will still be low because this accident will be a one time event with a limited duration of release, and because only a limited

Radionuclide	Half Life	Equilibrium Concentration at 2 MW (μCi/ml)	Concentration Reaching Ground Water (μCi/ml)
Aluminum-28	2.3 min	6.0×10^{-3}	0
Argon-41	1.8 hr	3.0×10^{-3}	6.17×10^{-12}
Hydrogen-3	12 yr	1.0×10^{-3} to 1.3×10^{-2}	1.0×10^{-3} to 1.3×10^{-2}
Magnesium-27	9.46 min	4.0×10^{-4}	0
Manganese-56	2.58 hr	4.7×10^{-4}	4.09×10^{-10}
Nitrogen-16	7.14 sec	131	0
Sodium-24	14.96 hr	2.6×10^{-3}	5.00×10^{-4}

Table 13-6 Predominant Radionuclides in Primary Coolant at Equilibrium and Upon Reaching Ground Water

amount of the 7,000 gallons of water potentially released from the reactor tank will likely escape from the radiography bays in the facility. There will obviously also be a reduction in the tritium concentration when the reactor tank water mixes with the ground water, and normally, chemical processes take place as water percolates through soil which result in partial removal of many radionuclides. While these processes are usually not as significant for tritium as they are for many other radionuclides, some small reduction in tritium concentration may occur.

The potential release of tritium between 1.0×10^{-3} μCi/ml and 1.3×10^{-2} μCi/ml also assumes the tritium concentration in the primary water reaches the predicted 2 MW levels. This may or may not occur and will definitely not occur rapidly. The Hydrogen-3 concentration will gradually build up as it is produced. Periodic monitoring of the primary coolant for this radionuclide (semi annually until the trend stabilizes) will allow continuous long-term assessment of the Hydrogen-3 concentration and its relation to 10 CFR Part 20 limits.

At the time the reactor tank water reaches the ground water, the Sodium-24 concentration will meet the 10 CFR Part 20 release limit for discharge into a sewer system, but will exceed the 10 CFR Part 20 effluent release concentration. However, after just 2.1 days of decay, the concentration of Sodium-24 in the ground water (ignoring dilution) will be within the NRC effluent concentration limit in 10 CFR Part 20. In addition, the Sodium-24 ground water concentration will continue to drop due to the continued rapid decay of this radionuclide. Therefore, Sodium-24 does not represent a significant source of potential radiation exposure to the general public.

13.2.3.2.2.12 Radiation Levels from the Uncovered Core

Even though there is a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered radiation doses from an uncovered core following 2 MW operations were calculated in Appendix B and are summarized here. Direct radiation doses were calculated for a person standing on the grating directly above the reactor core. The core, shut down and drained of water, was treated as a bare cylindrical uniform source of 1 MeV photons. No accounting was made of sources other than fission product decay gammas, and no credit was taken for gamma attenuation through the fuel element end pieces and the upper grid plate. The first of these assumptions is optimistic, the second conservative, and the net effect is conservative. The results are given in Table 13-7 and agree with results for the 2 MW Torrey Pines TRIGA® Reactor (Reference B.18).

Table 13-7 Dose Rates on the UCD/MNRC Reactor Top After a Loss of Pool Water Accident Following 2 MW Operations

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	3.64×10^4
1 hour	3.77×10^3
1 day	1.69×10^3
1 week	8.96×10^2
1 month	4.70×10^2

A second calculation was made to determine the dose rate to a person in the reactor room who is not in the direct beam from the exposed core but is still subject to scattered radiation from the reactor room ceiling. The dose point was chosen to be three feet above the reactor room floor at a distance of six feet away from the edge of the reactor tank. This is the furthest distance a person can get from the edge of the tank and still remain in the reactor room. The ceiling of the reactor room is about twenty four feet from the reactor top and is assumed to be a thick concrete slab. The concrete slab assumption gives the worst case scattering, but it should be carefully noted that the roof over the reactor is only corrugated metal and not a thick concrete slab. Therefore, in reality the scattering will not be as great as calculated because the radiation from the unshielded core will be collimated upward by the shield structure and will undergo minimal interaction with the roof, greatly reducing the actual dose rates away from the edge of the tank. The results of the calculated dose rates due to scatter in the reactor room are found in Table 13-8. These dose rates show that personnel could occupy areas within the reactor room shortly after the accident for a sufficient period of time to undertake mitigating actions without exceeding NRC occupational dose limits.

Table 13-8 Scattered Radiation Dose Rates in the UCD/MNRC Reactor Room After a Loss of Pool Water Accident Following 2 MW Operations

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	9.640
1 hour	1.000
1 day	0.449
1 week	0.238
1 month	0.124

A final calculation was carried out to estimate the dose rates to a person at the UCD/MNRC facility fence due to scattered radiation from the reactor room ceiling. The dose point was chosen to be three feet above the ground at the facility fence. This is the closest point a member of the public would be able to occupy. The calculated dose rates are presented in Table 13-9, but once again are overestimates because scatter off of the reactor room ceiling will be much less than assumed.

Table 13-9 Scattered Radiation Dose Rates at the UCD/MNRC Facility Fence After a Loss of Pool Water Accident Following 2 MW Operations

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	0.460
1 hour	0.047
1 day	0.021
1 week	0.011
1 month	0.006

13.2.4 Loss of Coolant Flow

13.2.4.1 Accident Initiating Events and Scenarios

Loss of coolant flow could occur due to failure of a key component in the reactor primary or secondary cooling system (e.g. a pump), loss of electrical power, or blockage of a coolant flow channel. Operator error could also cause loss of coolant flow.

Scenarios for loss of coolant flow events during operation are difficult to imagine since the bulk water temperature adiabatically increases at a rate of about 1.1 °C/min at a power level of 2 MW. Under these conditions, the operator has ample time to reduce the power and place the heat-removal system into operation before any abnormal temperature is reached in the reactor water. A core inlet temperature alarm at 35 °C and primary and secondary low flow alarms will alert the operator to an abnormal condition and should allow for corrective action prior to reaching the bulk water temperature limit.

13.2.4.2 Accident Analysis and Determination of Consequences

13.2.4.2.1 Loss of Coolant Flow Without Immediate Operator Action

If the reactor were operated without coolant flow for an extended period of time (and there was no heat removal by reactor coolant systems), voiding of the water in the core would occur and the water level in the tank would decrease because of evaporation. The sequence of events postulated for this very unlikely condition is as follows:

(a) The reactor would continue to operate at a power of 2 MW (provided that the rods were adjusted to maintain power) and would heat the tank water at a rate of about 1.1 °C/min until the water entering the core approached the saturation temperature (this would take 60 minutes, assuming an initial temperature near 35 °C and adiabatic conditions). At this time, voids in the core would cause power oscillations and the negative void coefficient of reactivity would cause a reduction in power if control rods were not adjusted to maintain power;

(b) If it is assumed that the operator or automatic control system maintained power at 2 MW, about 3180 kg/hr of water would be vaporized (assuming that the system is adiabatic except for the evaporation process), and the water level would decrease. It would take about 9 hours to heat and vaporize the entire tank at this rate. In fact, the reactor would shut down as the water level passed the top of the fuel.

It is considered inconceivable that such an operating condition would go undetected. Water level, water flow, and water temperature alarms would certainly alert the operator. Also, as the water level lowers, the reactor room radiation monitors will alarm. Because of all of these factors, water should be added to the tank to mitigate the problem.

13.2.5 Mishandling or Malfunction of Fuel

13.2.5.1 Accident Initiating Events and Scenarios

Events which could cause accidents in this category at the UCD/MNRC reactor include 1) fuel handling accidents where an element is dropped underwater and damaged severely enough to breach the cladding, 2) simple failure of the fuel cladding due to a manufacturing defect or corrosion, and 3) overheating of fuel with subsequent cladding failure during steady state operations or pulsing; overheating might occur due to incorrect loading of fuel elements with different ^{235}U enrichments in a mixed core.

13.2.5.2 Accident Analysis and Determination of Consequences

13.2.5.2.1 Single Element Failure in Water

At some point in the lifetime of the UCD/MNRC reactor, used fuel within the core will be moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is shut down. The most serious fuel-handling accident involves spent or used fuel that has been removed from the core and then dropped or otherwise damaged, causing a breach of the fuel element cladding and a release of fission products. As noted previously, the standard or accepted maximum hypothetical accident for TRIGA® reactors involves failure of the cladding of a single fuel element after extended reactor operations, followed by instantaneous release of the fission products directly into the air of the reactor room. A less severe, but more credible accident involving a single element cladding failure assumes the failure occurs underwater in the reactor tank 48 hours after reactor shutdown (i.e., 48 hours of decay has occurred). This accident has been analyzed in Appendix B and results in much lower doses to the public and the reactor staff than those estimated for the MHA.

Assumptions used for assessing the consequences of the single element failure in water are almost exactly the same as those used for the MHA, except for the presence of pool water which contains most of the halogens and thereby reduces the halogen dose contribution. The fission product release fraction to the cladding gap remains at 7.7×10^{-5} and the halogen release fraction from the fuel-cladding gap is still a very conservative 0.5. However, for the single element failure in water there are two assumptions which differ from the MHA. First, the fuel is assumed to have decayed for 48 hours prior to the accident and secondly it is assumed that most of the halogens released from the cladding gap remain in the water and are removed by the demineralizer. However, a small fraction, approximately 2.5% of the total halogens released to the cladding gap are, in this case, assumed to escape from the reactor tank water into the reactor room air, which is more conservative than assuming total (100%) solubility of the halogens as is sometimes done for TRIGA® reactors (Reference B.18). However, even assuming a 2.5% halogen release from the pool water will almost certainly result in an overestimate of the actual radioiodine activity released into the room because of the use of a 50% halogen gap release fraction rather than the 1.7% documented in References B.7 and B.8. In addition, about 50% of the airborne halogens released from the pool water are expected to plate out in the reactor building before reaching the outside environment. See

References B.4 and B.5. The experience at TMI-2, along with recent experiments, indicate that the 50% halogen release fraction is much too large. Smaller releases, possibly as little as 0.6% of the iodine reaching the cladding gap may be released into the reactor room air due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble than elemental iodine (Reference B.8).

As with the MHA, radiological consequence calculations were done using the Radiological Safety Analysis Computer Program (RSAC-5) version 5.2, 02/22/94 (Reference B.9). RSAC-5 calculates consequences of the release of radionuclides to the atmosphere and can generate a fission product inventory; decay and ingrow the inventory with time; model the downwind dispersion of the activity; and calculate doses to downwind individuals.

For this accident at the UCD/MNRC, dispersion coefficients (χ/Q values) for locations in the unrestricted area, 10 m (the UCD/MNRC perimeter fence line nearest the building which defines the interface of the restricted and unrestricted area) out to 100 m, were input directly into the code and were calculated using Regulatory Guide 1.145 methodology (Reference B.11). Calculations were performed for Pasquill weather classifications A through F. Diffusion coefficients were taken from Reference B.12 and are presented in Table B-2. Calculations were performed assuming a ground level release at an 800 cfm reactor room release rate without any credit for stack height or building wake effects, which would only improve mixing and lower projected doses. Furthermore, it was assumed that all of the fission products were released to the unrestricted area by a single reactor room air change, which would maximize the dose rate to persons exposed to the plume during the accident and minimize the exposure time to receive the highest estimated dose from this accident. These latter assumptions regarding release are very, very conservative since the reactor room is not at ground level and, rather than 10 meters, is closer to 30 meters from the perimeter fence. Furthermore, there are no normal or direct flow pathways to support an 800 cfm ground level flow from the reactor room to the unrestricted area.

The results of the RSAC-5 calculations are shown in Tables B-4 and B-6. Included are doses inside the reactor room (Table B-4) and doses at several locations in the unrestricted area outside the UCD/MNRC (10 to 100 m from the building) as a function of weather class (Table B-6). Results are reported for the Committed Dose Equivalent (CDE) to the thyroid, the Committed Effective Dose Equivalent (CEDE) due to inhalation, the Deep Dose Equivalent (DDE) due to air immersion, and the Total Effective Dose Equivalent (TEDE) resulting from adding the CEDE and the DDE.

As indicated by the results in Table 13-10, the occupational dose to workers who evacuate the reactor room within 5 minutes following the cladding failure of a single fuel element in water should be approximately 32.5 millirem Total Effective Dose Equivalent and 660 millirem Committed Dose Equivalent to the thyroid. If evacuation occurs within 2 minutes, as it no doubt will because the reactor room is small and easy to exit, the doses drop to 13.2 millirem TEDE and 260 millirem CDE. All of these doses are well within the NRC guidelines for occupational exposure as stated in 10 CFR 20.1201.

Accident: Cladding Failure in Water 48 Hours after Reactor Shutdown				
	CDE Thyroid (millirem)	CEDE (millirem)	DDE (millirem)	TEDE (millirem)
2 minute room occupancy	260	13	0.2	13.2
5 minute room occupancy	660	32	0.5	32.5

Table 13-10 Occupational Radiation Doses in the UCD/MNRC Reactor Room Following a Single Element Failure in Water.

Projected doses to the general public in the unrestricted area around the UCD/MNRC following a single element failure in water are shown in Table 13-11. To receive the indicated dose, a person must be exposed to the airborne plume from the reactor room for the entire 9.2 minute period it is being vented. Even using this exposure requirement at the closest distance to the UCD/MNRC building (10 m), and assuming the most unfavorable atmospheric conditions (Category F), the maximum TEDE to a member of the general public would be only 4.7 millirem. Although this accident and the corresponding radiation doses are very unlikely to occur, the maximum estimated dose to a member of the general public of 4.7 millirem is still well within the 100 millirem TEDE limit for the general public published in 10 CFR 20 (Reference 10 CFR 20.1301). It should also be noted that if one assumes a 50% halogen plateout in the reactor room, the TEDE drops to 2.4 millirem.

Distance (Meters)	CDE Thyroid (millirem)	CEDE (millirem)	DDE[1] (millirem)	TEDE (millirem)
10	97	4.7	0.0	4.7
20	76	3.7	0.0	3.7
40	52	2.5	0.0	2.5
80	30	1.4	0.0	1.4
100	24	1.1	0.0	1.1

Table 13-11 Radiation Doses to Members of the General Public Under Different Atmospheric Conditions and at Different Distances from the UCD/MNRC Following a Cladding Failure in Water 48 hours after Reactor Shutdown.

[1] Doses less than 0.1 mrem were entered as zero.

• CDE - Committed Dose Equivalent
• CEDE - Committed Effective Dose Equivalent
• DDE - Deep Dose Equivalent
• TEDE - Total Effective Dose Equivalent

13.2.5.2.2 Fuel Loading Error

Operation of the UCD/MNRC Reactor after a 20/20 fuel element has been loaded into the wrong grid position could result in increased temperatures in surrounding fuel elements. Neutronics calculations were done to identify the worst-case error for use in analyzing this type of accident. It was assumed that no fuel elements can be loaded in Rows A or B of the UCD/MNRC reactor because of the cutout in the upper grid plate. The highest power peaking would result from a fresh 20/20 fuel element being substituted for a graphite dummy element at a Row C flat (even numbered) position. Because of the surrounding 8.5/20 fuel environment, higher element power would be generated if this substitution were made in the mixed-fuel reference core than in the all-20/20 reference core. The worst case is fresh 20/20 fuel replacing the dummy element in position C10 of the MixJ Core loading. The loading error would increase the excess reactivity by $1.51 (18%) and would increase the peak element power by 11.5 kW (42%) to 39.1 kW. Accordingly, the loading error is assumed to result in a peak element power of 40 kW.

The RELAP5 steady-state thermal-hydraulic analysis described in Section 4.6 was repeated with the nominal inlet temperature (32.2 °C) and the peak element power increased to 40 kW (core radial peaking factor increased to 2.0). The peak fuel temperature was 734°C, which is still below the operational safety limit (the LCO) of 750°C. The critical heat flux ratio was 2.6, indicating that there is still ample margin before film boiling. Since the hot channel outlet void fraction was 5% and the core outlet subcooling was 8°C, it appears unlikely that any detectable chugging will occur. Should chugging occur, it will be easily detected and appropriate operational constraints established.

Operation in pulse mode with the maximum allowed reactivity insertion, $1.75, and the above loading error was also considered. The core-average fuel ΔT with this insertion is 161 °C. The four factors used to produce the total peaking factor were:

- Core radial peaking factor of 2.0, based on a peak element power of 40 kW;
- Axial and pin tilt factors of 1.27 and 1.5, respectively, from the worst MixJ Core in Section 4.3.3.7;
- 1.33 pin radial peaking factor, since the erroneously loaded fuel is the 20/20 type.

This leads to a peak fuel temperature of 837 °C, well below the 1100°C pulsing limit. Thus, pulse operation is also predicted to be benign.

13.2.6 Experiment Malfunction

13.2.6.1 Accident Initiating Events and Scenario

Improperly controlled experiments involving the UCD/MNRC reactor could potentially result in damage to the reactor, unnecessary radiation exposure to facility staff and members of the general public, and unnecessary releases of radioactivity into the unrestricted area. Mechanisms for these occurrences include the production of excess amounts of radionuclides with unexpected radiation levels, and creation of unplanned for pressures in irradiated materials which subsequently vent into reactor irradiation facilities or into the reactor building causing damage from the pressure release or an uncontrolled release of radioactivity. Other mechanisms for damage, such as corrosion and large reactivity changes, are also possible.

13.2.6.2 Accident Analysis and Determination of Consequences

Because of the potential for accidents which could damage the reactor if experiments are not properly controlled, there are strict procedural and regulatory requirements addressing experiment review and approval (Chapter 10). These requirements are focused on ensuring that experiments will not fail, but they also incorporate requirements to assure that there is no reactor damage and no radioactivity releases or radiation doses which exceed the limits of 10 CFR Part 20, should failure occur. For example, specific requirements in UCD/MNRC administrative procedures such as the Utilization of the University of California - Davis/McClellan Nuclear Radiation Center Research Reactor Facility (MNRC-0027) (Reference 11.7) establish detailed administrative procedures, technical requirements, and the need for safety reviews for all types of proposed reactor experiments.

Safety related reviews of proposed experiments usually require the performance of specific safety analyses of proposed activities to assess such things as generation of radionuclides and fission products (i.e., radioiodines), and to ensure evaluation of reactivity worth, chemical and physical characteristics of materials under irradiation, corrosive and explosive characteristics of materials, and the need for encapsulation. This process is an important step in ensuring the safety of reactor experiments and has been successfully used for many years at research reactors to help assure the safety of experiments placed in these reactors. Therefore, the process is expected to be an effective measure in assuring experiment safety at the UCD/MNRC reactor.

A specific limitation of less than \$1.00 on the reactivity of individual moveable experiments placed in the reactor tank has been established and is safe because analysis has shown that pulse reactivity insertions of \$1.75 in the 2 MW UCD/MNRC reactor result in fuel temperatures which are well below the fuel temperature safety limit of 930°C (Section 13.2.2). In addition, limiting the worth of each moveable experiment to less than \$1.00 will assure that the additional increase in transient power and temperature will be slow enough so that the fuel temperature scram will be effective. Likewise, an additional reactivity limitation of less than \$1.75 for any single secured experiment and an absolute total reactivity worth of \$1.92, including the potential reactivity which might result from malfunction, flooding or voiding, is safe because Section 13.2.2 shows that a maximum reactivity of \$1.92 can be safely inserted.

Limiting the generation of certain radionuclides in experiments and certain fission products in fueled experiments also helps to assure that occupational radiation doses (as well as doses to the general public) due to postulated experiment failure, with subsequent radionuclide or fission product release, will be within the limits prescribed by 10 CFR 20. A limit of 1.5 curies of I-131 through I-135 for a single fueled experiment and a limit of 40 curies of I-125 in the I-125 production facility glovebox are small compared to the approximately 8,500 curies of I-131 through I-135 which are present in the single fuel element failure analyzed in Section 13.2.1 (failure in air) and Section 13.2.5 (failure in water). In both cases, the occupational doses and the doses to the general public in the unrestricted area due to radioiodine are within 10 CFR 20 limits. Therefore, establishing conservative limits for radioiodine in experiments will result in projected doses well within 10 CFR 20 limits. Strontium-90 in a fueled experiment is limited to 0.005 curies which is far below the 34 curies present in the single fuel element failures mentioned above. Since no dose limits will be exceeded in the single element failure accidents, doses from experiments where the Strontium-90 is limited to 0.005 curies are expected to be safely within 10 CFR 20 limits.

Projected damage to the reactor from experiments involving explosives varies significantly depending on the quantity of explosives being irradiated and where the explosives are placed relative to critical reactor components and safety systems. For example, an explosives limit of 25 milligrams when irradiation is to be in the reactor tank carries the additional restriction that experiment containment must be able to withstand the pressure produced upon detonation. Based on the following discussion, containment of detonation pressure from this small quantity of explosives is possible using conventional materials and methods, and such containment will eliminate potential damage to reactor components or other experiments (Reference 13.15).

A 25 milligram quantity of explosives, upon detonation, releases approximately 25 calories (104.2 joules) of energy with the creation of 25 cm^3 of gas. For the explosive TNT, the density is 1.654 gm/cm^3 so that 25 mg represents a volume of 0.015 cm^3. If the assumption is made that the energy release occurs as an instantaneous change in pressure, the total force on the encapsulation material is the sum of two pressures. For a one cm^3 volume, the energy release of 104.2 joules represents a pressure of 1032 atmospheres. The instantaneous change in pressure due to gas production in the same volume adds another 25 atmospheres. The total pressure within a 1 cm^3 capsule is then 1057 atmospheres for the complete reaction of 25 mg of explosives.

Typical construction materials of capsules are stainless steel, aluminum and polyethylene. Table 13-12 lists the mechanical properties of these encapsulation materials.

Table 13-12 Material Strengths			
Material	Yield	Ultimate	Density
Stainless Steel (type 304)	35 ksi	85 ksi	7.98 g/cm^3 (500 lb/ft^3)
Aluminum (alloy 6061)	40 ksi	45 ksi	2.739 g/cm^3 (171 lb/ft^3)
Polyethylene	1.7 ksi	1.4 ksi	0.923 g/cm^3

Analysis of the encapsulation materials determines the material stress limits that must exist to confine the reactive equivalent of 25 mg of explosives. The stress limit in a cylindrical container with thin walls is one half the pressure times the ratio of the capsule diameter to wall thickness,

$$\sigma_{max} = \frac{pd}{2t} ;$$

where:

σ_{max} = maximum stress in container wall;

p = total pressure within the container;

d = diameter of the container;

t = wall thickness.

When evaluating an encapsulation material's ability to confine the reactive equivalent of 25 mg of explosives, the maximum stress in the container wall is required to be less than or equal to the yield strength of the material:

$$\frac{pd}{2t} \leq \sigma_{yield} ;$$

where σ_{yield} is the yield strength. Solving this equation for d/t provides an easy method of evaluating an encapsulation material:

$$\frac{d}{t} \leq \frac{2}{p} \sigma_{yield} .$$

Assuming an internal pressure of 1057 atmospheres (15,538 psi), maximum values of d/t are displayed in Table 13-13 for the encapsulation materials of Table 13-12. The results indicate that a polyethylene vial is not a practical container since its wall thickness must be 4.5 times the diameter. However, both the aluminum and the stainless steel make satisfactory containers.

Table 13-13 Container Diameter to Thickness Ratio

Material	d/t
Stainless Steel (type 304)	4.3
Aluminum (alloy 6061)	5.1
polyethylene (low density)	0.22

As a result of the preceding analysis, a limit of 25 milligrams of TNT equivalent explosives is deemed to be a safe limitation on explosives which may be irradiated in facilities located inside the reactor tank.

Irradiation of larger quantities of explosives in the reactor tank is not allowed. However, safety analyses have been performed which show that three pounds of TNT equivalent explosives may be safely irradiated in radiography Bays 1, 2, 3 and 4, provided the beam tube cover plates are at least 0.5 inch thick (Reference 13.16).

Southwest Research Institute (SRI) completed a safety analysis to determine the maximum amount of TNT equivalent explosive allowable in radiography Bay 3, (i.e., the amount that will not cause failure of the beam tube cover plate and will cause only repairable structural damage to the bay) (Reference 13.17). Bay 3 is the smallest in volume of all the radiography bays at the UCD/MNRC. The study concluded that Bay 3 can withstand a detonation of 6 pounds of TNT equivalent explosive with certain modifications. The study performed by SRI concluded that the Bay 3 door track must be strengthened. The recommended strengthening consists of welding three additional anchor bolt plates to the door track and bolting these plates into the wall with additional drilled anchor bolts. This strengthening assures that the door will respond in a ductile manner to an unexpected high blast load, absorbing the additional load with larger deflections, rather than responding in a brittle failure mode.

The UCD/MNRC completed a similar study to determine the maximum amount of TNT equivalent explosives allowable in all radiography bays (Reference 13.16). This study concluded that Bays 1, 2 and 4 can withstand a detonation of 6 pounds of TNT equivalent explosives without any damage provided the criteria in Table 13-14 are implemented in each bay. However, to meet category 1 protection requirements for 6 pounds of explosives, the west door of Bay 2 also requires modification by means of an additional wheel and post assembly. The analysis performed by the UCD/MNRC demonstrates that for 3 pounds of TNT equivalent explosives, no modifications are necessary to the radiography bay doors for Bays 1, 2 or 4. These doors will also respond in a ductile manner. As a result of the above studies, it is concluded that installation of beam tube cover plates with the thicknesses shown in Table 13-14 and implementing an explosives limitation of 3 pounds of TNT equivalent for each of the four radiography bays will satisfy the safety limitations established by the two previous safety analyses.

BAY	Cover Plate Thickness (in)	Explosive Location	Explosive distance* ref. @ 0°	Deflection (in)	Resistance (psi)	Ultimate Resistance (psi)
1	0.60	L/D = 100	13.00	0.294	43.6	108
2	0.60	L/D = 100	10.40	0.353	52.3	108
3	0.75	L/D = 100	8.80	0.433	125	168
4	0.60	L/D= 100	13.70	0.248	37	108

Table 13-14 Changes to Beam Tube Cover Plates (*Minimum distance from the beam tube cover plate to the explosive.)

The Argon-41 Production Facility (see Chapter 10) can produce argon-41 in excess of the amounts analyzed in Appendix A. However, if the system releases argon-41, the gas will be contained in the reactor room and the existing reactor room ventilation system will be used in recirculation mode to prevent the release of argon-41 to the environment by recirculating the gas until it decays. The existing stack continuous air monitor will also be used to verify that none has been released outside the UCD/MNRC boundary.

If the system had a catastrophic failure and 4 Curies of argon-41 were released to the volume of the reactor room, the argon-41 concentration in the reactor room would be 2×10^{-2} μCi/ml and the gamma dose rate in the reactor room would be approximately 22 R/hr (based on a semi-infinite cloud, see following calculation). Personnel would be evacuated from the reactor room and access would be restricted. The reactor room ventilation system (as described in Chapter 9) would be operated in the recirculation mode for approximately one day before the dose rate from argon-41 decays to less than 1 mR/hr. Therefore, the argon-41 discharge limit defined in the UCD/MNRC Technical Specifications will not be exceeded due to the recirculation mode of the reactor room ventilation system.

Another potential accidents include failure of the irradiation canister due to over-pressurization from the argon gas supply cylinder, since a new argon supply cylinder is typically delivered at a pressure of 2200 psi and the canister is rated for 1800 psi. However, this requires multiple failures and is considered non-credible: a) the operator would have to violate an operational procedure; b) the regulator would have to fail, and c) at the same time the pressure relief valve would have to fail. Also, another potential accident is that liquid nitrogen could spill into the reactor tank, causing expansion of the water and expelling a portion of tank water. To prevent this, a catch basin surrounds the Cold Trap, and the liquid nitrogen is supplied through a pipe in the reactor room wall connecting the trap to a supply container in the equipment room. A third accident could result if the pressure relief valve became choked with supersonic flow; however, the flow rates are estimated to be less than sonic as shown in the following calculation.

ARGON-41 CONCENTRATION IN REACTOR ROOM

Given:

1. Reactor room volume = 7.39×10^3 ft^3 (1)

2. 4 Ci Argon-41 in argon production system

3. $D(\gamma)_{\infty/2} = 0.25E_\gamma\chi$ (2)

where

$D(\gamma)_{\infty/2}$ = gamma dose rate from a semi-infinite cloud (rad/sec)
E_γ = average gamma energy per disintegration (Mev/dis)
= 1.2936 MeV/dis for Argon-41 (3)
χ = concentration of gamma emitting isotope in the cloud (Ci/m^3)

Therefore:

$\chi = (4Ci)/(7.39 \times 10^3\ ft^3)(1\ m^3/35.314\ ft^3) = 1.91 \times 10^{-2}\ Ci/m^3$

$D(\gamma)_{\infty/2} = 0.25E_\gamma\chi$
$= (0.25)(1.2936\ Mev/dis)(1.91 \times 10^{-2}\ Ci/m^3)$
$= (0.0062\ rads/sec)(3600\ sec/hr)$
$= 22.24\ rads/hr$

Since

$D = D_0e^{-\lambda t}$

$t = -(1/\lambda)\ln(D/D_0)$
$= -(T_{½}/\ln2)\ln(D/D_0)$

Then for:

$D = 1$ mrad/hr

$t = -(1.8\ hr/\ln2)\ln(1\ mrad/hr/22.240\ mrad/hr)$
$= 26$ hr

(1) See Figure 9.11
(2) Shleien, B., L. Slaback, and B. Birky, Handbook of Health Physics and Radiological Health, Williams & Wilkins, January 1997, p. 439.
(3) Nuclides and Isotopes, 14th edition, Chart of the Nuclides, GE Nuclear Energy, p. 22.

SONIC FLOW

Assume: Perfect Gas

Constants:

Property	Value	Units
R	208	N-m/kg-degK
K(cp/cv)	1.67	dimensionless

Problem: Determine if the pressure relief valve will experience choking due to supersonic flow.

Solution:

First calculate the speed of sound in argon at 40 °C and -200 °C given that c = speed of sound in a medium = $(kRTg_c)^{1/2}$

$c = [1.67 \times 208(\text{N-m/kg-degK}) \times (40+273)\text{K} \times 1(\text{kg-m/N-s}^2)]^{1/2}$
$= 329.7327$ m/s at 40 °C

$c = [1.67 \times 208(\text{N-m/kg-degK}) \times (-200+273)\text{K} \times 1(\text{kg-m/N-s}^2)]^{1/2}$
$= 159.2397$ m/s at -200 °C

Next calculate the velocity of the argon in the tubing at the pressure relief valve given volumetric flow rate V = velocity × area.

From technical data on valve, assume V = 1 ft^3/min, based on air and relief at 1125 psi.

$V = 1\ \text{ft}^3/\text{min} \times (12\ \text{in./ft})^3 \times (2.54\ \text{cm/in.})^3 \times 1\ \text{min}/60\ \text{sec} = 471.9474\ \text{cm}^3/\text{sec}$

$\text{area} = \pi r^2 = 3.14 \times (0.18\ \text{in}/2)^2 = 0.025434\ \text{in.}^2$ (based on ¼ in. tubing)
$= 0.16409\ \text{cm}^2$

velocity = V/area = 2876.15 cm/sec = 28.7615 m/s

mach number = velocity/c = 0.180618 at -200 °C
= 0.087227 at 40 °C

Conclusion: Gas velocity at the relief valve is less than the speed of sound in argon and therefore should not experience choking at the valve

Reference: Zucher, Robert D., Fundamentals of Gas Dynamics, Weber Systems, Incorporated, 1977, pp. 89, 130-133, 375.

Another potential accident involves the Central Irradiation Facility (see Chapter 10) since it may be considered similar to a control rod. Therefore, consider three potential scenarios for an uncontrolled reactivity insertion analogous to the Uncontrolled Withdrawal of a Control Rod (see Section 13.2.2.2.2). First, if the material in the sample can were of sufficiently different worth than the aluminum cylinder, the sample can would cause reactivity changes in the same fashion as a control rod, and either operator error or mechanical failure could cause an uncontrolled reactivity insertion. Second, if the aluminum cylinder failed to engage upon the sample can's insertion, a water void would be created in the central facility as the aluminum cylinder descended ahead of the sample can. Similarly, if the aluminum cylinder failed to replace the can upon removal from the central facility a water void would result.

All three of the above scenarios can be bounded by the Uncontrolled Withdrawal of a Control Rod analysis (Section 13.2.2.2.2). Specifically, the Central Irradiation Facility must have less reactivity and must drive slower than the control rod analyzed ($3.50 and 42 inches/minute, respectively). To that end, the reactivity of any material in the sample can shall be measured at low power to verify its worth is not only less than $3.50, but also less than $1.75, the reactivity limit for the Central Irradiation Facility (based on the Technical Specification limit of $1.75 for the pulsed reactivity insertion). For example, the worth of a silicon ingot in the previous 1 MW in-core experiment facility was measured at $0.73 positive (vs. Water, reference exp. #96-01, 1/30/96, reactor run #2411). The worth of an aluminum cylinder vs. Void and vs. Water has been analyzed by computer simulation (Reference 13.18). The most positive reactivity effect in the computer simulation is from Case 3 to Case 9, where the voided sample can is lowered 18 inches, resulting in an increase of about $0.06. The most negative effect is from Case 3 to Case 12, where in an accident the sample can not only floods but also the aluminum cylinder drops, resulting in a decrease of about $1.76. Thus, the worth of the sample can or the aluminum cylinder vs. Water is less than $3.50, and also less than the most reactive control rod (for example, a typical regulating rod worth is $2.57, measured 6/98). With respect to the drive mechanism, the maximum drive speed is identical to the rod speed analyzed in Section 13.2.2.2.2. Furthermore, in the event of failure of the aluminum cylinder to engage upon installation of the sample can, the base of the Central Thimble is designed (by sizing the hole in the base) to allow the aluminum cylinder to descend at no more than the analyzed 42 inches/minute. Therefore, the accident analysis for Uncontrolled Withdrawal of a Control Rod (Section 13.2.2.2.2) is sufficient to bound any accident associated with the Central Irradiation Facility since: a) the material in the sample can shall be measured and verified to be less than $1.75 (half of the analyzed $3.50); b) the drive speed cannot exceed the analyzed 42 inches/minute; and c) the aluminum cylinder cannot fall uncontrolled faster than the analyzed 42 inches/minute.

Finally, physical impact on the fuel is considered non-credible since the sample can is always contained in a guide tube or attached to a drive mechanism such that it is unlikely to drop onto the core (see description in Section 10.4.1.4).

The I-125 production facility is another one of the reactor's experiment facilities, in this case, a facility designed to produce curie levels of iodine-125. A complete facility description and diagram can be found in Chapter 10, Sections 10.4.6.1 through 10.4.6.11.

The I-125 production facility is an in-core experiment facility with an independent irradiation chamber and its associated components. The irradiation chamber is located in or near the center of the reactor core. The analysis (Ref 13.20.A) shows that the irradiation chamber can be placed in the center of the reactor core without overheating a fuel element. This analysis also shows that two irradiation chambers could be installed in the reactor without causing a fuel element to overheat. The maximum change of operational reactivity worth of this experiment is estimated to be $0.27 (Ref 13.20.A), and therefore, it is well within the limit of less than $ 1.75 for any single secured experiment and the limit of less than $1.00 for any single moveable experiment.

A typical irradiation for the production of I-125 begins with transferring xenon-124 (Xe-124) gas to the irradiation chamber. After a precalculated number of hours of irradiation, the activated Xe-124 gas, now containing up to 6,000 curies of xenon-125 (Xe-125), is transferred cryogenically to decay storage vessel 1. After a few days, most of the activated Xe-125 has decayed to I-125 and will plate-out inside the decay chamber. The remaining xenon gas is then transferred cryogenically to decay storage vessel 2 and the I-125 is removed in solution by sodium hydroxide (NaOH) washes. The sodium iodide solution is then packaged as a liquid and sent to an off-site user in an appropriate DOT container.

The I-125 production facility has a primary and a secondary containment. Under normal operating conditions, the primary containment's components are the only ones that interact with the xenon gas. The components of the primary containment are the irradiation chamber, tubing, pneumatically-operated bellows valves, pneumatically-operated diaphragm valves, transfer vessel, decay storage vessel 1, decay storage vessel 2, pressure transducers, vacuum transducers, iodine trap, and thermocouples. While the facility is in operation, the xenon gas will be located in one of three locations: the irradiation chamber and cold finger, decay storage vessel 1, or decay storage 2. During irradiation, the gas is typically at 180 psig, but the irradiation vessel is tested to 450 psig, providing a safety factor of 2.5.

Secondary containment is also provided around the primary containment to the irradiation chamber, and a secondary containment vessel houses the pneumatically-operated bellows valves, pneumatic operators, chamber for the cold finger, decay storage vessel 1, and decay storage vessel 2. This containment allows for recovery of the xenon gas and containment of any I-125 should a leak occur within the primary containment. The secondary containment will be filled with helium gas to approximately 1 to 5 psig, which will allow the xenon gas to be cryogenically-separated and recovered from the helium. The volume of the secondary containment will readily allow for up to three liters of xenon gas to be added to the secondary with only a small increase in secondary pressure. Pneumatic actuation of the valves within the secondary containment will be by helium pressure to ensure that helium is maintained within the secondary. Located in a fuel storage pit, the secondary containment vessel will utilize the pit's concrete shielding to reduce radiation exposure. Shielding may also be placed over the secondary containment vessel to reduce worker exposure and maintain whole body radiation levels below approximately 10 mrem/hr.

The I-125 production facility has been designed to prevent the uncontrolled release of Xe-125 and I-125. For example, the catastrophic uncontrolled release of radionuclides from the facility, which because of the process would be primarily Xe-125, is not considered to be a credible accident because it would require failure of both the primary and secondary containment (a simultaneous-double mode failure). However, it is possible to postulate a loss of xenon gas from the primary containment into the secondary containment. In evaluating such a scenario, consider that the secondary containment volume is approximately 80 liters while the volume of the primary containment is approximately one liter. With three liters of xenon gas charged into the primary system, a total loss of the primary gas to the secondary containment would result in a very small (about 2 psig) increase in pressure inside the secondary system. The secondary system will normally operate at 1 to 5 psig and so this increase in pressure is considered insignificant in terms of secondary system integrity.

Evaluating the radiological impact of the above situation, one could make the worst-case assumption that all of the Xe-125 from the primary containment leaks into the secondary containment and that the design leak rate for the secondary system allows Xe-125 to enter the reactor room. The Xe-125 release rate from the secondary containment, the reactor room air concentration, the maximum Xe-125 concentration in the unrestricted area and the corresponding radiation dose to personnel in the reactor room and the unrestricted area can be calculated as follows, assuming that:

1. 6,000 curies of Xe-125 are available in the primary system after irradiation (a worst-case assumption based on an expected delivery of 40 Ci of I-125 from the system),
2. The volume of the secondary containment is 80 liters,
3. The secondary containment system leak rate is 1×10^{-3} cm^3/sec,
4. The reactor room volume (V) is 2.09×10^8 cm^3,
5. The reactor room ventilation flow rate is 2.26×10^7 cm^3/min, and
6. 100% of the Xe-125 gas is released to the secondary containment system.

If G equals the Xe-125 release rate into the reactor room, then:

$$G = \left(\frac{6{,}000\text{ curies}}{80\text{liters}}\right)\left(\frac{1\text{liter}}{1{,}000\text{cm}^3}\right)\left(\frac{10^{-3}\text{ cm}^3}{\text{sec}}\right)\left(\frac{10^6\ \mu\text{curies}}{1\text{curie}}\right)\left(\frac{60\text{sec}}{1\text{min}}\right)$$

$$G = 4{,}500\ \ \mu\text{ curies/min}$$

If A_e equals the equilibrium radioactivity present in the reactor room based on the secondary containment design leak rate and radioactive decay, then:

$$A_e = \frac{G}{\lambda}, \text{ where } \lambda = \left(\frac{0.693}{T_{1/2}\,Xe-125}\right) = \left(\frac{0.693}{17 \text{ hours}}\right),$$

therefore,

If A equals the equilibrium radioactivity in the reactor room based on the secondary system leak rate, radioactive decay and reactor room exhaust ventilation, then:

$$A = A_e\left(\frac{\lambda}{\lambda + \lambda_V}\right),$$

where λ is as defined above and

$$\lambda_V = \left(\frac{\text{reactor room ventilation flow rate}}{\text{reactor room volume}}\right).$$

NOTE: Assuming the reactor room ventilation continues to operate results in a worst-case scenario for releasing Xe-125 into the unrestricted area. If the ventilation system switches over to the recirculation mode, as it is designed to do when the reactor room continuous air monitor alarms (See Section 9.5.2), and it would alarm in this situation, then all but a trace of the Xe-125 would be contained in the reactor room and the recirculation system, where the xenon and any I-125 would decay and be removed by the charcoal filters in the system.

From above, A would then

$$= (6.62\text{x}10^{6}\ \mu\text{curies})\frac{(6.79\text{x}10^{-4}\,/\text{min}}{(6.79\text{x}10^{-4}\,/\text{min} + 1.08\text{x}10^{-1}\,/\text{min})},$$

$A = 4.13 \times 10^4$ μcuries of Xe-125 in the reactor room at equilibrium.

If concentration = A / V, then the Xe-125 concentration in the reactor room would be:

$$\left(\frac{4.13x10^{4}\ \mu curies}{2.09x10^{8}\ cm^{3}}\right) = 1.97x10^{-4}\ \frac{\mu curies}{cm^{3}}$$

Actual evacuation time for the reactor room has been conservatively assumed to be two minutes (Section 13.2.1.2), but five minutes was also evaluated for dosimetry purposes. Based on the 10 CFR 20 derived air concentration (DAC) for Xe-125 of 2.0×10^{-5} μcuries per milliliter, a one-hour occupancy during the preceding situation would result in a deep dose equivalent (DDE) of approximately 25.0 millirem, for a five-minute occupancy, a DDE of about 2.1 millirem and for a two-minute occupancy, a DDE of less than one (approximately 0.8) millirem. Since these doses are from submersion in Xe-125, the DDE is also essentially equal to the total effective dose equivalent (TEDE), but, in any case, the doses are well within applicable 10 CFR 20 limits.

Continuing the scenario (assuming the reactor room ventilation does not go into the recirculation mode), the Xe-125 being exhausted from the reactor room by the ventilation system then mixes with the combined ventilation flow from the radiography bays and the preparation area fume hood, which reduces the reactor room concentration from 1.97×10^{-4} μCi/ml to 2.94×10^{-5} μCi/ml at the point of discharge from the 60-foot high UCD/MNRC stack (Reference Section 9.5.2 and Figure 9.11). Taking this concentration of Xe-125 and applying the atmospheric dispersion model and meterological data used to assess the release of argon-41 from the same stack (See Appendix A, Section A.4), the Xe-125 concentration at the point of maximum concentration in the unrestricted area is 6.8×10^{-9} μCi/ml. This value is about 9.7% of the applicable effluent concentration limit in 10 CFR 20, and corresponds to an annual DDE (based on Environmental Protection Agency dose conversion factors, Reference 13.21) of approximately 10 millirem, assuming continuous occupancy at the point of maximum concentration for a full year and the existence of the stated Xe-125 concentration for the full year. As stated, this dose assessment is based on a worst-case scenario which is never expected to occur, but the dose projected is well within 10 CFR 20 limits for the unrestricted area. As with the previous Xe-125 dose projection from reactor room occupancy, this dose is based on submersion in a semi-infinite cloud and, therefore, the DDE is essentially the same as the TEDE.

Release of Xe-125 from the primary containment system due to loss of electrical power and/or computer monitoring is another accident scenario considered to be non-credible. This conclusion is based on the fact that, even if all valves in the primary containment were simultaneously actuated in the open position, there would be no release of radioactivity into the reactor room. Releasing gas from this system requires that both manual and pneumatically-operated valves be open at the same time. If all the pneumatically-controlled valves were opened simultaneously, no gas would be released because the manual valves are normally closed. Also, all pneumatic valves are normally closed, and electric solenoid valves must be energized to supply pressure to the pneumatic valves to open them. In addition, if electrical power is lost, then all pneumatic valves close. Valve position is displayed on the console and is independent of the computer.

While there are no automatic features to assure safe valve-sequencing in the event of a computer failure, qualified personnel following approved procedures will perform the system's valve operation. Valve actuation and gas or liquid transfer involving either manual or remotely-operated valves is strictly dependent upon procedures. The production of I-125 is not an automatic operation and is dependent on personnel involvement and procedural compliance. A training program designed to ensure that only qualified personnel operate the controls of the I-125 production system is an integral part of this program.

The Iodine-125 system has two operating modes, a computer-monitored mode and a manual mode. In the computer-monitored mode, computer-driven software monitors the remotely-operated valve actuation, and prevents actuation of an improperly selected valve. In manual mode, operating personnel are able to manually-operate the valves without software intervention. All manually- or remotely-operated valves are initiated by a physical action, either at the valve or at the control panel. While the computer is operating, the software will prevent inadvertent opening of pneumatically-operated valves if they are out of sequence for a particular portion of the operation. The computer can never force a valve to open. To open a valve, the operator must set a control switch on the control panel to the open position. A "valve open" request signal is then sent to the computer. If this particular valve is next in the sequence, then the computer will respond with a "valve open" signal. The valve open signal is routed through the valve control switch and opens the valve. If the operator sets the valve control switch to the closed position, the valve closes.

Manual valves are operated at the glove box. The remotely operated-valves are operated at the control panel. In the event of a computer failure, valve position indication would remain the same as it was prior to the loss of the computer. If there is a loss of computer monitoring, the manual setting will allow the operator to continue operation without computer intervention. This allows the operator to place the system in a secure mode.

The basic design criteria for the I-125 production facility requires that all of the solenoid- and pneumatically-operated valves be normally-closed valves. For example, to operate a three-way solenoid valve, electrical power must be applied in order to energize the solenoid to allow the helium operating gas to pass through the valve. When power to the solenoid is lost or secured, spring tension causes the valve to close, which stops the helium gas flow. When the solenoid-operated valve reaches the shut or closed position, an internal bleed port is exposed that vents off any residual downstream helium gas to ensure that the pneumatically-operated valve supply line is depressurized. To operate a pneumatically-operated bellows valve, a minimum pressure on the pneumatic actuator is required. When energized, the three-way solenoid valve described above provides the necessary actuating pressure to operate the desired bellows valve. The high pressure bellows valves in the I-125 production system are normally-closed valves with an integral valve position indication. In the event of electrical power failure at the UCD/MNRC facility itself, the solenoid valves will de-energize and close. This action will isolate and vent off the helium operating gas from any open bellows valve(s) causing them to close. With the bellows valves closed, the xenon/iodine in the system is isolated and cannot be moved or transferred until electrical power is restored. Each section of the I-125 production facility is tested to pressures appropriate for the various sections of the system.

If I-125 is being dispensed in the glove box when a facility power failure occurs, procedures will require personnel to shut the dispensing valve and place the I-125 sample container in a safe/secured position. When electrical power is restored, the production system computer and the operating program must be restarted. With the computer's selector switch in "AUTO" (computer monitoring) or "OFF," none of the primary containment system's solenoid valves will energize or reposition any pneumatically-operated valve to "OPEN" (regardless of individual valve switch positions). The operator must confirm that all of the valve selector switches are in the closed position, and select the appropriate procedure from the program menu, before any remotely-operated valve can be operated. With the system selector switch in "MANUAL" when power is restored, the operator must position the system selector switch back to "OFF" (reset) before any solenoid can receive power. At that point, any valve that was open before the power outage will reopen, unless the operator moves the respective individual valve control switch to the shut/closed position prior to resetting the switch.

As noted above, loss of electrical power to the UCD/MNRC facility will result in the loss of the I-125 production system computer, the bellows valves position indication, solenoid valve control power and displays of pressure, vacuum, temperature, and metering valves. However, the reactor room glovebox and fume hood ventilation systems and the reactor room ventilation systems will remain operational on power provided by the UCD/MNRC facility's emergency propane generator. The basic design and operating features built into this system confirm that if a loss of facility power or a loss of power specifically to the I-125 production facility were to occur, there would be no release (above the design leak rate of 10^{-3} cc/sec) of any xenon/iodine to the reactor room or to the unrestricted area.

There are several accident scenarios that can be evaluated which are related to the handling of the I-125, especially during the dispensing and handling in the glovebox of the final quantity of I-125 obtained from a given production run. There are also several accidents that can be postulated during the processing of the quality assurance (QA) sample and the final quantity of I-125 to be shipped out while handling in the reactor room fume hood.

To evaluate the postulated worst-case accident that could occur in the glovebox, it is assumed that the maximum expected I-125 activity in a NaOH solution present at any given time in the glovebox will be 40 curies. The glovebox filtering system consists of an activated charcoal inlet filter and two activated charcoal outlet filters (in series). The glovebox is maintained at a negative pressure with respect to the outside atmosphere by means of a variable speed blower which exhausts to the reactor room exhaust ventilation system and is supported by backup power in the event of a conventional power failure.

Before the dispensing and processing of the I-125 solution in the govebox begins and during the entire process, the reactor room ventilation exhaust, including the ventilation flow from the glovebox and the reactor room fume hood, will be diverted through special additional filtration (which is the same filtration used when the reactor room ventilation goes into the recirculation mode). More specifically, the exhaust stream will pass through a moisture separator, a standard pre-filter, a HEPA filter and two activated charcoal filters before being mixed with the radiography bays' ventilation flow and discharged out the 60-foot high stack.

In addition to the enhanced air filtration described above, the exhaust flow from the glovebox, fume hood, and reactor room will be continuously monitored for I-125 before it is discharged from the stack. Should there be an increase in the I-125 air concentration sufficient to exceed the preset limit on the I-125 CAM, or on the reactor room CAM, then the entire reactor room ventilation flow will automatically go into a recirculating mode (See Section 9.5.2 for a more complete description of the ventilation system). In this mode, the exhaust air from the reactor room will no longer be discharged out the stack and the release of any radioactive material in this air effluent will thus be stopped. However, while recirculating, the air will continue to be filtered through the moisture separator, the pre-filter, the HEPA filter and the charcoal filters before it is returned to the reactor room.

In considering the specific accidents that could occur during the dispensing and handling of up to 40 curies of I-125 in the glovebox, two scenarios can be analyzed. First, as a worst-case scenario, it is assumed that there is an accident in the glovebox which, by some undefined mechanism, causes all 40 curies of I-125 to instantaneously volatilize, and with no iodine plateout, to leave the glovebox through the two charcoal filters in the glovebox exhaust system. It is further assumed that the I-125 in the glovebox exhaust follows the previously described special filtration path for reactor room exhaust during I-125 processing, which introduces two more charcoal filters. An additional assumption for this situation is that the I-125 CAM fails to respond properly and does not switch the reactor room ventilation into the recirculating mode, and that this failure results in the reactor room exhaust being mixed with the ventilation flow from the radiography bays and discharged out the facility stack into the unrestricted area. It is also assumed that this series of events occurs over a 30-second interval. Using these assumptions, as listed below, the concentration of I-125 in the unrestricted area and the corresponding dose to unmonitored personnel can be calculated as follows, assuming that:

1. All 40 curies of I-125 volatilizes and leaves the glovebox through the glovebox exhaust system with no internal iodine plateout,

2. The rated efficiency for each of the two glovebox activated charcoal filters is 99%, but, for conservatism, the efficiency will be assumed to be 90%,

3. The rated efficiency for each of the two reactor room activated charcoal filters (i.e., the same two charcoal filters used when the reactor room ventilation is in the recirculation mode) is 99.7%, but, for conservatism, the efficiency will be assumed to be 90%,

4. The reactor room ventilation flow rate is 2.26×10^7 cm^3/min (800 cfm) (See Figure 9.11),

5. The concentration reduction factor for mixing the reactor room exhaust with all other exhaust stack flow is 6.7 (See Section 9.5.2 and Figure 9.11),

6. The concentration reduction factor from the point of discharge at the 60-foot high stack to the point of maximum concentration in the unrestricted area is 4350 (based on the atmospheric dispersion model and meteorological data used to assess the release and dispersion of Argon-41 from the same stack (See Appendix A, Section

A.4),

7. The release from the glovebox and subsequent discharge out the stack occurs over a 30-second interval, and

8. The reactor room ventilation system fails to go into the recirculation mode.

First, the I-125 reduction factor for one pass through four 90% efficient activated charcoal filters will be:

$(0.1)(0.1)(0.1)(0.1) = 1 \times 10^{-4}$ of the original I-125 remains.

Next, based on the above assumptions (1-3) regarding air filtration and I-125 release, the amount of I-125 exhausted to the stack will be:

$(1 \times 10^{-4})(40 \text{ Ci}) = 4 \times 10^{-3}$ curies $= 4 \times 10^{3}$ μcuries.

Assuming that the 4,000 μcuries of I-125 is mixed with the reactor room exhaust air over a 30-second interval, the average I-125 air concentration coming from the reactor room into the stack for this 30-second period will be:

$$\frac{(4x10^{3}\,\mu curies)}{2.26x10^{7}\,cm^{3}/\min)(0.5)} = 3.53x10^{-4}\,\mu curies/cm^{3}$$

Based on the assumption that the reactor room ventilation system does not go into the recirculation mode, the I-125 exhausted from the reactor room into the stack mixes with the combined ventilation flow from the radiography bays and the equipment area fume hood, which reduces the I-125 concentration at the point of discharge from the stack to:

$$\frac{(3.53x10^{-4}\,\mu curie/cm^{3})}{6.7} = 5.26x10^{-5}\,\mu curie/cm^{3}$$

Applying the applicable atmospheric dispersion model and appropriate meteorological data (See Appendix A, Section A.4), the I-125 concentration at the point of maximum concentration in the unrestricted area is:

$$\frac{(5.26x10^{-5}\,\mu curie/cm^{3})}{4{,}350} = 1.20x10^{-8}\,\mu curie/cm^{3}$$

Assuming a person were exposed to this concentration of I-125 for the entire 30-second duration of this event, the CEDE to the thyroid is much less than 1 millirem (based on

Environmental Protection Agency dose conversion factors, Reference 13.21). Extrapolating this to a more extreme situation, if the exposure duration were to increase to 10 minutes (a factor of 20 increase) at the same concentration the estimated CEDE to the thyroid would be only about 2 millirem. In either case, the projected doses are well within 10 CFR 20 limits.

A second accident scenario that can be postulated for periods when I-125 is being dispensed or processed in the glovebox is similar to the preceding accident, but in this case it is assumed that the reactor room ventilation system is put into the recirculation mode due to the expected response of the I-125 CAM. In this situation, little or none of the I-125 will be discharged out the facility stack compared to the previous accident scenario and the focus is on exposure to occupationally-exposed individuals in the reactor room. Should the accident in the glovebox (again by some undefined mechanism) cause all 40 curies of I-125 to instantaneously volatilize and leave the glovebox through the glovebox exhaust system, the concentration of airborne I-125 in the reactor room and the subsequent occupational dose to workers in the room can be calculated as follows, assuming that:

1. All 40 curies of I-125 volatilizes and, with no internal iodine plateout, leaves the glovebox through the glovebox exhaust system,

2. The rated efficiency for each of the two glovebox activated charcoal filters is 99%, but, for conservatism, the efficiency will be assumed to be 90%,

3. The rated efficiency for each of the two reactor room recirculation system activated charcoal filters is 99.97%, but, for conservatism, the efficiency will be assumed to be 90%, and

4. The volume of the reactor room is 7.39×10^3 ft^3 or 2.09×10^8 cm^3 (See Figure 9.11).

NOTE: Under the postulated accident scenario, the air from the glovebox will first pass through two activated charcoal filters on the glovebox itself. It will then mix with the reactor room exhaust flow and pass through a HEPA filter and two more activated charcoal filters before it returns to the reactor room atmosphere where personnel uptake of I-125 could occur. In addition, since the reactor room is in a recirculation mode, the room air will continue to pass through two large activated charcoal filters so the I-125 concentration will continue to decrease. Nevertheless, if a worst-case I-125 air concentration in the reactor room is based on only one pass of the mixed glovebox and reactor room exhaust through the four sequential charcoal filters and a release of all 40 curies of I-125 from the glovebox into the reactor room recirculation system, the occupational dose for workers in the reactor room would be as shown below.

First, the I-125 reduction factor for one pass through four 90% efficient activated charcoal filters will be:

$$(0.1)(0.1)(0.1)(0.1) = 1 \times 10^{-4} \text{ of the original I-125 remains.}$$

Next, based on the above assumptions (1-3) regarding air filtration and I-125 release, the I-125 returned to the reactor room after the first pass through the recirculation system will be:

$$(1 \times 10^{-4})(40 \text{ Ci}) = 4 \times 10^{-3} \text{ curies} = 4 \times 10^{3}\ \mu\text{curies.}$$

Assuming that the 4,000 microcuries of I-125 are mixed uniformly in the reactor room air, the reactor room I-125 air concentration will be:

$$\left(\frac{4x10^{3}\ \mu Ci}{2.09\,x10^{8}\ cm^{3}} \right) = 1.91\,x10^{-5}\ \mu Ci\ /\ cm^{3}$$

Using the five-minute and two-minute reactor room occupancy times evaluated earlier in conjunction with a release of Xe-125 into the reactor room, and basing the committed effective dose equivalent (CEDE) for the thyroid on the 10 CFR 20 derived air concentration (DAC) for iodine-125, the thyroid CEDE for a five-minute occupancy would be approximately 134 millirem and, for a two-minute occupancy, the CEDE would be about 54 millirem. Doses to other organs and any external dose from the I-125 air concentration will be very small, and therefore the CEDE values are representative of the total effective dose equivalents that would result from this occurrence and are well within 10 CFR 20 values.

Liquid quality assurance aliquots from the I-125 sample in the glovebox will be contained in sealed serum glass vials, which will be placed in a mechanically-sealed metal pipe and passed out of the glovebox using a pass-through sleeve and an umbilical cut. All of this will then be placed in a plastic zip lock bag. This packaging will preclude any accidental release during transport of the sample to the iodine fume hood located in the reactor room.

After transferring the smaller I-125 QA samples onto a suitable counting medium inside the fume hood, the sample will be sealed with plastic tape and then double bagged in plastic. The sample is now ready for transport to the counting lab for QA measurement. While awaiting measurement, the sample will be stored in a shielded container for iodine sample use only. All QA samples and residual process liquids will be gathered and sealed in a metal container for decay in storage or possible future disposal by appropriate means. As noted in conjunction with work in the glovebox, prior to starting the processing of the I-125 QA sample in the reactor room fume hood and while the processing is taking place, the reactor room ventilation system will be subjected to the special additional filtration described previously in this section and detailed in Section 9.5.2.

The maximum amount of I-125 in the QA samples that will be transferred to the reactor room fume hood is 400 millicuries. The fume hood filtering system consists of two activated charcoal outlet filters (in series). The hood will be maintained at a negative pressure with respect to the outside atmosphere by means of a variable speed blower which exhausts to the reactor room exhaust ventilation system. The hood blower and the reactor room ventilation system are equipped with backup emergency power.

Analysis of occurrences in the fume hood similar to the two postulated for the glovebox can be carried out using essentially the same assumptions, since (1) the fume hood has two charcoal filters like the glovebox (with 90% plus iodine removal efficiency), and (2) it exhausts into the same ventilation system for the reactor room. The one difference is that, except for transferring sealed I-125 production samples of up to 20 curies maximum at one time from one shielded container to another for shipping, a maximum I-125 activity of only 400 millicuries in the form of QA samples will be allowed in the fume hood, and therefore, the doses to workers should all 400 millicuries volatilize into the hood ventilation system will be approximately 1% of those estimated for the glovebox events. Therefore, the CEDE for the thyroid will be less than 2.0 millirem.

In order to minimize contamination on the shipping shield, product vials are filled in the glove box and placed in a transfer container. Up to approximately 20 curies of I-125 may be contained in a single vial. This transfer container is passed out of the glove box and is then placed in the fume hood. The transfer container is opened in the fume hood and the product vial is passed out into a shipping/container. If all 20 curies of I-125 product plus the additional one (1) curie of I-125 also allowed in the hood were to be simultaneously released in the fume hood the consequences would be bounded by the analysis of a 40 curie release in the glove box, but numerically would equate to approximately one millirem CEDE for the thyroid in the unrestricted area and a thyroid CEDE of about 70 millirem based on (the maximum) five (5) minute assumed occupancy time in the reactor room. The radiation protection program outlined in Chapter 11 will support operation of the I-125 production facility. This program includes appropriate air monitoring, radiation level and contamination surveys, shielding, waste management, and bioassay program to assess thyroid uptake of radioiodine (see Chapter 11).

Up to this point, the most significant accidents analyzed for the dispensing and handling of I-125 have focused on the release of either 40 curies of I-125 into the glovebox or 21 curies of I-125 into the fume hood. The impact of these accidents has been evaluated for members of the general public in the unrestricted area as well as for those occupationally exposed in the reactor room. Relative to the previous accident evaluations, there is one final accident scenario that can also be evaluated, which involves the simultaneous release of 40 curies of I-125 into the glovebox and 21 curies into the fume hood (a 61 curie total release into the two containments). The assumptions used to assess the radiological consequences of the isolated release of I-125 into either the glovebox or the fume hood have been clearly stated, are very conservative, and both the assumptions and the subsequent radiation doses from these previously analyzed accidents can be used to determine the impact of the accident involving simultaneous release into both the glovebox and the fume hood. Although this accident is considered to be highly unlikely because the production process does not normally create a situation where there will be 40 curies of I-125 in the glovebox and 21 curies in the fume hod at the same time, and it requries the simultaneous failure of containment barriers at two separate location in the production sequence, the doses, as shown below, will still be well within 10 CFR 20 values. For example, should this accident occur, the CEDE for the thyroid in the unrestricted area based on the most extreme exposure assumptions (a 10 minute exposure duration) would be only about 3.0 millirem, and for those exposed in the reactor room for the maximum assumed occupancy time of five minutes the CEDE for the thyroid would be about 205 millirem. Clearly, both of these projected doses are with 10 CFR 20 limits.

13.2.7 Loss of Normal Electrical Power

13.2.7.1 Accident-Initiating Events and Scenarios

Loss of electrical power to the UCD/MNRC could occur due to many events and scenarios which routinely affect commercial power.

13.2.7.2 Accident Analysis and Determination of Consequences

Since the UCD/MNRC does not require emergency backup systems to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. A backup power system is present at the UCD/MNRC which mainly provides conditioned power to the reactor console and control instrumentation. Therefore, the reactor will not automatically scram when there is a loss of normal electrical power. In fact, the backup power system is capable of providing electrical power for reactor control and various operational measurements for a period of time after loss of normal electrical power and until its battery power supply is exhausted.

Loss of normal electrical power during reactor operations is addressed in the reactor operating procedures, which require that upon loss of normal power an orderly shutdown is to be initiated by the operator on duty. The battery backup power will allow monitoring of the orderly shutdown of the reactor and verification of the reactor's shutdown condition.

13.2.8 External Events

13.2.8.1 Accident Initiating Events and Scenarios

Hurricanes, tornadoes and floods are virtually nonexistent in the area around the UCD/MNRC reactor. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. In addition, seismic activity in Sacramento is low relative to other areas of California (Chapter 2). Seismic activity has already been mentioned in connection with postulated reactor tank damage in Section 13.2.3.

The UCD/MNRC facility is surrounded by a security fence and a physical security plan is continuously in force for personnel and activities inside the fence. The reactor site is located in an Industrial Park on a former U.S. Air Force Base where access and overall security is far stricter than the surrounding civilian business and residential areas. Therefore, accidents caused by human controlled events which would damage the reactor, such as explosions or other unusual actions, are considered to be of very low probability.

Since the UCD/MNRC reactor is located at the edge of the runway at the former McClellan AFB, airplane crashes involving the reactor may potentially cause reactor damage.

13.2.8.2 Accident Analysis and Determination of Consequences

A study of the probability of aircraft crashes which could cause reactor damage at the UCD/MNRC was conducted by GA Technologies as a part of the original Stationary Neutron Radiography System Proposal (Reference 13.19). The conclusions show that the calculated reactor damage probability due to aircraft accidents is 5×10^{-8} per reactor year. This value was obtained using conservative assumptions and the "best estimate" value is expected to be considerably lower than 5×10^{-8}. Safety analyses of nuclear power reactors have generally concluded that a reactor damage probability due to an aircraft accident which is less than 1×10^{-7} per year does not represent a significant contribution to the overall reactor risk. Therefore, it is concluded that no specific aircraft accident and no radiological consequences need to be considered for the UCD/MNRC reactor.

13.2.9 Mishandling or Malfunction of Equipment

13.2.9.1 Accident Initiating Events and Scenarios

No credible accident initiating events were identified for this accident class. Situations involving an operator error at the reactor controls, a malfunction or loss of safety related instruments or controls and an electrical fault in the control rod system were anticipated at the reactor design stage. As a result, many safety features, such as control system interlocks and automatic reactor shutdown circuits, were designed into the overall TRIGA® Control System (Chapter 7). TRIGA® fuel also incorporates a number of safety features (Chapter 4)

which together with the features designed into the control system assured safe reactor response, including in some cases reactor shutdown.

Malfunction of confinement or containment systems would have the greatest impact during the maximum hypothetical accident (MHA), if they were used to lessen the impact of such an accident. However, as shown in Section 13.2.1, no credit is taken for confinement or containment systems in the analysis of the MHA for the UCD/MNRC reactor. Furthermore, no safety considerations at the UCD/MNRC depend on confinement or containment systems, although simple confinement devices like a fume hood might be used as part of normal operations.

Rapid leaks of liquids have been previously addressed in Section 13.2.3. Although no damage to the reactor occurs as a result of these leaks, the details of the analyses provide a more comprehensive explanation.

13.3 Summary and Conclusions

Chapter 13 of the Safety Analysis Report contains a conservative analysis of many different types of hypothetical accidents as they relate to the UCD/MNRC reactor and the surrounding environment. Beginning with the maximum hypothetical accident and continuing on through an entire array of other accidents, it has been shown that the consequences of such accidents will not result in occupational radiation exposure of the UCD/MNRC staff or radiation exposure of the general public in excess of applicable NRC limits in 10 CFR Part 20. Furthermore, there is no projected significant damage to the reactor as an outcome of the accidents evaluated, except the damage or malfunction assumed as part of the different accident scenarios analyzed. Details of the assumptions used for each accident scenario and the specific consequences of each accident are presented in the text of this Chapter.

Ref 13.20.A

MEMORANDUM

DATE: August 13, 2002

TO: Chuck Heidel

FROM: H. Ben Liu HBL

SUBJECT: Safety Analysis Report for Two I-125 Irradiation Chambers in the UCD/ MNRC's Reactor Core

Monte Carlo (MCNP) code, version 4B2, with continuous neutron/ photon cross-section data of ENDF/ B-V and B-VI versions, has been used to evaluate the inclusion of two aluminum irradiation chambers for I-125 production in the UCD/ MNRC's reactor core. There are three optional plans, Plan I, II, and III, evaluated and analyzed in the following report.

Plan I:

Locations for these irradiation chambers are at E6 position (in referenced C-ring) and G6 position (in the central irradiation facility). (See attached Figure 1 for locations). Currently, the E6 and G6 positions are occupied by an 8.5/20 fuel element and tank water, respectively. The reactor is operated up to 2 MW power but normally run at 1.8 MW power.

The general conditions for reactor operation are:

1. The transient rod (D4) and all other control rods (D7, G3, G9, J4, J7) are 80% withdrawn. (See attached Figure 1)

2. The average burn-up for all 8.5/20 and 20/20 type fuel elements is 25%. The most recent fuel configuration for the reactor core consists twenty-four 8.5/20 fuel elements, including five fuel-followed control rods and the element at E6 position, and eighty-seven 20/20 fuel elements.

3. The reactor core is operated at 2.0 MW power though the normal operation power is 1.8 MW.

4. Each of the following six cases was run for 3,000 cycles and 3,000 particles per cycle. The statistical error estimate for each K_{eff} value is ± 4¢

and for each flux intensity value is < 1%. The same statistical error estimate applies to all other cases run for Plan II and Plan III.

The general conditions for irradiation chambers are:

1. The 1st aluminum irradiation chamber (see attached Figure 2) is double encapsulated (1.5" w/ 0.125" wall and 1.125" w/ 0.035" wall) with an internal volume of about **300 cm^3**, 1.055" in diameter and 21.1" long. This chamber will be positioned at E6 position. (See attached Figure 1)

2. The 2nd aluminum irradiation chamber will also be double encapsulated (1.75" w/ 0.125" wall and 1.375" w/ 0.035" wall) with an internal volume of about **328 cm^3**, 1.305" in diameter and 15.0" long. This chamber will be positioned at G6 position. (See attached Figure 1)

3. 90% enriched Xe-124 gas will be added to either of the irradiation chambers at a gas pressure of about 180 psig before and during the irradiation. The estimated amount of Xe-124 is **11.7 gm** and **12.0 gm** in the 1st and 2nd irradiation chambers, respectively. The irradiations will not be run on both chambers simultaneously.

4. General neutron/ photon cross-section data for xenon (Xe) are available, 54000.35c in MCNP, instead of specific data for the Xe-124 nuclide. Approximation to weigh on the cross-section data was made to simulate the Xe-124 enriched gas in the irradiation chambers.

Table 1. All cases in Plan I.

		Case-0	Case-1	Case-2	Case-3	Case-4	Case-5
E6	8.5/20	X					
	Void		X		X		X
	Xe-124			X		X	
G6	Water	X	X	X			
	Void				X	X	
	Xe-124						X

The above table explains the changes of reactor core configuration at E6 and G6 positions for the evaluated six cases. Case-0 is the base case without any irradiation chambers. In Case-1, the void 1st irradiation chamber replaces the 8.5/20 fuel element at E6 position. In Case-2, the 1st irradiation chamber at E6 position is filled with Xe-124 gas. While the void 1st irradiation chamber is at E6 position, the void 2nd irradiation chamber is placed at G6 of the central irradiation facility in Case-3. With both irradiation chambers being installed, Case-4 and Case-5 evaluate the reactivity and flux changes when either of the irradiation chambers is filled with Xe-124 gas.

The detailed results for all six cases are shown in the following table.

Table 2. Detailed results for all cases in Plan I.

	Case-0	Case-1	Case-2	Case-3	Case-4	Case-5
K_{eff}	1.0381	1.0344	1.0332	1.0368	1.0361	1.0351
Difference vs. Case-0	xxxxx	- 50¢	- 66¢	**- 18¢**	- 27¢	- 41¢
Difference vs. Case-1	-----	xxxxx	- 16¢	+ 32¢	-----	-----
Difference vs. Case-3	-----	-----	-----	xxxxx	- 9¢	**- 23¢**
Peak Element Heating	**27.8 kW**	28.2 kW	27.9 kW	28.3 kW	28.2 kW	**28.4 kW**
E6-$\phi_{Thermal}$ ($n/cm^2.sec$)	-----	1.84e13	**1.63e13**	1.75e13	**1.56e13**	1.68e13
E6-ϕ_{RI} ($n/cm^2.sec$)	-----	1.39e12	**1.34e12**	1.46e12	**1.40e12**	1.43e12
G6-$\phi_{Thermal}$ ($n/cm^2.sec$)	-----	-----	-----	2.29e13	2.24e13	**2.01e13**
G6-ϕ_{RI} ($n/cm^2.sec$)	-----	-----	-----	1.90e12	1.90e12	**1.80e12**

$\phi_{Thermal}$: Thermal neutron flux with neutron energies < 0.1 eV.

ϕ_{RI}: Resonance integral (RI) per unit lethargy with neutron energies between 0.1 eV and 100 keV.

Xe-124 is activated to become Xe-125 through the (n, gamma) reaction, which then decays to I-125. The half-life for Xe-125 is 17.1 hours while the half-life for I-125 is 59.4 days. The neutron cross-section data for this reaction is 168 b and 2600 b for thermal and resonance integral, respectively, based on Chart of the Nuclides by Lockheed Martin and GE Nuclear Energy. However, based on Tables for Neutron Activation Analysis by M. Glascock at MURR, the neutron cross-section data are 106 b and 3600 b instead for thermal and resonance integral, respectively. The following table estimates the Xe-125 and I-125 activities, based on the average cross-section data published from the above, could be produced at E6 and G6 locations at different irradiation times when the reactor is operated at 1.8 MW power.

Table 3. Xe-125 and I-125 production at E6 and G6 positions at 1.8 MW power.

	Irradiation Time (Hour)	$1-e^{-\lambda T}$	E6	G6
Xe-125 (Ci)	8	0.2769	1460	1920
	16	**0.4771**	**2520**	**3320**
	24	0.6219	3280	4320
	32	0.7266	3830	5050
	40	0.8023	4230	5580
I-125 (Ci)*	8		10.7	14.1
	16		**18.6**	**24.5**
	24		24.2	31.9
	32		28.3	37.3
	40		31.2	41.1

* I-125 activity is estimated 3.5 days after the end of irradiation.

Observations and Discussions:

1. Peak element heating is slightly increased (< 3%) after both irradiation chambers are installed. The peak element heating is < 30 kW in all cases when the reactor core is operated at 2 MW power. Normally, the reactor is operated at 1.8 MW power. The increase of fuel element heating is quite insignificant.

2. The inclusion of the 1st irradiation chamber, replacing an 8.5/20 fuel element, would lose about 50 ¢ of reactivity worth. To the contrary, when the 2nd irradiation chamber is added to the central irradiation facility, an estimated 32¢ is gained. Therefore, the reactivity change for installing both irradiation chambers is a loss of 18¢ in total (Case-3 vs. Case-0).

3. Once both irradiation chambers are installed, the maximum change of operation reactivity worth of an irradiation is estimated to be **23¢**, or **$0.23**

based on Case-5 vs. Case-3. This is well within the limit of < $1.75 for any single secured experiment and the limit of < $1.00 for any single moveable experiment.

4. Thermal neutron flux suppression before and after the irradiation chamber is filled with Xe-124 gas is approximately 11%. Thermal neutron flux at G6 position is **29% higher** than that at E6 position when the chamber is filled with Xe-124 gas. (Case-5 vs. Case-4)

5. From theoretical estimate, a collection of 1,000 Ci Xe-125 would lead to approximate 11.2 Ci I-125 after 3.5 days of decay from the end of irradiation. Based on our prior irradiation experiences, actual I-125 production, seen in Table 3, is estimated for different irradiation times (from 8 to 40 hrs) based on an overall efficiency of 66%. As an example, a **16-hr irradiation at E6 irradiation chamber could yield 18.6 Ci I-125 while 24.5 Ci I-125 could be collected if the same irradiation takes place at G6 irradiation chamber**.

Plan II:

The I-125 production will begin with the 1st irradiation chamber at E6 position. (See Plan I) In the meantime, a new design will be made to replace the existing graphite sleeve to accommodate two irradiation chambers at X1 and X2 positions. (See attached Figure 3) Both aluminum irradiation chambers will be double encapsulated (1.75" w/ 0.125" wall and 1.375" w/ 0.035" wall) with an internal volume of about **328 cm^3**, 1.305" in diameter and 15.0" long.

The following table explains the configuration changes of the new graphite insert in the central irradiation facility. The base case remains the same as Case-0 (see Plan I, where **K_{eff} = 1.0381**) with an 8.5/20 fuel element at E6 position and tank water at G6 position. In Case-6, a new graphite insert with two void aluminum irradiation chambers will be placed in the central irradiation facility. In Case-7, chamber X1 is still void while chamber X2 is filled with Xe-124 gas. In Case-8 and Case-9, in the event of service need for one of the chambers, a graphite plug can be used for X1 position while chamber X2 is either void or filled with Xe-124 gas. In Case-10 and Case-11, in the event of a plan change to use X1 position for other irradiation applications, X1 is now filled with tank water while X2 is either void or filled with Xe-124 gas.

Table 4. All cases in Plan II.

		Case-6	Case-7	Case-8	Case-9	Case-10	Case-11
X1	Void	X	X				
	Graphite			X	X		
	Water					X	X
X2	Void	X		X		X	
	Xe-124		X		X		X

The detailed results for all cases in Plan II are shown in the following table.

Table 5. Detailed results for all cases in Plan II.

	Case-6	Case-7	Case-8	Case-9	Case-10	Case-11
K_{eff}	1.0389	1.0369	1.0395	1.0381	1.0377	1.0362
Difference vs. Case-0	+ 10¢	- 16¢	+ 18¢	0	- 5¢	- 25¢
Difference vs. Case-6	xxxxx	**- 27¢**	-----	-----	-----	-----
Difference vs. Case-8	-----	-----	xxxxx	- 19¢	-----	-----
Difference vs. Case-10	-----	-----	-----	-----	xxxxx	- 20¢
Peak Element Heating	27.8 kW	27.6 kW	27.8 kW	27.8 kW	27.8 kW	27.6 kW
X1-$\phi_{Thermal}$ (n/cm^2.sec)	2.05e13	1.95e13	2.12e13*	1.98e13*	2.88e13†	2.76e13†
X1-ϕ_{RI} (n/cm^2.sec)	1.89e12	1.85e12	1.96e12*	1.92e12*	1.87e12†	1.84e12†
X2-$\phi_{Thermal}$ (n/cm^2.sec)	2.05e13	**1.84e13**	2.10e13	**1.86e13**	2.28e13	**2.01e13**
X2-ϕ_{RI} (n/cm^2.sec)	1.89e12	**1.81e12**	1.92e12	**1.82e12**	1.88e12	**1.77e12**

$\phi_{Thermal}$: Thermal neutron flux with neutron energies < 0.1 eV.

ϕ_{RI}: Resonance integral (RI) <u>per unit lethargy</u> with neutron energies between 0.1 eV and 100 keV.

* Average fluxes in 15"-long graphite.

† Average fluxes in 15"-long water.

Observations and Discussions:

1. Peak element heating is little changed (< 1%) after the existing graphite sleeve in the central facility is redesigned and replaced to accommodate two aluminum irradiation chambers. The peak element heating is < 30 kW in all cases when the reactor core is operated at 2 MW power. Normally, the reactor is operated at 1.8 MW power. The change of fuel element heating is quite insignificant.

2. The inclusion of the new graphite insert has insignificant impact to the core reactivity worth compared to Case-0. When either chamber is filled with Xe-124 gas for production run, an estimated maximum **27¢** of operation reactivity worth is lost (Case-7 vs. Case-6). This is well within the limit of < $1.75 for any single secured experiment and the limit of < $1.00 for any single moveable experiment.

3. Thermal neutron flux suppression before and after the irradiation chamber is filled with Xe-124 gas is approximately 11% once again. Thermal neutron flux at either X1 or X2 positions is **18% higher** than that at E6 position and **8% lower** than that at G6 position when either chamber is filled with Xe-124 gas.

4. Based on an overall efficiency of 66% for actual I-125 production runs, a 16-hr irradiation using either X1 or X2 irradiation chambers could yield 23.4 Ci I-125 which is **25% higher** than that at E6 position and < **4% lower** than that at G6 position.

Plan III:

The 3rd optional plan combines Plan I and Plan II in the event that a dedicated central irradiation facility for other irradiation applications is still required. Under the circumstances, the 1st irradiation chamber will remain at E6 position while the new graphite insert will be made to replace the existing graphite sleeve. Either X1 or X2 will become the 2nd irradiation chamber for I-125 production run while the other position becomes the future central irradiation facility for other irradiation applications. This unoccupied position can be either filled with a graphite plug or left with tank water flowing.

The following table explains the configuration changes at E6 position and the new graphite insert (X1 and X2 positions) in the central irradiation facility. The base case remains the same as Case-0 (see Plan I, where $\mathbf{K_{eff} = 1.0381}$) with an 8.5/20 fuel element at E6 position and tank water at G6 position. In Case-12 through Case-14, a new graphite insert is added with X1 filled with a graphite plug. In Case-15 through Case-17, X1 is filled with tank water. Two aluminum irradiation chambers are placed at E6 and X2 positions. In Case-12 and Case-15, both chambers are void. In Case-13 and Case-16, E6 is filled with Xe-124 gas while X2 is void. In Case-14 and Case-17, E6 is void while X2 is filled with Xe-124 gas.

Table 6. All cases in Plan III.

		Case-12	Case-13	Case-14	Case-15	Case-16	Case-17
E6	Void	X		X	X		X
	Xe-124		X			X	
X1	Graphite	X	X	X			
	Water				X	X	X
X2	Void	X	X		X	X	
	Xe-124			X			X

The detailed results for the above six cases are shown in the following table.

Table 7. **Detailed results for all cases in Plan III.**

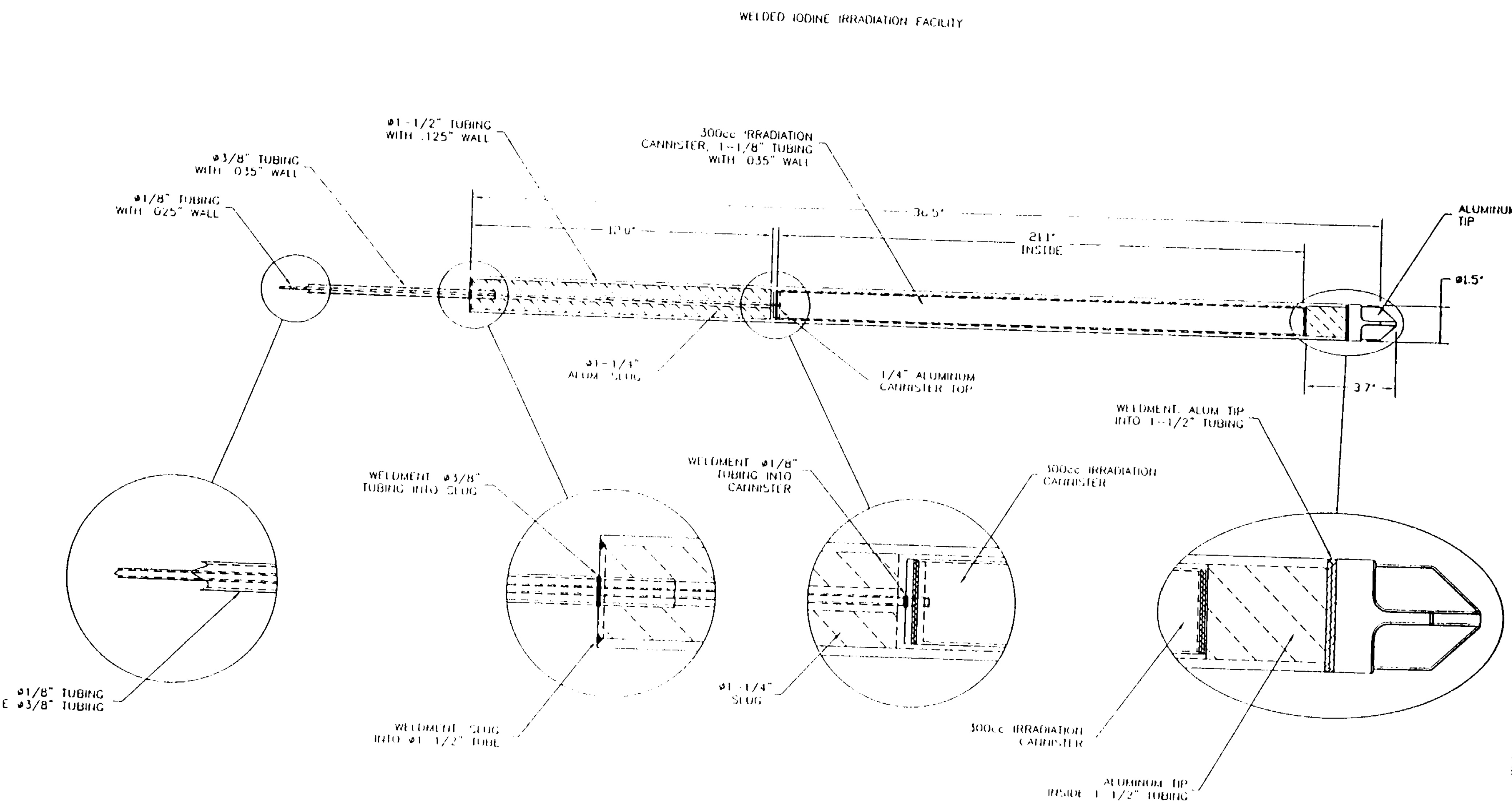
	Case-12	Case-13	Case-14	Case-15	Case-16	Case-17
K_{eff}	1.0357	1.0347	1.0342	1.0341	1.0333	1.0321
Difference vs. Case-0	**- 32¢**	- 46¢	- 52¢	**- 54¢**	- 65¢	- 81¢
Difference vs. Case-12	xxxxx	- 13¢	- 20¢	-----	-----	-----
Difference vs. Case-15	-----	-----	-----	xxxxx	- 10¢	**- 27¢**
Peak Element Heating	28.0 kW	28.1 kW	28.0 kW	28.0 kW	**28.3 kW**	27.9 kW
E6-$\phi_{Thermal}$ ($n/cm^2.sec$)	1.69e13	**1.50e13**	1.64e13	1.77e13	**1.58e13**	1.76e13
E6-ϕ_{RI} ($n/cm^2.sec$)	1.45e12	**1.39e12**	1.43e12	1.40e12	**1.34e12**	1.40e12
X1-$\phi_{Thermal}$ ($n/cm^2.sec$)	2.11e13*	2.05e13*	2.01e13*	2.85e13†	2.79e13†	2.75e13†
X1-ϕ_{RI} ($n/cm^2.sec$)	1.90e12*	1.87e12*	1.87e12*	1.82e12†	1.81e12†	1.82e12†
X2-$\phi_{Thermal}$ ($n/cm^2.sec$)	2.08e13	2.04e13	**1.86e13**	2.26e13	2.22e13	**2.01e13**
X2-ϕ_{RI} ($n/cm^2.sec$)	1.89e12	1.89e12	**1.82e12**	1.86e12	1.84e12	**1.77e12**

$\phi_{Thermal}$: Thermal neutron flux with neutron energies < 0.1 eV.

ϕ_{RI}: Resonance integral (RI) <u>per unit lethargy</u> with neutron energies between 0.1 eV and 100 keV.

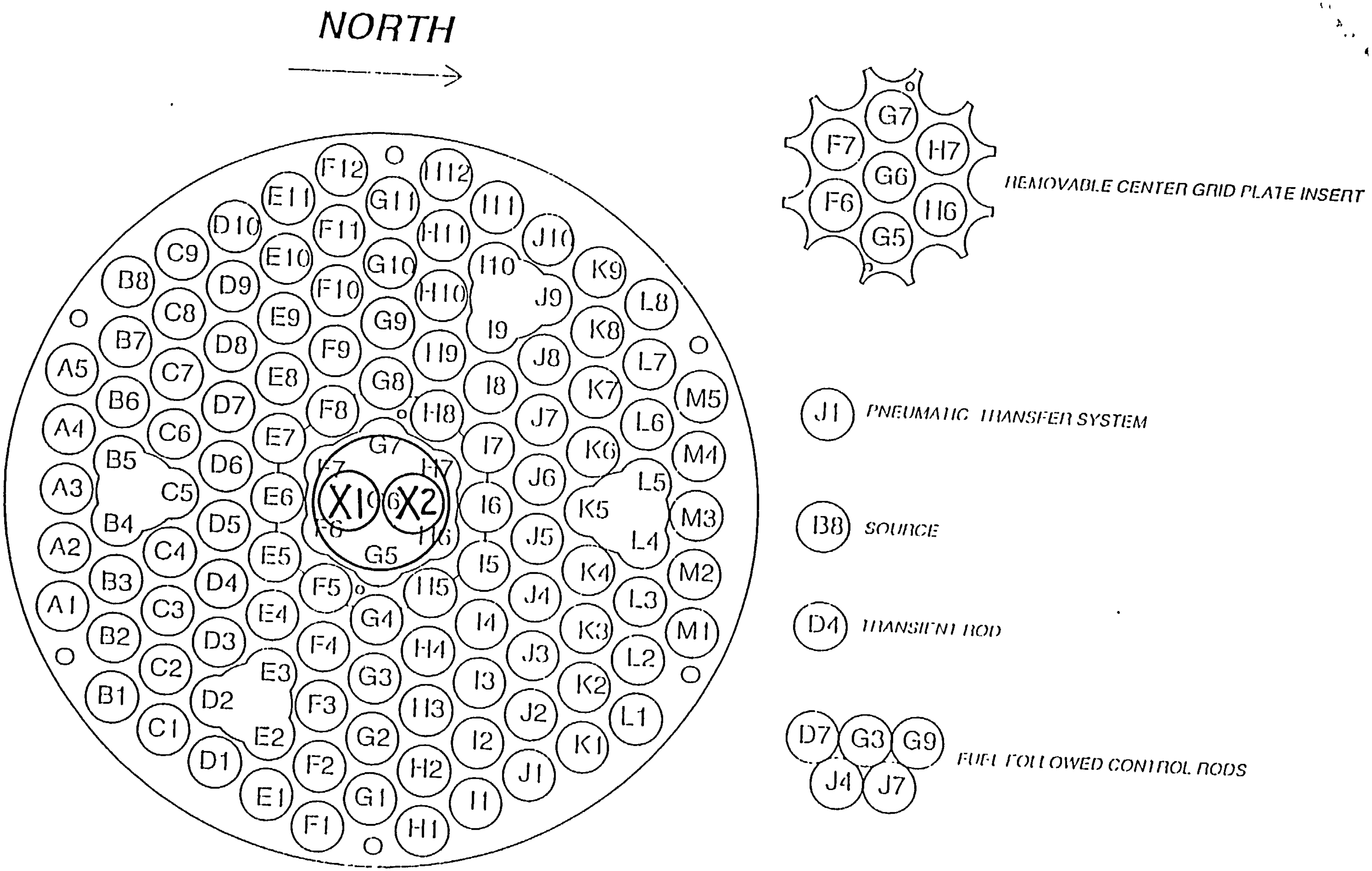
* Average fluxes in 15"-long graphite.

† Average fluxes in 15"-long water.



WELDED IODINE IRRADIATION FACILITY
ø1-1/2" TUBING WITH .125" WALL
300cc IRRADIATION CANNISTER, 1-1/8" TUBING WITH .035" WALL
ø3/8" TUBING WITH .035" WALL
ø1/8" TUBING WITH .025" WALL
ALUMINUM TIP
36.5"
12.0"
21.1" INSIDE
ø1.5"
ø1-1/4" ALUM. SLUG
1/4" ALUMINUM CANNISTER TOP
3.7"
WELDMENT, ALUM TIP INTO 1-1/2" TUBING
WELDMENT ø3/8" TUBING INTO SLUG
WELDMENT ø1/8" TUBING INTO CANNISTER
300cc IRRADIATION CANNISTER
ø1/8" TUBING
ø3/8" TUBING
ø1-1/4" SLUG
WELDMENT SLUG INTO ø1-1/2" TUBE
300cc IRRADIATION CANNISTER
ALUMINUM TIP INSIDE 1-1/2" TUBING

Figure 2



MCCLELLAN NUCLEAR RADIATION CENTER

Figure 3

Observations and Discussions:

1. Peak element heating is slightly increased (< 2%) after both irradiation chambers are installed at E6 and X2 positions. The peak element heating is < 30 kW in all cases when the reactor core is operated at 2 MW power. Normally, the reactor is operated at 1.8 MW power. The increase of fuel element heating is quite insignificant.

2. The inclusion of the both irradiation chambers, replacing an 8.5/20 fuel element and the existing graphite sleeve, would lose about **32 ¢** of reactivity worth when X1 is filled with a graphite plug (Case-12 vs. Case-0). It would lose about **54 ¢** of reactivity worth when X1 is filled with tank water (Case-15 vs. Case-0).

3. Once both irradiation chambers are installed, the maximum change of operation reactivity worth of an irradiation is estimated to be **27¢**, or **$0.27** based on Case-17 vs. Case-15. This is well within the limit of < $1.75 for any single secured experiment and the limit of < $1.00 for any single moveable experiment.

4. Flux distributions are quite similar to cases run for Plan I (Case-13 and Case-16 vs. Case-4) (Case-14 and Case-17 vs. Case-5). The Xe-125 and I-125 production at E6 and G6 positions in Table 3 (Plan I) can be used to estimate the similar production, within a 3% difference, at E6 and X2 positions for Plan III. For a similar example based on an overall efficiency of 66%, a 16-hr irradiation at E6 irradiation chamber could yield 18.6 Ci I-125 while 24.5 Ci I-125 could be collected if the same irradiation takes place at X2 irradiation chamber.

Conclusions:

Dependent upon future needs and requirements, all three plans proposed above are safe alternatives. None of the plans analyzed poses any unacceptable risks by overheating a fuel element. The change in reactivity worth during installation or actual irradiations is well within the safe limit established in the SAR.

The production of I-125 in any plan would meet the target set for near 20 Ci per 16-hr irradiation run per irradiation chamber at 1.8 MW power.

Cc. J. Ching
W. Richards