**From:** Marvin Mendonca
**To:** George Miller
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Attached for your information

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF CALIFORNIA, IRVINE
TRIGA MARK I NUCLEAR REACTOR

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## 1.0 DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications.

1.1 Reactor Shutdown - The reactor is in a shutdown condition when sufficient control rods are inserted so as to assure that it is subcritical by at least $1.00 of reactivity.

1.2 Reactor Secured - The reactor is secured when all the following conditions are satisfied:

a. The reactor is shutdown;

b. Power to the control rod magnets and actuating solenoids is off, and the key removed;

c. No work is in progress involving fuel or in-core experiments or maintenance of the core structure, control rods, or control rod drive mechanisms.

1.3 Reactor Operation - The reactor is in operation when it is not secured.

1.4 Standard Control Rod - A standard control rod is one having rack and pinion, electric motor drive, and scram capability.

1.5 Transient Control Rod

a. Adjustable Transient Rod - an adjustable transient rod is one having both pneumatic and electro-mechanical drives and with scram capability.

b. Fast Transient Rod - A fast transient rod is one that is pneumatically operated and has scram capability.

1.6 Operable - A system or device is operable when it is capable of performing its intended functions in a normal manner.

1.7 Cold Critical - The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures the same (~20°C).

1.8 Steady-State Mode - The reactor is in the steady-state mode when the reactor mode selection switch is in the steady-state or automatic position. In this mode, reactor power is held constant or is changed on periods greater than three seconds.

1.9 Pulse Mode - The reactor is in the pulse mode when the reactor mode selection switch is in the pulse position. In this mode, reactor power is increased on periods less than one second by motion of the transient control rod(s).

1.10 Experiment - An experiment is:

a. Any apparatus, device or material placed in the reactor core region, in an experimental facility, or in-line with a beam of radiation emanating from the reactor;

b. Any operation designed to measure reactor characteristics.

1.11 Untried Experiment - An untried experiment is any experiment not previously performed in this reactor.

1.12 Experimental Facilities - Experimental facilities are the pneumatic transfer systems, central thimble, rotary specimen rack, and the in-core facilities (including single element positions, three-element positions, and the seven element position).

1.13 Abnormal Occurrence - An abnormal occurrence is any of the following:

a. Any actual safety system setting less conservative than specified in the Limiting Safety System Settings section of the Technical Specifications;

b. Operation in violation of a limiting condition for operation;

c. An engineered safety system component failure which could render the system incapable of performing its intended function;

d. Release of fission products from a fuel element;

e. An uncontrolled or unanticipated change in reactivity;

f. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

1.14 Standard Thermocouple Fuel Element - A standard thermocouple fuel element is a standard fuel element containing three sheathed thermocouples imbedded near the axial and radial center of the fuel element.

1.15 Measured Value - The measured value of a process variable is the value of the variable as it appears on the output of a channel.

1.16 Measuring Channel - A measuring channel is the combination of sensor, lines, amplifiers and output device which are connected for the purpose of measuring the value of a process variable.

1.17 Reactor Safety System - The reactor safety system is that combination of channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

1.18 Operating - Operating means a component or system is performing its intended function in its normal manner.

1.19 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or methods measuring the same variable.

1.20 Channel Test - A channel test is the introduction of a signal into the channel to verify that it is operable.

1.21 Channel Calibration - A channel calibration is an adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures.

1.22 Reference Core - A reference core is a core with a configuration similar to the core configuration existing at the initial start-up of the reactor.

1.23 Ring - A ring is one of the six concentric bands of fuel elements surrounding the central opening of the core. The rings are designated by the letters B through G, with the letter B used to designate the innermost band.

1.24 Three Element Positions - Two generally triangular-shaped sections cut out of the upper grid plate, one encompassing ring holes D5, E6 and E7 and the other D14, E18 and E19. When fuel elements are placed in these locations a special fixture provides lateral support. With the fixture and fuel removed, an experiment up to 2.4 in. in diameter may be inserted.

1.25 Seven Element Position - A hexagonal section which can be removed from the upper grid plate for insertion of specimens up to 4.4 in. in diameter after relocation of the six B-ring elements and removal of the central thimble.

1.26 Closed Packed Array - A closed packed array is a fuel loading pattern in which the fuel elements are arranged in the core by filling the inner rings first.

1.27 Surveillance Activities - Activities required at pre-defined intervals to assure performance of reactor and safety related components. During prolonged periods when the reactor remains shutdown, Technical Specification Surveillance Requirements 4.1 (fuel element dimensions), 4.2 (control rod integrity), and 4.3 (fuel temperature safety limit) may be deferred. However, they must be completed prior to reactor start-up except for 4.2 (a), 4.3 (d), and 4.3 (f) which require reactor operation in order to be accomplished and must be completed within 30 days of restarting reactor operations.

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 Safety Limit - Fuel Element Temperature

Applicability

This specification applies to the fuel element temperature.

Objective

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

Specification

The temperature in a stainless steel clad, high hydride fuel element shall not exceed 1000°C under any conditions of operation.

Bases

The safety limitations of the TRIGA fuel are described in the Safety Analysis Report (SAR) for the UC Irvine TRIGA. The important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification since it can be measured. A loss in the integrity o£ the fuel element cladding could arise from an excessive build-up of pressure between the fuel moderator and the cladding. The pressure is caused by the presence of fission product gases and the dissociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel moderator temperature.

The safety limit for the stainless steel clad, high hydride ($ZrH_{1.7}$) fuel element is based on data which indicate that the stress in the cladding (due to the hydrogen pressure from the dissociation of the zirconium hydride) will remain below the yield stress provided the temperature of the fuel does not exceed 1000°C.

### 2.2 Limiting Safety System Settings

Applicability

This specification applies to the trip setting for the fuel element temperature channel.

Objective

The objective is to prevent the safety limit from being exceeded.

Specification

For a core composed entirely of stainless steel clad, high hydride fuel elements, limiting safety system settings apply according to the location of the standard thermocouple fuel element which shall be located in the B- or C-ring as indicated in the following table:

| Location | Limiting Safety System Setting |
|---|---|
| B-ring | 800°C |
| C-ring | 755°C |

Bases

Stainless steel clad, high hydride fuel element: The limiting safety system settings that are indicated represent values of the temperature, which if exceeded, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by a fuel element designed for this purpose, the limiting settings are given for different locations in the fuel array. Under these conditions, it is assumed that the core is loaded so that the maximum fuel temperature is produced in the B-ring. If the fuel element temperature is measured in the C-ring, the respective temperature is the limiting safety system setting.

## 3.0 LIMITING CONDITIONS FOR OPERATION

### 3.1 Reactivity

Applicability

These specifications apply to the reactivity condition of the reactor, and the reactivity worths of control rods and experiments, and apply for all modes of reactor operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit will not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin referred to the cold, xenon-free condition, with the highest worth rod fully withdrawn, is greater than $0.50;

b. The total reactivity worth of the two transient control rods is less than $3.00;

c. Any experiment with a reactivity worth greater than $l.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor;

d. The excess reactivity is less than $3.00;

e. The reactivity worth of an individual experiment is not more than $3.00;

f. The total reactivity worth of all experiments is limited so that the shutdown margin referred to the cold xenon-free condition with all rods in is at least $0.50;

g. The total of the absolute values of the reactivity worth of all experiments in the reactor is less than $3.00;

h. The drop time of a standard control rod from the fully withdrawn position to 90 percent of full reactivity insertion is less than one second; and

i. The neutron power level indication on the startup channel is greater than $1 \times 10^{-7}$ % of full power.

Bases

The shutdown margin required by specification 3.1a is necessary so that the reactor can be shut down from any operating condition and remain shutdown after cooldown and xenon decay even if one control rod (including a transient control rod) should stick in the fully withdrawn position.

Specification 3.1b is based on the SAR. The power level at which a pulse could be initiated in an accident may be as high as 100°C. At 100 kw, the peak temperature of the fuel will be ll5°C. The calculations indicate that a $3.00 pulse will result in a peak temperature of only 502°C, well below the safety limit.

Specification 3.1c is based on the same calculations. By restricting each

experiment to $1.00, an additional margin is provided to allow for considerable uncertainty in experiment worth.

Specification 3.1c through 3.lg are intended to provide additional margins between those values of reactivity changes encountered during the course of operations involving experiments and those values of reactivity which, if exceeded, might cause a safety limit to be exceeded.

Specification 3.lh is intended to assure prompt shutdown of the reactor in the event a scram signal is received.

Specification 3.li is intended to assure that sufficient neutrons are available in the core to provide a signal at the output of the startup channel during approaches to criticality.

## 3.2 Pulse Operation

Applicability

These specifications apply to operation of the reactor in the pulse mode.

Objective

The objective is to prevent the fuel temperature safety limit from being exceeded during pulse mode operation.

Specifications

The reactor shall not be operated in the pulse mode unless, in addition to the requirements of Section 3.1, the following conditions exist:

a. The transient rods are set such that their reactivity worth upon withdrawal is less than $3.00; and

b. The steady-state power level of the reactor is not greater than 1 kilowatt.

Bases

Specification 3.2a is based on the SAR which shows that the temperature rise expected for a pulse insertion of $3.00 is less than 500°C.

Specification 3.2b is intended to prevent inadvertent pulsing from a high steady-state power level such that the final peak temperature might approach the safety limit.

## 3.3 Reactor Instrumentation

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

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 measuring channels described in the following table are operable and the information is displayed in the control room:

| | Minimum Number Operable | Operating Mode in which Required |
|---|---|---|
| Measuring Channel | | |
| | | |
| Fuel Element Temperature | 1 | All Modes |
| Reactor Power Level | 2 | Steady-State |
| Reactor Power Level (high range) | 1 | Pulse Mode |
| Startup Power Level | 1 | During Reactor Startup |
| Area Radiation Monitors | 2 | All Modes |
| Continuous Air Radiation Monitor | 1 | All Modes |

Bases

The fuel temperature displayed at the control console gives continuous information on the process variable which has a specified safety limit.

The neutron detectors assure that measurements of the reactor power level are adequately covered at both low and high power ranges.

The radiation monitor provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

3.4 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels that must be operable in order to assure that the fuel temperature safety limit is not exceeded.

Specification

The reactor shall not be operated unless the safety system channels described in the following table are operable.

| Measuring Channel | Minimum Number Operable | Function | Operating Mode in which Required |
|---|---|---|---|

| | | | |
|---|---|---|---|
| Fuel Element Temperature | 1 | Scram | All Modes |
| Reactor Power Level | 1 | Scram | Steady-State Mode |
| Reactor Power Level | 1 | Prevent transient rods firing when power is >1 kilowatt | Pulse Mode |
| Manual Button | 1 | Scram | All Modes |
| Seismic Switch | 1 | Scram | All Modes |
| Startup Power Level | 1 | Prevent control rod withdrawal when power level indication is less than 1 x $10^{-7}$ %. | Reactor Startup |
| Standard Control Rod Position | 1 | Prevent application of air to fast transient rod when all other rods are not fully inserted | Steady-State Mode |
| Adjustable Transient Cylinder Position | 1 | Prevent application of air to adjustable transient rod unless cylinder is fully down | Steady-State Mode |

Bases

The interlocks which prevent the firing of the transient rods in the steady-state mode or if the power level is greater than 1 kilowatt prevent inadvertent pulses. The interlock to prevent startup of the reactor with less than 1 x $10^{-7}$ % indicated on the startup channel assure that sufficient neutrons are available to assure proper operation of the startup channel.

The fuel temperature scram provides the protection to assure that if a condition results in which the limiting safety system setting is exceeded, an immediate shutdown will occur to keep the fuel temperature below the safety limit. The power level scram is provided as added protection against abnormally high fuel temperature and to assure that reactor operation stays within the licensed limits. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. The seismic switch will shut down the reactor if major earth movement (M.M. VI or above) occurs in case the operator is prevented from operating the manual scram at the time.

## 3.5 Release of Argon 41

Applicability

This specification applies to the release of radioactive argon 41 from the facility exhaust system.

Objective

The objective is to assure that exposures to the public resulting from the release of argon 41 generated by reactor operation, will not exceed the limits of 10 CFR Part 20 for unrestricted areas.

Specification

Releases of argon 41 from the reactor room exhaust shall not be made in concentrations greater than 4 x $10^{-8}$ µc/ml averaged over a year.

Basis

It is shown in the SAR that the release of argon 41 will be diluted by a factor of at least 40 in reaching the potential exposure site even in the poorest dispersion conditions. At a concentration level of 1 x 10-9 µc/ml, for constant immersion, the maximum conceivable annual exposure will be 5 millirem to an individual and is well within acceptable limits.

3.6 Ventilation System

Applicability

This specification applies to the operation of the reactor facility ventilation system.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation,

Specification

The reactor shall not be operated unless the facility and building ventilation system is in operation and the emergency exhaust shutdown system has been verified to be operable within the preceding 30 days. An exception may be made for periods of time not to exceed two days to permit repairs to the system. During such periods of repair:

a. The reactor shall not be operated in the pulse mode; and

b. The reactor shall not be operated with experiments in place whose failure could result in the release of radioactive gases or aerosols.

Basis

It is shown in the SAR that operation of the emergency exhaust system reduces off-site doses to below 10 CFR Part 20 limits in the event of a TRIGA fuel element failure, and that operation of the normal system adequately dilutes the argon 41 released even under unusual experimental operations. The specifications governing operation of the reactor while the ventilation system is undergoing repair preclude the likelihood of fuel element failure during such times. It is shown in the SAR that, if the reactor were to be operating at full steady-state power, fuel element failure will not occur even if all the reactor tank water were to be lost immediately.

3.7 Pool Water Level

Applicability

This specification applies to the pool water level.

Objective

The objectives are to assure that an adequate level of water is maintained above the core and that prompt corrective action will be initiated in the unlikely event that pool-water leaks from the tank.

Specification

The pool water level shall normally be maintained approximately 19 feet above the reactor top grid plate. A pool water level measuring channel shall sound an alarm at the UCI Police Dispatch Desk if the water level in the reactor tank drops to 13 feet above the top grid plate. The measuring channel shall be operable except during periods of maintenance on the channel. If the measuring channel is inoperable, the level of the pool water shall be verified to be normal by visual observation at least every ten (10) hours. Whenever the duration of inoperability exceeds five (5) consecutive days, the reactor shall not be operated until repairs are completed and normal operation of the water level measuring channel has been verified. If either the alarm actuates or visual observation indicates that water level is not normal, prompt corrective action shall be taken.

Basis

The SAR discusses the results of loss of pool water from the Irvine TRIGA reactor and shows that fuel cladding rupture is unlikely even following operation at full licensed power. Calculations in the SAR indicate that ten hours after a leak develops in the pool or five hours after the water level (13 ft) alarm sounds, the radiation levels in the room above the reactor facility would be 0.028 mr/hr with the reactor shutdown. Both instrument and visual monitoring at the intervals specified will provide adequate time for corrective action. Written procedures, approved in accordance with Specification 6.3, shall define emergency actions to be taken.

3.8 Limitations on Experiments

Applicability

This specification applies to experiments placed in the reactor and its experimental facility.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. Fueled experiments are limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 0.3 curies and the Strontium 90 inventory is not greater than 1 microcurie;

b. The quantity of known explosive materials to be irradiated is less than 25 milligrams and the pressure produced in the experiment container upon accidental detonation of the explosive has been experimentally determined to be less than the design pressure of the container; and

c. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials or liquid fissionable materials are doubly encapsulated.

Basis

It is shown in the SAR that a release of 0.024 curies of iodine activity will result in a maximum dose to the thyroid of a person in an unrestricted area of less than 1/20 of the permissible dose. The limit on iodine inventory is set at 10 times this value. The limit for Strontium 90 is that which corresponds to the iodine yield of 0.3 curies for a given number of fission events and would be no hazard. Specifications 3.8b and 3.8c reduce the likelihood of damage to reactor components resulting from experiment failure.

## 4.0 SURVEILLANCE REQUIREMENTS

### 4.1 Fuel

Applicability

This specification applies to the surveillance requirement for the fuel elements.

Objective

The objective is to assure that the dimensions of the fuel elements remain within acceptable limits.

Specifications

a. The standard fuel elements shall be measured for length and bend at intervals separated by not more than 500 pulses of magnitude greater than $1.00 of reactivity, but the intervals shall not exceed 60 months. Fuel follower control rods shall be measured for bend at the same time interval. However, full surveillance shall be carried out before further operations are conducted if any significant changes are observed in pool water conductivity, pool water radioactivity, control rod drop times, control rod reactivity worths, or core reactivity worths such that it could be concluded that fuel element or control rod integrity may be compromised.

b. A fuel element indicating an elongation greater than 1/10 of an inch over its original length or a lateral bending greater than 1/16 of an inch over its original bending shall be considered to be damaged and shall not be used in the core for further operation.

   A fuel follower control rod shall be considered to be damaged and shall not be used for further operation if it indicates a lateral bending greater than 1/16 of an inch over the fuel containing portion of the rod.

c. Fuel elements in the B- and C-ring shall be measured for possible distortion in the event that there is indication that fuel temperatures greater than the limiting safety system setting on temperature may have been exceeded.

Bases

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 have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at other TRIGA installations. The surveillance interval is selected based on the past history of more frequent, uneventful, inspections for over 20 years at this facility and experience at other TRIGA facilities with similar power levels, fuel type, and operational modes. It is also designed to reduce the possibilities of mechanical failures as a result of handling elements, and to minimize potential radiation exposures to personnel.

## 4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods.

Objective

The objective is to assure the integrity of the control rods.

Specifications

a. The reactivity worth of each control rod shall be determined annually, but at intervals not to exceed eighteen months.

b. Control rod drop times shall be determined annually, but at intervals not to exceed eighteen months.

c. The control rods shall be visually inspected for deterioration at intervals not to exceed five years.

d. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed.

   Annually, at intervals not to exceed eighteen months, the transient (pulse) rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The visual inspection of the control rods and measurement of their drop times are made to determine whether the control rods are capable of performing properly. The surveillance intervals are selected based on the past history of more frequent, uneventful, inspections for over 20 years at this facility and experience at other TRIGA facilities with similar power levels, fuel type, and operational modes. They are also designed to reduce the possibilities of mechanical failures as a result of handling control rods, and to minimize radiation exposures to personnel.

## 4.3 Reactor Safety System

Applicability

This specification applies to the surveillance requirements for the measuring channels of the reactor safety system.

Objective

The objective is to assure that the safety system will remain operable and will prevent the fuel temperature safety limit from being exceeded.

Specifications

a. A channel test of each of the reactor safety system channels shall be performed prior to each day's operation or prior to each operation extending more than one day.

b. A channel check of the fuel element temperature measuring channel shall be performed daily whenever the reactor is in operation or when pulse operation is planned.

c. A channel check of the power level measuring channels shall be performed daily whenever the reactor is in operation.

d. A channel calibration by the calorimetric method shall be made of the power level monitoring channels annually, but at intervals not to exceed eighteen months.

e. A calibration of the temperature measuring channels shall be performed annually, but at intervals not to exceed eighteen months. This calibration shall consist of introducing electric potentials in place of the thermocouple input to the channels.

f. A verification of the original calibration of the temperature measuring channels shall be performed annually, but at intervals not to exceed eighteen months. This verification shall consist of comparing the measured temperature in a reference core at a known power level with the temperature measured in the reference core during the initial startup of the reactor.

Basis

The daily tests and channel checks will assure that the safety channels are operable. The annual calibrations and verifications will permit any long-term drift of the channels to be corrected. The history of operations at this facility over the last 20 years has shown that annual checks will allow correction for the very small amounts of drift observed.

## 4.4 Pool Water Level Channel

Applicability

This specification applies to the pool water level channel required by Section 3.7 of these specifications.

Objective

The objective is to assure that the channel is operable.

Specifications

The pool water level measuring channel shall be verified to be operable at intervals not to exceed two months.

Basis

This verification will assure that a continued warning system for a loss-of-coolant accident is maintained.

## 4.5 Radiation Monitoring Equipment

Applicability

This specification applies to the radiation monitoring equipment required by Section 3.3 of these specifications.

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification

The alarm set points for the radiation monitoring instrumentation shall be verified daily during periods when the reactor is in operation.

Basis

Surveillance of the equipment will assure that sufficient protection against radiation is available.

## 4.6 Maintenance

Applicability

This specification applies to the surveillance requirements following maintenance of control or safety system.

Objective

The objective is to assure that a system is operable before being used after maintenance has been performed.

Specification

Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable prior to its return to service.

Basis

This specification assures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

# 5.0 DESIGN FEATURES

## 5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

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

Specifications

a. Standard Fuel Element: The standard fuel element shall contain uranium-zirconium hydride, clad in 0.020 inch of 304 stainless steel. It shall contain a maximum of 9.0 weight percent uranium which has a maximum enrichment of 20 percent. There shall be l.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

b. Loading: The elements shall be placed in a closely packed array except for experimental facilities or for single positions occupied by control rods and a neutron start-up source.

Basis

These types of fuel elements have a long history of successful use in TRIGA reactors.

## 5.2 Reactor Building

Applicability

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

Objective

The objective is to assure that provisions are made to restrict the amount of release of radioactivity from the reactor facility.

Specifications

a. The reactor shall be housed in a closed room designed to restrict leakage when in operation, when the facility is unmanned, or when spent fuel is being handled exterior to a cast.

b. The minimum free volume of the reactor room shall be 1,000 cubic feet.

c. The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 70 feet above ground level.

Basis

In order that the movement of air can be controlled, the reactor area contains no windows that can be opened. The room air is exhausted through an independent exhaust and discharged at roof level with other exhausts to provide dilution.

## 5.3 Fuel Storage

### Applicability

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

### Objective

The objective is to assure that fuel which is being stored will not become supercritical and will not reach unsafe temperatures.

### Specifications

a. All fuel elements shall be stored in a geometrical array where the $k_{eff}$ is less than 0.8 for all conditions of moderation.

b. Irradiated fuel elements and fuel devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed 800°C.

### Basis

New fuel is stored in their shipping containers. Hot fuel is stored in pits described in the submittal dated June 5, 1969. These pits are designed to hold 19 elements, an amount which cannot form a critical array. Very hot fuel is stored in racks in the main tank where cooling water is provided.

## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 Organization

a. The reactor facility shall be an integral part of the School of Physical Sciences of the University of California, Irvine. The reactor shall be related to the University structure as shown in Chart I.

b. The reactor facility shall be under the direction of the Reactor Administrator who shall be a tenure member of the UCI faculty and supervised by a Reactor Supervisor who shall be a qualified licensed senior operator for the facility. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Operations Committee.

c. There shall be a Radiation Safety Officer responsible for the safety of operations from the standpoint of radiation protection. The Radiation Safety Officer shall report to the Office of Environmental Health and Safety which is an organization independent of the reactor operations organization as shown in Chart I.

CHART I

```
                                                School of Physical Science
                                                          |
                                              Reactor Operations Committee
                                                          |
Office of Environmental Health and Safety          Reactor Administrator
                  |                                       |
     Radiation Safety Officer  <-------------------------->  Reactor Supervisor
                  ^                                       |
                  |-------------------------------------->  Reactor Operations
```

### 6.2 Review

a. There shall be a Reactor Operations Committee which shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

b. The responsibilities of the Committee include, but are not limited to the following:

1. Review and approval of experiments utilizing the reactor facilities;
2. Review and approval of all proposed changes to the facility, procedures, and Technical Specifications;
3. Determination of whether a proposed change, test, or experiment

would constitute an unreviewed safety question or a change in the Technical Specifications;

4. Review of the operation and operational records of the facility;

5. Review of abnormal performance of plant equipment and operating anomalies;

6. Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR 20 and 10 CFR 50; and

7. Approval of individuals for the supervision and operation of the reactor.

c. The Committee shall be composed of at least five members, one of whom shall be a health physicist designated by the Office of Environmental Health and Safety of the University. The Committee shall be proficient in all areas of reactor operation and reactor safety. The membership of the Committee shall include at least one member who is not associated with the School of Physical Sciences.

d. The Committee shall have a written statement defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee. Minutes of all meetings of the Committee shall be kept.

e. A quorum of the Committee shall consist of not less than a majority of the full Committee and shall include the chairman or his designee.

f. The Committee shall meet at least semi-annually, at intervals not to exceed nine months.

6.3 Operating Procedures

Written procedures, reviewed and approved by the Reactor Operations Committee, shall be in effect and followed for the following items. The procedures shall be adequate to assure the safety of the reactor but should not preclude the use of independent judgment and action should the situation require such.

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

b. Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

c. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, and abnormal reactivity changes.

d. Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

e. Maintenance procedures which could have an effect on reactor safety.

f. Periodic surveillance of reactor instrumentation and safety systems, area monitors and continuous air monitors.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Operations Committee. Temporary changes to the procedures that do not change their original intent may be made by the Reactor Supervisor. All such temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Operations Committee.

6.4 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded, or thought to have been exceeded:

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

b. An immediate report of the occurrence shall be made to the Chairman of the Reactor Operations Committee, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications.

c. A report shall be made which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of reoccurrence. This report shall be submitted to the Reactor Operations Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.5 Action to be Taken in the Event of an Abnormal Occurrence

In the event of an abnormal occurrence, as defined in Section 1.13 of the specifications, the following action shall be taken:

a. The Reactor Supervisor shall be notified and corrective action taken prior to resumption of the operation involved.

b. A report shall be made which shall include an analysis of the cause of the occurrence, efficacy of corrective action and recommendations for measures to prevent or reduce the probability of reoccurrence. This report shall be submitted to the Reactor Operations Committee for review.

c. Where appropriate, a report shall be submitted to the NRC in accordance with Section 6.7 of these specifications.

6.6 Plant Operating Records

a. In addition to the requirements of applicable regulations, and in no way substituting therefore, records and logs shall be prepared and retained for a period of at least 5 years of the following items, as a minimum:

1. Normal plant operation;
2. Principal maintenance activities;
3. Abnormal occurrences;
4. Equipment and component surveillance activities;
5. Gaseous and liquid radioactive effluents released to the environs;
6. Off-site environmental monitoring surveys;
7. Fuel inventories and transfers;

8. Facility radiation and contamination surveys;

9. Radiation exposures for all personnel;

10. Experiments performed with the reactor.

b. Updated, corrected, and as-built drawings of the facility shall be retained for the facility life.

6.7 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made to the NRC as follows:

a. An immediate report (by telephone and telegraph to the NRC Headquarters Office) of:

1. Any accidental off-site release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury or exposure; and

2. Any violation of a safety limit.

b. A report within 24 hours (by telephone or telegraph to the NRC Headquarters Office) of:

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. Incidents or conditions relating to operation of the facility which prevented or could have prevented the performance of engineered safety features as described in these specifications; and

3. Any abnormal occurrences as defined in Section 1.13 of these specifications.

c. A report within 10 days (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) of:

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. Incidents of conditions relating to operation of the facility which prevented or could have prevented the performance of engineered safety features as described in these specifications; and

3. Any abnormal occurrences as defined in Section 1.13 of these specifications.

d. A report within 30 days (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) of:

1. Any substantial variance from performance specifications contained in these specifications or in the Safety Analysis Report;

2. Any significant change in the transient or accident analyses as described in the Safety Analysis Report;

3. Any changes in facility organization; and

4. Any observed inadequacies in the implementation of administrative or procedural controls.

e. A report within 60 days after criticality of the reactor (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

1. Total control rod reactivity worth;

2. Reactivity worth of the single control rod of highest reactivity worth;

3. Total and individual reactivity worths of any experiments inserted in the reactor; and

4. Minimum shutdown margin both at room and operating temperatures.

f. A routine report in writing to the (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) within 60 days after completion of the first six months of facility operation and at the end of each 12-month period thereafter, providing the following information:

1. A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period;

2. A tabulation showing the energy generated by the reactor (in megawatt hours), the amount of pulse operation, the number of hours the reactor was critical;

3. The number of emergency shutdowns and inadvertent scrams, including the reasons therefore;

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

5. A summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10 CFR 50;

6. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

7. A description of any environmental surveys performed outside the facility; and

8. A summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility.

6.8 Review of Experiments

a. All proposed experiments utilizing the reactor shall be evaluated in writing by the experimenter and the Reactor Supervisor (and the Radiation Safety Officer when appropriate) to assure compliance with the provisions of the utilization license, the Technical Specifications, and 10 CFR 20. If, in the judgment of the Reactor Supervisor, the experiment meets with the above provisions and is not an "untried experiment" the experiment shall be scheduled. Otherwise he shall submit it to another member of the Reactor Operations Committee for written evaluation and thence to the Reactor Operations Committee for final approval as indicated in Section 6.2 above. When pertinent, the evaluation shall include:

1. The reactivity worth of the experiment;
2. The integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition;
3. Any physical or chemical interaction that could occur with the reactor components; and
4. Any radiation hazard that may result from the activation of materials or from external beams.

b. Prior to the performing of an experiment not previously performed in the reactor, it shall be reviewed and approved in writing by the Reactor Operations Committee. Their review shall consider the following information:

1. The purpose of the experiment;
2. A procedure for the performance of the experiment; and
3. The written evaluations made as in Paragraph a. above.

c. A request for radioisotopes or the irradiation of materials shall be handled in the same manner as any other experiment except that a series of irradiations can be approved as one experiment. The expiration date for such approvals shall be one year or the expiration date of the applicant's appropriate radioactive materials license. For each irradiation, the applicant shall submit an "Irradiation Request" to the Reactor Supervisor. This request shall contain information on the target material including the amount, chemical form, and packaging. For the purposes of Paragraph a. above, routine irradiations, which do not contain nuclear fuel or known explosive materials and which do not constitute a significant threat to the integrity of the reactor or to the safety of individuals, may be classified as "tried experiments".

d. In evaluating experiments, the following assumptions shall be used for the purpose of determining whether failure of the experiment would cause the appropriate limits of 10 CFR 20 to be exceeded:

1. If the possibility exists that airborne concentrations of radioactive gases or aerosols may be released within the facility, 100 percent of the gases or aerosols will escape;

2. If the effluent exhausts through a filter installation designed for greater than 99 percent efficiency for 0.3 micron particles, at least 10% of gases or aerosols will escape; and

3. For a material whose boiling point is above 550C and where vapors formed by boiling this material could escape only through a column of water above the core, at least 10 percent of these vapors will escape.

Dr. Michael R. Gottfredson
Executive Vice Chancellor
University of California, Irvine
Irvine, CA 92697-1000

SUBJECT: ISSUANCE OF Amendment No. 7 TO FACILITY LICENSE NO. R-116 - UNIVERSITY OF CALIFORNIA, IRVINE TRIGA MARK I PULSING RESEARCH REACTOR (TAC NO. MA0226)

Dear :

The U.S. Nuclear Regulatory Commission has issued Amendment No. 7 for Facility License No. R-116 for the University of California, Irvine TRIGA Mark I pulsing research reactor in response to the application for renewal dated October 18, 1999, as supplemented on April 24, June 2, and xxxxxxxx xx, 2000. This amendment renews the operating license for 20 years from its date of issuance.

In accordance with our practice, we have restated the license in its entirety, incorporating all the changes and amendments made since issuance of the original license.

Enclosed with the amended license is a copy of the notice of renewal (xx FR xxxxx-xxxxx) and a copy of the Environmental Assessment (65 FR 53236-53237) that were published in the Federal Register, and the safety evaluation for the renewal.

Sincerely,

Marvin M. Mendonca, Senior Project Manager
Events Assessment, Generic Communications and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-326

Enclosures:

1. Amendment No. 7
2. Notice of Renewal
3. Environmental Assessment
4. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Service
P. O. Box 942732
Sacramento, CA 94234-7320

Dr. Richard Chamberlain, Chair
Department of Chemistry
University of California, Irvine
Irvine, CA 92697-2025

Dr. George E. Miller
Department of Chemistry
University of California, Irvine
Irvine, CA 92697-2025

Test, Research, and Training
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-326

UNIVERSITY OF CALIFORNIA, IRVINE

RENEWAL OF THE FACILITY OPERATING LICENSE

Amendment No. 7
License No. R-116

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

   A. The application for amendment by the University of California, Irvine dated October 18, 1999, as supplemented on April 24, June 2, and xxxxxxxxx xx, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10, Chapter I, Code of Federal Regulations (10 CFR).

   B. Construction of the facility was completed in substantial conformity with Construction Permit No. CPRR-19 dated May 5, 1969, the provision of the Act, and the rules and regulations of the Commission;

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

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

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

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

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

I. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including Sections 30.33, 70.23 and 70.31.

2. Facility Operating License No. R-116 is hereby amended in its entirety to read as follows:

1. This license applies to the TRIGA Mark I pulsing research reactor (herein "the reactor"), owned by The Regents of the University of California and located on its campus in Irvine, California, and which is described in the application for license dated October 18, 1999, and supplements thereto dated April 24, June 2, and xxxxxxxxx xx, 2000, (herein referred to as "the application"), and authorized for operation of License No. R-116.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses The Regents of the University of California, Irvine:

A. Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the reactor as a utilization facility in accordance with the procedures and limitations described in the application and in this licensee;

B. Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to 4.5 kilograms of uranium 235 in connection with the operation of the reactor; and

C. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess and use a 3-curie sealed americium-beryllium neutron source for reactor startup; and to possess, but not to separate, such byproduct material as may be produced by the operation of the reactor.

3. This license shall be deemed to contain and be subject to the conditions specified in Parts 20, 30, 40, 50, 51, 55, 70, and 73 of 10 CFR to all applicable provisions of the

Act, and to the rules, regulations and orders of the Commission now, or hereafter in effect, and to the additional conditions specified below:

A. Maximum Power Level

The licensee may operate the reactor at steady-state power levels up to a maximum of 250 kilowatts (thermal).

B. Technical Specification

The Technical Specifications contained in Appendix A, as revised through Amendment No. 7, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications. No changes shall be made in the Technical Specifications unless authorized by the Commission as provided in Section 50.59 of 10 CFR Part 50.

C. Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission's approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of the "Physical Security Plan, Nuclear Reactor Facility for University of California, Irvine," Revision 3, dated May 31, 2000, withheld from pubic disclosure pursuant to 10 CFR 2.790(d), submitted by letter dated June 2, 2000, under License R-116.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure:

Appendix A, Technical
Specifications

Date of Issuance:

Tech Specs

notice of renewal

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

UNIVERSITY OF CALIFORNIA, IRVINE

DOCKET NO. 50-326

NUCLEAR RESEARCH REACTOR

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering the issuance of a license amendment to Facility Operating License No. R-116, issued to University of California, Irvine (the licensee) for operation of their research reactor.

## ENVIRONMENTAL ASSESSMENT

### Identification of the Proposed Action

The proposed action would allow renewal of the license for 20 years for the University of California, Irvine Nuclear Reactor Facility (UCINRF). The proposed action is in accordance with the licensee's application for amendment dated October 18, 1999, as amended on April 24, and June 2, 2000. The licensee submitted an Environmental Report for license renewal.

### Need for the Proposed Action

The proposed action is needed to allow continued operation of the UCINRF in order to continue education, training, research and development using neutrons and radioisotopes for experimental purposes beyond the current term of the license.

Environmental Impact of the Proposed Action

____The research reactor is on the campus of the University of California, Irvine in Rowland Hall. Rowland Hall has research and teaching laboratories, lecture halls, classrooms, offices and workshops. It is surrounded by similar facilities in the immediate area.

The UCINRF is authorized by a NRC license to operate at steady-state thermal power levels up to a maximum of 250 kilowatts (KW). The reactor can also be operated in a pulse mode with reactivity addition of up to $3 in a short period from power levels of 1 KW or less. The construction permit was issued on May 5, 1969, and the operating license was issued on November 24, 1969. The reactor has operated less than 218 effective full-power days over the approximate 30-year license period as indicted in SAR Section 1.3.2. Facility modifications have been minor as, outlined in the SAR Section 1.4. The licensee has not indicated any plans to significantly change the design or the level of usage. Since initial operation, the gaseous Argon-41 radiological release has been conservatively estimated to be less than $5.9 \times 10^{9}$ becquerels per year (0.160 curies per year). Average concentrations of Argon-41 were conservatively estimated by the licensee as $2.4 \times 10^{-9}$ microcuries/milliliter. This concentration is well below the 10 CFR 20 Appendix B Table 2 limit of $1.0 \times 10^{-8}$ microcuries/milliliter. Since 1992, the facility has had no radiological liquid or solid radiological releases. Material has been stored as required. Releases of radioactive material have been transferred and disposed of in accordance with the requirements of the licensee's byproduct license. Any necessary releases will be similarly treated. Currently, there are no plans to change any operating or radiological release practices or characteristics of the reactor during the license renewal period.

The NRC concludes that conditions are not expected to change and that the radiological effects of the continued operation will continue to be minimal. The radiological exposures for facility operations have been within regulatory limits and should continue to remain so.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

As for potential non-radiological impacts, the proposed action does not involve any historic sites. It does not affect non-radiological effluents and has no other environmental impact. Therefore, no significant non-radiological environmental impacts and associated with the proposed action.

In addition, the environmental impact associated with operation of research reactors has been generically evaluated by the staff and is discussed in the attached generic evaluation. This evaluation concludes that no significant environmental impact is associated with the operation of research reactors licensed to operate at power levels up to and including 2 megawatts thermal. We have determined that this generic evaluation is applicable to operation of the UCINRF and that there are no special or unique features that would preclude reliance on the generic evaluation.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

The alternative to the proposed action for the research reactor facility is to deny the application. If the NRC denied license renewal, UCINRF operations would stop and decommissioning would be required with, likely, a small impact on the environment. The environmental impacts of the proposed action and alternative are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the safety analysis and evaluation for construction permit issuance and operating license issued in 1969.

Agencies and Persons Contacted

On July 25, 2000, the staff consulted with the California Department of Health Official, Steve Hsu, regarding the environmental impact of the proposed action. The State officials had no comment.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated October 18, 1999, as amended on April 24, and June 2, 2000. A hard copy is available for public inspection at the NRC's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555. Publicly available records will also be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 25th day of August 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications, and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

# ENVIRONMENTAL CONSIDERATIONS REGARDING THE LICENSING OF RESEARCH REACTORS AND CRITICAL FACILITIES

## Introduction

This discussion deals with research reactors and critical facilities which are designed to operate at low power levels, 2 MWt and lower, and are used primarily for basic research in neutron physics, neutron radiography, isotope production, experiments associated with nuclear engineering, training and as a part of a nuclear physics curriculum. Operation of such facilities will generally not exceed a 5-day week, 8-hour day, or about 2000 hours per year. Such reactors are located adjacent to technical service support facilities with convenient access for students and faculty.

Sited most frequently on the campuses of large universities, the reactors are usually housed in already existing structures, appropriately modified, or placed in new buildings that are designed and constructed to blend in with existing facilities. However, the environmental considerations discussed herein are not limited to those which are part of universities.

## Facility

There are no exterior conduits, pipelines, electrical or mechanical structures or transmission lines attached to or adjacent to the facility other than for utility services, which are similar to those required in other similar facilities, specifically laboratories. Heat dissipation is generally accomplished by use of a cooling tower located on the roof of the building. These cooling towers typically are on the order of 10' x 10' x 10' and are comparable to cooling towers associated with the air-conditioning systems of large office buildings.

Make-up for the cooling system is readily available and usually obtained from the local water supply. Radioactive gaseous effluents are limited to Ar-41 and the release of radioactive liquid effluents can be carefully monitored and controlled. Liquid wastes are collected in storage tanks to allow for decay and monitoring prior to dilution and release to the sanitary sewer system. Solid radioactive wastes are packaged and shipped offsite for

storage at NRC-approved sites. The transportation of such waste is done in accordance with existing NRC-DOT regulations in approved shipping containers.

Chemical and sanitary waste systems are similar to those existing at other similar laboratories and buildings.

Attachment

Environmental Effects of Site Preparation and Facility Construction

Construction of such facilities invariably occurs in areas that have already been disturbed by other building construction and, in some cases, solely within an already existing building. Therefore, construction would not be expected to have any significant effect on the terrain, vegetation, wildlife or nearby waters or aquatic life. The societal, economic and aesthetic impacts of construction would be no greater than those associated with the construction of a large office building or similar research facility.

Environmental Effects of Facility Operation

Release of thermal effluents from a reactor of less than 2 MWt will not have a significant effect on the environment. This small amount of waste heat is generally rejected to the atmosphere by means of small cooling towers. Extensive drift and/or fog will not occur at this low power level.

Release of routine gaseous effluents can be limited to Ar-41, which is generated by neutron activation of air. Even this will be kept as low as practicable by using gases other than air for supporting experiments. Yearly doses to un-restricted areas will be at or below established guidelines in 10 CFR Part 20 limits. Routine releases of radioactive liquid effluents can be carefully monitored and controlled in a manner that will ensure compliance with current standards. Solid radioactive wastes will be shipped to an authorized disposal site in approved containers. These wastes should not require more than a few shipping containers a year.

Based on experience with other research reactors, specifically TRIGA reactors operating in the 1 to 2 MWt range, the annual release of gaseous and liquid effluents to unrestricted areas should be less than 30 curies and 0.01 curies, respectively.

No release of potentially harmful chemical substances will occur during normal operation. Small amounts of chemicals and/or high-solid content water may be

released from the facility through the sanitary sewer during periodic blowdown of the cooling tower or from laboratory experiments.

Other potential effects of the facility, such as aesthetics, noise, societal or impact on local flora and fauna are expected to be too small to measure.

Environmental Effects of Accidents

Accidents ranging from the failure of experiments up to the largest core damage and fission product release considered possible result in doses that are less than 10 CFR Part 20 guidelines and are considered negligible with respect to the environment.

Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation involve the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of these unavoidable effects.

Alternatives to Construction and Operation of the Facility

To accomplish the objectives associated with research reactors, there are no suitable alternatives. Some of these objectives are training of students in the operation of reactors, production of radioisotopes, and use of neutron and gamma ray beams to conduct experiments.

Long-Term Effects of Facility Construction and Operation

The long-term effects of research facilities are considered to be beneficial as a result of the contribution to scientific knowledge and training. Because of the relatively small amount of capital resources involved and the small impact on the environment, very little irreversible and irretrievable commitment is associated with such facilities.

Costs and Benefits of Facility Alternatives

The costs are on the order of several millions of dollars with very little environmental impact. The benefits include, but are not limited to, some combination of the following: conduct of activation analyses, conduct of neutron radiography, training of operating personnel, and education of students. Some of these activities could be conducted using particle accelerators or radioactive sources which would be more costly and less efficient. There is no reasonable alternative to a nuclear research reactor for conducting this spectrum of activities.

Conclusion

The staff concludes that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 MWt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities.

**SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION**

**SUPPORTING AMENDMENT NO. 7 TO**

**FACILITY OPERATING LICENSE NO. R-116**

**UNIVERSITY OF CALIFORNIA, IRVINE, TRIGA MARK I PULSING RESEARCH REACTOR**

**DOCKET NO. 50-326**

# 1 INTRODUCTION

## 1.1 Overview

The University of California, Irvine (UCI or the licensee) submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for a 20-year renewal of the Class 104c Facility Operating License (NRC Docket No. 50-326). The renewal application consisted of a letter and supporting documentation dated October 18, 1999, as supplemented on April 24, June 2, and xxxxxxxxx xx, 2000. This license renewal would authorize continued operation of the UCI TRIGA Mark I pulsing research reactor as a NRC-licensed facility.

The application included financial qualifications, the safety analysis report (SAR), proposed technical specifications (TS), the operator requalification program, the emergency plan, the security plan and the environmental report. Except for the security plan, this material is available for review in the Commission's Public Document Room. The facility's security plan is protected from public disclosure under 10 CFR Part 2.790.

In conducting its safety review, the NRC staff evaluated the facility against the requirements of 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides (RGs); relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series; and NRC guidance documents, such as NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." The NRC staff review used NUREG-1537 as the primary guidance document.

The purpose of this safety evaluation (SE) is to summarize the findings of the NRC staff's safety review for operation of the UCI TRIGA Mark I pulsing research reactor. This SE will be part of the bases for issuing an NRC license renewal. The licensee authorizes operation at steady-state thermal power levels up to a maximum of 250 kilowatts (KW). The license also authorizes operation in the pulse mode with rapid reactivity additions up to $3 from power levels of 1 KW or less.

The sections of this SE are as follows:

- Section 1 summarizes the conclusions regarding the principal safety considerations of the NRC staff SE, the history and general description of the reactor facility, information on shared facilities and equipment, comparison with similar facilities, and how the licensee complies with the Nuclear Waste Policy Act of 1982.

- Section 2 is on the site and applicable site characteristics, including demography, interaction with nearby installations and facilities, climatology and meteorology, geology and seismology, and hydrology.

- Section 3 is on the design bases of significant structures, systems, and components.

- Section 4 is on the review of the design bases of the reactor core and related components.

- Section 5 is on the design bases of the reactor coolant and associated systems, including the primary and secondary coolant systems, and the coolant makeup and purification systems.

- Section 6 is on the design bases of the ventilation system to mitigate consequences of postulated accidents at the facility.

- Section 7 is on the design bases of instrumentation and control (I&C) systems and subsystems at the facility.

- Section 9 is on the design bases of auxiliary systems, such as fuel handling and storage, and fire protection.

- Section 9 is on the design bases of auxiliary systems, such as fuel handling and storage, warning and communication, and fire protection.

- Section 10 is on the design bases of the experimental facilities and programs.

- Section 11 is on the design bases of the radiation protection and radioactive waste management programs and facilities.

- Section 12 is on the conduct of facility operations. This includes consideration of management structure and responsibilities, review activities, and other required functions such as the procedures, reporting, and operator requalification, security and emergency plans.

- Section 13 is on the bases, scenarios, and accident analyses at the reactor facility.

- Section 14 is on the TSs operating limits, conditions and other requirements for the facility.

- Section 15 is on the financial qualifications of the licensee for continuing operations and decommissioning.

- Section 16 is on previous reactor utilization.

- Section 17 summarizes the major conclusions of the NRC staff's review of the license renewal application.

### 1.2 Summary and Conclusions Regarding the Principal Safety Considerations

As part of its SE, the NRC staff considered information submitted by the licensee including past operating history. The staff also considered inspection observations. On the basis of this SE, the NRC staff reached the following conclusions:

(1) The design, testing, and performance of the research reactor structures, systems and components important acceptably continue to ensure the safe operation of the facility.

(2) The licensee's management organization, qualifications, training, experimental activities, and security measures continue to maintain the research reactor acceptably so that the facility employees, the public, and the environment are not subject to any significant radiological risk.

(3) The licensee and the NRC staff have considered the expected consequences of several postulated accidents emphasizing those likely to cause a loss of integrity of fuel-element cladding. The NRC staff confirmed conservative analyses of the most serious, hypothetically credible accidents. As a result, the NRC staff determined that the calculated potential radiation doses are within applicable criteria of 10 CFR Part 20 and 10 CFR Part 100.

(5) The radiation protection program, and the radioactive materials and wastes control program acceptably continue to control radiological exposures and concentrations within the limits and principles specified 10 CFR Part 20 including the principle of as low as reasonably achievable (ALARA).

(6) The licensee's TSs give a high degree of assurance that the facility will continue to operate acceptably according to the assumptions and analyses in the SAR. No significant degradation of equipment and administrative requirements has occurred. The TSs will

continue to ensure no significant degradation of equipment and administrative requirements.

(7) The licensee continues to have access to sufficient revenues to cover operating costs and eventually to decommission the reactor facility acceptably.

(8) The licensee's program for physical protection the facility and its special nuclear materials continues to comply with the requirements of 10 CFR Part 73.

(9) The licensee's procedures for operator requalification continue to give reasonable assurance that UCI will operate the reactor acceptably.

(10) The licensee's emergency plan provides assurance that the licensee is prepared to assess and respond to emergency events acceptably.

Based on these findings, the NRC staff concludes that the licensee can operate its TRIGA Mark I pulsing research reactor in accordance with its application without endangering the health and safety of their staff, the public and the environment.

### 1.3 History

The research reactor was originally designed and constructed for research and teaching in the field of radio-analytical chemistry and for other assigned tasks. The Construction Permit was issued on May 5, 1969, and the operating license was issued on November 24, 1969. The reactor operated less than 218 full power effective days over approximately 30 years as indicted in SAR Chapter 1.3.2. Facility modifications have been minor as outlined in the SAR Chapter 1.4. The licensee has not suggested any plans to change the design or the level usage.

### 1.4 Reactor Description

The UCI TRIGA Mark I is a heterogeneous, pool-type, pulsing research reactor. The core is in highly purified light water in an open aluminum pool 10 feet wide by 15 feet long by 25 feet deep. The pool is surrounded and supported by reinforced concrete.

All core components are between the top and bottom aluminum grid plates. The core is designed for natural convection cooling. The reactor core is reflected by light water and graphite. The reactor coolant circulates through an external heat removal and purification system.

The fuel design is similar to that used by other NRC-licensed TRIGA reactors. The uranium enrichment is less than 20 percent in the U-235 isotope. The reactor exhibits a large prompt

negative temperature coefficient, which is typical of TRIGA fuel. Four control rods control reactivity.

The reactor experimental facilities include a rotary 'lazy Susan' irradiation device, three pneumatic transfer systems, and a dry tube 'central thimble' irradiation position. Removal of certain portions of the upper grid plate and rearrangement of fuel elements in accordance with the Technical Specification limits creates other experimental facilities.

This general description from SAR Chapter 4, "Reactor Description," is consistent with the design as previously licensed and operated by the licensee.

### 1.5 Facilities and Equipment

This general description from SAR Chapter 3, "Design of Rowland Hall and Reactor Facility," is consistent with the design as previously licensed by the NRC and operated by the licensee.

### 1.6 Comparison with Similar Facilities

The UCI TRIGA Mark I pulsing research reactor is similar to other research reactors licensed to operate by the NRC. NRC licenses about 17 other TRIGA research reactors that use fuel similar or identical to that of UCI. Of these TRIGA research reactors, 5 are licensed for maximum power level approximately equal to that at UCI. About 9 of the TRIGAs are licensed at power levels greater than at UCI. NRC licenses research reactors at about 27 other nonprofit educational institutions. Of these nonprofit educational institutions, about 11 are TRIGA reactors. Several of these are sited in multipurpose educational building situations similar to that at UCI.

### 1.7 Nuclear Waste Policy Act of 1982

Section 302(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the licensee will have entered an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R. L. Morgan, DOE, informed H. Denton, NRC, that DOE had determined that universities and other government agencies operating non-power reactors have entered contracts with the DOE, if DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. Appendix C to the SAR is a letter from E. D. Houck, Idaho National Engineering and Environmental Laboratory Spent Nuclear Fuel Program, which confirms that UCI has entered such an agreement. As specified in Chapter 11.8.2 of the SAR, the plan identifies the year 2029 as final use year, and 2032 for shipment of the fuel.

### 1.8 Conclusions

The NRC staff concludes as follows:

- The design bases and safety considerations of the facility are consistent with similar non-power reactors in fuel type, thermal power level, and siting considerations. The history of these facilities shows consistently safe operation that is acceptable to the NRC staff.

- The licensee's design does not differ in any substantive way from similar facilities that have been found acceptable by the NRC.

- The licensee has used data and operational experience from similar reactor facilities, including its own, to provide assurance that the facility can operate safely as designed.

## 2 SITE CHARACTERISTICS

This section is on the site and applicable site characteristics, including demography, interaction with nearby installations and facilities, climatology and meteorology, geology and seismology, and hydrology. Site characteristics are primarily discussed in SAR Chapter 2.

### 2.1 Reactor Site

As indicated in SAR Chapter 2.1, the research reactor is on the UCI campus property. It is in Orange County about 64 km (40 miles) southeast of the center of Los Angeles, California. The UCI TRIGA Mark I pulsing research reactor is about 6.4 km (4 miles) from the coast of the Pacific Ocean. The Regents of the University own about 600 hectares (1500 acres) of land. The research reactor is in Rowland Hall. The building has research and teaching laboratories, lecture halls, classrooms, offices and workshops. Similar facilities in the immediate area surround the reactor.

### 2.2 Demography

Section 2.2 of the SAR shows Newport Beach (to the southwest), Costa Mesa (to the northwest), and Irvine within 10 km (6.2 mi) of UCI. Estimated population of this area was about 631,000 in 1999 with a projected increase to 705,000 by the year 2020. The estimated population of UCI in 1998 was about 5,600 students and about 700 faculty and staff with projected growth to 10,400 and 1,100, respectively, in 2005. The closest dwelling to the reactor is about 150 meters (500 feet) to the west.

Daytime population on the campus is about 2,500 faculty and staff, and 18,000 students with projected growth by 2005 to 4,600 faculty and staff, and 25,000 students. At other times, the population on campus is substantially below these values.

## 2.3 Nearby Industrial, Transportation, and Military Facilities

These topics consider SAR Chapters 2.1 and 2.3.

### 2.3.1 Industry

The industry in the area is similar to that occurring in UCI research facilities of Rowland Hall and adjacent buildings. Several reservoirs are near the campus with the closest being 1.6 km (1 mile) away. These facilities have not affected the research reactor.

### 2.3.2 Transportation

Highway Transportation. The campus borders state highway 73. Interstate highway I-805 is about 1.6 km (1mile) from the campus.

Airports. Orange County Airport is about 3.7 km (2.3 miles) away. There were 3.5 million takeoffs and landings in 1995 and this number is expected to rise to 5.9 million by 2005.

There is reasonable assurance that operation of these facilities will not affect reactor operation based on experience and the distances and nature of these transportation facilities.

### 2.3.3 Military Facilities

El Toro Marin Base about 11 km (7 miles) away is being converted to civilian use. Santa Ana Marine Corps Air Facility is closed. Based on experience and the distances and lack of potential impact on the reactor due to the structure around the pool as discussed for seismic considerations in this SE, the NRC staff does not expect these facilities to impact future operations of the research reactor.

## 2.4 Climatology and Meteorology

Climatology and meteorology have not affected the research reactor.

### 2.4.1 Climatology

Section 2.3 of the SAR shows that extremes in the climate around the UCI campus are rare. From the licensee presented site hazard analysis, the only windstorm in the area was about 3.4 km (2.1 miles) southwest of the campus in the 1970- to 1990-time frame. The windstorm was measured at between about 130 and 160 km per hour (80 and 100 miles per hour). Other climatological hazards (i.e., hail storms, tornados and hurricanes) from the licensee referenced data sources were examined by the NRC staff from a historical perspective. This data confirmed a very low possibility that such hazards could affect the facility.

### 2.4.2 Meteorology

The licensee showed that temperature inversions are frequent. These inversions could affect atmospheric dispersion. The NRC staff confirmed that the licensee conservatively dealt with the effect of inversions in the dose assessment by estimating doses at the point of release to the environment. Also, the license conservatively estimated the atmospheric dispersion and further demonstrated acceptable radiological consequences. Based on experience of the facility and the NRC staff review of related data, meteorological conditions are not expected to affect future operations of the research reactor.

## 2.5 Geology and Seismology

### 2.5.1 Site Geology

In SAR Chapter 2.4.3 the licensee referenced research reactor site test borings. These borings in addition to the reference material in SAR Chapter 2.4, “Geology,” confirmed a stable, well-founded site geology with no ground water.

### 2.5.2 Seismology

The licensee’s SAR showed that the region is “relatively” stable and an official hazard map showed it to be a low hazard area. Further, as described earlier in this report, the UCI TRIGA Mark I pulsing research reactor is in Rowland Hall. The building was designed and built to meet or exceed building code requirements in 1964. These building code requirements for seismic consideration are being reevaluated as said in SAR Chapter 3.1. The Rowland Hall structures may suffer some damage from a seismic event of the highest possible yield, but the resultant radiological doses would be within the ranges evaluated in Section 13, Accident Analysis. This is based on the evaluation that the core and pool damage would not occur because of the large reinforced concrete structure and supporting structure as documented in Section 3.1 of this SE.

However, Chapter 2.5, “Seismology,” of the SAR says that a seismic hazard analysis update from the previous analysis is planned. XXXXXXXXXXXXXX new submittal by licensee XXXXXXXXXXXXXXX

## 2.6 Hydrology

SAR Chapter 2.5 shows the research reactor site is not subject to flooding potential. The NRC staff confirmed this through review of the referenced material in the SAR.

## 2.7 Conclusions

The NRC staff concludes as follows:

- The licensee has provided sufficient information to describe the geography and demography surrounding the research reactor accurately, and the information is sufficient to assess the radiological impact resulting from the location and operation of the reactor. There is reasonable assurance that no geographic or demographic features will render the site unsuitable for continued operation of the reactor.

- The licensee has discussed or shown nearby manmade facilities and activities (i.e., industrial, transportation, and military) that have a potential to pose a hazard to reactor operations. There is reasonable assurance that operation of these facilities will not affect reactor operation.

- Meteorological history and projections were factored into the design of the reactor building, such that no weather-related event is likely to cause damage to the reactor and a release of radioactive material. The meteorological information is sufficient to evaluate dispersion calculations conservatively and calculate the consequences of releases from routine operations or postulated accidents.

- UCI provided information on the geology and hydrology of the UCI site in sufficient detail and would not affect the design bases for structures, systems, and operating characteristics of the reactor.

- Information in the UCI SAR shows that damaging seismic activity at the reactor site during the term of the license is very unlikely. Considering the facility design if seismic damage to the building was to occur, radiological consequences would be bounded as analyzed in Chapter 13 of the SAR. The UCI SAR shows that there is no significant likelihood that the licensee's staff, the public and the environment would be subject to undue radiological risk from seismic activity; therefore, the site remains suitable for the research reactor.

# 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This section is on the design bases of significant structures, systems, and components. The licensee's discussion of the design of structures, systems and components is primarily in SAR Chapter 3.

## 3.1 Description of the Reactor Facility

The research reactor facility includes five rooms (i.e., a supervisor's office, the control room, a "wet" laboratory counting room, a "dry" laboratory counting room, and the reactor room). The "wet" laboratory has a fume hood and a pneumatic transfer system send and receive station. The control room has an emergency exit door. The control room has monitoring and control instrumentation plus a door and window to the reactor room.

The research reactor core is under about 6.1 meters (20 feet) of water in a pool that is below ground level in the reactor room. The area immediately above the pool has historical dose rates less than 0.0005 mSv/hr (0.05 mrem/hr). The aluminum pool liner is 0.635 cm (0.25 in) thick on the sides and 1.27 cm (0.5 in) thick on the bottom. The reinforced concrete shield around the pool varies from about 0.762 to 1.37 meters (2.5 to 4.5 feet) thick. The licensee considered that the design is more than adequate to endure an earthquake induced ground motion. The basic purpose of these massive concrete structures is to provide biological shielding for personnel working in and around the facility. These massive structures also provide acceptable protection for the reactor core against external man made and natural phenomena.

## 3.2 Air Confinement/Ventilation System and Radiation Monitoring System

In accordance with SAR Chapter 3.6, the reactor room design maintains a negative air pressure difference of at least 0.037 kPa (0.15 inches of water). The ventilation system exhausts air from the reactor room to the top of Rowland Hall, the unrestricted environment. TS 5.2 requires the reactor room to be closed to restrict leakage when in operation, the facility is unmanned, or spent fuel is being handled exterior to a cask. Also, TS 5.2 limits minimum free volume in the reactor room to 1000 cubic feet. Further, this TS limits the ventilation exhaust point to at least 70 feet above ground level. The ventilation system design controls the reactor room exhaust through normal or emergency operations. Doors from the reactor room have rubber seals to ensure confinement. This design and operational modes ensure that potential releases of radiation satisfy the requirements of 10 CFR Part 20. They also ensure acceptable radiological concentrations at the point of release as specified in TS 3.6 basis. The design of the radiation monitoring system ensures actuation of the emergency mode as specified in SAR Chapter 3.7. The system has also continued to ensure that the temperature and relative humidity have been acceptable through operations since original licensing.

### 3.3 Electro-Mechanical Systems and Components

The control rod motors, electromagnets, gear boxes, switches, and wiring are all above the pool water level and are readily accessible for visual inspection, testing, and maintenance. The licensee's preventive maintenance and surveillance program ensures that all mechanical systems and components important to safety continue to meet the performance requirements of the TSs and SAR.

### 3.4 Conclusions

The NRC staff concludes as follows:

- The design bases and operation since original licensing of the facility structures, systems and components give reasonable assurance that they will function as designed to ensure the continued safe operation and safe shutdown of the reactor.

- Surveillance activities proposed in the TSs continue to acceptably ensure the safety-related functions of the facility structures, systems and components.

## 4 REACTOR

This section is on the review of the design bases of the reactor core and related components presented in SAR Chapter 4.

### 4.1 Introduction

The UCI TRIGA Mark I is a fixed-core, pool-type, pulsing research reactor. It uses light water as the moderator, coolant, and shield. The NRC authorizes UCI to operate the reactor in the steady-state mode at thermal power levels up to and including 250 kilowatts. In addition, the NRC authorizes pulse mode of operation with a maximum reactivity addition of $3.

The reactor USES's standard TRIGA low enriched fuel with stainless steel cladding. The core itself is near the bottom of the pool. Inserting or withdrawing four neutron-absorbing control rods suspended from the drive mechanisms controls reactor power. A fixed structure that supports the control rod systems, reactor instrumentation, and some experimental facilities spans the pool.

## 4.2 Reactor Core

The UCI TRIGA Mark I pulsing research reactor core consists of the fuel-moderator assemblies (including the instrumented element), reflector assemblies, grid plates, neutron source, control rods, and experimental facilities as described in SAR Chapter 4.

### 4.2.1 Fuel-Moderator Element

The active part of each fuel-moderator element has a diameter of approximately 3.63 cm (1.43 in) and is 38.1 cm (15 in) long. The reactor fuel is a homogeneous mixture of a uranium-zirconium hydride alloy containing 8.5 wt% uranium enriched to less than 20% U-235. The hydrogen-to-zirconium atom ratio within the UCI fuel is approximately 1.7. The hydrogen in the alloy is a neutron moderator. This moderator is mixed with the fuel in a solid form which results in the moderator having the same operating conditions as the fuel. This design feature of the TRIGA fuel contributes to the ability to safely pulse the reactor.

Each element is clad with a stainless steel can that is 0.0508 cm (0.020 in) thick. The can contains two sections of graphite, one above and one below the fuel, to serve as top and bottom neutron reflectors for the core. Each element has stainless steel end fixtures, making the fuel-moderator element approximately 72.06 cm (28.37 in) long. Fuel element physical dimensions, transverse bend and elongation, are limited as specified in TS 4.1. The total weight of a fully loaded fuel element with end fixtures is about 3.4 kg (7.5 lb).

An instrumented fuel-moderator element has three thermocouples embedded in the fuel (the fuel-moderator design in this element is the same as the other fuel elements in the core). This element monitors fuel temperature which is the variable upon which the safety limit is placed. The sensing tips of the fuel element thermocouples are about halfway between the zirconium rod and the cladding around the horizontal center of the fuel section. TS 3.3, requires that the instrumented fuel element be operable for all operating modes. The thermocouple wires pass through a seal in the upper end fixture. A tube provides a watertight conduit that carries the thermocouple wires above the surface of the water in the reactor pool.

### 4.2.2 Reflector

The neutron reflector is a ring-shaped block of graphite that surrounds the core radially. The graphite has a diameter of approximately 53 cm (20.875 in) and a height of about 57.15 cm (22.5 in). A leak-tight welded-aluminum can protects the graphite from water penetration. A well is in the top for the rotary specimen rack. The reflector assembly rests on an aluminum platform about 0.61 cm (2 feet) from the bottom of the pool.

### 4.2.3 Grid Plates and Safety Plate

The top grid plate is an aluminum plate. The plate provides accurate lateral positioning for the core components. The top grid plate is on a ring welded to the top inside surface of the reflector container. The 126 holes for placement of the fuel elements are in six concentric rings around a central hole.

The bottom grid plate is an aluminum plate that supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements.

A welded in place, safety plate prevents control rods falling out of the core if all other support is lost.

### 4.2.4 Graphite Reflector Elements

Graphite reflector elements may fill grid positions not filled by the fuel-moderator elements or other core components.

### 4.2.5 Neutron Source

An americium-beryllium neutron source is for reactor startup. The source material is double encapsulated. The neutron source fits in specific locations.

## 4.3 Control Rods and Drives

### 4.3.1 Control Rods

The UCI research reactor has four control rods. For steady state operation, two motor driven rods (one shim rod and one regulating rod) control reactor power. The uppermost section is graphite, followed by 38.1 cm (15 in) of graphite impregnated with a boron carbide. Immediately below this neutron absorber section is a fuel section consisting of 38.1 cm (15 in) of standard fuel material. The bottom section is 16.51 cm (6.5 in) of graphite. Stainless steel cans that are approximately 109.22 cm (43 in) long with a diameter of about 3.4925 cm (1.375 in) contain these sections. TS 3.1.h specifies maximum allowable drop times to ensure a timely shutdown.

The fast transient rod is for pulses and acts as a scrammable safety rod for steady state operations. A pneumatic system drives the fast transient rod. The fast transient rod has no travel adjustment capability. A double length borated graphite section is in an aluminum can. Therefore, initial acceleration has no significant change in reactivity.

The adjustable transient rod provides fine control for potential pulse operations. This rod is about 93.98 cm (37 in) long with a 38.1 cm (15 in) section of borated graphite neutron absorber and a 53.34 cm (21 in) long air space. It has a pneumatic-electro-mechanical drive system with an adjustable travel. The adjustable transient rod is also a scrammable safety rod.

The shim rod, regulating rod, and the adjustable transient rod have travels of about 35.56 cm (14 in). The fast transient rod has about a 76.2 cm (30 in) travel. The adjustable and fast transient rods are designed to be used together for increased pulsing capability.

### 4.3.2 Control Rod Drives

The rods are attached to drive assemblies mounted on a raised bridge. For the shim and regulating rods, the drive assembly consists of a motor-driven rack and pinion. The rods and rod extensions are connected to the rack through an electromagnet and armature. On power failure or scram signal, the electromagnet is de-energized and the rods are driven into the core by gravity. Rod insertion and withdrawal speeds and travel limits are established. Position indication is also provided.

The fast transient rod can be rapidly removed from the core using compressed air. A hydraulic dashpot provides deceleration at the end of the stroke. The rod is designed to fall freely back into the core after a pulse. The rod can also be used as a safety rod.

The adjustable control rod drive mechanism is pneumatic/electro-mechanical system. The pneumatic portion drives a piston in a cylinder to an anvil of a shock absorber to give the power pulse. The anvil is position by the electro-mechanical portion of the system through a motor

driven ball nut assembly and worm gear. This rod can be used as a steady-state control and safety rod as well as a transient rod.

### 4.4 Reactor Pool

The UCI research reactor core is located in an aluminum pool surrounded by a reinforced concrete structure as previously discussed in Section 3.1 of this SE. The reactor pool is a welded aluminum vessel. The pool is welded for water tightness. Chapter 3.4.1 of the SAR, extensive leak testing procedures coupled with weld radiography at 20 percent of the welds assured integrity of the pool liner. Additionally, the outside wall of the pool liner was double wrapped with hot tarred felt to provide a water barrier to the surrounding concrete is coated with a tar material for corrosion protection. No indication of significant pool integrity loss has been observed since the reactor was originally licensed.

### 4.5 Reactor Instrumentation

The UCI research reactor instrumentation is similar to that found on research reactor installations at other locations. The control console and associated instruments are typical of those in use at several TRIGA research reactors that were licensed in the time frame that UCI was originally licensed. The nuclear instrumentation gives the operator the necessary information for proper manipulation of the controls. The following instrumentation functions are provided and are discussed in more detail in Section 7 of this SE.

- Wide Range Linear Monitor
- Wide Range Monitor
- Power Range Monitor

### 4.6 Nuclear Design

#### 4.6.1 Steady State Operations

The licensee analyzed the nuclear response of the core. The codes used for these calculations had been benchmarked against measurements from other TRIGA reactors. The calculations showed a negative reactivity response with increasing power. This negative temperature coefficient allows operational flexibility in steady-state operation as the effect of accidental reactivity changes occurring from experimental devices or other incidences is immediately and significantly mitigated. This temperature coefficient primarily arises from a change in the fuel utilization factor resulting from the heating of the uranium-zirconium hydride fuel-moderator elements (less neutrons available to cause fission). The coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise simultaneously. The heating of the moderator mixed with the heating of the fuel causes the spectrum to harden more in the fuel than in the water, which increases the leakage of neutrons from the fuel into the water moderator surrounding the fuel, where they are

absorbed preferentially. This yields a loss of reactivity. An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of the uranium-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances. The calculations also showed substantial margin to the limiting safety system setting (TS 2.2) in operating temperatures. The calculated maximum temperature was approximately 200°C (392°F) compared to the limiting safety system settings of 755°C (1391°F) for fuel temperature measurements in the C ring (or 800°C (1472°F) for B ring measurements).

The licensee estimated fuel element worth ranging from $1.16 (0.81 percent Δk/k) in the B ring to $0.21 (0.15 percent Δk/k) in the G ring. The licensee also presented control rod worth data as of February 2, 1999. The regulating rod worth was $2.79 (1.95 percent Δk/k), the shim rod worth was $3.70 (2.59 percent Δk/k), the fast transient rod worth was $0.70 (0.49 percent Δk/k), and the adjustable transient rod worth was $1.82 (1.27 percent Δk/k). The licensee also made estimates of reactivity worth for various experimental facilities. Various TSs limit reactivity worth to ensure safe operation and that potential consequences are bound by the accident analyses discussed later in this SE.

### 4.6.2 Pulse Mode of Operation

The basic parameter which allows the TRIGA reactor system to operate safely with large step insertions of reactivity is the previously discussed prompt negative temperature coefficient associated with the TRIGA fuel and core design. TS 3.1 limits reactivity for the transient rods (pulses) to $3.00 (2.1 percent Δk/k). The licensee has calculated in SAR Chapter 4.5.6 the effects of pulses on a TRIGA. Fuel temperature was less than 450°C (842°F). This provides considerable margin to the TS 2.1 safety limit of 1000°C (1832°F). TRIGA reactors have been acceptably pulsed with reactivity additions up to $5.00 (3.5 percent Δk/k).

### 4.6.3 Shutdown Margin, Excess Reactivity, and Experiment Reactivity Worth

The TS 3.1.a limit on the minimum shutdown margin ensures that the reactor can be safety shutdown from any operational configuration, even if the highest worth control rod remains stuck out of the core. This minimum shutdown margin of $0.50 (0.35 percent Δk/k) will ensure that the reactor can be shut down and remain shut down. This minimum shutdown margin must be met with the reactor in any core condition, with the most reactive control rod assumed to be fully withdrawn. The value of $0.50 is a standard value for shutdown margin that is measurable.

The total excess reactivity that UCI is authorized to have loaded into the TRIGA reactor during operation is $3.00 (TS 3.1.b). This amount provides for the various negative reactivity effects associated with operation and use of the reactor, as well as allowing some operational flexibility. The limit on excess reactivity helps ensure that the safety analysis report assumptions and analyses are applicable to all operational cores.

The TS 3.1 limits the combined absolute reactivity worth of all experiments or one individual experiment to less than $3.00 (2.1 percent Δk/k). This limit ensures any experimental failure would be within the previously analyzed limits for pulsed operation and those analyzed in accident analyses in SAR Chapter 13. Further, TS 3.1.c limits the absolute reactivity worth of unsecured experiments to less than $1.00 (0.7 percent Δk/k) per experiment. The restraining forces for experiments greater than $1.00 must prevent unplanned removal from or insertion into the reactor. Secured experiments are limited by the proposed TS to a worth of $1.75. This worth is less than the positive reactivity insertion limit of the pulse analyzed in the SAR Chapter 13 (i.e., pulse that would be needed to reach the fuel temperature safety limit). Finally in this regard, the total reactivity worth of all experiments is limited by TS 3.1.f so that the shutdown margin at cold xenon-free condition with all rods in is at least $0.50 (0.35 percent Δk/k).

### 4.7 Conclusions

The NRC staff concludes as follows:

- The design of the core-related components for the UCI facility are acceptable and should continue to permit safe operation and shutdown of the reactor.

- The reactor pool and attachments are designed to ensure continued safe reactor operation and minimize the possibility of a pool failure that could result in loss of coolant. The design features of the pool and operation since original licensing offer reasonable assurance of continued reliable performance for the period of the license.

- The licensee has proposed limits on pulsing the reactor. The maximum reactivity addition for pulsing will ensure that the reactor can be safely pulsed without fuel damage.

- The licensee has discussed and proposed minimum shutdown margin and excess reactivity limits that are acceptable to the NRC staff. The minimum shutdown margin ensures that the reactor can be shutdown from any operating condition with the highest worth control rod stuck out of the core. The limit on excess reactivity allows operational flexibility while limiting the reactivity available for reactivity addition accidents.

- Reactivity limits on experiments have been proposed by the licensee. The licensee has proposed values that are bounded by the pulse reactivity addition analysis. Therefore, failure of experiments will not add unacceptable amounts of reactivity to the reactor.

- The fuel and core design are acceptable, when considered with the restrictions and requirements on the operation of the reactor. Given the conditions required by TSs, there is reasonable assurance that the UCI TRIGA research reactor will continue to be operated safely at power levels up to 250 kilowatts thermal and with reactivity additions in the pulse mode of up to $3.00, as limited by the license and TS requirements.

- The information provided in the UCI SAR gives reasonable assurance that the reactor can continue to be operated at its licensed power level without undue risk to the health and safety of the licensee's staff, the public and the environment.

## 5 REACTOR COOLANT SYSTEMS

This section is on the design bases of the reactor coolant and associated systems, including the primary and secondary coolant systems, and the coolant makeup and purification systems. The licensee described the reactor coolant systems in SAR Chapter 5.

### 5.1 Primary Coolant System

The reactor core is cooled by the natural circulation of water in the reactor pool. The pool water is cooled on an as needed basis by the external primary and secondary cooling system. The licensee administratively limits pool water temperature to less than 30°C (86°F) and has maintained it below 25°C (77°F).

TS 3.7 requires that the water level be maintained at least 5.8 m (19 ft) above the reactor top grid plate to provide necessary shielding for the core. TS 4.4 requires an operable pool level measuring channel.

The water is pumped through a filter and ion exchanger to maintain clarity and purity. Purity is assessed through conductivity measurements. Make up water is from Rowland hall de-ionized water system.

The accident analysis for loss of reactor pool water (SAR Chapter 8.4) indicates that the cooling system and purification system pool suctions are both less than 0.912 m (3 ft) below the normal pool water level. Thus, pumping out water would only reduce water level by about 0.912 m. All components of the cooling and purification systems are above pool level, so that syphoning out of the pool water is not possible. Additionally, alarms are monitored to detect potential loss of pool water.

This system also provides flow to a Nitrogen-16 diffuser as describe in SAR Chapter 6.3.2.1, and discussed in this Section 11 of this SE.

### 5.2 Conclusions

The NRC staff concludes the following:

- There is reasonable assurance that credible and postulated malfunctions of the cooling system will not lead to uncontrolled loss of primary coolant.

- The Technical Specifications provide reasonable assurance that the cooling system will continue to operate as designed and be acceptable for normal reactor operations as described in the SAR.

## 6 ENGINEERED SAFETY FEATURE

This section is on the design bases of the ventilation system to mitigate consequences of postulated accidents at the facility.

### 6.1 Ventilation System

The ventilation system as described in SAR Chapter 3.6.2 has design features that are incorporated for accident mitigation that make the system an engineered safety feature. The ventilation system normal exhaust is closed and emergency purge is initiated on detection at preset radiation levels. Air exhaust rate is reduced and filtered, i.e., an independent fan exhausts the air through a filter. The pneumatic transfer system is also shut down. Doors from the reactor room have rubber seals to ensure confinement. The system is designed to operate at negative pressure and ensure control of radioactive material under potential accident conditions. These features are activated when the continuous air monitor measures radioactive gamma exceeding preset limit in the reactor room. TS 3.6 provide limiting conditions of operation for the ventilation system that reduce the potential for radiological release and provide acceptable assurance that potential accident conditions would be within limits as reviewed in Section 13 of this SE.

### 6.2 Conclusions

The NRC staff concludes as follows:

- The ventilation system will continue to limit the spread of radioactive contamination, and provide the means for isolating, recirculating, and filtering the air in the reactor room under potential accident conditions.

# 7 INSTRUMENTATION AND CONTROL

## 7.1 Introduction

Instrumentation and control (I&C) systems comprise the sensors, electronic circuitry, displays, and actuating devices that provide the information and the means to safely control the reactor and to avoid or mitigate accidents. The I&C system is primarily described in SAR Chapter 7 and have acceptably functioned since original licensing.

## 7.2 Control Console and Display Information, and Reactor Control System

The control console is a desk-type with controls or indications for control rod drives, facility interlocks, and neutron detectors. An adjacent cabinet provides indication or control for remote area radiation monitoring and other alarms.

Control console power, except to the 25 volt direct current power supply, the clock and the ion chamber power supply, is provided through the “POWER ON” switch. The power supply and clock are left on even when the console is not in operation through a separate circuit breaker.

A mode selector switch can be used to operate the reactor in mode 1, manual or automatic (not presently connected) operations, or in mode 2, pulse operations. The pulse mode involves the conditions that require the use of the transient (pulse) rods previously discussed in Section 4. The manual mode is used for steady-state operation, reactor startup and changes in power level, while the automatic mode has the potential to be used for steady-state operation. A percent demand control provides a power level control set point in the automatic mode. All instruments, safety, and interlock circuitry for manual operation apply to operation in automatic mode.

Instruments and components are provided to monitor, indicate, record and/or control neutron flux density, fuel temperature, coolant temperature, and radiation intensities in selected areas around the reactor.

A key switch controls the current to the recorder for logarithmic and linear indication of reactor power level, and to the rod magnets. This key switch must be on to withdraw any of the rods. Manual rod control is accomplished by switches. Position indication is provided for the regulating, shim and adjustable transient rods. Controls for pulsing the transient rods are through separate arm switches and a fire switch. A delay circuit is provided to ensure coordination between the two transient rods when used together for a pulse. Indication for the transient rods is provided for armed and up conditions.

## 7.3 Reactor Protection System

Interlocks ensure neutron indication (minimum source strength), prevent withdrawal of two rods simultaneously, prevent pulses at greater than 1 kilowatt, and ensure acceptable mode and rod positions for pulses. Interlocks do not inhibit inserting the control rods. Annunciation is provided for the minimum source range interlock, and operation of the cooling, purification and lazy susan systems. Annunciators also provide indication of seven scrams from the wide range monitor channel, wide range linear monitor channel, the power range monitor channel, the manual scram bar, the loss of detector voltage, loss of console power, and seismic switch.

TS 3.4 requires scrams on actuation of fuel element temperature, reactor power level, manual scram button, and seismic switch.

The instrumented fuel-moderator element previously discussed is connected to the water fuel temperature meter on the console. This meter indicates fuel temperature during operations as required by TS 3.3. A scram circuit is associated with this circuit on high fuel temperature as required by TS 3.4.

Reactor power level indication and scrams are provided by the wide range linear monitor, the wide range monitor, and the power range monitor.

The wide range monitor receives its input from a fission counter. The circuit converts the signal to logarithmic value over the range from $10^{-8}$ to 200 percent of full power. This monitor provides input for the source range interlock required by TS 3.4 for “Startup Power Level.” It also provides several indications of power level including the recorder value previous mentioned. The wide range monitor also provides digital period indications and a adjustable period trip (although this trip is not required by TS). It also trips the reactor during calibration or on loss of the detector high voltage (the detector high voltage is indicated on the console).

The wide range linear power monitor gets its input from a compensated ion chamber. Its signal is displayed in several indications of reactor power to the operator including the previously mentioned recorder. Indication is from below source range to full power. The operator can switch range manually or can select auto so it switches range automatically (except in the top range). The wide range linear power monitor can provide input to the automatic reactor flux control, although it is not currently connected. This monitor also provides two adjustable power level scram set points. One set at 105 percent of range and one to prevent pulsing at power levels greater than 1 kilowatt (TS 3.4). It also trips the reactor during calibration or on loss of the detector high voltage (the detector high voltage is indicated).

The power range channel gets it signal from an uncompensated ion chamber. It provides indication from 0 to 125 percent power on a digital indicator and a bar graph. An adjustable high power level trip is provided. It can be used to measure pulse energy level. It trips the reactor on loss of detector high voltage.

The seismic switch is set to approximately a modified Mercalli Intensity III event motion. Need description of instrument?

TS 4.3 provides surveillance requirements to ensure reactor protection system function.

### 7.4 Essential Safety Feature Actuation System

The emergency operation of the ventilation system is automatically initiated on detection of gamma radiation by a continuous air monitor signal as discussed in SAR Chapter 3.7.1. The continuous air monitor located next to the reactor bridge draws air through a tube under the bridge over the pool water. It provides local audio and visual alarms, and send a high level alarms to the security monitoring station (monitored continuously). Electric power is backed up by the emergency diesel generator. TS 3.3 requires the continuous air monitor in all modes of operation and TS 4.5 requires daily verification of alarm set points for operation.

### 7.5 Radiation Monitoring System

The radiation monitoring system is discussed in Chapter 3.7.2 of the SAR. It has sensors located throughout the facility to monitor potential radioactive releases. Each sensor operated independently and has failure, alert and alarm settings and audio and visual indications. High level alarms are also sent to the security monitoring station (monitored continuously). Electric power is backed up by the emergency diesel generator. TS 3.3 requires at least two of the six sensors to be operable in all modes of operation and TS 4.5 requires daily verification of alarm set points for operation.

### 7.6 Conclusions

The NRC staff concludes as follows:

- The I&C system, and its subsystems and components will continue to give all necessary information to the operator or to automatic devices to maintain planned control for the full range of normal reactor operations and analyzed potential accident conditions.

- The SAR describes equipment, systems, and devices that will give reasonable assurance that all radioactive sources will be identified and accurately evaluated, and acceptably controlled.

- The I&C design is such that credible malfunctions in its components would not prevent the reactor protection system from performing necessary functions, nor prevent safe shutdown of the reactor.

- The locking system on the control console reasonably ensures that the reactor facility will continue to be operated by authorized personnel.

- Discussions of testing, checking, and calibration provisions, and the bases of technical specifications including surveillance tests and intervals give reasonable confidence that the I&C system will continue to function as designed.

- The I&C system is designed to prevent or mitigate hazards to the reactor or escape of radiation. It is desinged so that there will be no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

## 8 ELECTRICAL POWER

### 8.1 Normal Electrical Power System

The routine electric power is supplied by the UCI system. This system has at least two major feed. UCI maintaining a reliable electric power source is discussed in SAR Chapter 8.

### 8.2 Emergency Electrical Power System

Emergency electrical power is provided to the continuous air monitoring system, the radiation monitoring system, the security monitoring system, associated alarms, and emergency lighting in the control room and reactor room. This power is provided by an emergency diesel generator. The emergency diesel generator continues to be maintained and tested to ensure that it operates on demand. Loss of electric power results in reactor scram and no cooling of the fuel is required after normal shutdown or under analyzed potential accident conditions.

### 8.3 Conclusions

The NRC staff concludes as follows:

- The design bases and functional characteristics of the electrical power systems will continue to provide all required electrical services.

- The design of electrical systems provides reasonable assurance that use or malfunction could not cause reactor damage or prevent continued safe reactor operation or shutdown conditions.

# 9 AUXILIARY SYSTEMS

This Section is on the design bases of auxiliary systems, such as fuel handling and storage, and fire protection.

## 9.1 Fuel Storage and Handling

Most fuel storage is in the pool. Underwater lights are provided illumination. Storage racks in the pool have the capacity for more than a full core offload. The racks are positioned to ensure an acceptable minimum water shielding for irradiated fuel. The racks are separated and arranged to prevent potential criticality. Five fuel storage pits in the reactor room floor are capable of storing 19 elements. The pits are designed to ensure shielding and prevent criticality. TS 5.3 ensures sub-critical conditions ($k_{eff}$<0.8 under all conditions of moderation) and cooling for fuel storage.

A grappling tool is provided for fuel movement. The grappling tool has a positive locking with a release on the tool handle. A hoist, as described in SAR Chapter 6.2, is installed to assist in fuel movement to potential transport or storage casks.

## 9.2 Fuel Inspection Tool

The tool measures fuel element bow and elongation to satisfy TS 4.1 requirements. The tool uses a machined cylinder to provide a go/no-go indication for bend. It also measures elongation through the displacement of the length of the fuel element. The tool is calibrated for length using a “standard” dummy element.

## 9.3 Fire Protection

Doors from the reactor room have rubber seals to ensure confinement. Walls are 1 hour fire resistant standard plaster and metal stud except the west wall that is reinforced concrete design. Fire detection system is continuously monitored when no one is present at the facility. The fire protection program and facilities meet applicable fire codes.

## 9.4 Conclusions

The NRC staff concludes as follows:

- The design of the auxiliary systems ensures that the fuel will continue to be stored, handled and inspected safely. Further, the TS and their bases proposed in the SAR give reasonable assurance that the auxiliary systems will continue to operate as required.

# 10 EXPERIMENTAL FACILITIES

This Section is on the design bases of the experimental facilities and programs.

## 10.1 In-Pool and In-Core Facilities

The research reactors open pool allows irradiation of experiments submerged adjacent to or in the core in watertight containers. The location of irradiation will depend on sample size, and the needed type and intensity of radiation fields. Such experimental facilities have been acceptably used, and will be controlled by the potential reactivity effect and other TS requirements as discussed below.

## 10.2 Rotary Rack Facility (Lazy Susan)

A rotary, multiple-position specimen rack for irradiation of samples is in an indented raceway in the top of the graphite reflector. All positions in this rack are exposed to neutron fluxes of comparable intensity. Samples are loaded from the reactor bridge through a tube offset to avoid direct radiation streaming. This experimental facility can be rotated manually by a motor drive. SAR Chapter 6.3.2.2 show that radiation releases are below measurable levels.

## 10.3 Pneumatic Transfer Systems

The pneumatic transfer systems allow small sealed samples to be transported rapidly between the core and a laboratory adjacent to the reactor room. These experimental facilities permit studies involving short-lived isotopes. Both the moderate and fast pneumatic transfer systems are fixed so they can not be moved while the reactor is operating. SAR Chapter 6.3.2.2 show that radiation releases are within 10 CFR Part 20 limitations.

For the moderate speed system, a blower exhausts air from the system through a filter to the reactor room exhaust, and the associated vacuum is used to move samples in and out of the core. A timer and control is located at the send receive station. Air flow is controlled by a solenoid valve. A permanent area radiation monitor is located at the send-receive station.

The fast pneumatic transfer system is similar, except the size of capsules for samples is smaller, and compressed nitrogen gas is used rather than air. Also, different is the location of the send-receive station which is in the reactor room for this system.

## 10.3 Central Thimble

The reactor is equipped with a central thimble for conducting experiments or irradiating small samples in the core. The central thimble is a 0.38 mm (1½ inch) outside diameter aluminum tube. The tube is constructed so samples are irradiated at the core center-line. A padlock is

used to control access to this experimental facility. SAR Chapter 6.3.2.2 show that radiation releases are below detectable levels.

### 10.4 Limits, Reviews, and Conduct of Experiments

TS 3.1.c requires that experiments with reactivity worth greater than $1.00 be securely fastened to prevent movement in the reactor. TSs 3.1.e and g limit reactivity worth of an individual experiment or of all experiments, respectively, to less than $3.00. TS 3.1.f limits reactivity worth of all experiments so that the shutdown margin at cold, xenon-free conditions with all rods in is at least $0.50. These reactivity limitations acceptably reduce the potential for a reactivity event caused by an experiment and ensure that any reactivity related event would be within the limits in the SAR.

TS 3.8.a limits iodine 131 through 135 and strontium 90 isotope inventories. This TS limits potential radiological consequences from these isotopes to acceptable values.

TS 3.8.b limits the known explosives to be irradiated to less than 25 milligrams. It also limits the pressure produced by accidental detonation of the explosive to less than the design pressure of the container. TS 3.8.c required double encapsulation for experiments that contain materials that are corrosive to reactor components, compounds that are highly reactive with water, materials that are potentially explosive, or liquids that are fissionable. These specifications acceptably ensure that experiments would not damage any safety function that limits potential radiological doses.

TS 6.2.b includes requirements for the Reactor Operations Committee review of new experiments including consideration if there is an unreviewed safety question or a change to TS is required. TS 6.2.c requires at least five members to compose the Reactor Operations Committee, with one being a health physicist from the Office of Environmental Health and Safety and one not associated with the School of Physical Sciences. TS 6.2.c also requires the Reactor Operations Committee to be proficient in all areas of reactor operations and safety. TS 6.3 requires reviewed and approved procedures for installation of experiments and experimental facilities. TS 6.8 specified requirements for review of experiments to ensure compliance with TS and 10 CFR Part 20 requirements. TS 6.8 also requires consideration of reactivity, integrity considering temperature, pressure or chemical conditions, and interactions with reactor components. TS 6.8 also requires consideration of radiation conditions and potential release assumptions.

Chapter 12.2.5 of the SAR discusses authorization of experiments. Experiments are divided into two classes by the Reactor Supervisor. One for repeat experiments or experiments of minor modification from previously authorized experiments. Experiments in this classification can be authorized by any Senior Reactor Operator. All other experiments must be sent to Reactor Operations Committee for review and approval. Therefore, experiments will be controlled and authorized to ensure acceptable conduct and conditions.

SAR Chapter 12.2.3 specifies that the “Operator-in-Charge” is responsible to ensure that any experiment is correctly authorized. This includes Reactor Safety Committee recommendations, and Radiation Use Authorization. Sample removed from the reactor are monitored, activity levels recorded, and barriers or shielding used to minimize personnel exposure. The experimenter and the Reactor Supervisor are informed of any unusual or unexpected incident, apparent equipment or instrument failure or malfunction.

SAR Chapter 12.2.5 indicates that any failure of an experiment or release of material which might damage the reactor in any way, the reactor shall be physically inspected to determine the consequences and need for corrective action before proceeding with any further experiments.

**10.5 Conclusions**

The NRC staff concludes as follows:

- Experimental facilities operation and utilization will continue to meet NRC regulatory requirements for the reactor facility during normal operations. Acceptable technical specifications have been maintained, proposed and justified for experiments.

- The design and functional information in the SAR gives reasonable assurance that the experimental facilities are capable of retaining necessary integrity during all anticipated operations and postulated accidents. Also, experimental facilities are controlled acceptably. The configuration of the experimental facilities is such that they would not impact pool water level.

- The reactivity insertion by malfunction of any experimental facility is limited to acceptable limits. Rapid reactivity insertions of this limit have been evaluated, and the resultant reactor behavior does not exceed acceptable conditions.

- The administrative controls are acceptable to continue to protect the operations personnel, experimenters, the general public and the environment from potential experimental radiation hazards. The radiation doses should continue to be less than the limits of 10 CFR Part 20.

- The consequences of a malfunction or failure of an experimental system are bound by reactor accidents as analyzed in SAR Chapter 13.

- The Reactor Operations Committee will continue to have diverse and independent membership as well as acceptable experience and expertise for experimental facility review.

- The procedures and methods used by the facility ensure a detailed review of all potential safety and radiological risks that an experiment may pose to the reactor facility staff, the public and the environment.

- The administrative controls ensure that all proposed new or changed experiments proposed will be reviewed in accordance with the requirements of 10 CFR 50.59.

- Technical Specifications ensure acceptable implementation of the review and approval of experiments.

## 11 RADIATION PROTECTION AND WASTE MANAGEMENT PROGRAM

This Section is on the design bases of the radiation protection and radioactive waste management programs and facilities.

### 11.1 Radioactive Sources and Potential for Contamination

Material balance inventory forms (NRC/DOE Forms 741 and 742) have been reviewed during inspections and verified quantities and locations of special nuclear material consistent with the SAR descriptions. Similarly, inspection observations have verified potential radioactive materials and conditions are contained and controlled as described in the SAR. SAR Chapter 6.2 indicates that the use of non-porous flooring in the facility reduces the potential for long-term contamination. Escorted and unescorted access to the facility is controlled in accordance with SAR Chapter 12.2.2.

### 11.2 Radiation Protection Program

The radiation protection program includes health physics staffing and procedures, monitoring programs for personnel exposures and effluent releases, and assessment and control of radiation doses both to workers, the public and the environment. The facility program for maintaining radiation exposures and releases as low as reasonably achievable (ALARA) is described in this Section.

As specified in SAR Chapter 11, the research reactor uses the UCI Radiation Safety Program. This program incorporates the requirements of 10 CFR Part 20. The Reactor Supervisor is responsible to implement this program for research reactor authorized activities and conditions.

Radiological safety oversight is provided by the UCI's Environmental Health and Safety (EH&S). EH&S's staff includes individual experienced in radiological safety. EH&S performs independent surveys, review of record, radioisotope inventory control, and direct supervision of unusual shipments. Technical Specification 6.1.c specifies a Radiation Safety Officer is

responsible for safety of operations from the standpoint of radiation safety, and reports to EH&S. Annually, EH&S evaluates overall radiological safety during the renewal of the Reactor Supervisors radiation use authorization.

The Reactor Operations Committee (ROC) also has oversight to ensure public safety as required by TS 6.2. TS 6.2.d requires a health physicist designated by EH&S as a member of the ROC.

SAR Chapter 12.1.4 describes the Radiation Safety Committee for UCI. It is appointed by the Executive Vice Chancellor. The Radiation Safety Committee is responsible for administration of the UCI Radiation Safety Program. The committee is chaired by an UCI Academic Senate faculty member whose research work involves expertise in the use of radioisotopes. The Reactor Supervisor and Radiation Safety Officer are ex-officio members.

TS 6.6 ensure that records include personnel exposures, and radiation surveys are maintained. SAR Chapter 12.1.7 specifies that Radiation Safety Program training is also provided to individuals that support the rector facility.

These provisions provide for independent oversight and control of radiation safety matters.

The ALARA program at the facility has been reviewed by the staff. The policies and the bases for procedures give reasonable assurance that doses to occupational workers, the public and the environment will be maintained below regulatory limits and ALARA. The controls and procedures for limiting access and personnel exposure (including allowable doses, effluent releases, ALARA principles are used for the action levels in radiation alarm systems) meet the applicable radiation protection program requirements and provide reasonable assurance that radiation doses to the environment, public, and facility personnel will be ALARA. The ALARA program is acceptablely supported at the highest levels of management for the facility. The provisions incorporated for personal dosimetry, shielding, ventilation, remote handling, and decontamination equipment provide reasonable assurances that radiation doses are maintained ALARA and within applicable regulations.

The licensee's radiatioin safety program has established and implemented a policy requiring that all facility operations be planned and conducted in a manner that limits radiation exposures to ALARA levels. Guidelines are developed to ensure uniform application of this policy. The licensee has committed to review all proposed experiments and procedures at the reactor for ways to limit potential exposures. All unanticipated or unusual reactor-related exposures are investigated by both the health physics and reactor operations staffs to ascertain the cause and to develop methods for preventing recurrences.

### 11.3 Radiation Monitoring

Section 7.5 of this SE found the radiation monitoring system acceptable. Additionally the licensee specified the use of calibrated portable radiation monitors for use in the facility. These monitors should detect potential radioactive material releases or radiation conditions.

Personnel dosimetry for experimenters is provided by commercial beta-gamma thermo-luminescent devices. The licensee specified this dosimetry is changed quarterly. Pocket dosimeters are used for visitors.

Area radiation surveys are performed in accordance with facility procedures. Neutron and gamma monitoring are also performed to ensure no significant change in facility conditions in accordance with procedures. Wipe tests for removable contamination are regularly taken. UCI also has commercial environmental packs to monitor for radiation conditions at 10 locations on campus.

### 11.4 Radiation Exposures

Assumptions in SAR Chapter 6.3.2.2.1 on the release of Argon-41 to the atmosphere conservatively assumed that the concentration was the same as at the point of release from the ventilation system to the environment. The results were acceptable with regard to 10 CFR Part 20, Appendix B limits. This assumption is extremely conservative considering that dispersion was estimated to reduce the concentration by a factor of one-seventh. Actual measurements show values of about background and calculations show values of 0.7 millirem/year.

SAR Chapter 6.3.1 conservatively estimated radiation exposures of individuals in the public and working at the reactor. These exposures were well within regulatory limits. Chapter 11.3 of the SAR indicates that tour groups have been monitored resulting in no measurable radiation exposure.

The dose rate at the tank surface as a result of Nitrogen-16 is greatly reduced by the operation of a diffuser incorporated in the primary coolant system as described in SAR Chapter 6.3.2.1. The diffuser operates anytime a primary coolant pump is running. The diffuser discharge above the reactor and directs flow across the top of the reactor. This flow diversion increases the amount of time for the Nitrogen-16 to reach the surface and thus limits radiations exposures.

### 11.5 Radioactive Waste Management

The program for radioactive waste management includes the control and disposal of radiological waste from both reactor operations and experimental programs. Chapter 11.8 of the SAR specifies that all low level radioactive waste in the research reactor facility is contained in containers as required by the radiation safety program (which as previously indicated explicitly incorporates the requirements of 10 CFR Part 20). When moved from the research reactor facility, low level waste is transferred to ES&H and a State license.

### 11.6 Conclusions

The NRC staff concludes as follows:

- The SAR identifies the potential radiation safety hazards and this provides an acceptable basis for the development and independent review of the radiation protection program. This program will continue to limit radiation exposure and contamination, and control access to radiation areas acceptably.

- The radiation protection program complies with applicable requirements and gives reasonable confidence that licensee management's commitment to radiation protection in all activities will continue to protect the health and safety of the licensee's staff, the public and the environment. It provides for independent oversight of research reactor radiation safety.

- The radiation monitoring system gives reasonable assurance of acceptable radiation conditions to personnel and the environment.

- Exposures have been and will continue to be well within regulatory limits.

- Radioactive waste management will be in accordance with the requirements of 10 CFR Part 20.

## 12 CONDUCT OF OPERATIONS

This Section is on the management structure and responsibilities, review activities, and other required functions such as the procedures, reporting, and operator requalification, security and emergency plans.

### 12.1 Organization

Responsibility for the safe operation of the reactor facility is vested with the UCI Executive Vice Chancellor. The reactor is operated by the School of Physical Sciences. The organizational structure (Chart 1 TS 6.1) shows the lines of authority from the School of Physical Sciences down through the Reactor Operations Committee, the Reactor Administrator, the Reactor Supervisor, to reactor operations.

The chart also shows lines of communications between the Reactor Supervisor and reactor operations, and the independent UCI Radiation Safety Officer. The Radiation Safety Officer reports to the UCI Office of Environmental Health and Safety and is an experience health physicist. The Radiation Safety Officer is responsible for the radiological health and safety of the community, and works with the Reactor Supervisor to ensure safe research reactor operations and conditions. The Radiation Safety Officer has additional staff member trained as an emergency response team and can provide additional consultation regarding radiological safety.

The Reactor Administrator will be a tenured UCI faculty member in accordance with TS 6.1. The Reactor Administrator appoints the Reactor Supervisor and delegates implementation of operations of the research reactor to the Reactor Supervisor.

The Reactor Supervisor will be a qualified licensed senior operator and all other reactor operators and trainees report to the Reactor Supervisor. The Reactor Supervisor has responsibility for implementation of operations of the research reactor facility. The Reactor Supervisor is a faculty or staff member of the University.

### 12.2 Review Activities

The Reactor Operations Committee TS 6.2 reviews include experiments, changes to the facility, procedures or Technical Specifications, operational records, abnormal occurrences, and approval of individuals for supervision and operation of the research reactor. TS 6.2.c requires at least 5 members on the committee, and at least one member not associated with the School of Physical Sciences. TS 6.2.d requires a written statement of authority, purview and other administrative provisions as needed, and that minutes of all meetings will be kept. The committee meets semi-annually (not to exceed every 9 months) in accordance with TS 6.2.f. TS 6.2.d defines a quorum as a majority of the full Committee including the Chairman or a Chairman designee. SAR Chapter 12.1.3 indicates that the Reactor Supervisor and Radiation Safety Officer are ex-officio members. Members of the Committee are experienced in reactor or radiation physics, medical physics or chemistry. The Reactor Operations Committee approves startup of the research reactor following major modifications or repairs and has the authority to stop operation “if it does not believe that everything is in order for safe operation.”

### 12.3 Procedures

TS 6.3 requires acceptable operating procedures to ensure safety of the research reactor. These procedures include startup, operation, shutdown, alarm response, emergency, maintenance, surveillance, and fuel, control rod, experiment and experimental facility movement. Temporary changes that do not change the intent may be made by the Reactor Supervisor. Temporary changes will subsequently be reviewed by the Reactor Operations Committee. Otherwise, changes to procedures may be made by the Reactor Supervisor with approval of the Reactor Operations Committee. Changes that relate to aspects of the radiation safety program must be approved by the UCI Radiation Safety Officer and the UCI Radiation Safety Committee. SAR Chapter 12.2.3 specifies that an “Opeator-in-Charge” for each reactor operation. This Operator is responsible to ensure compliance to Standard Operating Procedures and the Technical Specifications, and that log entries are made for all operations. This Operator is also responsible to ensure that the provisions of the UCI Radiation Safety Program are followed.

### 12.4 Event Reporting and Review Procedures

TS 6.4 specifies actions to be taken in the event of a exceeding a safety limit. They include reactor shutdown until operation is authorized by the NRC, notification of the NRC and the

Chairman of the Reactor Operations Committee, and a report with an analysis of the causes, damage, and corrective actions. TS 1.13 defines abnormal occurrence to include any actual setting less conservative than the TS Limiting Safety System Settings, violation of a limiting condition of operation, an engineered safety system component failure which could have rendered the system incapable, fission product release from a fuel element, uncontrolled or unanticipated reactivity change, or inadequacy in administrative or procedural controls that could have resulted in unsafe operation of the reactor. TS 6.5 specifies notifying the Reactor Supervisor of an abnormal occurrence prior to restart, a report to the Reactor Operations Committee on the causes and corrective actions, and a report to the NRC. These requirements meet or exceed the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 15.1-1990, "The Development of Technical Specifications for Research Reactors." Further, routine reporting requirements provided in the TS are also consistent with this standard.

### 12.5 Records

In addition to the regulatory requirements, TS 6.6 specifies 5 year retention of records for operations, maintenance, surveillance, abnormal occurrences, gaseous and liquid radioactive effluents to the environs, environmental monitoring, fuel inventories and transfers, radiation and contamination surveys, personnel radiation exposures, and experiments. Also, updated, corrected and as-build drawings of the facility will be retained for the life of the facility.

### 12.6 Emergency Planning

SAR Chapter 12.1.2 specifies that the Emergency Plan contains a clear succession system when the Reactor Supervisor is absent. The emergency plan dated May 31, 2000, provides for this succession. The Radiation Safety Officer and his trained staff are available for emergency response and consultation. Emergency exit may be made through the double doors to the loading dock or through the control room as specified in SAR Chapter 12.2.2. Persons authorized for unescorted access must have knowledge of safety and emergency procedures within the research reactor area. Emergency instructions require a member of the reactor staff or EH&S will be contacted prior to entry by Police or Firemen except in cases of extreme life-threatening emergency. The licensee maintains emergency supplies in various locations to ensure the materials need to respond to potential radiological emergencies. EH&S also maintains additional equipment and supplies for emergency response. The effectiveness of the plan has not been changed from that previously approved.

### 12.7 Security Plan

The security plan dated May 31, 2000, provides for normal access control to the facility. It requires both a door key and key card. The Reactor Supervisor issues keys and key cards to authorized individuals in accordance with the Security Plan. Double doors to the loading dock are normally closed and locked. The plan considers the threat from deliberate activities as specified in SAR Chapter 13.9, and that procedures are in place to alert staff to potential threats, and evacuate the buildings as necessary. No effective change has been made to the plan.

## 12.8 Operator Requalification

Reactor Operators are licensed by NRC. Reactor Operators are trained in accordance with UCI's Radiation Safety Program, submitted April 24, 2000, and 10 CFR Part 55. Further details on the operator training program is provided in SAR Chapter 12.3. Reactor Operators are required to maintain qualifications as specified in UCI's Licensed Operator Requalification Plan as described in SAR Chapter 12.4 and submited on April 24, 2000.

## 12.9 Conclusions

The NRC staff concludes as follows:

- The organizational structure continues to describe the facility organization. All organizational relationships important to safety have been shown, including that of the review function and the radiation safety function. The organization is consistent with American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 15.1-1990, "The Development of Technical Specifications for Research Reactors." The licensee has described the responsibilities of the persons in the organizational structure. The responsibility for safe operation of the facility and for the protection of the health and safety of the licensee's staff, the public and the environment has been shown. Radiation protection training and specialized training will be conducted at an acceptable level. The radiation safety organization has direct access to upper management and the review committee to express concerns, if necessary.

- The review function for the reactor facility continues to be performed by the Reactor Operations Committee. The committee membership is required to be qualified, with a wide spectrum of expertise. The committee membership is to include a person from outside the School of Physical Sciences. The committee has a written statement how often the committee meets, how the committee conducts business, and requirements for a quorum. The list of items that the committee will review is specified. The staff has determined the review functions for this research reactor continue to be acceptable.

- The required procedures are acceptable for continued safe operation of the facility as proposed in the SAR. The review and approval process for procedures continues to be acceptable including the method for making minor and substantive changes to existing procedures and for the temporary deviation from procedures during operations. The staff has determined that the process and method described in the SAR will continue to ensure management control and review of procedures.

- The incidents defined as reportable events and the required actions to be taken are consistent with the applicable standard. The definition of reportable events gives reasonable assurance that safety-significant events will be reported and that actions necessary to protect the health and safety of the licensee's staff, the public and the environment will be taken.

- The records that will be retained by the facility and the period of retention to ensure that important records will be acceptably retained.

- The Emergency Plan continues to provide acceptable preparedness for potential accident conditions.

- The Security Plan continues to provide an acceptable level of effectiveness.

- Reactor operators will be trained and requalified in a program that meets standards for non-power reactors and the requirements of the regulations.

# 13 ACCIDENT ANALYSES

## 13.1 Introduction

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" specifies accidents for consideration. The licensee considered the potential consequences on the reactor fuel and on the radiological health and safety of the licensee's staff, and the public for the applicable NUREG-1537 scenarios:

(1) Maximum Hypothetical Accident
(2) insertion of excess reactivity
(3) loss of coolant
(4) loss of coolant flow
(5) mishandling or malfunction of fuel
(6) experiment malfunction
(7) loss of normal electrical power
(8) external events, and
(9) mishandling or malfunction of equipment.

Those potential events that could result in the release of radioactive materials were evaluated in the SAR until it was possible to reach the conclusion that a particular event was, or was not, the limiting event. The NRC has prepared an independent analysis of credible accidents for TRIGA reactors. This study was documented in NUREG/CR-2387, "Credible Accident Analysis for TRIGA and TRIGA-Fueled Reactors." The staff has used applicable information as a basis for evaluating some of the information (e.g., U.S. Nuclear Regulatory Commission, "Credible Accident Analysis for TRIGA and TRIGA-Fueled Reactors," NUREG/CR-2387, April 1982, and U.S. Nuclear Regulatory Commission, "High-Uranium Content, Low-Enriched Uranium Zirconium Hydride Fuels for TRIGA Reactors," NUREG-1282, Docket No. 50-163, August 1987).

## 13.2 Maximum Hypothetical Accident (MHA)

The MHA for the UCI research reactor is a cladding failure of a single irradiated element in air in the reactor room analyzed in SAR Chapter 13.2.2.

The failed fuel element was assumed to have operated at the highest core power density for an infinite time at the maximum authorized power level with a 70 element core. Independent verification of selected fission product inventories were made based on U.S. Atomic Energy Commission, “Calculation of Distance Factors for Power and Test Reactor Sites,” TID-14844, March 1962, page 20, Table 1. This verified the selected iodine fission product inventories were acceptable.

The fuel release fraction was based on data analyses for fuel like that at the UCI research reactor. The assumed release fraction is consistent with that assumed and used in other TRIGA MHA analyses (e.g., McClellan Air Force Base, McClellan Nuclear Radiation Center, Safety Analysis Report, July 1997).

With this fission product release, conservative dose estimates were made. The dose estimates conservatively assumed instantaneous distribution of the released fission products, no reactor room ventilation exhaust for potential occupational exposures, and evacuation in 2 minutes of the alarm. The potential exposures to members of the public assumed emergency purge through the ventilation system for release to the environment. The results were acceptable in accordance with 10 CFR Parts 20 and 100.

## 13.3 Insertion of Excess Reactivity

A credible generic accident is the inadvertent rapid insertion (pulse insertion) of positive reactivity which, if large enough, could 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 a slower event to occur because of the uncontrolled withdrawal of multiple control rods. Flooding or removal of beam tube inserts could also have a positive effect on reactivity but not as severe as the pulse insertion of positive reactivity. The inherent prompt negative temperature response characteristics of TRIGA fuels clearly is a safety factor for this type of postulated accident.

The licensee has presented in the SAR Chapter 13.3, a conservative analysis of the rapid reactivity insertion accident from full power. This pulse insertion of positive reactivity from full power would require violation of the operating license and TS conditions, and several interlocks and scrams. As previously discussed in Section 4.6.2 of this SE, pulsing from low power (less than 1 kilowatt) has been and can be controlled to provide considerable margin to the safety limit. In SAR Chapter 13.3, the licensee performed a calculation with an insertion of the maximum allowed excess reactivity. The calculation showed that the maximum fuel temperature would be 570°C (1058°F). The results again show significant margin to the fuel element temperature safety limit of 1000°C (1832°F) in TS 2.1.

## 13.4 Loss of Coolant

Although as the licensee indicated the likelihood of a loss of coolant is “highly improbable,” the licensee postulated and analyzed a loss of coolant to evaluate (1) the integrity of the fuel, and (2) radiological consequences from draining the water.

For the integrity of the fuel, the licensee conservatively assumed a long operating period at full power, a 62 element core, instantaneous loss of water, and a temperature distribution equal to that of the hottest fuel element. The licensee also assumed no energy release from delayed neutrons (a relatively small factor), and no entrance or exit pressure loss on air flow. Licensee calculations show that the peak stress on cladding material is 985 psi which is at least a factor of 20 below allowable material conditions.

For the radiological condition, the licensee assumed that water from the tank would leak at a conservative rate. Licensee calculations show that the core would be uncovered about 10 hours after the low water level alarm actuates. The licensee’s calculation of radiological consequences with the pool drained conservatively assumed that: the reactor was operated for 1000 hours at full power prior to the potential loss of coolant, the core was a point source, and attenuation was negligible except for core components. The licensee also assumed that the reactor was shutdown as soon as the leak develops and that photon energy is only from fission decay with an average energy of 1.0 million electron volts (MeV). The calculations on SAR pages 13-15 through 13-19 show an accident radiological dose of about 550 mrem/hr. This dose rate is at the floor of the classroom above the reactor room when the core would be potentially uncovered. This would result in an exclusion area boundary dose well below the 25,000 mrem in two hours allowed by 10 CFR Part 100. Because of the relative long time frame (about 10 hours) between initial alarm and any potential significantly increased radiological consequences emergency procedures can be readily implemented, as indicated by the licensee, to maintain accident radiological dose within acceptable limits.

## 13.5 Loss of Coolant Flow

The licensee indicated in SAR Chapter 13.5 that the reactor is designed to operate under natural convection conditions. The licensee indicates that shutdown and repair would be initiated following any potential flow restriction. The licensee also indicated that pool temperature is limited during operations. The licensee also verifies that the core flow area is clear prior to operations. These conditions ensure acceptable coolant flow.

## 13.6 Mishandling or Malfunction of Fuel

The licensee indicated in Chapter 13.6 of the SAR that mishandled fuel would be inspected prior to use in the core. Malfunction of fuel is controlled by procedures to identify and remove potentially damaged fuel. The worst case analysis for mishandling or malfunction of fuel would be the MHA, failure of cladding on a single element while in air as previously discussed.

## 13.7 Experiment Malfunction

TS 3.8 limits experimental radioactive materials, and explosives. Further licensee procedures and controls as discussed in Section 10 of this SE provide assure that radiological consequences will be limited to well below acceptable levels. Based on these analyses, the provisions to address experiment malfunction are acceptable.

## 13.8 Loss of Normal Electrical Power

On loss of normal electrical power, the control rod electromagnets de-energize and control rods are driven into the core by gravity. Reactor shutdown is thus assured in accordance with Chapter 13.8 of the SAR and the earlier discussion in this SE in Section 4.3.2. The licensee also points out that facility radiation area monitors, continuous air monitors, alarms, security monitors and alarms are wired to the building emergency diesel generator. The diesel picks up emergency lighting and power automatically on loss of normal electrical power. The licensee further points out that the diesel generator is likely to operate under most conditions because of the location in Rowland Hall. The diesel generator unit is under a surveillance and maintenance program to provide additional assurance that it will provide the desired information to the licensee on loss of normal electrical power. These provisions are acceptable for loss of normal electrical power.

## 13.9 External Events

The licensee discussed the design to mitigate potential external events in Chapter 13.9 of the SAR. The licensee indicated that the concrete and reinforced concrete structure provide significant protection from external events. The licensee discussed the protection the building provides from flying debris or falling aircraft, and the drain system provides against flooding. External events as documented in Section 2 of this SE provide minimal risk to the facility and are acceptably considered in the facility design.

## 13.10 Mishandling or Malfunction of Equipment

As specified in SAR Chapter 13.10, licensee procedures require correct equipment use. Additionally, TS 6.3 requires operating, maintenance, surveillance, experimental, fuel movement, abnormal and emergency procedures to prevent and mitigate potential mishandling or malfunction of equipment. The licensee points out that their staff is qualified as required of a large university. Supervision of personnel and activities is provided by licensed reactor operators, trained radiation safety specialist, and the management chain. Procedures and training require suspension of activities following any perceived malfunction. These provisions have been acceptably used at this facility and at similar facilities. Further, the NRC staff finds that these provisions should continue to be acceptable as the licensee plans no change in this regard.

## 13.11 Conclusions

The NRC staff concludes as follows:

- The licensee has postulated and analyzed sufficient accident-initiating events and scenarios to demonstrate that the reactor design, management, operating limits, and procedures are planned in a manner that radiation exposure to the licensee staff and the public will not exceed NRC limits, and will avoid inadvertent reactor damage that could prevent safe shutdown.

- Under the least favorable atmospheric conditions, the maximum hypothetical accident of the failure of a fuel element cladding in air will not result in occupational radiation exposure of the licensee staff or radiation exposure of the general public in excess of acceptable NRC limits.

- For accidents involving insertions of excess reactivity, loss of coolant, loss of coolant flow, mishandling or malfunction of fuel, experiment malfunction, external events and mishandling or malfunction of equipment, the licensee has demonstrated that there is no projected significant damage to the reactor.

## 14 TECHNICAL SPECIFICATIONS

### 14.1 Summary

In the course of this licensing action as discussed in this SE, the staff has reviewed and evaluated the TSs submitted by the licensee. These TSs define certain features, characteristics, and conditions governing the operation of the facility and will be explicitly included in the license as Appendix A. In addition, the staff reviewed the format and content of the TS using guidance from ANSI/ANS 15.1-1990, “The Development of Technical Specifications for Research Reactors,” and the guidance in applicable sections of NUREG 1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” dated 1996. TS are essentially the same as previously used at UCI for over 30 years. The only change is to take out references to SAR chapter numbers as they have changed and to avoid the need for TS change with SAR changes.

### 14.2 Conclusion

The NRC staff finds the TSs acceptable and hence concludes that normal plant operation within the limits of the TSs will not result in offsite radiation exposures in excess of the limits specified in 10 CFR Part 20. Furthermore, the limiting conditions for operation and surveillance requirements will continue to limit the likelihood of malfunctions and mitigate the consequences to the staff, public and environment in regard to potential accidents.

## 15 FINANCIAL QUALIFICATIONS

The staff reviewed the financial status of the licensee and concludes that the necessary funds will be made available to support continued operations and, when necessary, to shut down the facility and carry out decommissioning activities. Operating costs will be paid from an annual allocation of funds governed by the University of California, Irvine in accordance with the licensee's application letter dated October 18, 1999. Also, the University as a state agency of California plans to request funds for decommissioning when the facility is permanently shutdown. In accordance with 10 CFR 50.75e(2)(iv), since the University of California, Irvine, as part of the State of California, is the source of funds and since both operating and decommissioning costs are at levels that should not present funding problems, no additional analysis or verification of the adequacy of funding is required.

## 16 PRIOR UTILIZATION

The operating license was issued on November 24, 1969. The reactor has operated less than 218 full power effective days over the approximate 30 year license period as indicted in SAR Chapter 1.3.2. During that time, the reactor was used primarily by the University of California Department of Chemistry for research and teaching in the field of radio-analytical chemistry. In addition, some other users from both on and off campus departments and organizations carried out analytical determinations using neutron activation analysis. There were no significant incidents of radioactive material releases or occupational exposures above the limits of 10 CFR Part 20.

The staff concluded that the UCI TRIGA Mark I pulsing reactor was initially designed and constructed to operate safely. During the license application review, the staff considered whether prior operation would cause significant degradation in the capability of components and systems to continue to perform their safety functions. Because fuel cladding is the component most responsible for preventing release of fission products to the environment, the staff considered mechanisms that could possibly lead to detrimental changes in cladding integrity. The mechanisms include radiation degradation of cladding integrity, high fuel temperature and temperature cycling effects on the mechanical properties of the cladding, corrosion, damage from handling or experimental use, and degradation of safety components or systems.

For the UCI TRIGA reactor, the factors which could result in changes to cladding integrity, such as power density and maximum fuel temperatures, coolant flow rates and temperatures, conductivity and pH of primary coolant, are comparable to those of other operating TRIGA reactors. The UCI TRIGA reactor is typical of a large number of TRIGA reactors operating both in the United States and overseas. Experience with this reactor and others show that the control of these factors has acceptably maintained cladding integrity. To ensure this, the licensee staff is required to perform regular surveillance activities and preventive maintenance. The staff concludes that there has been no significant degradation of equipment and that facility management will continue to maintain and operate the reactor so that there is no significant increase in the radiological risk to facility staff, the public or the environment.

## 17 CONCLUSIONS

On the basis of its evaluation of the application as set forth in the previous sections of this SE, the NRC staff has reached the following conclusions:

- The application filed by the University of California, Irvine for renewal of the operating license for their TRIGA research reactor complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), as well as the Commission's regulations set forth in 10 CFR Chapter I.

- The facility will operate in conformance with the application, as well as the provisions of the Act and the rules and regulations of the Commission.

- The licensee has provided reasonable assurance that (a) the activities authorized by the operating license can continue to be conducted without endangering the health and safety of the licensee's staff, the public and the environment and (b) such activities will be conducted in compliance with the Commission's regulations as set forth in 10 CFR Chapter I.

- The licensee continues to be technically and financially qualified to engage in the activities authorized by the license in accordance with the Commission's regulations as set forth in 10 CFR Chapter I.

- The renewal of this license will not be inimical to the common defense and security or to the health and safety of the licensee's staff, the public and the environment.

Principal Contributor: Marvin M. Mendonca

Date: xxxxx xx, xxxx

Need copy of 10/21/99 letter referenced in SAR section 15?

Need an environmental report. For example, the NC State one sent. 5/1

Need operator requalification program. 5/1

Need Emergency Plan. 6/1

Need Security Plan 6/1

There are three sections 2.3?

Section 3.1 says building declared "POOR" Need to address this? Section 3.4.1 says design more than adequate to endure earthquake induced ground motion. SAR Appendix A, page 4 in part says "Although it is not anticipated that an earthquake would result in damage to the reactor itself, it is likely that the facility walls, ceilings, and doors, which act as a confinement system for any potential radioactive gas or liquid releases, would be significantly damaged. In this event, low-level radioactivity could be released..." Make statements consistent. If release, provide estimate of what the low-level radioactivity that could be released and associated radiological consequences and is it less than calculated in previously accident analyses.

Were Figures 3-2 through 3-5 included?

Appendix C INEL letter says 30 year relicense?

Section 3.4.1 says five fuel storage pits. Descibe the use of these pits. Fresh and/or spent fuel storage? Number of elements per pit? Shielding design? Criticality design? 10 CFR 40444 requirement? Is this consistent with TS 5.3?

Page 4-8 is it true that the fast transient rod can provide pulsed operations and also acts as a safety rod for steady state operations. Does the adjustable transient rod also act as safety rod? Does that mean they fall into the core on scram?

What do you do for verification that the pool is not leaking? No indication of significant pool integrity loss has been observed since the reactor was originally licensed.

Section 6.3.1 says small neutron leakage above lazy susan drive shaft. ARe finger ring dosimeters needed for handling materials and working around there? Indicate what monitoring and/or controls are taken.

The terms used in section 7.1 for

- Wide Range Linear Monitor (fission chamber)
- Wide Range Monitor (Compensated Ion Chamber)
- Power Range Monitor (Un-Compensated Ion Chamber)

seem to be different than used in 7.2.4, 5, and 7.4.1?

Section 7.1.1 what are the three positions on the key switch?

Page 7-2 describe annuciators for COOLING, PURIFICATION AND LAZY SUSAN.

Page 7-2 what effect will the magnet current supplies and indicator modules have when activated? Describe or make clear won't be installed without proper 50.59 or NRC review as appropriate.

Page 7-2 what power supply is left on when not in ops? The 25 vdc, clock and or the ion chamber? Clock not

Pages 7-3 does the adjustable transient rod scram?

Page 7-4, what causes the fast transient rod to scram? Air vented off on scram signal?

Page 7-4, section 7.2.3

> says square wave operation uses the fast transient rod? Where is it specified that square wave operations is allowed?

Page 7-5 says auto rx flux control not used or connected? True? Does SAR need changes where it says it used auto mode?

Section 7.3 refers to block diagrams, but I could not find them in this section?

Section 7.3.1 says that trips are enabled on all three channels to scram the reactor if excessive power is indicated on any one. However, section 7.2.4 on the log channel only seems to say that it trips on period or has the source range interlock. Does it have a high power trip?

Page 7-7 refers to Figure 7-7 but I can't find it?

Section 7.4.3 refers to FMA in Appendix IV but I can't find it? Not

Section 7.4.1 mentions seismic trip.  Need description of instrument.

Section 11.3 is Neutron and gamma monitoring are also performed approximately on an annual schedule to ensure no significant change in facility conditions in accordance with procedures.

Section 11.5 are wipe test per procedure?

Section 11, provide description of ALARA program or is it included in RP program.

Section 12.2.2 Who is E.H. and S. Officer?  RSO?  Consistent terminology or is it just someone from EH&S?  Clarify or not.

Section 12.2.3 what does “All members reactor operators...” mean?

Page 13-5 refers to figure 13-1?  Where is it?

Section 13.2.2.4 when I do calc using calculated concentrations based on inventory times release fraction divided by volume to get microcuries/millilitter, them do ratio to DAC which corresponds to 25 rem, I get a lot lower dose than the .45 rads/sec?

Page 13-19, section 13.5, provide verification of procedures, policies, and anything that prevent introduction of foreign material to core area.

Section 13.8 says diesel is NOT considered part of reactor facility yet we rely on it for some things?  Emphasize that the licensee (UCI) ensures it will function for the reactor when needed.

TS 3.6 ensures ventilation system for operations.  What does for fuel movement?

Page 13-15 & 16 has sections 8.4 vice a section 13 number?

Page 13-16 refers to Darcy’s Law with a reference, but no reference appears provided?

Page 13-16 says “permissible” 0.24 mr/hr.  Where does this come from?

Page 13-17 says quarterly doses per 20.101 which is no longer applicable?

Page 13.17 for 15.2 hr, zero water over core says “ca 550" mr/hr direction radiation at floor of classroom.  What does “ca” mean?

Page 13-18 should the reactor shutdown be from time of alarm (level or radiation) rather than from as soon as leak develops?

Tech Spec Bases have to match sections quoted in SAR:

Bases for 3.1 refers to Section 8.5 of SAR but no such thing?

Basis for 3.6 refers to section 8.7.5, 8.4.4 and 8.4.5 of SAR, but these sections are not in SAR?

Similarly basis for 3.7 refers to section 8.6 of the SAR but not right reference?

TS 2.1 refers to SAR section 8 and pages 8.38 through 8.40?

TS2.2 specifies LSSS of 800 and 755 degrees C,basis

Bases for TS 3.1b states “The power level at which a pulse could be initiated in an accident may be as high as 100°C At 100kw, the peak temperature of the fuel will be 115°C. The calculations indicate that a $3.00 pulse will result in a peak temperature of only 502°C, well below the safety limit.”

What does the 100°C refer to?

What does the sentence “[a]t 100kw, the peak temperature of the fuel will be 115°C” mean?

For a $3.00 (2.1 percent Δk/k) pulse, SAR section 4.5.6 calculated the effects of pulses on a TRIGA. Fuel temperature was less than 450°C (842°F). From where did 502°C come?

Bases for “Specification 3.2a is based on Figure 7-4...” However 7-4 doesn’t match up with this?

TS 3.4 doesn’t specify setpoint for Reactor Power Level Scram in steady-state? Where specified or specify?

The basis for 3.5 refers to Section 8.4.4 of the SAR, pages 8-18 through 8-23, but these are not appropriate? Further, this basis refers to a factor of 40 dispersion, but it is not obvious where in the SAR this is proved. Provide calculation or evaluation to show appropriateness of 40.

XXXXXXXXXXXXXXX

Does TS 3.6 require that the CAM be verified operable as part of emergency ventilation?

Basis for TS 3.6 refers to Section 8.7.5, 8.4.4, 8.4.5 & 8.6 of SAR but these are not appropriate?

TS 3.7 says provide alarm to UCI Police Dispatch Desk but shouldn't it also require alarm at control console?

TS 3.8a specifies limits for some iodine isotopes and Strontium 90 for fueled experiments. Where is it proven that these are limiting? Should it be limited to Part 20 doses?

Basis for TS 3.8 revers to SAR p. 8.53 which is incorrect? Where is reference for the iodine activity dose and strontium yield that is made in this bases?

TS 4.1c says "Fuel elements in the B- and C-ring shall be measured for possible distortion in the event that there is indication that fuel temperatures greater than the limiting safety system setting on temperature may have been exceeded." The last part of phrase is imprecise should it be something like "Fuel elements in the B- and C-ring shall be measured for possible distortion in the event that there is indication of fuel temperatures greater than the limiting safety system setting."?

TS 4.2d says "...a functional performance check of the transient (pulse) rod system shall be performed. Annually, at intervals not to exceed eighteen months, the transient (pulse) rod drive cylinder and the associated air supply system shall be inspected, cleaned , and lubricated as necessary." Does this include both transient rods? Provide clarification to ensure it is understood that it covers both transient rods.

TS 4.3 a says "...prior to each operating extending more than one day." What does that mean? Every 24 hours?

TSs 4.3e and f are they both necessary?

TS 4.5 says "...alarm -set points..." should it be "...alarm set points"

TS 5.2.a says "cast" versus cask?

TS 5.3 basis refers to submittal dated June 5, 1969. Is this still applicable reference?

TS 6.1b says "UCIfaculty" should have space?

TS 6.2.c has two periods on the end of it?

TS 6.2.f is not indented like the previous ones.

TS 6.5c says “Where appropriate.” What does this mean?

TS 6.7.a says “immediate report” It is vague and not same as ANSI guidance (i.e., no later than the following working day). Should this be same as ANSI/ANS guidance or more specific like within X hours?

TS6.7.f is this needed anymore?

TS 6.8.a says “...theTechnical Specifications...” Should be space after “the”

TS 6.8.c says “many other experiment” and should experiment be plural? It also says “Thisd” and should it be “This”?

Need copy of 10/21/99 letter referenced in SAR section 15? I have 10/18/99 letter

Section 2.3 has “whne” in one sentence? In the last sentence, first word, “the” is not capitalized? Also the word “polciy” is used?

Page 4-26 says typical power pulse and fuel temperature for 2.1% delta k/k addition sown in Figure 7-4. Should it be figure 4-17?

Page 7-1, section 7.1 refers to Fig. 7-2 on range of reactor power but I didn’t see it or any other figures referred to in this section?

Page 13-10 has “nobel” rather than “noble”?

Page 13-11 “citred” rather than “cited.”

Section 13.6 says “ios” instead of “is” and “rost” instead of “worst.”

Page 13-13 sentence right after box of data has “coorrected”? There are other typos throughout that should be corrected.?