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THE UNIVERSITY OF UTAH

December 30, 2009

United States Nuclear Regulatory Commission
Docket Control Desk
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***Subject*: UUTR Technical Specification Amendment Request – Clarification of the Administration Organization for UUTR Operation (License No. R-126, Docket No. 50-407) and - Notification of new Administrator (Level 2)**

This letter is to provide clarification of the University of Utah 100 kW TRIGA Reactor (License No. R-126, Docket No. 50-407) (UUTR) Administrative Organization chart existent in the Technical Specification (Chapter 14) submitted together with SAR in 2005. This clarification is submitted as amendment #7, attached. The clarified organization chart is Figure 6.1, in Section 6.0 Administrative Controls, in Chapter 14 Technical Specification, pg.26. In the attached file, Figure 6.1 is redrawn to more clearly show the facility organization and management, lines of communication and lines of responsibility and reporting. This is designed to be in full compliance with the ANSI/ANS 15.1-2007 and ANSI/ANS 15.4-2007 regulations. Section 6.0 Administrative Controls was also rearranged to follow NUREG 1537 Part 1 & 2, and ANSI/ANS 15.1-2007 and ANSI/ANS 15.4-2007 regulations. The clarifications in Section 6.0 do not reflect any substantive changes in our administrative organization.

This letter is also to provide you with notification of our new Level 2 Administrator. In a letter dated May 26, 2009, we confirmed that Dong-Ok Choe would be serving as Interim Reactor Administrator (Level 2) under our old organizational chart. In August 2009, we hired **Chaired Professor Tatjana Jevremovic** as the new Director (Administrator) of the Utah Nuclear Engineering Program (UNEP) and Nuclear Engineering Facilities (NEF). We would now like to confirm that she will be serving as the Level 2 Administrator (with responsibilities based on the revised Section 6.0 Administrative Controls), effective immediately. She will now be the primary point of contact between the University and the NRC.

Also, the Level 1 administrator is now Associate Vice President for Research, Dr. Cynthia Furse. She is replacing Dr. Ron Pugmire in this role. Please also update your contact for our Vice President of Research. This is now Dr. Thomas Parks, replacing Dr. Ray Gesteland.

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Contact information for Drs. Jevremovic, Furse and Parks is given below:

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Sincerely,

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Chapter 14

Technical Specifications

Note: changes from the TS submitted in 2005 are in CHAPTER 6 [Fig.6.1 is clarified; whole Chapter is rearranged to follow the ANSI/ANS 15.1-2007 and NUREG 1537 Part 1 & 2]

FACILITY LICENSE R-126

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE UNIVERSITY OF UTAH

TRIGA REACTOR

DOCKET 50-407

TECHNICAL SPECIFICATIONS AND BASES FOR THE UNIVERSITY OF UTAH TRIGA NUCLEAR REACTOR

This document constitutes the Technical Specifications for the Facility License No. R-126 and supersedes all prior Technical Specifications. Included in these Technical Specifications are the "Basis" to support information the selection and significance of the specification. The bases are included for information purposes only. They are not part of the technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere. Furthermore, the dimensions, measurements and other numeric values given in these specifications may differ slightly from actual values because of normal construction and manufacturing tolerances, or normal degree of accuracy of the instrumentation.

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1.0 DEFINITIONS

The following frequently used terms are herein explicitly defined to ensure uniform interpretation of the Technical Specifications.

1.1 Reactor Operating Conditions

Abnormal Occurrence: An abnormal occurrence is defined for the purposes of reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (PL 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

Cold Critical: The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperature both below 40°C.

Reactor Operation: Reactor operation is any condition wherein the reactor is not secured.

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

(1) The reactor is shut down.

(2) The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area.

(3) No work is in progress involving incore fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of incore experiments.

Reactor Shutdown: The reactor is shut down when the reactor is subcritical by at least $1.00 of reactivity.

Reportable Occurrence: A reportable occurrence is any of the following that occur during reactor operation:

(1) Operation with any safety system setting less conservative than specified in Section 2.2. "*Limiting Safety System Settings.*"

(2) Operation in violation of a limiting condition for operation.

(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function.

(4) Any unanticipated or uncontrolled change in reactivity greater than $1.00.

(5) An observed inadequacy in the implementation of either administrative or procedural controls, to such a degree that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits.

(6) A measurable release of fission products into the environment.

Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that (1) the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating conditions; and (2) the reactor will remain subcritical without further operator action.

1.2 Reactor Experiments and Irradiations

Experiment: Experiment shall mean: (1) any apparatus, device, or material which is not a normal part of the core or experimental facilities, but which is inserted into these facilities or is in line with a beam of radiation originating from the reactor; or (2) any operation designed to measure reactor parameters or characteristics.

(1) Routine Experiment. Routine experiments are those which have been previously tested in the course of the reactor program.

(2) Modified Routine Experiment. Modified routine experiments are those which have not previously been performed but are similar to routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiment.

(3) Special Experiment. Special experiments are those which are not routine or modified routine experiments.

Experimental Facilities: Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, incore irradiation baskets or tubes, pneumatic transfer systems, and any other inpool irradiation facilities.

Irradiation: Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an irradiation facility so that the device or material is exposed to a significant amount of radiation available in that irradiation facility.

Irradiation Facilities: Any inpool experimental facility that is not a normal part of the core and that is used to irradiate devices and materials.

Secured Experiment: A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, is not readily removable from the reactor, and that requires one of the following actions to permit removal:

(1) removal of mechanical fasteners

(2) use of underwater handling tools

(3) moving of shield blocks or beam port components

1.3 Reactor Components

Flux Trap: A flux trap is any region within the core whose composition is modified to change the neutron flux.

Hexagonal Ring: A hexagonal ring is one of the concentric hexagonal bands of fuel elements surrounding the central position of the core referred to as the A position. They are designated by letters starting with B for the innermost hexagonal ring.

Instrumented Fuel Rod: An instrumented fuel rod is a special fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating at the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Regulating Control Element: Regulating control element shall mean a low worth control element that may be positioned either manually or automatically by means of an electric motor-operated positioning system and that need not have a scram capability.

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

Standard Control Element: Standard control element shall mean any control element that has a scram capability that is utilized to vary the reactivity of the core, and that is positioned by means of an electric motor-operated positioning system.

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 to 20 weight percent of uranium with a U-235 enrichment of less than 20%.

1.4 Reactor Instrumentation

Channel Calibration: A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond with acceptable accuracy to known values of the measured variables.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison with independent channels measuring the same variable or other measurements of the variables.

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

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

Limiting Safety Systems Settings: Limiting safety systems settings are the settings for automatic protective devices related to those variables having significant safety functions.

Measured Value: The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, interconnecting cables, or lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a variable.

Operable: A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

Reactor Safety Systems: Reactor safety systems are those systems, including their associated input circuits, designed to initiate a scram for the primary purpose of protecting the reactor or to provide information that requires protective action to be initiated.

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

Safety limits: Safety limits are limits on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit – Fuel Element Temperature

Applicability: This specification applies to the maximum temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications:

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

(2) The temperature in an aluminum clad low hydride fuel element shall not exceed 530°C under any conditions of operation.

Bases: The important parameter for a TRIGA reactor is the fuel rod temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel rod cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding, if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the high hydride TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed l000°C and the fuel cladding is water cooled. ~~See GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227.~~

The safety limit for the low hydride fuel elements is based on avoiding the phase change in the zirconium hydride which might cause excessive distortion of a fuel element. This phase change takes place at 530°C, ~~as shown by the phase diagram on page 5-13 of the University of Utah SAR.~~ Additional information is given in Technical Foundations of the TRIGA Report GA-471, pages 63-72, August 1958.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings that prevent the safety limit from being reached.

Objective: The objective is to prevent the safety limits from being exceeded.

Specifications:

(1) For a core composed entirely of stainless steel clad, high hydride fuel elements or a core composed of stainless steel clad, high hydride fuel elements with low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in the following table:

Location of Instrumented Fuel Rod	Limiting Safety System Setting
B-hexagonal ring	800°C
C-hexagonal ring	755°C
D-hexagonal ring	680°C
E-hexagonal ring	580°C

(2) For a core with low hydride fuel elements installed in other than the F or G hexagonal ring, limiting safety system settings apply according to the location of the instrumented fuel rod, as indicated in the following table:

Location of Instrumented Fuel Rod	Limiting Safety System Setting
B-hexagonal ring	460°C
C-hexagonal ring	435°C
D-hexagonal ring	390°C
E-hexagonal ring	340°C

(3) For a core containing flux traps, the limiting safety system settings given in either of the above two tables must be applied to the anticipated hottest fuel element in the core.

Bases: Stainless steel clad, high hydride fuel element: The limiting safety system settings that are indicated represent values of the temperature, which if exceed, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by fuel elements 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-hexagonal ring.

The margin between the safety limit of 1000°C and the limiting safety system setting of 800°C in the B-hexagonal ring was selected to assure that conditions would not arise which would allow the fuel element temperature to approach the safety limit. The safety margin of 200°C accounts for differences between the measured peak temperature and calculated peak temperature encountered during operation and for uncertainty in temperature channel calibration. The thermocouples that measure the fuel-moderator temperature are located nominally midway between the fuel axial centerline and the fuel edge.

During steady-state operations, the equilibrium temperature is determined by the power level, the physical dimensions and properties of the fuel element, and the parameters of the coolant. Because of the interrelationship of the fuel-moderator temperature, the power level, the changes in reactivity required to increase or maintain a given power level, any unwarranted increase in the power level would result in a relatively slow increase in the fuel-moderator temperature. The margin between the maximum setting and safety limit would assure a shutdown before conditions could result that might damage the fuel elements.

Low hydride fuel element: The 460°C maximum limit for the safety system setting gives an ample margin that assures that the safety limit would not be reached through errors in measurement. Temperatures of 460°C have been shown to be safe through extensive operating experience. The surveillance requirement on the measurements of the fuel dimensions will give control over changes that result from the thermal cycling during operation.

The temperatures shown for C. D and E hexagonal ring locations were derived using the power distributions from report GA-4339.

The maximum fuel element temperatures calculated on page III-8 of the UUTR SAR 1985 were 245°C at 100 kW and 440°C at 250 kW. However experimental data based upon 363 critical reactor operations indicate a maximum fuel temperature of 114°C at 100 kW and an extrapolated maximum fuel temperature of 314°C at 300 kW.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 Normal Operation

Applicability: This specification applies to the energy generated in the reactor during normal operation

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded during normal operation

Specifications: The reactor power level shall not deliberately be raised above ~~100~~ 250 kW under any conditions of operation

Basis: Thermal and hydraulic calculations performed by the vendor indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.2 Reactivity limitations

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worth of control elements and experiments.

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

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

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

(2) The rate of reactivity insertion by control rod motion shall not exceed $0.30 per second;

(3) Any experiment with a reactivity worth greater than $1.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor;

(4) The excess reactivity for the cold critical, xenon free condition is less than $2.80;

(5) The reactivity worth of an individual experiment is not more than $2.80.

Bases: The shutdown margin required by specification 3.2(1) is necessary so that the reactor can be shutdown from any operating condition and remain shutdown after cool down and xenon decay even if one control rod should remain in the fully withdrawn position.

Specification 3.2(2) assures that power increases caused by rod motion will be terminated by the reactor safety system before the fuel temperature safety limit is exceeded.

It is assumed that the worst reactivity insertion accident from unrestrained motion of the control rods is initiated from a condition corresponding to reactor startup, between 1 and 100 milliwatts, with a subcritical condition corresponding to the source level. Then 30¢/sec of reactivity insertion continues until the power level scram is tripped. A further delay of 0.1 seconds is used until the control rods begin to insert reactivity at the rate of $3/sec until the rods are inserted. Assuming no thermodynamic feedback occurs (making the calculation quite conservative) it is found that 1.7 mW-sec of energy is produced by the excursion, raising the fuel temperature by 20°F in the peak fuel flux location of the core. This temperature rise is far below the allowable rise to the fuel damage point starting from ambient conditions at startup.

Specifications 3.2(3) is based on Section 8.5 of the UUTR SAR 1985 which indicates that as much as $3.00 reactivity could be inserted in a pulse from a power level of 3 Mwt without violation of the fuel temperature safety limit. By restricting each experiment to reactivity worth of one dollar, an ample margin is provided to allow for uncertainties in the information and the uncertainty in the worth of an experiment.

Specifications 3.2(3) through 3.2(5) 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.

3.3 Control and Safety System

3.3.1 Scram Time

Applicability: This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the safety system setting.

Objective: The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification: The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, the above specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds.

Basis: This specification ensures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

3.3.2 Reactor Control System

Applicability: This specification applies to the information that 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 ensure safe operation of the reactor.

Specification: The reactor shall not be operated unless the measuring channels listed in the following table are operable.

Measuring Channel	Minimum Number Operable
Fuel Element Temperature	1(b)
Reactor Power Level	2
Startup Count Rate	1
Reactor Tank Water Level	1
Area Radiation Monitor	1(a)
Continuous Air Radiation Monitor	1(a)

(a) For periods of time for maintenance to the radiation monitoring systems, the intent of this specification will be satisfied if the installed systems are replaced with portable gamma-sensitive instruments having their own alarms or which shall be kept under visual observation.

(b) For periods of time for maintenance to the standard instrumented fuel element, the reactor shall be in the shutdown condition with all control rods fully inserted, and, power to the control-rod magnets and actuating solenoids has been switched off and the key removed.

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 specifications on the reactor power level indication are included in this section since the power level is closely related to the fuel temperature as shown in Section 5 and Appendix III of the UUTR SAR 1985.

The reactor tank water level detector provides early information of a possible leak in the reactor cooling system or tank.

The radiation monitors provide information to operating personnel of an emergency 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.

3.3.3 Reactor Safety System

Applicability: This specification applies to the reactor safety system channels.

Objective: The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation.

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

Safety System Measuring Channel	Minimum Number Operable	Function
Fuel element temperature	1(a)	Scram at or below Limiting Safety System Setting
Reactor power level	2	Scram at 120 percent of full licensed power
Manual console scam button	1	Manual scram
Magnet current key switch	1	Manual scram
Console power supply	1	Scram on loss of electrical power
Reactor tank water level	1	Scram at one foot below normal operating level
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second
Control rod withdrawal interlocks	All control rods	Prevent manual withdrawal of more than one control rod simultaneously

(a) For periods of time for maintenance to the standard thermocouple fuel element, the reactor shall be in the shutdown condition with all control rods fully inserted, and power to the control-rod magnets and actuating solenoid has been switched off and the key removed.

Bases: The fuel temperature scrams provide 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 scrams are provided as redundant protection against abnormally high fuel temperature and to assure that the reactor operation stays within the licensed limits. The equivalent operation with scrams at 120 percent of full power or ~~120~~ 300 **kW** assures that the reactor operation will terminate well below that power level above which safe cooling may not be available and also at a level below the level temperature scram trip. The manual scrams allow the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure of the console power supply, the console power supply scram provides that operation will not continue without adequate instrumentation. The reactor tank water leak occurs in the primary system or when water level is too low (the result of any cause) for adequate radiation shielding.

3.4 Argon-41 Discharge Limit

Applicability: This specification applies to the concentration of argon-41 that may be discharged from the TRIGA reactor facility.

Objective: To ensure that the health and safety of the public are not endangered by the discharge of argon-41 from the TRIGA reactor facility.

Specification: The concentration of argon-41 released from the facility to the environment shall not exceed 4×10^{-8} μCi/ml averaged over one year.

Basis: It is shown in Appendix B of the UUTR SAR 2005 that the release of argon-41 at the above concentration will not result in exposure in unrestricted areas in excess of the limits of 10 CFR Part 20.

3.5 Engineered Safety Feature - Ventilation System

Applicability: This specification applies to the operation of the facility ventilation system.

Objective: The objective is to ensure 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 ventilation system is operable, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in the dilution mode to prevent the release of a significant quantity of airborne radioactivity from the facility.

Basis: It is shown that during normal operation of the ventilation system the concentration of argon-41 in unrestricted areas is below ~~MPC~~DAC. In the event of a substantial release of fission products, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials (UUTR SAR 1985 Section 8.7.5).

3.6 Limitations on Experiments

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

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

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

(1) Fueled experiments are limited such that the total inventory of iodine isotopes 131 through 135 in the experiment are not greater than 10 millicuries;

(2) 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

(3) Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, or liquid fissionable materials are doubly encapsulated and able to withstand any overpressure condition deemed likely to occur.

Bases: It is shown in the UUTR SAR 1985 (Section 8.7.2) that the limits of Specification 3.6(1) prevent the dose in unrestricted areas resulting from experiment failure from exceeding 10 CFR Part 20 limits from a single accidental exposure averaged on a yearly basis. Specifications 3.6(2) and 3.6(3) are intended to reduce the likelihood of damage to reactor components resulting from experiment failure.

3.7 As Low As Reasonable Achievable (ALARA) Radioactive Effluent Releases

Applicability: This specification applies to the measures required to ensure that the radioactive effluents released from the facility are in accordance with ALARA criteria.

Objective: The objective is to limit the annual population radiation exposure owing to the operation of the TRIGA reactor to a small percentage of the normal local background exposure.

Specifications:

(1) In addition to the radiation monitoring specified in Section 5.4, an environmental radiation monitoring program shall be conducted

to measure the integrated radiation exposure in and round the environs of the facility on an annual basis.

(2) The annual radiation exposure due to reactor operation, at the closest offsite point of extended occupancy shall not, on an annual basis, exceed the average local offsite background radiation by more than 20%.

(3) In the event of a significant fission product leak from a fuel rod or a significant airborne radioactive release from a sample being irradiated, as detected by the continuous air monitor, the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may be operated on a short-term basis as needed to assist in determining the source of the leakage.

Basis: The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site.

3.8 Primary Coolant Conditions

Applicability: This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements, and (2) to minimize neutron activation of dissolved materials.

Specifications:

(1) Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm

(2) the pH of the pool water shall be between 5.0 and 8.0.

Bases: A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

4.0 SURVEILLANCE REQUIREMENTS

4.1 General

Applicability: This specification applies to the surveillance requirements of any system related to reactor safety.

Objective: The objective is to verify the proper operation of any system related to reactor safety.

Specification: Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the control element drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Committee. A system shall not be considered operable until after it has been successfully tested.

Basis: This specification relates to changes in reactor systems that could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, it can be assumed that they meet the presently accepted operating criteria.

4.2 Safety Limit - Fuel Element Temperature

Applicability: This specification applies to the surveillance requirements of the fuel element temperature measuring channel.

Objective: The objective is to ensure that the fuel element temperatures are properly monitored.

Specifications:

(1) Whenever a reactor scram caused by high fuel element temperature occurs, the peak indicated fuel temperature shall be examined to determine whether the fuel element temperature safety limit was exceeded.

(2) The fuel element temperature measuring channel shall be calibrated semi-annually or at an interval not to exceed 8 months by the substitution of a known signal in place of the instrumented fuel element thermocouple.

(3) a channel check of the fuel element measuring channel shall be made each time the reactor is operated by comparing the indicated instrumented fuel element temperature with previous values for the core configuration and power level.

Basis: Operational experience over the past 8 years with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperature have been sufficiently reliable to ensure accurate indication of this parameter.

4.3 Limiting Conditions for Operation

4.3.1 Reactivity Requirements

Applicability: These specifications apply to the surveillance requirements for reactivity control.

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

Specifications:

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

(2) The control rods shall be visually inspected for deterioration at intervals not to exceed 2 years.

Basis: The reactivity worth of the control rods is measured to ensure the required shutdown margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor.

4.3.2 Control and Safety System

Applicability: These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective: The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specifications:

(1) The scram time shall be measured annually but at intervals not to exceed 15 months.

(2) A channel check of each of the reactor safety system channels shall be performed before each day's operation or before each operation extending more than 1 day, except for the pool level channel which shall be tested monthly.

(3) A channel calibration shall be made of the power level monitoring channels by either nuclear or calorimetric methods annually, but at intervals not to exceed 15 months.

(4) A channel test of the temperature measuring channel shall be performed semiannually, but at intervals not to exceed 8 months.

Basis: Measurements of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly. The channel tests will ensure that the safety system channels are operable on a daily basis or before an extended run. The power level channel calibration will ensure that the reactor will be operated at the proper power levels.

4.3.3 Radiation Monitoring System

Applicability: This specification applies to the surveillance requirements for the area radiation monitoring equipment and the continuous air monitoring system.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification: The area radiation monitoring system and the continuous air monitoring system shall be calibrated annually and shall be verified to be operable at monthly intervals.

Basis: Experience has shown that monthly verification of area radiation and air - monitoring system setpoints in conjunction with annual calibration is adequate to correct for any variation in the system caused by a change of operating characteristics over a long time span.

4.3.4 Ventilation System

Applicability: This specification applies to the reactor room 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 reactor room ventilation system is in operation, establishing a negative air pressure within the reactor room, except for periods of time not to exceed 48 hours to permit repair of the system.

Basis: It is shown that during normal operation of the ventilation system the concentration of argon-41 in unrestricted areas is below ~~MPC~~DAC. In the event of a substantial release of fission products, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials (UUTR SAR 1985 Section 8.7.5).

4.3.5 Experiment and Irradiation Limits

Applicability: This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities and for irradiations performed in the irradiation facilities.

Specifications:

(1) A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed by the Reactor Supervisor and reviewed by the Reactor Safety Committee. Minor modifications to reviewed and approved experiments may be made at the discretion of the senior operator responsible for the operations provided that the hazards associated with the modifications have been reviewed and a determination has been made that the modifications do not create a significantly different, a new, or a greater hazard than the original approved experiment.

(2) An irradiation of a new type of device or material shall not be performed until an analysis of the irradiation has been performed and reviewed by the Reactor Supervisor.

Basis: It has been demonstrated over a number of years that experiments and irradiations reviewed by the reactor staff and the Reactor Safety Committee, as appropriate, can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

4.4 Reactor Fuel Elements

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

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

Specifications: All fuel elements shall be inspected visually for damage or deterioration every two years. Any fuel element which appears damaged shall be measured for length and bend. The reactor shall not be operated with damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

(1) in measuring the transverse bend, its sagitta exceeds 0.125 inches over the length of the cladding,

(2) in measuring the elongation, its length exceeds its original length by 0.250 inches,

(3) a clad defect exists as indicated by release of fission products. However, the reactor may be operated on a short-term basis as needed to assist in determining the source of the leakage.

Bases: The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding. The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to ensure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to ensure adequate coolant flow.

4.5 Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary water quality.

Objective: The objective is to ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured monthly and shall be as follows:

(1) *conductivity < 5 x 10-6 mhos/cm.*

(2) pH between 5.0 and 8.0

Bases: Section 3.8 ensures that the water quality is adequate during reactor operation. Section 4.5 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

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 ensure 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 physical and nuclear characteristics.

Specifications:

Standard TRIGA Fuel - Each individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

a. High Hydride Fuel Element - Each high hydride fuel element shall contain uranium-zirconium hydride and be clad with 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 920.0 weight percent uranium which has a maximum enrichment of less than 20 percent and 1.5 to 1.8 hydrogen atoms to 1.0 zirconium atom.

b. Low Hydride Fuel Element - Each low hydride fuel element shall contain uranium-zirconium hydride and be clad with 0.030 inch of aluminum or 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 9 weight percent uranium which has a maximum enrichment of less than 20 percent and 0.9 to 1.6 hydrogen atoms to 1.0 zirconium atom.

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

5.2 Reactor Core

Applicability: This specification applies to the configuration of fuel and incore experiments.

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

Specifications:

(1) The core shall be in arrangement of TRIGA uranium-zirconium hydride fuel elements positioned in the reactor grid plate.

(2) The reflector, excluding experiments and experimental facilities, shall be a combination of graphite, aluminum, and light water and heavy water.

Bases: Standard TRIGA cores have been used for years and their characters are well documented. Mixed cores of standard fuel have been tested and operated at a number of university reactors. Calculations, as well as measured performance of mixed cores have shown that such cores may be safely operated.

The core will be assembled in the reactor grid plate which is located in a pool of light water. Light water in combination with heavy water and graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Elements

Applicability: This specification applies to the control elements used in the reactor core.

Objective: The objective is to ensure that the control elements are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) The standard control element shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding.

(2) The regulation control element need not have scram capability and shall be a stainless steel element or contain the materials as specified for standard control elements.

Basis: The poison requirements for the control elements are satisfied by using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid insertion of the control element which is the primary safety feature of the reactor.

5.4 Radiation Monitoring System

Applicability: This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective: The objective is to describe the radiation monitoring equipment that is available to the operator to ensure safe operation of the reactor.

Specifications:

(1) Function of Area Radiation Monitor (gamma-sensitive instruments): Monitor radiation fields in key locations, alarm and readout at control console.

(2) Function of Continuous Air Radiation Monitor (beta-, gamma-sensitive detector with particulate collection capability): Monitors concentration of radioactive particulate activity and radioactive gases including Argon-41 in the building exhaust, alarm and readout at control console.

Basis: The radiation monitoring system is intended to 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.

5.5 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 ensure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications:

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

(2) Irradiated fuel elements and fueled devices shall be stored in an array, which will permit sufficient natural convection cooling by water or air, so that the fuel element or fueled device temperature will not exceed design values.

Basis: The limits imposed by the Specifications 5.5(1) and 5.5(2) are conservative and ensure safe storage of reactor fuel.

5.6 Reactor Building and Ventilation System

Applicability: This specification applies to the building that houses the reactor.

Objective: The objective is to ensure that provisions are made to restrict the amount of radioactivity released into the environment.

Specifications:

(1) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 5×10^8 cm^3.

(2) The reactor building shall be equipped with, a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 40 feet from ground level.

Basis: The facility is designed so that the ventilation system will normally maintain a negative pressure with respect to the atmosphere to minimize uncontrollable leakage to the environment. The free air volume within the reactor ~~building~~ room is confined when there is an emergency shutdown of the ventilation system. Proper handling of airborne radioactive materials (in emergency situations) can be effected with a minimum of exposure to operating personnel.

5.7 Reactor Pool Water Systems

Applicability: This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective: The objective is to ensure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications:

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 ft below the surface of the pool shall have adequate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate a loss of coolant if the pool level drops more than 2 ft below the normal level.

(4) The reactor shall not be operated with less than 18 ft of water above the top of the core.

Bases: This specification is based on, thermal and hydraulic calculations which show that the TRIGA core can operate in a safe manner at power levels up to 2700 kW with natural convection flow of the coolant water. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for standard cores.

In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool.

Loss of coolant alarm after 2 ft of loss requires corrective action. This alarm is observed in the reactor control room, and at the campus police station.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

1) The UUTR is an integral part of the Nuclear Engineering Facilities (NEF) of the University of Utah. The organization of the facility management and operation shall be as shown in Figure 6.1. The responsibilities and authority of each member of the operating staff shall be defined in writing.

2) As indicated in Fig. 6.1, the Reactor Safety Committee shall report to Level 1. Radiation safety personnel shall report to Level 2. Additional description of levels follows:

   a. *Level 1*: Individual responsible for the reactor facility's licenses, i.e., the Associate Vice President for Research in the Office of Vice President for Research; The Vice President for Research will assign which of the Associate Vice Presidents for Research will be the responsible Level 1 individual.

   b. *Level 2*: Individual responsible for reactor facility operation, i.e., the facility director shall be the Director of the Utah Nuclear Engineering Program, who shall also be the Director of the Nuclear Engineering Facilities (NEF) of the University of Utah.

   c. *Level 3*: Individual responsible for day-to-day operation or shift shall be the reactor supervisor (RS). This person shall be a senior reactor operator (SRO).

   d. *Level 4*: Operating staff shall be senior reactor operators, reactor operators, and trainees.

3) Transition plan shall be defined as follows:

   a. In the absence of the Level 2 individual [sabbatical leave, sudden leave]: Level 1 shall have the authority to automatically appoint the RS as an interim reactor administrator (RA) with direct access to Level 1 and communication to U.S.N.R.C; Radiation safety personnel shall report to RA unless Level 1 decides differently.

   b. In the absence of the Level 3 individual [sabbatical leave, sudden leave]: Level 2 shall have the authority to automatically appoint a new RS who shall be a most senior Level 4 individual (SRO).

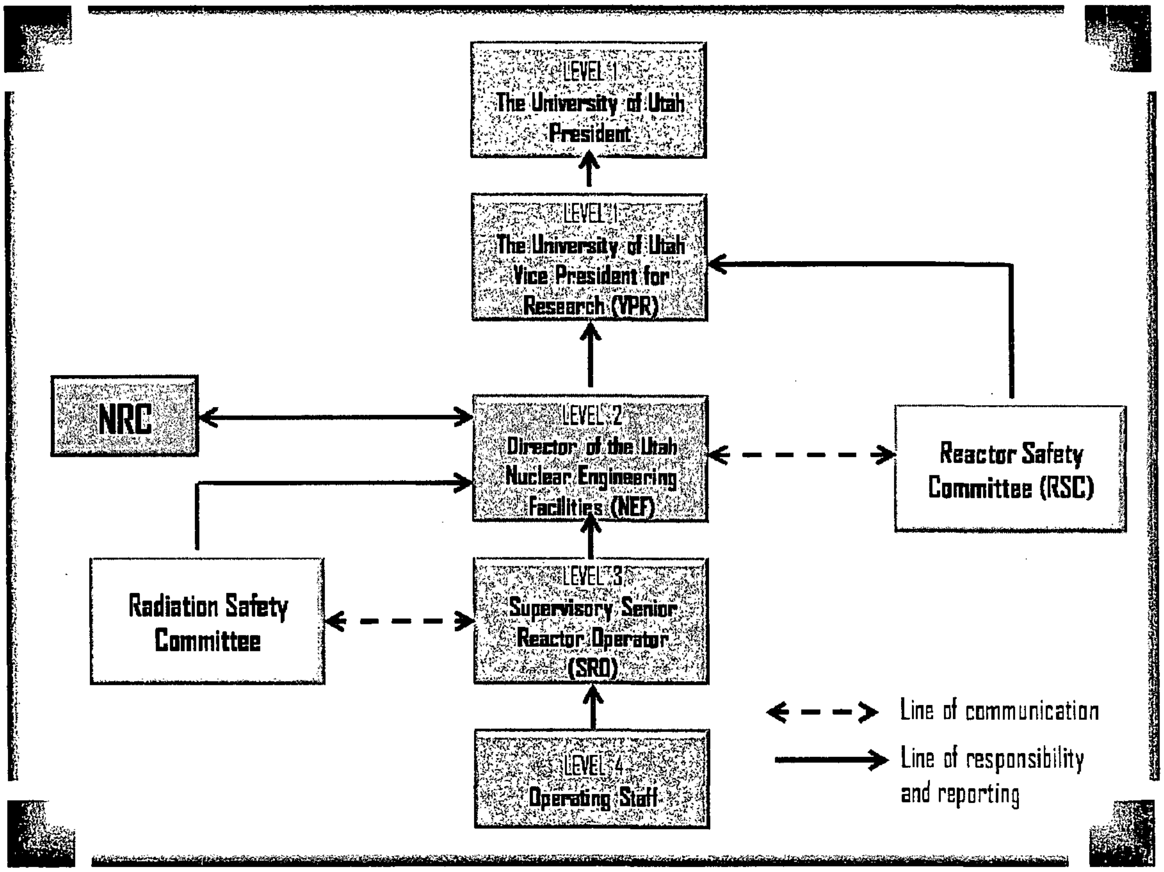

Figure 6.1

University of Utah Administrative Organization for Nuclear Reactor Operations [Clarified - December 2009; in compliance with ANSI/ANS 15.1-2007]

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Fig. 6.1 following ANSI/ANS 15.1-2007:

1) Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license or charter and technical specifications.

2) In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

3) The reactor facility shall be under the direct control of a licensed Senior Reactor Operator (SRO) designated by the Reactor Supervisor (RS) who is also a licensed Senior Reactor Operator. The SRO shall be responsible to the RS for the overall facility operation including the safe operation and maintenance of the facility and associated equipment. The SRO shall also be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, federal and state regulations, and requirements of the Reactor Safety Committee.

6.1.3 Staffing

1) When the reactor is not secured, the minimum staff shall consist of:

    a. A licensed reactor operator in the control room, e.g., Reactor Operator (RO) (may be the SRO or RS).

    b. A designated Senior Reactor Operator (SRO) on call but not necessarily on site.

    c. Another person present at the facility complex who is able to carry out prescribed written instructions.

2) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. Level 1/2The list shall include:

    a. management personnel,

    b. radiation safety personnel,

    c. other operations personnel;

3) Events requiring the presence at the facility of the senior reactor operator are:

    a. initial startup and approach to power,

b. all fuel or control-rod relocations within the reactor core region,

c. relocation of any experiment with reactivity worth greater than one dollar;

d. recovery from unplanned or unscheduled shutdown or significant power reduction.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard "Selection and Training of Personnel for Research Reactors," ANSI0ANS-15.4-1988 ~R1999, Sections 4 through 7.

The Reactor Supervisor shall be responsible for the facility's Requalification Training Program and Operator Training Program.

6.2 Reactor Safety Committee

6.2.1 Function

The Reactor Safety Committee (RSC) shall function to provide an independent review and audit of the facility's activities including:

1) reactor operations
2) radiological safety
3) general safety
4) testing and experiments
5) licensing and reports
6) quality assurance

6.2.2 Composition and Qualifications

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members shall collectively represent a broad spectrum of expertise in the appropriate reactor technology.

The members of the committee shall include the Reactor Supervisor and faculty and staff members designated to serve on the committee. The University's Radiation Safety Officer shall be an ex officio member of the RSC.

Members and alternates shall be appointed by and report to Level 1 management. Individuals may be either from within or outside the operating organization (univ. Qualified and approved alternates may serve in the absence of regular members.

6.2.3 Operation

The Reactor Safety Committee shall operate in accordance with a written charter, including provisions for:

1) meeting frequency: not less than once per calendar year and more frequently as circumstances warrant, consistent with effective monitoring of facility activities

2) voting rules and quorums: a quorum shall be made up of the chairman or his designate and voting members such that at least half of the voting body is represented, and where the Level 2 facility director and anyone who reports directly to him/her (such as Level 3 and Level 4 personnel) does not constitute a voting majority

3) use of subcommittees

4) review, approval, and dissemination of minutes in timely manner

6.2.4 Review Function

The responsibilities of the RSC or designed subcommittee(s) thereof shall include, but is not limited to, the following:

1) review and approval of all new experiments utilizing the reactor facility

2) review and approval of all proposed changes to the facility license by amendment, and to the Technical Specifications

3) review of the operation and operational records of the facility

4) review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that effect nuclear safety

5) review and approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or on unreviewed safety questions as defined by 10 CFR 50.59

6) review of reportable occurrences and the reports filed with the Commissions for

said occurrences

7) review and approval of all standard operating procedures and changes thereto

8) biennial review of all standard procedures, the facility emergency plan, and the facility security plan.

A written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 and the review and audit group members in a timely manner after the review has been completed.

6.2.5 Audits

The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. The semiannual audit shall include at least the following:

1) review of the reactor operating records
2) inspection of the reactor operating areas
3) review of unusual or abnormal events
4) radiation exposures at the facility and adjacent environs

6.2.6 Records

The activities of the RSC shall be documented by the committee and the RSC shall maintain a file of the minutes of all meetings.

6.3 Radiation Safety

The Radiation Safety Office of the University of Utah shall be assigned responsibility for implementing the radiation protection program at the reactor using the guidelines of American National Standard "Radiation Protection at Research Reactor Facilities," ANSI/ANS-15.11-1993 (R2004). The Radiation Safety Officer shall report to Level 2 [Fig. 6.1].

6.4 Operating Procedures

Written operating procedures shall be adequate to ensure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

1) performing irradiations and experiments
2) startup, operation, and shutdown of the reactor
3) emergency situations including provisions for building evacuation, earthquake, radiation emergencies, fire or explosion, personal injury, civil disorder, and bomb threat
4) core changes and fuel movement
5) control element removal and replacement
6) performing preventive maintenance and calibration tests on the reactor and associated equipment
7) power equipment

Substantive changes to the above procedures shall be made only with the approval of the Level 2. Minor modifications to the original procedures that do not change their original intent may be made by Level 3 or higher, but the modifications must be approved by Level 2. Temporary changes to the procedures that do not change their original intent may be made by a licensed SRO. All such changes shall be documented and subsequently reviewed by the Reactor Safety Committee. Such deviations shall be documented and reported within 24 hours or the next working day to the Level 2.

6.5 Experiments Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures as described in 6.4.

6.6 Required Actions

6.6.1 Action To Be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded:

1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission (U.S.N.R.C.).

2) An immediate report of the occurrence shall be made to Level 2 and to the Chairman of the Reactor Safety Committee, and reports shall be made to the U.S.N.R.C in accordance with Section 6.7 of these specifications.

3) A report shall be prepared that 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 recurrence. This report shall be submitted to the Reactor Safety Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

4) A report shall be made to the U.S.N.R.C. in accordance with Section 6.7 of these specifications.

6.7 Reports

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

1) A report within 24 hours by telephone to the Project Manager U.S.N.R.C., of

   a. any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

   b. any violation of the safety limit;

   c. any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

2) A report within 10 days in writing to the Document Control Center, U.S.N.R.C., Washington, D.C., with a copy to the U.S.N.R.C. Operations, of

   a. any accidental release or radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

   b. any violation of safety limit,

   c. any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

3) A report within 30 days in writing to the Document Control Center, U.S.N.R.C., Washington, D.C., with a copy to the U.S.N.R.C. Operations, of

   a. 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,

   b. any significant change in the transient or accident analysis is described in the Safety Analysis Report,

   c. any significant changes in facility organization,

   d. any observed inadequacies in the implementation of administrative or procedural controls.

4) A report within 60 days after completion of startup testing of the reactor (in writing to the Director, Office of Nuclear Reactor Regulation, 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 describing the measured values of the operating conditions including:

   a. an evaluation of facility performance to date in comparison with design predictions and specifications,

   b. a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

5) An annual report within 60 days following the 30^{th} of June of each year (in writing) to the Document Control Center, U.S.N.R.C., Washington, D.C. with a copy to the U.S.N.R.C. Operations, providing the following information:

   a. a brief narrative summary of (i) operating experience (including experiments performed), (ii) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (iii) results of surveillance tests and inspections:

   b. tabulation of the energy output (in Megawatt-days) of the reactor, hours reactor was critical and the cumulative total energy output since initial criticality;

   c. the number of emergency shutdowns and inadvertent scrams, including reasons for them;

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

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

f. 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 before the point of such release or discharge:

Liquid Waste (summarized on a monthly basis)

i. radioactivity discharged during the reporting period

- total estimated quantity of radioactivity released (in curies) an estimation of the specific activity for each detectable radionuclide present if the specific activity of the released material after dilution is greater than 1×10^{-7} µCi/cc.

- summary of the total release (in curies) of each nuclide determined just above for the reporting period based on representative isotopic analysis,

- estimated average concentration of the released radioactive material at the point of release for the reporting period in terms of µCi/cc and fraction of the applicable DAC or Effluent concentration

ii. total volume (in gallons) of effluent water (including dilutant) released during each period of release.

Gaseous Waste (summarized on a monthly basis)

i. radioactivity discharged during the reporting period (in curies)

- total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method,

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

- estimated average atmospheric diluted concentration of argon-41 released during the reporting period in terms of µCi/cc and fraction of the applicable DAC value.

- total estimated quantity of radioactivity in particulate form with half-lives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system.

- average concentration of radioactive particulates with half-lives greater than 8 days released in µCi/cc during the reporting period, and

- an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of µCi/cc and fraction of the applicable DAC-value for the reporting period if the estimated release is greater than 20% of the applicable DAC.

Solid Waste (summarized on an annual basis)

i. total amount of solid waste packaged (in cubic feet),

ii. total activity in solid waste (in curies),

iii. the dates of shipment and disposition (if shipped off site).

g. An annual summary of the radiation exposure received by facility personnel and visitors in terms of the average radiation exposure per individual and greater exposure per individual and greater exposure per individual in the two groups. Each significant exposure in excess of the limits of 10 CFR 20 should be reported, including the time and data of the exposure as well as the circumstances that led up to the exposure.

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

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

6.8 Facility Operating Records

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, records and logs shall be prepared for at least the following items and retained for a period of at least 5 years for items 1) through 6) and indefinitely for items 7) through 11).

1) normal reactor operation

2) principal maintenance activities

3) abnormal occurrences

4) equipment and component surveillance activities required by the Technical Specifications

5) experiments performed with the reactor

6) gaseous and liquid radioactive effluents released to the environs

7) offsite inventories and transfers

8) fuel inventories and transfers

9) facility radiation and contamination surveys

10) radiation exposures for all personnel

11) updated, corrected, and as-built drawings of the facility

6.9 Quality Assurance

In accordance with Regulatory Guide 2.5 and ANSI 402, "Quality Assurance Program Requirements for Research Reactors," Section 2.17, the facility shall not be required to prepare quality assurance documentation for the as-built facility. Quality assurance (QA) requirements will still be limited to those specified in Section 2.17 as follows:

"All replacements, modifications, and changes to systems having a safety related function shall be subjected to a QA review. Insofar as possible, the replacement, modification, or change shall be documented as meeting the requirements of the original system or component and have equal or better performance or reliability."

"The required audit function shall be performed as specified in Section 6.2.5."