Appendix: Technical Specifications

This appendix is included in response to question 38.

FACILITY LICENSE NO. R-76

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE

WASHINGTON STATE UNIVERSITY

MODIFIED TRIGA REACTOR

DOCKET NO. 50-27

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TECHNICAL SPECIFICATIONS AND BASES FOR THE
WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR

This document constitutes the Technical Specifications for Facility License No. R-76 and supersedes all prior Technical Specifications. Included in these Technical Specifications are the "Bases" to support the selection and significance of the specification. These 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 numerical values given in these specifications may differ slightly from actual values because of normal construction and manufacturing tolerances, or normal degree of accuracy or instrumentation.

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 the 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.

Pulse Mode: Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

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 in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core 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 of operation listed in Section 3.0

(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 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) 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.

Steady-State Mode: Steady-state mode operation shall mean any operation of the reactor with the mode selector switch in the steady-state position.

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.

Experimental Facilities: Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, in-core irradiation baskets or tubes, pneumatic transfer systems, and any other in-pool 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 the radiation available in that irradiation facility.

Irradiation Facilities: Any in-pool 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, that 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 Component

TRIGA 30/20 LEU Fuel: 30/20 LEU fuel is TRIGA fuel that contains a nominal 30 weight percent of uranium with a nominal ^{235}U enrichment of 19.75% and erbium, a burnable poison.

Fuel Bundle: A fuel bundle is a cluster of three or four fuel rods fastened together in a square array by a top handle and bottom grid plate adapter.

Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard or 30/20 LEU fuel.

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.

Mixed Core: A mixed core is a core arrangement containing Standard and 30/20 LEU fuels with at least 22 30/20 LEU fuel rods located in the central positions in the core.

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.

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 Core: A standard core is any arrangement of all-Standard fuel.

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

Transient Control Element: Transient control element shall mean any control element that has the capability of being rapidly withdrawn from the reactor core by means of a pneumatic drive, that is capable of being positioned by means of an electric motor-operated positioning system, and that has scram capabilities.

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 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 Setting: 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.

PTR (Peak-to-Measured-Fuel Temperature Ratio): The PTR is defined as the ratio between the maximum calculated fuel temperature in a given core arrangement to that measured by the instrumented fuel element.

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 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:

(1) The maximum temperature in a Standard TRIGA fuel rod shall not exceed 1000°C under any condition of operation.

(2) The maximum temperature in a 30/20 LEU TRIGA fuel rod shall not exceed 1150°C under any condition 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 30/20 LEU fuel is based on data that indicate that the stress in the cladding because of the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1150°C and the fuel cladding is water cooled.* The safety limit for the Standard 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 1000°C and the fuel cladding is water cooled.*

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 reached.

Specifications: The limiting safety system settings shall be 500°C as measured in an instrumented fuel rod located in the central region of the core. For a mixed core, the instrumented rod shall be located in the region of the core containing the 30/20-type fuel rods.

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*GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227.

Bases: The limiting safety system setting is the measured instrumented fuel rod temperature that, if exceeded, shall initiate a scram to prevent the fuel temperature safety limit from being exceeded.

The temperature safety limit for LEU fuel is 1150° C (2100° F). Due to various errors in measuring temperature in the core, it is necessary to arrive at a Limiting Safety System Setting (LSSS) for the fuel element safety limit that takes into account these measurement errors. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. Although a lesser contributor to error, a minimum safety margin of 10% was applied on an absolute temperature basis. Adjusting the fuel temperature safety limit to degrees Kelvin, K, and applying a 10% safety margin results in a safety limit reduction of 150° C. Applying this first margin of safety, the safety setting would be 1000° C for LEU. However, to arrive at the final LSSS it is also necessary to allow for the difference between the measured temperature value and the peak core temperature, which is a function of the location of the thermocouple elements within the core. For example, if one of the thermocouple elements were located in the hottest position in the core, location D4NE, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at WSU the IFEs are located in core locations D6NW and C4NW. Calculations indicate that, for these cases, the true temperature at the hottest location in the core will differ from the measured temperatures by no more than 16%. When applying this 16% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a limit temperature of about 850° C (1562° F) is obtained. Finally a margin of 350° C (660° F) was imposed in setting the LSSS temperature at 500° C (930° F).

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will not limit the peak power generated during the pulse because of the relatively long response time of the temperature channel as compared with the width of a pulse. On the other hand, the temperature scram would limit the total amount of energy generated in a pulse by cutting off the "tail" of the energy transient in the event that the fuel temperature limit is exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation to protect the fuel in the event of such conditions as sticking of the transient control element in the withdrawn position after a pulse.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 Steady-State Operation

Applicability: This specification applies to the energy generated in the reactor during steady-state operation.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded during steady-state operation.

Specifications: The reactor power level shall not exceed 1.3 MW under any condition 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. They apply for all modes of operation.

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 shutdown margin provided by control elements shall be 0.25$ or greater with:

(1) the highest worth nonsecured experiment in its most reactive state

(2) the highest worth control element and the regulating element (if not scrammable) fully withdrawn

(3) the reactor in the cold critical condition without xenon

Basis: The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the highest worth rod should remain in the fully withdrawn position. If the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

3.3 Pulse Mode Operation

Applicability: This specification applies to the peak fuel temperature in the reactor as a result of a pulse insertion of reactivity.

Objective: The objective is to ensure that fuel element damage does not occur in any fuel rod during pulsing.

Specifications: The maximum reactivity inserted during pulse mode operation shall be such that the peak fuel temperature in any fuel rod in the core does not exceed 830°C. The maximum safe allowable reactivity insertion shall be calculated annually for an existing core and prior to pulsing a new or modified core arrangement.

Basis: TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for 30/20 LEU fuel and 1.65 for Standard. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000° C. However, after extensive steady-state operation at 1 MW, the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874° C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow larger with each pulse. The expansion of the fuel stresses and distorts the fuel rod material which, in turn, can cause overall swelling and distortion of the cladding and entire fuel rod. The pulsing limit of 830° C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874° C to 830° C reduces hydrogen pressure by a factor of two, which provides an acceptable safety factor. This phenomenon does not alter the steady-state safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

3.4 Maximum Excess Reactivity

Applicability: This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective: The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

Specifications: The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.6% $\Delta k/k$.

Basis: Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, that specification does not address the total reactivity available within the core. This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.5 Core Configuration Limitation

Applicability: This specification applies to mixed cores of 30/20 LEU and Standard types of fuel.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded as a result of power peaking effects in a mixed core.

Specifications:

(1) The 30/20 LEU-fueled region in a mixed core shall contain at least 22 30/20 LEU fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the 30/20 LEU region shall be limited to nonadjacent single-rod holes.

(2) The PTR as defined in Section 1.4 shall not exceed 1.5 for an operational core.

Bases: The limitation on the allowable core configuration as set forth in Section 4.1 of the FLIP fuel conversion safety analysis report limits power peaking effects. The limitation on power peaking effects ensures that the fuel temperature safety limit will not be exceeded in a mixed core.

A 500° C Limiting Safety System Setting and a 1.5 PTR limit the maximum possible steady-state fuel temperature in the 30/20 LEU region to less than 750° C.

3.6 Control and Safety System

3.6.1 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.

Specifications: 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.

3.6.2. 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.
Specifications: The reactor shall not be operated in the specified mode of operation unless the measuring channels listed in Table 3.1 are operable.

Table 3.1 Measuring Channels

Measuring Channel	Min. no. operable	Effective mode	
		SS	pulse
Fuel element temperature	1	X	X
Linear power level	1	X	
Log power level	1	X	
Integrated pulse power	1		X

Note: SS = steady-state

Bases: Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation. The specifications on reactor power level indication are included in this section since the power level is related to the fuel temperature.

3.6.3 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.

Specifications: The reactor shall not be operated unless the safety channels described in Table 3.2 are operable.

Table 3.2 Minimum Reactor Safety Channels

		Number operable in specified mode	
Safety Channel	Function	SS	Pulse
Fuel temperature	Scram if fuel temperature exceeds 500° C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	1	
Manual scram	Manually initiated scram	1	1
Wide range	Prevent initiation of a pulse above 1 kW		1
	Prevent control element withdrawal when neutron count is less than 2 cps	1	
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Pulse-mode switch	Prevent withdrawal of standard control and regulation elements in pulse mode		1
Preset timer	Transient rod scram 15 seconds or less after pulse		1
Pool level	Alarm if pool level falls below 16 ft over the core	1	1
Transient rod control	Prevent application of air unless fully inserted	1	

Note: SS = steady-state

Bases: The fuel temperature and power level scrams provide protection to ensure that the reactor can be shut down before the safety limit on the fuel element temperature has been exceeded. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety chambers, operation of the reactor without adequate instrumentation is prevented. The preset timer ensures that the reactor power level will reduce to a low level after pulsing. The interlock to prevent startup of the reactor with less than 2 cps ensures that sufficient neutrons are available for proper startup.

The interlock to prevent the initiation of a pulse above 1 kW is to ensure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent withdrawal of the standard or regulating control elements in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The pool level alarm is intended to alert the operator to any significant decrease in the pool level.

3.7 Radiation Monitoring System

Applicability: This specification applies to the radiation monitoring information which must be available to the reactor operator during reactor operation.

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

Specifications: The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.3 are operable. Each channel shall have a readout in the control room and be capable of sounding an audible alarm that can be heard in the reactor control room.

Basis: The radiation monitors inform operating personnel about 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.

Table 3.3 Minimum Monitoring Channels

Channel*	Function	No.
Area radiation monitor	Monitor radiation level on the bridge	1
Area radiation monitor	Monitor radiation level in the beam room	1
Continuous air monitor	Monitor the activity of the pool room air	1
Exhaust gas monitor	Monitor the Argon-41 activity in the exhaust	1

*For periods of time for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma-sensitive instruments having their own alarms or that shall be kept under visual observation.

3.8 Argon-41 Discharge Limit

Applicability: This specification applies to the concentration of ^{41}Ar that may be discharged from the WSU TRIGA reactor facility.

Objective: To ensure that the health and safety of the public are not endangered by the discharge of ^{41}Ar from the WSU TRIGA reactor facility.

Specification: The concentration of ^{41}Ar in the effluent gas from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, shall not exceed 1×10^{-8} μCi/ml averaged over one year.

Basis: The maximum allowable concentration of ^{41}Ar in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 1×10^{-8} μCi/ml. Section 6.5 of the safety analysis report for conversion of the WSU TRIGA reactor to FLIP fuel substantiates a 3.4×10^{-3} atmospheric dilution factor for a 4.4 mph wind speed. A somewhat more conservative value of 4×10^{-3} has been selected for the calculation of ^{41}Ar dilution.

3.9 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.

Specifications: 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: During normal operation of the reactor and the ventilation system, the concentration of ^{41}Ar and other airborne radionuclides discharged from the facility is below the applicable maximum air effluent concentration (AEC) values. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in a dilution mode as appropriate. This action will permit minimizing the concentration of airborne radioactive materials discharged to the environment until it is within the appropriate AEC value. In addition, operation of the reactor with the ventilation system shut down for short periods of time to make system repairs or tests does not compromise the control over the release of airborne radioactive materials. Moreover, radiation monitors within the building, independent of the ventilation system, will give warning of high levels of radiation that might occur during operation with the ventilation system secured.

3.10 Limitations on Experiments

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities (defined in Section 1.2).

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) Nonsecured experiments shall have reactivity worths less than 1.00$.

(2) The reactivity worth of any single experiment shall not exceed 2.00$.

(3) Total worth of all experiments will not exceed 5.00$.

(4) Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 mg shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

(5) Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the

experiment, shall be limited in activity so that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the applicable limits of Appendix B of 10 CFR 20.

In calculations pursuant to item 5 above, the following assumptions shall be used:

- If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.
- If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 μm particles, at least 10% of these particles can escape.
- For materials whose boiling point is above 60°C and in cases where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.
- An atmospheric dilution factor of 4×10^{-3} for gaseous discharges from the facility.

(6) Each fueled experiment shall be controlled so that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Ci.

(7) If a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means, that material shall be removed and physically inspected to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the senior operator responsible for the operation and must be determined to be satisfactory before operation of the reactor is resumed.

Bases:

(1) This specification is intended to provide assurance that the worth of a single unsecured experiment will be limited to such a value that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted.

(2) The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since an experiment of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase so that the reactor protective systems would act to prevent power levels from exceeding the safety limits.

(3) The total worth of all experiments is limited to ensure that the reactor will remain subcritical in the event of a simultaneous removal of all of the experiments with one safety control element withdrawn.

(4) This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

(5) This specification is intended to reduce the likelihood that radioactive airborne particles in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility.

(6) The 1.5 Ci limitation on iodine isotopes 131 through 135 ensures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR 20 for an unrestricted area.

(7) Operation of the reactor with the reactor fuel or structure damaged is prohibited (to avoid release of fission products).

3.11 Limitations on Irradiations

Applicability: This specification applies to irradiations performed in the irradiation facilities contained in the reactor pool as defined in Section 1.2, "Irradiation Facilities." Irradiations are a subclass of experiments that falls within the specifications hereinafter stated in this section. The surveillance requirements for irradiations are given in Section 4.3.5(2).

Objective: The objective is to prevent damage to the reactor, excessive release of radioactive materials, or excessive personnel radiation exposure during the performance of an irradiation.

Specifications: A device or material shall not be irradiated in an irradiation facility under the classification of an irradiation unless all the following conditions exist:

(1) The irradiation meets all the specifications of Section 3.10 for an experiment.

(2) The expected radiation field produced in air by the device or sample upon removal from the reactor pool is not more than 10 rem/hr beta and gamma equivalent at 1 ft; otherwise, it shall be classed as an experiment.

(3) The device or material is encapsulated in a suitable container.

(4) The reactivity worth of the device or material is 0.25$ or less; otherwise, it shall be classed as an experiment.

(5) The device or material does not remain in the reactor for more than a 15-day period; otherwise, it shall be classed as an experiment.

Basis: This specification is intended to provide assurance that the special class of experiments called irradiations will be performed in a manner that will not permit any safety limit to be exceeded.

3.12 As Low As Reasonably 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 WSU 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 around the environs of the facility on a quarterly basis.

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

(3) Whenever practicable, the reactor shall be operated 4 in. or more from the thermal column in order to minimize the production of ^{41}Ar.

(4) The total annual discharge of ^{41}Ar into the environment shall not exceed 20 Ci per year.

(5) 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 continue to be operated on a short-term basis as needed to assist in determining the source of the leakage.

(6) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for their beta-gamma activity content. The total annual quantity of liquid effluents released (above background) shall not exceed 1 Ci per year.

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.

3.13 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 7.5.

Basis: 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.

3.14 Sealed Sources in the Reactor Pool

Applicability: This specification applies to any and all sealed sources stored or used in the reactor pool.

Objective: The objectives of this requirement are to ensure that: 1) any sealed source or sources that are stored or used in the pool do not constitute any type of significant hazard to the operation of the reactor, 2) any such sealed source or sources do not create a significant environmental or personal radiation exposure hazard, and 3) any such sealed source or sources do not compromise the ALARA criteria of the facility.

Specifications:

(1) Sealed sources shall not at any time be stored or used closer than five (5) feet away from the face of an operating reactor core. The total activity of all sealed sources stored in the pool shall not exceed 100,000 curies. All sealed source configurations shall be designed so that a loss of pool water accident will not precipitate a sealed source incapsulation integrity problem and the sources shall be stored in an appropriate shield so as not to produce a significant radiation hazard in the event of a loss of reactor pool water accident.

(2) All storage of sealed sources greater than 100 curies in the reactor pool shall be considered as an experiment and shall be reviewed and approved by the Reactor Safeguards Committee. A written operating procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

(3) The radionuclide content of the reactor pool water shall be monitored monthly at an interval not to exceed six (6) weeks in order to detect a significant leak in the sources stored in the reactor pool. If the specific radionuclide content of the pool water for radionuclides from a sealed source stored in the reactor pool exceeds one-third (1/3) the 10 CFR 20 Appendix B, Table 3 value, steps shall be taken to isolate the source of the activity and to mitigate the problem.

Basis:

(1) Limiting the proximity of sealed sources to five (5) or more feet away from the surface of the reactor core minimizes the effect of such sources on the reactor and the operation of the reactor upon the sources. The neutron flux at a distance of five (5) feet from the core surface is insignificant and thus could not cause activation of the sources and any associated shielding. The presence of the sources in the pool would have no impact upon the D.B.A. which is the rupture of the cladding on one fuel element. However, the presence of sources in the pool could contribute to the radiation hazard associated with a loss of pool water accident. The dose rate 25 feet above an unshielded core in the event of a loss of pool water accident would only be increased by less than 2% with the presence of 100,000 curies of ^{60}Co stored in the irradiation unit in the reactor pool.

(2) Classifying the storage of sealed sources in the reactor pool as an experiment mandates that such storage be reviewed by the Reactor Safeguards Committee.

(3) The 10 CFR 20 Appendix B, Table 3 limit for ^{60}Co is 3×10^{-5} µCi/mℓ. At this limit the entire pool could be dumped into the WSU sewage system without taking advantage of the dilution factor associated with the discharge volume of the WSU sewage system. The detection limit for ^{60}Co in the reactor-pool water depends upon the system used but in the worst case would be at least 1×10^{-7} µCi/mℓ, or 100 pCi/ℓ, or about one-three-hundredth of the 10 CFR 20 limit stated above. Setting a limit of 100 times the detection limit and one-third the discharge limits provides the facility with ample time to take corrective action in the event the limit is exceeded and does not compromise ALARA considerations.

3.15 Generation of Boron Neutron Capture Facility Beam

Definitions:

(1) For the purpose of this technical specification, the term "BNC facility" shall refer to the boron neutron capture facility which includes the beam, bridge moving system, beam monitoring equipment, beam shielding room, access gate and experimental area viewing equipment. The experimental bench, positioning equipment, and other equipment used for the beam targets are not considered part of the BNC facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

(2) The term "BNC experiment" shall refer to a boron neutron capture experiment involving the neutron irradiation of biological cells enriched with boron.

(3) The term "calibration check" refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.

(4) The term "functional check of the beam monitors" shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common reactor neutronic power level.

(5) The term "characterization" refers to the process of obtaining the dose-versus-depth profile in phantoms. The dose-versus depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the target volume to be irradiated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.

(6) The term "calibration of the beam monitors" refers to the process whereby the beam monitors are calibrated against instruments that measure dose including a tissue equivalent chamber and a graphite or magnesium wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.

(7) The term "design modification" as applied to the BNC facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and

(b) to a change that has the potential to increase significantly the amount of activation products in the BNC facility.

(8) The term "radiation fluence" means the total fluence of neutrons and gamma radiation that is emitted in the BNC facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

Applicability: This specification applies solely to the generation of the BNC facility beam for BNC experiments. It does not apply to any other use of the BNC facility and/or its beam. Surveillances listed in this specification are required only if BNC experiments are planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of BNC experiments during the interval in question.

Objective: To acquire testing and operational experience in use of a facility developed specifically for Boron Neutron Capture Technology.

Specifications:

(1) It shall be possible to initiate a scram of the reactor from a control panel located in the BNC facility area. In the event that the BNC facility scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.

(2) Access to the BNC facility shall be controlled by means of the access gate located at its entrance.

(3) The following features and/or interlocks shall be operable:

(a) An interlock shall prevent moving the bridge from the retracted position unless the BNC facility"s access gate is closed.

(b) The reactor shall scram and the bridge shall move to the retracted position automatically upon opening the treatment room"s access gate.

(c) The bridge shall be designed to move to the retracted position automatically upon failure of facility electric power or low voltage on the backup batteries that power the bridge motor.

(d) Bridge movement that controls beam delivery shall be designed for manual movement to the retracted position.

(e) It shall be possible to move the bridge to the retracted position from within the BNC facility.

(f) A BNC facility lockdown near the access gate shall inhibit blade withdrawal when the key is not inserted and turned to the locked position.

(4) Bridge shall be equipped with a position readout that indicates the status of the bridge. A bridge position readout shall be visible at the BNC facility"s local control panel. In the event of a bridge position readout malfunction, it shall be acceptable to use an alternate means of verifying position such as a video camera in the pool room providing a signal to a monitor at the BNC facility"s local control panel. Use of this alternate means of bridge position verification is limited to seven consecutive working days.

(5) The BNC facility shall be equipped with a read out display of the reactor log-power and the linear power on the BNC facility control console just outside of the shielding.

(6) The BNC facility shall be equipped with a monitor that provides a visual indication of the radiation level within the facility, that indicates both within the facility and at the local control panel, and that provides an audible alarm both within the facility and at the local control panel.

(a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.

(b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any BNC experimentation.

(c) This radiation monitor shall be calibrated quarterly.

(d) The audible alarm shall be set at or below 50 mR/hr. This monitor and/or its alarm may be disabled once the BNC room has been searched and secured, such as is done immediately prior to initiation of BNC experimentation. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the BNC facility access gate.

(e) In the event that this monitor is inoperable, personnel entering the BNC facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the WSU Research Reactor"s radiation protection program and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.

(7) An intercom or other means of two-way communication shall be operable both between the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility shielding.

(8) It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access gate manually.

(9) It shall be possible to observe the BNC experiment by means of two independent closed-circuit TV cameras. Both cameras providing visualization shall be operable at the outset of any BNC experiment. Should either fail during the irradiation, the experiment may be continued at the discretion of the experimenter. Adequate lighting

to permit such viewing shall be assured by the provision of emergency lighting and backup power for one TV camera and monitor.

(10) The following interlocks or channels shall be tested at least monthly and prior to a BNC experiment if the interlock or channel has been repaired or deenergized:

Interlock or Channel		Surveillance
a)	The reactor scrams and the bridge retracts upon BNC facility scram	Scram test
b)	Bridge will not move from the retracted position unless access gate is closed	Operational test
c)	Upon opening the BNC room's access gate the reactor scrams and the bridge moves to the retracted position	Operational test
d)	The bridge moves toward the retracted position on loss of electrical power and low voltage on the bridge motor batteries	Operational test
e)	Manual movement of bridge	Operational test
f)	Bridge can be moved manually by someone standing on the reactor bridge	Operational test
g)	Bridge position indicator and status lights	Operational test
h)	Radiation monitor alarm	Operational test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Operational test
j)	Intercoms	Operational test
k)	BNC facility TV cameras, monitors and its power backup	Operational test
l)	BNC facility emergency lighting	Operational test
m)	BNC facility lockdown blade inhibit	Operational test

In addition to the above, the BNC facility scram shall be tested prior to reactor startup if the reactor has been shut down for more than sixteen hours.

(11) Manual operation of the BNC facility's access gate in which the door is opened fully shall be verified semi-annually.

(12) Use of the BNC facility beam shall be subject to the following:

(a) A calibration check of the beam and a functional check of the beam monitors shall be made weekly for any week that the beam will be used for BNC experiments. These checks shall be made prior to any BNC experiment for a given week. In addition, a calibration check shall be performed prior to any BNC experiment in the event that any component of a given beam design has been replaced. Finally, a calibration and a functional check shall be performed prior to any BNC experiment in the event of a design modification.

(b) A characterization of the beam shall be performed every six months for any six-month interval that the beam will be used for BNC experiments. This six-month characterization shall be made prior to any BNC experiment for a given six-month interval. A characterization shall also be performed prior to any BNC experiment in the event of a design modification. As part of the characterization process, the proper response of the beam monitors shall be verified.

(c) A calibration of the beam monitors shall be performed at least once every two years for any two-year interval that the beam will be used for BNC experimentation. The two-year calibration shall be made prior to any BNC experimentation during any given two-year interval.

(d) A scram from full power initiated when the reactor is positioned against the BNC facility filter shall be performed every six months or in the event of a design modification. The BNC room radiation monitor reading shall not exceed 50 mR/hr, 30 seconds after the initiation of the scram and bridge retraction.

(13) Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the WSU Research Reactor. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59.

(14) Personnel who are not licensed to operate the WSU Research Reactor but who are responsible for either the BNC or the beam's design including construction and/or modification may operate the controls for the BNC facility beam provided that:

(a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

(b) Instructions are posted at the BNC facility's local control panel that specify the procedure to be followed:

(i) to ensure that only the appropriate target is in the irradiation facility before turning the primary beam of radiation on to begin an irradiation;

(ii) if the operator is unable to turn the primary beam of radiation off with controls outside the BNC facility, or if any other abnormal condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

(c) In the event that bridge movement affects reactivity, personnel who are not licensed on the WSU Research Reactor but who have been trained under this provision may initiate bridge movement provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the WSU Research Reactor"s training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

<u>Basis</u>: The requirement that it be possible to initiate a scram from a control panel located in the BNC facility area assures the experimenter of the capability to terminate the irradiation immediately should the need arise. The provision that access to the BNC facility be limited to a single gate ensures that there will be no inadvertent entries. The various interlocks for the bridge movement system that controls beam delivery ensure that exposure levels in the BNC facility will be minimal prior to entry by personnel. The bridge position indicator and status lights serve to notify personnel of the beam's status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the BNC facility prior to entry. The purpose of this monitor's audible alarm is to alert personnel to the presence of elevated radiation levels. This monitor and/or its alarm may be disabled once the BNC facility has been searched and secured so that it will not distract attending personnel. The monitor and/or its alarm are interlocked with the access gate so that they are made functional upon opening that gate, and hence prior to any possible entry to the BNC facility. One intercom provides a means for the prompt exchange of information between the experimenter(s) and the reactor operator(s).

The provision for manual operation of the BNC facility's access gate ensures access to the experimental area in the event of a loss of electrical power. The presence of the closed-circuit TV cameras provide the experimenter(s) with the opportunity to monitor the target area visually as well as through the use of various instruments. The emergency lighting and the backup power for a TV camera and monitor will permit visual surveillance of the target area in the event of a power failure.

The surveillance requirements for beam calibration checks and characterizations provide a mechanism for ensuring that the BNC facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the beam monitors ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence.

The specifications on maintenance and repair of the BNC facility ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the BNC facility beam. (Note: Licensed reactor operators may, of course, operate the BNC facility beam.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

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.

Specifications: 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 Safeguards 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 semiannually or at an interval not to exceed 8 months by the substitution of a thermocouple simulator 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 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 of experiments and systems.

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 reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

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

(4) The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary semiannually at intervals not to exceed 7.5 months.

(5) The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity.

Basis: The reactivity worth of the control rods is measured to ensure that the required shut-down 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. The reactor is pulsed at suitable intervals and a comparison is made with previous similar pulses to determine if changes in fuel or core characteristics are taking place.

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 for the intended mode of operation shall be performed before each day's operation or before each opera-tion 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 the calorimetric method annually, but at intervals not to exceed 15 months.

(4) A channel test of each item in Table 3.2, other than measuring channels, shall be performed semiannually, but at intervals not to exceed 7.5 months.

Basis: Measurement 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. Transient control element checks and semiannual maintenance ensure proper operation of this control element.

4.3.3 Radiation Monitoring System

Applicability: This specification applies to the surveillance monitoring for the area monitoring equipment, Argon-41 monitoring system, and continuous air monitoring system.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating properly and capable of performing its intended function, and that the alarm points are set correctly.

Specifications: All radiation monitoring systems shall be verified to be operable at least monthly at an interval not to exceed 45 days. In addition, the following surveillance activities shall be performed on an annual basis at intervals not to exceed 15 months: 1) the area radiation monitoring system shall be calibrated using a certified source; 2) a calibration of the Ar-41 system shall be done using at least two different calibrated gamma-ray sources; 3) a calibration shall be performed on the CAM in terms of counts per unit time per unit of activity using calibrated beta sources.

Basis: Experience has shown that monthly verification of Radiation Monitoring Systems' operability in conjunction with an annual more thorough surveillance is adequate to correct for any variations in the systems caused by a change of operating characteristics over a long timespan.

4.3.4 Ventilation System

Applicability: This specification applies to surveillance requirements for the pool room ventilation system.

Objective: The objective is to ensure the proper operation of the pool room ventilation system in the isolation and dilution modes, which would be used in controlling the release of radioactive material to the uncontrolled environment in the event of an emergency.

Specifications: The operation of the pool room system shall be checked monthly (at intervals not to exceed 6 weeks) by cycling the system from the "normal" to the "isolate" and "dilution" modes of operation. The positions of the associated dampers, indicator display, and fan operation shall be visually checked to ensure correspondence between the device performance and selected mode of operation. The pressure drop across the absolute filter in the pool ventilation system shall be measured at least twice a year. The absolute filter shall be changed whenever the pressure drop across the filter increases by 1 in. of water.

Basis: Experience has shown that the only reliable method of testing the ventilation is to cycle the system into the various modes and visually check each portion of the system for proper operation in that mode.

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 and reviewed for compliance with "Limitations on Experiments," Section 3.10, by the Reactor Safeguards Committee. Minor modifications to a reviewed and approved experiment may be made at the discretion of the senior operator responsible for the operation, provided that the hazards associated with the modifications have been reviewed and a determination has been made and documented 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 for compliance with "Limitations on Irradiations," Section 3.11, by a licensed senior operator qualified in health physics, or a licensed senior operator and a person qualified in health physics.

Basis: It has been demonstrated over a number of years that experiments and irradiations reviewed by the reactor staff and the Reactor Safeguards 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 and measured for length and bend at intervals not to exceed the sum of 3,500.00$ in pulse reactivity. 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 in. over the length of the cladding

(2) in measuring the elongation, its length exceeds its original length by 0.125 in.

(3) a clad defect exists as indicated by release of fission products

Basis: The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known.

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 at least once every 2 weeks, and shall be as follows:

(1) conductivity $\leq 5 \times 10^{-6}$ mhos/cm

(2) pH between 5.0 and 7.5

Basis: Section 3.3 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:

(1) TRIGA 30/20 LEU Fuel. The individual unirradiated TRIGA 30/20 LEU fuel elements shall have the following characteristics

- uranium content: maximum of 30 wt%; enriched to nominal 19.75% uranium-235
- hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 hydrogen atoms to 1.0 zirconium atoms
- natural erbium content (homogeneously distributed): nominal 0.90 wt%
- cladding: 304 stainless steel, nominal 0.5 mm thick

(2) Standard TRIGA Fuel - The individual unirradiated Standard TRIGA fuel elements shall have the following characteristics:

- uranium content: 8.5 wt%; enriched to a nominal 19.75% uranium-235
- hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 hydrogen atoms to 1.0 zirconium atoms
- cladding: 304 stainless steel, nominal 0.5 mm thick

Basis: The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2%.

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 wt%. (However, the quantity of erbium in the full core must not deviate from the design value by more than –3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than two percent.

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than for a hydrogen-to-zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the rupture strength of the clad.

5.2 Reactor Core

Applicability: This specification applies to the configuration of fuel and in-core 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 an arrangement of TRIGA uranium-zirconium-hydride fuel-moderator bundles positioned in the reactor grid plate.

(2) The TRIGA core assembly may be composed of Standard fuel, 30/20 LEU fuel, or a combination thereof (mixed cores) provided that the 30/20 LEU fuel region contains at least 22 30/20 LEU fuel rods located in a contiguous block in the central region of the core.

(3) The reactor fueled with a mixture of fuel types shall not be operated with a core lattice position vacant in the 30/20 LEU fuel region. Water holes in the 30/20 LEU region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel bundle.

(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite, aluminum and water.

Basis: Standard TRIGA cores have been used for years and their characteristics are well-documented. Mixed cores of 30/20 LEU and Standard fuel have been tested by General Atomics Co.. Calculations, as well as measured performance of mixed cores (FLIP and 8.5/20 TRIGA fuel) in the WSU reactor, the Texas A&M reactor, and the University of Wisconsin reactor, have shown that such cores may be safely operated.

In mixed cores, it is necessary to arrange 30/20 LEU elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

Vacant core lattice positions in the Standard fuel region will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacant core positions are not permitted in the 30/20 LEU fuel region as specified by Section 3.5.

The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with 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.

(3) The transient control element shall have scram capability and contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient element shall have an adjustable upper limit to allow a variation of reactivity insertions. This element may incorporate a nonfueled follower.

Basis: The poison requirements for the control elements are satisfied by using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds. Since the regulating element normally is a low worth element, its function could be satisfied by using solid stainless steel. 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 inser-tion of the control element which is the primary safety feature of the reactor. The transient control element is assigned for a reactor pulse. The nuclear behavior of the nonfueled follower which may be incorporated into the transient element is similar to a void.

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): Monitor radioactive particulate activity in the pool room air, alarm and readout at control console.

(3) Function of Argon-41 Stack Monitor (gamma-sensitive detector): Monitor ^{41}Ar content in reactor exhaust air, alarm and readout at 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 Specifications 5.5(1) and 5.5(2) are conservative and ensure safe storage.

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 10^9 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 20 ft from ground level.

(3) Emergency shutdown controls for the ventilation system shall be located outside the pool and control room areas and the system shall be designed to shut down in the event of a substantial release of airborne radioactivity within the facility.

(4) The pool room ventilation system shall have a dilution mode of operation in which air from the pool room is mixed and diluted with outside air before being discharged from the facility.

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 is confined when there is an emergency shutdown of the ventilation system. Emergency controls for startup, isolation, dilution, and normal operation of the ventilation system are located external to the control and pool rooms. 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 ade-quate 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 15 ft of water above the top of the core.

Basis: This specification is based on thermal and hydraulic calculations which show that the TRIGA-30/20 LEU core can operate in a safe manner at power levels up to 2000 kW with natural convection flow of the coolant water. A comparison between operation of the TRIGA-

30/20 LEU and standard TRIGA MARK III has shown them to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for TRIGA-FLIP and 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, at the office, and at the campus police station.

5.8 Physical Security

The Licensee shall maintain in effect and fully implement all provisions of the NRC staff-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 documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled, "Washington State University, Pullman, Washington TRIGA Reactor Security Plan."

6.0 ADMINISTRATIVE CONTROL

6.1 Responsibility

The facility shall be under the direct control of a licensed Senior Reactor Operator (SRO) designated by the Director of the WSU Nuclear Radiation Center. The SRO shall be responsible to the Director 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 Safeguards Committee.

6.2 Organization

(1) The reactor facility shall be an integral part of the Nuclear Radiation Center of Washington State University. 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) When the reactor is not secured, the minimum staff shall consist of:

- (a) Reactor Operator (RO) at the controls (may be the SRO)
- (b) 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

6.3 Facility Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors," for comparable positions.

6.4 Training

The licensed Senior Reactor Operator designated by the Director as being responsible for the facility also shall be responsible for the facility's Requalification Training Program and Operator Training Program.

6.5 Reactor Safeguards Committee (RSC)

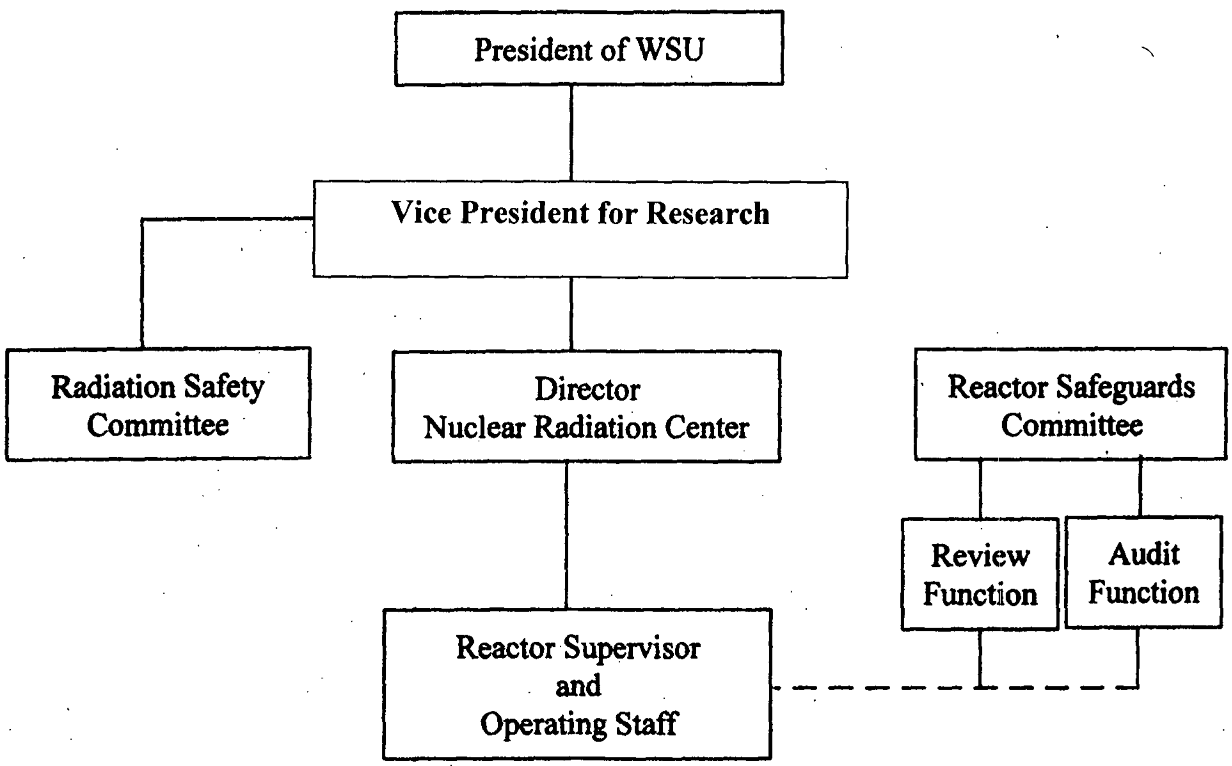
6.5.1 Function

The 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

Figure 6.1 Facility organization

6.5.2 Composition and Qualifications



The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members of the Committee shall include one facility Senior Reactor Operator and WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual. The University's Radiation Safety Director shall be an ex officio member of the Committee.

6.5.3 Operation

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

(1) meeting frequency: the full committee shall meet at least semiannually and a subcommittee thereof shall meet at least semiannually

(2) voting rules

(3) quorums: chairman or his designate and two members

(4) method of submission and content of presentations to the committee

(5) use of subcommittees

(6) review, approval and dissemination of minutes

6.5.4 Reviews

The responsibilities of the RSC or designated subcommittee thereof shall include, but are 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 affect 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 an unreviewed safety question as defined by 10 CFR 50

(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

(9) annual review of the radiation protection program

6.5.5 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 occurrences
(4) radiation exposures at the facility and adjacent environs

6.5.6 Records

The activities of the RSC shall be documented by the secretary of the Committee and distributed as follows:

(1) A written report of all audits performed under Section 6.5.5 shall be prepared and forwarded within 30 days to the Dean of the Graduate School and Facility Director.

(2) A written report of all reviews performed under Section 6.5.4 shall be prepared and forwarded to the Facility Director within 30 days following the completion of the review.

(3) The secretary of the RSC shall maintain a file of the minutes of all meetings.

6.6 <u>Quality Assurance</u>

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, modification, 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 by the RSC specified in Section 6.5."

6.7 <u>Action To Be Taken in the Event a Safety Limit Is Exceeded</u>

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 (NRC).

(2) An immediate report of the occurrence shall be made to the Chairman of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.10 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 Safeguards 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 NRC in accordance with Section 6.10 of these specifications.

6.8 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 calibration

Substantiative changes to the above procedures shall be made only with the approval of the licensed SRO directly in charge of the facility. Temporary changes to the procedures that do not change their original intent may be made by a licensed SRO. All such temporary changes shall be documented and subsequently reviewed by the licensed SRO directly in charge of the facility.

6.9 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) off-site 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

6.10 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, reports shall be made to the Nuclear Regulatory Commission as follows:

(1) A report within 24 hours by telephone to the NRC Operations Center, 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 USNRC Document Control Desk, Washington, D.C. 20555, 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 a 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 USNRC Document Control Desk, Washington, D.C. 20555, 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 as 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 USNRC Document Control Desk, 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 30th of June of each year in writing to the USNRC Document Control Desk, Washington, D.C. 20555, 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, the cumulative total energy output since initial criticality, and number of pulses greater than 1.00$;

(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) monthly radioactivity discharged

- total estimated quantity of radioactivity released (in curies),
- an estimation of the specific quantity for each detectable radionuclide in the monthly release,
- fraction of 10 CFR 20 table 3, appendix B limit for each detectable radionuclide taking into account the dilution factor from the total volume of sewage released by the licensee into the sewage system,
- sum of the fractions for each radionuclide reported above.

(ii) total quantity of radioactive material released by the facility into the sewage system during the year period of the report

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 ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system,
- estimated average atmospheric diluted concentration of ^{41}Ar released during the reporting period in terms of μCi/ml 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/ml 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/ml 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 in the two groups. Each significant exposure in excess of the limits of 10 CFR 20 should be reported, including the time and date 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;

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

6.11 Written Communications

All written communications with the Nuclear Regulatory Commission shall be made in accordance with the requirements of 10 CFR 50.4 "Written Communications."

References

6.5-1 MITR Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J., "Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor," Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.