Appendix A

FACILITY LICENSE NO. R-101

TECHNICAL SPECIFICATIONS

FOR THE

TRIGA MARK III

BERKELEY RESEARCH REACTOR

OF

UNIVERSITY OF CALIFORNIA

BERKELEY, CALIFORNIA

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1209 161

TABLE OF CONTENTS

1209 162

Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, 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. Reference NRC Regulatory Guide 1.16 and ANSI N378-1974.

1.0 DEFINITIONS

REACTOR OPERATING CONDITIONS

1.1 REACTOR SHUTDOWN

The reactor is shut down when the reactor is subcritical by at least one dollar of reactivity.

1.2 REACTOR SECURED

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

a. The reactor is shut down,

b. The console key switch is in the "off" position and the key is removed from the console and stored in a locked cabinet, and

c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms.

1.3 REACTOR OPERATION

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

1.4 COLD CRITICAL

The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures both below 40°C with equilibrium samarium present.

1.5 STEADY STATE MODE

Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

1.6 PULSE MODE

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

POOR ORIGINAL

1.7 SQUARE WAVE MODE

Square wave mode operation shall mean any operation of the reactor with the mode selector switch in the square wave mode position.

1.8 SHUTDOWN MARGIN

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

1.9 ABNORMAL OCCURRENCE

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

1.10 REPORTABLE OCCURRENCE

A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with any safety system setting exceeding that specified in Section 2.1.

b. Operation in violation of a Limiting Condition for Operation;

c. Failure of a required reactor or experiment safety system component which could render the system incapable of performing its intended safety function;

d. Any unanticipated or uncontrolled change in reactivity greater than one dollar;

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits: and

f. Release of fission products from a fuel element.

REACTOR EXPERIMENTS

1.11 EXPERIMENT

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

1209 165

1.12 EXPERIMENTAL FACILITIES

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, hohlraum, central thimble and other incore irradiation facilities, exposure room, rotary specimen rack and pneumatic transfer systems.

REACTOR COMPONENTS

1.13 SAFETY ROD

The safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled follower section.

1.14 SHIM ROD

The shim rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually or by a servo-controller. It may have a fueled follower section.

1.15 TRANSIENT ROD

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It may have a voided follower.

1.16 REGULATING ROD

The regulating rod is a control rod that has scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.17 FUEL ELEMENT

A fuel element is a single TRIGA fuel rod of either 8.5 wt% or 12 wt% standard type, i.e., 20% enriched in uranium-235.

1.18 INSTRUMENTED ELEMENT

An instrumented element is a special fuel element in which at least one thermocouple is embedded in the fuel.

1.19 CORE LATTICE POSITION

The core lattice position is that region in the core normally occupied by a fuel-element, a control rod, an experiment, a reflector element, or an ionchamber.

1.20 STANDARD CORE

A standard core contains an arrangement of standard TRIGA fuel in the reactor grid plate.

POOR ORIGINAL

1.21 MIXED CORE

A mixed core contains an arrangement of standard 8.5 wt% and 12 wt% TRIGA fuel elements.

1.22 OPERATIONAL CORE

An operational core may be a standard core, or mixed core, for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirement of the Technical Specifications.

REACTOR INSTRUMENTATION

1.23 SAFETY LIMIT

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

1.24 LIMITING SAFETY SYSTEM SETTING

Limiting safety systems setting is a setting for automatic protective devices related to those variables having significant safety functions.

1.25 OPERABLE

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

1.26 REACTOR SAFETY SYSTEMS

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

1.27 EXPERIMENT SAFETY SYSTEMS

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

1209 167

POOR ORIGINAL

1.28 MEASURED VALUE

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

1.29 MEASURING CHANNEL

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers and other necessary electrical, mechanical and output devices which are connected for the purpose of measuring the value of a variable.

1.30 SAFETY CHANNEL

A safety channel is a measuring channel in the reactor safety system.

1.31 CHANNEL CHECK

A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

1.32 CHANNEL TEST

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

1.33 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 variable.

1209 168

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 element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specifications

The temperature in a standard 8.5 wt% and 12 wt% TRIGA fuel element shall not exceed 1830°F (1000°C) under any conditions of operation.

Bases

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a build-up 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 dissociation 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. Experiments performed by G.A. have shown that the measured pressure rise was only 24 psi at a calculated temperature of 1100°C.[1]

If we assumed that the cladding temperature is in equilibrium with the fuel meat, this pressure of 24 psi is far below the rupture pressure of 450 psi at 800°C.[2] Of course it is unreasonable to expect that the cladding will be as hot as the fuel meat. Calculation shows [2] that the maximum clad temperature is about 2/3 of the maximum fuel temperature. Thus, even at a calculated fuel temperature of 1100°C, the cladding does not exceed 733°C.

2.2 LIMITING SAFETY SYSTEM SETTINGS IN STEADY STATE MODE OF OPERATION

Applicability

The specification applies to the alarm settings which indicate that the safety limit has been reached during steady state and square wave modes of operation.

Objective

The objective is to provide an alarm so that operator action can be taken to prevent the safety limit fuel element temperature of 1830°F (1000°C) from being reached.

1209 169

POOR ORIGINAL

Specification

The limiting safety system setting shall be 930°F (500°C) as measured in a B or C-ring instrumented fuel element.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause an alarm to be initiated prior to the safety limit being exceeded. A setting of 930°F (500°C) provides a safety margin of 900°F (500°C) for the standard TRIGA fuel elements. A part of the safety margin is used to account for the difference between the true and measured temperatures from the actual location of the thermocouple. Locating the thermocouple in either the B or C-rings will reduce the difference to a small percentage. Calculations have shown that if the thermocouple element were located on the periphery of the core, the true temperature at the hottest location would differ from the measured temperature by less than a factor of two. Therefore, placing the thermocouple in the B or C-rings will ensure that when the alarm setting of 930°F (500°C) is reached, there is a sufficient safety margin to permit operator action prior to reaching the safety limit.

(1) G. B. West, et al, "Kinetic Behavior of TRIGA Reactors", General Atomic Publication, GA-7882, March 31, 1967, page 2.

(2) C. O. Coffer, et, "Characteristics of Large Reactivity Insertion in a High Performance TRIGA U-ZrH Core", General Atomic Publication, GA-6216, April 12, 1965, pp. 23-26.

POOR ORIGINAL

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 STEADY STATE AND SQUARE WAVE MODE OPERATION

Applicability

This specification applies to the energy generated in the reactor during steady state or square wave mode operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded during steady state or square wave mode operation.

Specification

The reactor power level shall not exceed 1.3 megawatts under steady state or square wave mode operation. The normal steady state or square wave mode operating power level of the reactor shall be 1.0 megawatts. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 megawatts during the testing period.

Bases

Thermal and hydraulic calculations indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 megawatts with natural convection cooling.

3.2 REACTIVITY LIMITATIONS

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the shutdown margin provided by control rods shall be greater than 0.25 dollar with:

a. the highest worth experiment in its most reactive state,

b. the highest worth control rod fully withdrawn,

c. the reactor in the cold critical condition without xenon,

d. the maximum excess reactivity above cold, clean critical plus samarium poisoning shall be 4.9% $\Delta k/k$, and

e. the maximum rate of reactivity insertion associated with movement of a standard control rod shall be no greater than 0.15% $\Delta k/k$ per second.

Bases

a. The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position.

3.3 PULSE MODE OPERATION

Applicability

This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective

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

Specification

a. The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical stop on the pulse rod, such that the reactivity insertion will not exceed 3.0 dollars (2.1% $\Delta k/k$).

b. Only one pulsing control rod may be used in the core. This rod shall contain aluminum or stainless steel clad, borated graphite poison. The pulse rod shall be designed to release and fall upon initiation of a scram signal. The maximum worth of the poison section with respect to water shall be 2.9% $\Delta k/k$.

c. The peak neutron flux shall be recorded for every pulse. Peak power levels during pulsing that exceed 2800 megawatts shall be investigated to determine the reason for the pulse magnitude. Conclusions shall be submitted to the Reactor Hazards Committee for evaluation. Pulsing will be discontinued until resumption is approved by the Committee.

Bases

Measurements performed on the Berkeley Research Reactor TRIGA MARK III reactor using 8.5 wt% fuel indicated that a pulse insertion of reactivity of 3.0 dollars resulted in a maximum temperature rise of approximately 450°C. With an ambient water temperature of approximately 25°C, the maximum fuel temperature would be approximately 475°C resulting in a safety margin of 525°C for standard 8.5 wt% fuel. The maximum calculated temperature rise in a 12 wt% fuel, under the conditions described in Section 5.1 is approximately 650°C.[1] The safety margin in this case is approximately 325°C. These margins allow amply for uncertainties due to the accuracy of measurement or location of the instrumented fuel element or due to the extrapolation of data from the BRR reactor.

POOR ORIGINAL

1209 172

3.4 CONTROL AND SAFETY SYSTEM

3.4.1 Scram Time

Applicability

This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective

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

Specification

The scram time measured from the instant a simulated signal reaches the value as described in Table I to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds.

Basis

This specification assures 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 assure the safety of the reactor.

3.4.2 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

(1) W. Froloff, "Analysis of Partial Refueling with 12 wt% Uranium Fuel for the Berkeley Research Reactor", MS Thesis, Department of Nuclear Engineering, University of California, Berkeley, 1977.

POOR ORIGINAL

1209 173

Objective

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

Specification

The reactor shall not be operated unless the safety channels described in Table 1 are operable.

Bases

The power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual and the magnet current key switch scrams allow 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 insures that the reactor power level will reduce to a low level after pulsing.

The interlock to prevent startup of the reactor at power levels less than 4×10^{-3} watts, which corresponds to approximately 2 cps, assures that sufficient neutrons are available to initiate a self-sustaining chain reaction.

The interlock to prevent the initiation of a pulse above 1 kW is intended to assure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted, is intended to prevent pulsing the reactor in the steady state mode. The interlock to prevent withdrawal of the shim, safety or regulating rod in the pulse mode is designed to prevent changing the critical state of the reactor just before pulsing. The earthquake scram will trigger a shutdown upon receipt of a horizontal acceleration.

3.5 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 assure that suffient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in the following table are operable.

TABLE 1

Minimum Reactor Safety Channels

Safety Channel	Number Operable	Function	Effective Mode S.S.	Pulse	Sq.W.
Linear (Power Level)	1	SCRAM @ 110%	X		X
Safety (Power Level)	1	SCRAM @ 1.1 megawatt	X		X
Console Scram Button	1	SCRAM	X	X	X
Log Power	1	SCRAM @ < 3 seconds	X		
Linear and Safety Detector Power Supply	1	SCRAM on loss of supply voltage	X	X	X
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X	
Startup Channel	1	Prevent rod withdrawal at less than two neutron induced counts per second	X		
Log Power	1	Prevent pulsing above 1 kW		X	
Transient Rod Position	1	Prevent application of air unless fully inserted	X		
Rod Drive Control	1	Prevent withdrawal of any rod except transient rod		X	
Fuel Element Temperature	1	Alarm @ 500°C	X		X
Earthquake Detector	1	SCRAM	X	X	X
Magnet Current Key Switch	1	SCRAM	X	X	X
Pool Water Level	1	2 feet above top of grid bridge	X	X	X
Rod Drive Control	1	Prevent simultaneous manual withdrawal of 2 rods	X		X
Pool Bulk Water Temperature	1	Alarm @ > 120°F	X	X	X

POOR ORIGINAL

1209 175

Radiation Monitoring Channels	Function	Number
Area Radiation Monitor*	Monitor radiation levels within the reactor room	1
Area Radiation Monitor*	Monitor radiation levels around reactor bay	1
Continuous Air Particulate Monitor**	Monitor concentration of radioactive particulate activity within the reactor room	1
Exhaust Gas Radiation Monitor*	Monitor radiation levels in the exhaust air stack	1

Bases

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

3.6 ARGON-41 DISCHARGE LIMIT

Applicability

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

Objective

To insure that the health and safety of the public is not endangered by the discharge of Argon-41 from the TRIGA reactor facility.

Specification

The concentration of Argon-41 at the facility stack exhaust shall not exceed 3.12×10^{-6} μCi/ml averaged over one year.

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*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 which shall be kept under visual observation.

**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 beta-gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Bases

As shown in the SAR, the continuous emission at the stack exhaust provides a yearly dose of 0.13 rem in still air, for the 2.4% of the time in which calm weather occurs. Also shown is that at the point of nearest west wind habitation, 100 m from the stack, on the centerline of a plume whose diameter of 40 m provides the largest possible dose, the same continuous emission would provide a yearly dose of 0.45 rem, for the 29% of the time that the westerly winds prevail. No credit is taken in the latter calculation for plume dispersion or for finite wind velocity. Thus, in the case of either still air or of prevailing westerlies, a person in continuous residence at the worst location could not receive more than the yearly dose allowed under 10 CFR 20.

3.7 ENGINEERED SAFETY FEATURE - VENTILATION SYSTEM

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the normal ventilation system and glove box scrubber exhaust system are operable except for periods of time necessary to permit repair of the system. In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically by a signal from an exhaust air radiation monitor.

Bases

In the event of a substantial release of airborne radioactivity, the ventilation systems will be secured automatically. Therefore, operation of the reactor with the ventilation systems shut down for short periods of time to make repairs insures the same degree of control of release of radioactive materials. Moreover, radiation monitors within the room, independent of those in the ventilation systems, will give warning of high levels of radiation that might occur during operation with the ventilation systems secured.

3.8 LIMITATIONS ON EXPERIMENTS

Applicability

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

Objective

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

Specifications

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

a. The reactivity worth of any single experiment shall be less than 3.0 dollars.

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

c. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such 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 limit of Appendix B of 10 CFR Part 20.

d. In calculations pursuant to c. above, the following assumptions shall be used:

(1) 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.

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

(3) For materials whose boiling point is above 130°F and 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.

POOR ORIGINAL

1209 178

e. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies.

f. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Supervisor or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Bases

a. 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.

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

c. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

d. The 1.5 curie limitation on iodine 131 through 135 assures 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 Part 20 for an unrestricted area.

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

3.9 IRRADIATIONS

This specification applies to irradiation performed in the experimental facilities as defined in Section 1.12. Irradiations are a subclass of experiments that fall within the specifications hereinafter stated in this section. The surveillance requirements for irradiations are given in Section 4.2.5.

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.

POOR ORIGINAL

1209 179

Specifications

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

a. The irradiation meets all the specifications of Section 4.2.5 for an experiment,

b. The expected radiation field produced by the device or sample upon removal from the reactor is not more than 10 rem/hr at one foot, otherwise it shall be classed as an experiment.

c. The device or material is encapsulated in a suitable container,

d. The reactivity worth of the device or material is $0.25 or less, otherwise it shall be classed as an experiment, and

e. The device or material does not remain in the reactor for a period of over 15 days, otherwise it shall be classed as an experiment.

Bases

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.

4.0 SURVEILLANCE REQUIREMENTS

4.1 GENERAL

POOR ORIGINAL

Applicability

This specification applies to the surveillance requirement 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 rod 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 Hazards Committee. A system shall not be considered operable until after it is successfully tested.

Bases

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

4.2 LIMITING CONDITIONS FOR OPERATION

4.2.1 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

a. The reactivity worth of each control rod and the shutdown margin shall be determined bi-annually but at intervals not to exceed 28 months.

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

c. The control rods shall be visually inspected for deterioration bi-annually but at intervals not to exceed 28 months.

d. The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary semi-annually at intervals not to exceed 8 months

e. The reactor shall be pulsed semi-annually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.

f. Control rod drop times shall be verified to be less than one second. If the rod drop time is found to be greater than one second, the rod shall not be considered operable.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Past experience with the BRR TRIGA Mark III reactors gives assurance that measurement of the reactivity worth on a bi-annual basis is adequate to insure 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 made with previous similar pulses to determine if changes in fuel or core characteristics are taking place.

4.2.2 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

a. The scram time shall be measured annually but at intervals not to exceed 14 months.

b. A Channel Test of each of the reactor safety system channels with scram capability for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day.

c. A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method annually but at intervals not to exceed 14 months.

d. A functional check shall be made of all reactor control and reactor position interlocks described in Table 1.

Bases

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 assure that the safety system channels are operable on a daily basis or prior to an extended run. The power level channel calibration will assure that the reactor will be operated at the proper power levels. Transient control rod checks and semiannual maintenance insure proper operation of this control rod.

4.2.3 Radiation Monitoring System

Applicability

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

Objective

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

Specification

The area radiation monitoring system and the continuous air monitoring system shall be calibrated annually but at intervals not to exceed 14 months and shall be verified to be operable at weekly intervals or after an extended period of reactor shutdown.

Basis

Experience has shown that weekly verification of area radiation and air monitoring system set points in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.

4.2.4 Ventilation System

Applicability

This specification applies to the building confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment.

Specification

It shall be verified weekly or after an extended period of reactor shutdown that the ventilation system is operable.

Bases

Experience accumulated over several years of operation has demonstrated that the tests of the ventilation system on a weekly basis are sufficient to assure the proper operation of the system and control of the release of radioactive material.

4.2.5 Experiment and Irradiation Limits

Applicability

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

Objective

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

Specifications

a. A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed and evaluated for compliance with the Limitations on Experiments, Section 3.8, by the Reactor Supervisor. The new experiment shall be submitted for comments to the Reactor Health Physicist prior to approval by the Reactor Supervisor. Certain experiments as defined by the Experiment Reviewal Procedure shall be referred to the Reactor Hazards Committee. The Experiment Reviewal Procedure and any changes to that procedure shall be approved by the Reactor Hazards Committee. Minor modifications to a reviewed and approved experiment may be made at the discretion of the senior reactor operator responsible for the operation, provided that the hazards associated with the modifications have been reviewed and a determination made and documented that the modifications do not create a new, or a greater hazard than that in the original approved experiment.

b. 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 the Limitations on Irradiations, Section 3.9, by the Reactor Supervisor.

1209 184

POOR ORIGINAL

Bases

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

4.3 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

Each fuel element and fuel follower shall be checked for transverse bend and longitudinal elongation after the first 100 pulses of any magnitude and then after the first 500 pulses above 1.5% $\Delta k/k$. If any element distorts to or beyond the maximum limits during the first 500 pulses above the 1.5% $\Delta k/k$, inspection of each element shall be made after the next series of 500 pulses above 1.5% $\Delta k/k$. If no element distorts beyond maximum limits after a series of 500 pulses above 1.5% $\Delta k/k$, the number of pulses above 1.5% $\Delta k/k$ between inspections may be increased to 1000. If an element is found to distort beyond maximum limits after a series of 1000 pulses above 1.5% $\Delta k/k$, the next inspection interval shall be reduced to 500 pulses above 1.5% $\Delta k/k$. The limit of transverse bend shall be 0.125 inch over the total length of the element. The limit on longitudinal elongation shall be 0.125 inch. The reactor shall not be operated with elements which have been found to exceed these limits. Any element which is exhibiting a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation of routine operation.

Bases

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 assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow.

5.0 DESIGN FEATURES

5.1 REACTOR FUEL

POOR ORIGINAL

Applicability

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

Objective

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

Specifications

The core fuel elements may consist of an arbitrary mixture of the following kinds of fuel:

1. Fuel, when unirradiated, consisting of zirconium-uranium hydride, in which uranium is contained to 8.5 wt% and is enriched to 20% in uranium-235, and in which the hydrogen-to-metal ratio is 1.7.

2. Fuel, when unirradiated, consisting of zirconium-uranium hydride, in which uranium is contained to 12 wt% and is enriched to 20% in uranium-235, and in which the hydrogen-to-metal ratio is 1.7.

3. Cladding: 304 stainless steel, nominally 0.02 inch thick

Basis

The Safety Analysis Report shows that with a core composed exclusively of Type (1) fuel, peak (B-ring) fuel temperature was predicted to be 350°C above ambient at 1.0 MW, and less than 500°C during production of a 23 MW-s pulse. Both are well below the value of 1000°C which is known to be safe for such fuel.

The maximum power gradient in a core composed of a mixture of Type (1) and Type (2) fuels, arises when Type (2) fuel fills the B ring and Type (1) fuel fills all other core positions. Analysis[1] of this configuration shows that the B-ring peak fuel temperature will be 408°C as compared with a prediction of 320°C for an all Type (1) core, at a power of 1.0 MW. Analysis of a $3.00 pulse predicts a peak B-ring fuel temperature of 650°C as compared with a value of 490°C indicated for the all Type (1) core. Further analysis shows that the Type (2) fueled B-ring operates at a heat flux well below that necessary to produce film boiling. Thus, for this type of mixed core, all fuel temperatures are still well below 1000°C, and in other types of mixed cores, peak temperatures will be lower than in the case analysed.

[1] W. Frolloff, "Analysis of Partial Refueling with 12 wt% Uranium Fuel for the Berkeley Research Reactor", MS Thesis, Department of Nuclear Engineering, University of California, Berkeley, 1977.

1209 186

5.2 REACTOR CORE

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Objective

The objective is to assure 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

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

b. The TRIGA core assembly may be standard, 8.5 wt% or 12 wt%, or a combination thereof (mixed core).

c. Single positions may be occupied by control rods, neutron startup source, ionization chamber, or by in-core experimental facilities. A maximum of three separated experiment positions in the D through G rings, each occupying a maximum of three fuel element positions, may be arranged in the core.

Bases

Standard TRIGA cores have been in use for years and their characteristics are well documented.

5.3 CONTROL RODS

Applicability

This specification applies to the control rods used in the reactor core.

Objective

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

Specification

a. The shim or safety control rods 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. These rods may incorporate fueled followers.

b. The regulating control rod has scram capability and is a stainless rod that contains the materials as specified for shim or safety control rods. This rod may incorporate a fueled follower,

c. The transient control rod 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 rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum or air follower.

Bases

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron and its compounds. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core and increase the worth of the control rod. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse. The nuclear behavior of the air or aluminum follower which may be incorporated into the transient rod is similar to a void. A voided follower may be required in certain core loadings to reduce flux peaking values.

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 assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in the following table will be available for reactor operation.

Radiation Monitoring Channel and Function

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

Continuous Air Radiation Monitor (beta, gamma sensitive detector with air particulate collection capability)

Function - Monitor concentration of radioactive particulate activity in reactor room, alarm and readout at monitor and alarm at control console and in reception room.

Gas Monitor (gamma sensitive detector)

Function - Monitor concentration of radioactive gases in building exhaust, alarm and readout at control console and alarm in reception room.

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 assure that fuel which is being stored will not become critical and will not reach an unsafe temperature.

Specifications

a. All fuel elements shall be stored in a geometrical array where the k-effective is less than 0.8 for all conditions of moderation.

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

Basis

The limits imposed by Specifications 5.5.a and 5.5.b are conservative and assure safe storage.

5.6 REACTOR BUILDING AND VENTILATION SYSTEM

Applicability

This specification applies to the building which houses the reactor.

Objective

The objective is to assure that provisions are made to restrict the amount of release of radioactivity into the environment.

Specifications

a. The reactor shall be housed in a facility design co leakage. The minimum free volume in the facility shall be 300,000 cubic feet.

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

c. Emergency shutdown controls for the ventilation system shall be located in the reception room and the system shall be designed to shut down in the event of a substantial release of fission products.

Bases

The facility is designed such that the ventilation system will normally maintain a negative pressure with respect to the atmosphere so that there will be no uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Controls for emergency filtering and normal operation of the ventilation system are located in the reception room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reception room 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 assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

a. The reactor core shall be cooled by natural convective water flow.

b. During reactor operation the pool water level shall be monitored and shall be at least 14 feet above the top grid plate.

c. A pool level alarm shall indicate loss of coolant if the pool level drops to approximately 2 feet above the top of the core.

d. A pool temperature alarm shall indicate when the maximum bulk temperature exceeds 120°F.

e. Loss of coolant alarm requires corrective action. This alarm is observed in the reactor control area and outside the reactor room in the reception room.

5.8 REACTOR SUPPORT BRIDGE

Applicability

This specification applies to the reactor support bridge.

Objective

The objective is to assure that the reactor support bridge be provided with electrical as well as mechanical stops to prevent bridge over-travel and to assure that the bridge-travel speed is within a safe limit.

Specifications

a. Limit switches shall be provided to stop the drive motor of the support bridge when the bridge has reached either of its two extreme limits of travel.

b. Mechanical stops shall be mounted at both ends of the track to prevent bridge over-travel.

c. The maximum speed of the bridge shall be 6.5 feet per minute.

Basis

Our past experience proves that the limit switches and the mechanical stops provided as well as the speed limit specified in 5.8.c are adequate.

5.9 REACTOR AUXILIARY ELECTRICAL SYSTEM

Applicability

This specification applies to the emergency generator system.

Objective

The objective is to assure there is a source of electrical power in the event of loss of normal electrical power to the reactor laboratory and associated building.

Specification

The emergency generator shall be tested quarterly (not to exceed four months) and verified to assume and carry anticipated emergency reactor laboratory electrical loads and an equivalent maximum anticipated building load.

Basis

The auxiliary electrical system is designed such that there will be sufficient electrical power available in emergency situations where loss of normal electrical power may occur.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

a. The facility shall be under the direct control of the Reactor Supervisor or a licensed senior operator designated by him to be in direct control. The Reactor Supervisor shall be responsible to the Reactor Administrator for safe operation and maintenance of the reactor and its associated equipment. The Reactor Supervisor or his appointee shall review and approve all experiments and experimental procedures prior to their use in the reactor. He shall enforce rules for the protection of personnel against radiation.

b. The safety of operation of the Berkeley Research Reactor shall be related to the University Administration as shown in Figure 1.

6.2 REVIEW AND AUDIT

a. A Reactor Hazards Committee (RHC) of at least three (3) members knowledgeable in fields which relate to Nuclear Safety shall review, evaluate, and approve safety standards associated with the operation and use of the facility. The Reactor Health Physicist shall be an ex-officio member of the Reactor Hazards Committee. The jurisdiction of the RHC shall include all nuclear operations in the facility and general safety standards.

b. The operations of the Reactor Hazards Committee shall be in accordance with a written charter, including provisions for:

(1) Meeting frequency,

(2) Voting rules,

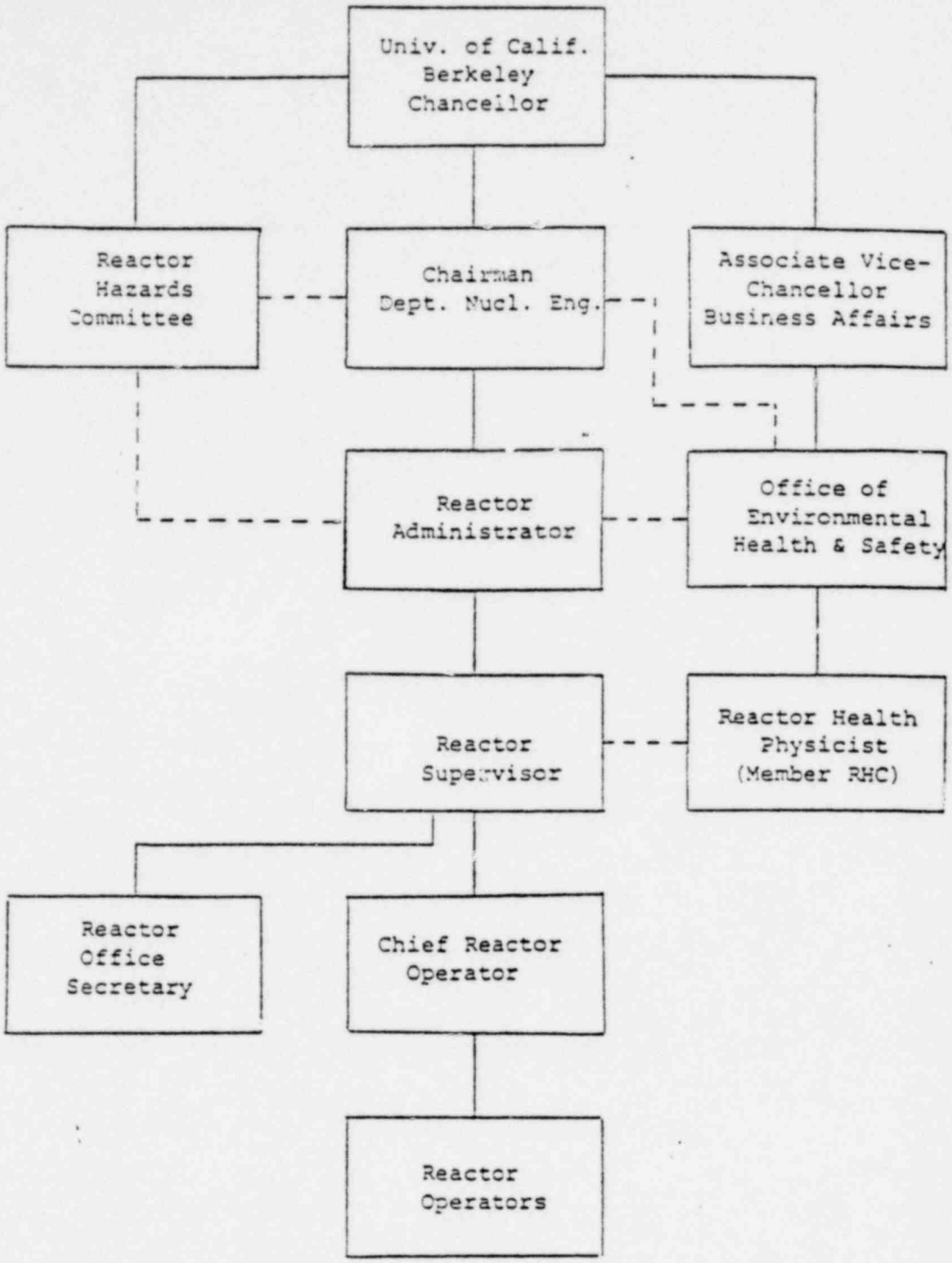
(3) Quorums,

(4) Method of submission and content of presentation to the Committee,

(5) Use of subcommittees, and

POOR ORIGINAL

1209 193



——————— Line of Responsibility

— — — — — — — — Line of Communication

OPERATION ORGANIZATION OF THE BERKELEY RESEARCH REACTOR

POOR ORIGINAL

1209 194

(6) Review, approval, and dissemination of minutes.

c. The RHC or a Subcommittee thereof shall audit reactor operations at least quarterly but at intervals not to exceed four months.

d. The responsibilities of the Committee or designated Subcommittee thereof include, but are not limited to, the following:

(1) Review and approval of experiments utilizing the reactor facilities, as described in the Experiment Reviewal Procedure,

(2) Review and approval of all proposed changes to the facility, procedures, and Technical Specifications, as described in the charter,

(3) Review of the operation and operational records of the facility,

(4) Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50,

(5) Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or a change in the Technical Specifications,

(6) Review of abnormal performance of facility equipment and operating anomalies, and

(7) Review and approval of the implementation of the physical security plan and the emergency plan.

6.3 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a safety limit is exceeded:

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

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

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

1209 195

POOR ORIGINAL

6.4 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE

In the event of a reportable occurrence, the following action shall be taken:

a. The Reactor Supervisor or his designated alternate shall be notified and corrective action taken with respect to the operations involved,

b The Reactor Supervisor or his designated alternate shall notify the Chairman of the Reactor Hazards Committee,

c. A report shall be made to the Reactor Hazards Committee which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and

d. A report shall be made to the NRC in accordance with Section 6.7 of these specifications.

6.5 OPERATING PROCEDURES

Written operating procedures shall be in effect for the following items:

a. Testing and calibration of reactor operating instrumentation and controls, area radiation monitors, and air particulate monitors;

b. Reactor startup, operation, and shutdown;

c. Emergency and abnormal conditions, including provisions for evacuation, reentry, recovery, and medical support;

d. Fuel element loading or unloading;

e. Control rod removal or replacement;

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

g. Actions to be taken to correct specific malfunctions of systems or components, including responses to alarms, and

h. Civil disturbances on or near the facility site.

Substantive changes to the above procedures will require the approval of the Reactor Supervisor and Reactor Administrator or the Reactor Hazards Committee as described in 6.2.d(2). Temporary changes to the procedures that do not change their

original intent may be made by the Reactor Supervisor or his designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Administrator.

6.6 FACILITY OPERATING RECORDS

In addition to the requirements of applicable regulations, and in no way substituting therefor, records and logs shall be prepared of at least the following items and retained for a period of at least five years for items a through f and indefinitely for items g through k.

a. Normal reactor operation,

b. Principal maintenance activities,

c. Reportable occurrences,

d. Equipment and component surveillance activities required by the Technical Specifications,

e. Experiments performed with the reactor,

f. Gaseous and Liquid radioactive effluents released to the environs,

g. Environmental monitoring surveys,

h. Fuel inventories and transfers,

i. Facility radiation and contamination surveys,

j. Radiation exposures for all personnel, and

k. Updated, corrected, and as-built drawings of the facility.

6.7 REPORTING REQUIREMENTS

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the NRC Region V, Office of Inspection and Enforcement as follows:

a. A report within 24 hours by telephone and telegraph.

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

(2) Any violation of the safety limit; and

(3) Any reportable occurrences as defined in Section 1.10 of these specifications

b. A report within 10 days in writing of:

(1) Any accidental release of 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;

(2) Any violation of a safety limit; and

(3) Any reportable occurrence as defined in Section 1.10 of these specifications.

c. A report within 30 days in writing of:

(1) Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;

(2) Any significant change in the transient or accident analysis as described in the Safety Analysis Report;

(3) Any changes in facility organization; and

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

6.7.1 A report within 90 days after completion of startup testing of the reactor 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 or characteristics of the reactor under the new conditions including:

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

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

6.7.2 An annual report covering the operation of the unit during the previous calendar year submitted prior to March 31 of each year providing the following information:

a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

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

c. The number of emergency shutdowns and inadvertent scrams, including reasons therefor;

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

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

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

Liquid Waste

(1) Radioactive waste packaged and shipped (summarized on an annual basis)

(a) Total volume shipped (in gallons)

(b) Total activity shipped (in curies)

(c) Dates of shipment and disposition

(2) Radioactive waste discharged direct to sewer during the reporting period

(a) Total volume discharged (in gallons)

(b) Total radioactivity discharged (in curies)

(c) Date of discharge

(d) Average concentration at point of release (in microcuries/cc) during the reporting period.

Gaseous Waste (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period (in curies)

(a) Total estimated quantity of radioactivity release (in curies) determined by an appropriate sampling and counting method.

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

(c) Estimated average atmospheric diluted concentration of Argon-41 released during the reporting period in terms of microcuries/cc and fraction of the applicable MPC value.

(d) Total estimated quantity of radioactivity in particulate form (in curies) released during the reporting period as determined by an appropriate particulate monitoring system.

(e) Average concentration of radioactive particulates released in microcuries/cc during the reporting period.

(f) An estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of microcuries/cc and fraction of the applicable MPC value for the reporting period if the estimated release is greater than 20% of the applicable MPC.

Solid Waste (summarized on an annual basis)

(1) Total amount of solid waste packaged (in cubic feet)

(2) Total activity in solid waste (in curies)

(3) 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 greatest 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 name of the individual and the circumstances leading up to the exposure.

h. An annual summary of the radiation levels and 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.