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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 18, 1979

Docket No.: 50-224

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Mr. Thomas H. Pigford
Reactor Administrator
Department of Nuclear Engineering
The Regents of the University of
  California
Berkeley, California 94720

Dear Mr. Pigford:

We have completed a thorough review of your license renewal application and find that additional information is required. These items were discussed in detail in your conversation with Mr. Ramos on Wednesday, January 3, 1979, and are reiterated herein.

We informed you by letter on October 17, 1978, that the physical security plan (PSP) for your reactor facility had been reviewed and that although our evaluation showed it was acceptable, requested that it be reconciled and resubmitted in a single document in loose-leaf format. In your discussions with Mr. Ramos, we were to send you a copy of proposed Regulatory Guide 5.XX, "Standard Format and Content for the License RSP . . ." Because of some changes being made to the Regulatory Guide, we are deferring sending it to you at this time. In addition, pursuant to 10 CFR 50.54, we plan to include your PSP in the license conditions by reference. Because of the sensitive nature of the security plan, the actual PSP will not be attached to the license and will be withheld from public disclosure in accordance with 10 CFR 2.790(d). The following is an example of such a license condition:

> "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 2.790, collectively titled, "University of California at Berkeley Security Plan," as follows:
>
> Original, submitted with letter dated May 31, 1973
> Revision 1, submitted with letter dated November 26, 1973
> Revision 2, submitted with letter dated January 14, 1974
> Revision 3, submitted with letter dated March 11, 1974"

This, of course, is only an example and does not reflect your actual PSP.

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The second item discussed relates to the Department of Energy and State Department program to implement the Nonproliferation Act of March 10, 1978, by reducing the enrichment of fuels in nonpower reactors. Concomitant to this, the proposed Regulation § 73.47 is designed to implement the US/IAEA Agreement when approved by the Senate. Both of these actions are keyed to the enrichment of fuel and other SNM. Therefore, your license, which authorizes certain maximum possession limits of SNM (U235, Pu, U233), should be changed to reflect not only the total amount of SNM, but the percent enrichment of each; the amount of SNM exempt and how exempt (i.e., 10 CFR 73.6(b)); and the amount of SNM non exempt. This will form the basis for establishing the level of protection of your PSP.

For your information, in September 1975, a letter was sent to all licensees authorized to possess SNM in excess of 10 CFR 73.1(b) quantities requesting that they review their requirements and provide justication for the "lowest acceptable quantity" necessary to sustain current operations and those projected for the ensuing twelve months. There are still a number of licensees that are authorized to possess quantities in excess of 73.1(b) quantities.

In view of the foregoing, you are requested to review your SNM requirements and provide:

1. The maximum amounts of SNM and types.
2. The enrichment of each item in 1.
3. The amounts of each SNM exempt and how exempt.
4. The amounts of each SNM non exempt, to be included in your license.

To reiterate one item of our discussion relating to exempt and non exempt SNM, this definition in Part 73 refers to whether or not the SNM is exempt or not exempt from meeting certain regulations in Part 73. These definitions do not apply to possession limits.

The third item discussed was your Technical Specifications (TS). Our review reveals that your TS's should be upgraded to current standards. As agreed, we are providing a copy of the Texas A&M/Washington State University TS for you to use as a guide in revising yours.

The final item discussed was the need to update your safety hazards report. This will not hold up the license renewal, but will be an item left open. It was agreed that this updating would be due following completion of the license renewal.

As we agreed, you will submit the Technical Specifications as the first priority item and the PSP second. It is requested that the Technical Specifications and your SNM limits be submitted within 60 days, if possible.

Please do not hesitate to contact Steve Ramos (301-492-7435) regarding these matters.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Sample Technical Specifications
2. Guidance on Administrative Controls
3. Draft ANS 15.18 Standard for Administrative Controls

APPENDIX A

CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. R-83

FOR THE

NUCLEAR SCIENCE CENTER REACTOR

OF

TEXAS A&M UNIVERSITY

DOCKET NO. 50-128

These Technical Specifications have been modified to incorporate guidance from NRC Regulatory Guide 1.16 and salient features of Washington State University Technical Specifications for their TRIGA Docket No. 50-27, License No. R-76, Also attached is Guidance for Section 6.0 Administrative Controls which differs from Section 6.0 herein.

Transmitted w/Amendment No. 4, dated 6/26/73

TABLE OF CONTENTS

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 under the control of a licensed operator or stored in a locked storage area, and

c. 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.

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.

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.

1.7 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.

1.8 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.

1.8a REPORTABLE OCCURRENCE

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

a. Operation with any safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings:

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

1.9 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.

1.10 EXPERIMENTAL FACILITIES

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

REACTOR COMPONENTS

1.11 SHIM-SAFETY ROD

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

1.12 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.

1.13 REGULATING ROD

The regulating rod is a low worth control rod that need not have scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.14 FUEL ELEMENT

A fuel element is a single TRIGA fuel rod of either standard or FLIP type.

1.15 FUEL BUNDLE

A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

1.16 CORE LATTICE POSITION

The core lattice position is that region in the core (approximately 3" x 3") over a grid plug hole. It may be occupied by a fuel bundle, an experiment, or a reflector element.

1.17 INSTRUMENTED ELEMENT

An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.18 STANDARD CORE

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

1.19 MIXED CORE

A mixed core is an arrangement of standard TRIGA fuel elements with at least 23 TRIGA-FLIP fuel elements located in a central region of the core.

1.20 FLIP CORE

A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.

1.21 OPERATIONAL CORE

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

REACTOR INSTRUMENTATION

1.22 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.

1.23 LIMITING SAFETY SYSTEM SETTING

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

1.24 OPERABLE

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

1.25 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.

1.26 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.

1.27 MEASURED VALUE

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

1.28 MEASURING CHANNEL

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

1.29 SAFETY CHANNEL

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

1.30 CHANNEL CHECK

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

1.31 CHANNEL TEST

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

1.32 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.

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

a. The temperature in a TRIGA-FLIP fuel element shall not exceed 2100°F (1150°C) under any conditions of operation.

b. The temperature in a standard 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.

The safety limit for the TRIGA-FLIP fuel element is based on data which indicate that the stress in the cladding due to the

hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature of the fuel does not exceed 2100°F (1150°C) and the fuel cladding is water cooled. (SAR II, pg. 4)*

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 due to hydrogen pressure from the dissociation of zirconium hydride will remain below the utlimate stress provided that the temperature of the fuel does not exceed 1830°F (1000°C) and the fuel cladding is water cooled. (SAR II, pg. 4)

2.2 LIMITING SAFETY SYSTEM SETTINGS

Applicability

This specification applies to the scram settings which prevent the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specification

The limiting safety system settings shall be 400°C (750°F) as measured in an instrumented fuel element. For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 400°C provides a safety margin of 750°C for FLIP type fuel elements and a margin of 600°C for standard TRIGA fuel elements. A part of the safety margin is used to account for the difference between

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* References to the Safety Analysis Report and its amendments will be abbreviated as:

SAR - Safety Analysis Report, August 1967
SAR I - Amendment I to SAR, April 1968
SAR II - Amendment II to SAR, December 1972.

the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the thermocouple element is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, when the temperature in the thermocouple element reaches the trip setting of 400°C, the true temperature at the hottest location would be no greater than 800°C providing a margin to the safety limit of at least 200°C for standard fuel elements and 350°C for FLIP type elements. These margins are ample to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation. For a mixed core (i.e., one containing both standard and FLIP type elements), the requirement that the instrumented element be located in the FLIP region of the core provides an even greater margin of safety since the peak to average power ratio within that region will be smaller than over an entire core composed of elements of the same type.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting of the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

3.0 LIMITING CONDITIONS FOR 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 assure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specifications

The reactor power level shall not exceed 1.3 megawatts under any condition of operation. The normal steady state 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.

Specifications

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 non-secured experiment in its most reactive state,

b. the highest control rod and the regulating rod (if not scrammable) fully withdrawn, and

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

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. 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 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

The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block on the pulse rod, such that the reactivity insertion will not exceed 2 dollars.

Bases

Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of reactivity of 2 dollars resulted in a maximum temperature rise of approximately 400°C. With an ambient water temperature of approximately 100°C, the maximum fuel temperature would be approximately 500°C resulting in a safety margin of 500°C for standard fuel and 650°C for FLIP type fuel. 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 PRNC reactor.

3.4 CORE CONFIGURATION LIMITATION

Applicability

This specification applies to maxed cores of FLIP and standard types of fuel.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in a mixed core.

Specifications

a. The FLIP fueled region in a mixed core shall contain at least 22 FLIP fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the FLIP region shall be limited to single rod holes.

b. The PTR as defined by 2.36 and as calculated by the method used in Amendment I to the S.A.R. shall not exceed 1.5 for an operational core.

Bases

a. The limitation of the allowable core configurations as set forth in Section 5.0 of Amendment I to the W.S.U. TRIGA reactor S.A.R. limits power peaking effects. The limitation on power peaking effects insures that the fuel temperature limit will not be exceeded in a mixed core.

b. A 500°C safety system setting and a 1.5 PTR limit the maximum possible steady state fuel temperature in the FLIP region to below 800°C.

3.5 CONTROL AND SAFETY SYSTEM

3.5.1 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 of the LSSS 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.

3.5.2 Reactor Control System

Applicability

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

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

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

Measuring Channel	Min. No. Operable	Effective Mode S.S.	Pulse
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	
Log Power Level	1	X	
Integrated Pulse Power	1		X

Bases

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors assure 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.

3.5.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

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

Specification

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

Bases

The fuel temperature and 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 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 insures that the reactor power level will reduce to a low level after pulsing.

TABLE I

Minimum Reactor Safety Channels

Safety Channel	Number Operable	Function	Effective Mode S.S.	Effective Mode Pulse
Fuel Element Temperature	1	SCRAM @ LSSS	X	X
Safety #1 (Power Level)	2	SCRAM @ 125%	X	
Console Scram Button	1	SCRAM	X	X
Safeties #1 & #2 Detector Power Supply	1	SCRAM on loss of supply voltage	X	
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X
Log Power	1	Prevent withdrawal of shim-safeties at <4 x 10^{-3} watts	X	
Log Power	1	Prevent pulsing above 1 kW		X
Transient Rod Position	1	Prevent application of air unless fully inserted	X	
Shim-safeties & Regulating Rod Position	1	Prevent withdrawal		X
Pool Level	1	Alarm at 90% normal operating level	X	X

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 for proper startup.

The interlock to prevent the initiation of a pulse above 1 kW is 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 to prevent pulsing the reactor in the steady state mode. The interlock to prevent withdrawal of the shim-safeties or regulating rod 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 of any significant decrease in the pool level.

3.6 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 sufficient 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.

Radiation Monitoring Channels*	Function	Number
Area Radiation Monitor	Monitor radiation levels within the reactor room	1
Continuous Air Radiation Monitor	"	1
Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Exhaust Particulate Radiation Monitor	"	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 which shall be kept under visual observation.

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.

3.7 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 in the effluent gas from the facility as diluted by atmospheric air in the lee of the facility due to the turbulent wake effect shall not exceed 4.8×10^{-8} uci/ml averaged over one year.

Bases

The maximum allowable concentration of Argon-41 in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 4.8×10^{-8} Ci/ml. Section 6.5 of Amendment 1 to the S.A.R. for the TRIGA reactor facility 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 the Argon-41 dilution.

3.8 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 facility ventilation system is 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

During normal operation of the ventilation system, the concentration of Argon 41 in unrestricted areas is below MPC (SAR, pg. 101). In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system 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 building independent of those in the ventilation system will give warning of high levels of radiation that might occur during operation with the ventilation system secured.

3.9 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. Non-secured experiments shall have reactivity worths less than 1 dollar.

b. The reactivity worth of any single experiment shall be less than 2 dollars.

c. 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.

d. 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 atomsphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

e. In calculations pursuant to d. 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.

f. 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.

g. 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 Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Bases

a. This specification is intended to provide assurance that the worth of a single unfastened experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted (SAR II, pg. 24).

b. 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 experiments 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 such that the reactor protective systems would act to prevent high power levels from being attained (SAR II, pg. 21).

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

d. 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.

e. 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.

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

3.10 IRRADIATIONS

Applicability

This specification applies to irradiations performed in the irradiation facilities contained in the reactor pool as defined in Section 2.10. 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 5.3.5.b.

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 the following conditions exist:

a. The irradiation meets all the specifications of Section 4.9.1 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

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 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 Safety Board. A system shall not be considered operable until after it is successfully tested.

Bases

This specification relates 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 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 assure that the fuel element temperatures are properly monitored.

Specifications

a. Whenever a reactor scram caused by high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature safety limit was exceeded.

b. A calibration of the temperature measuring channels shall be performed semi-annually but at intervals not to exceed 8 months.

c. A Channel Check of the fuel element temperature measuring channel shall be made daily whenever the reactor is operated by recording a measured value of a meaningful temperature indication.

Bases

Operational experience with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperatures have been sufficiently reliable to assure 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

a. The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 14 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 at intervals not to exceed 2 years.

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

e. The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value or the reactor shall not be pulsed until such comparative pulse measurements are performed.

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 TRIGA reactors gives assurance that measurement of the reactivity worth on an 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.

4.3.2 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

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 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, except for the pool level channel which shall be tested weekly.

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.

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.

4.3.3 Radiation Monitoring System

Applicability

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

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.

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.

4.3.4 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 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.

4.3.5 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.

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 reviewed for compliance with the Limitations on Experiments, Section 3.6, by the Reactor Safeguards Board. 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 significantly different, a new, or a greater than 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 3.6, by a licensed senior operator qualified in health physics, or a licensed senior operator and a person qualified in health physics.

Bases

It has been demonstrated over a number of years of experience that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safeguards Board 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 dollars 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:

a. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding,

b. In measuring the elongation, its length exceeds its original length by 0.125 inch, or

c. A clad defect exists as indicated by release of fission products.

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

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

a. TRIGA-FLIP Fuel

The individual unirradiated FLIP fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.

(3) Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.

(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.

(5) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b. Standard TRIGA fuel

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

(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.

(3) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Bases

a. A maximum uranium content of 9 Wt-% in a TRIGA-FLIP element is about 6% greater than the design value of 8.5 Wt-%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by at most ten 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 the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings. The accidental rotation of fuel bundles containing standard and FLIP elements can be detected by visual inspection. Should this occur, however, studies of a single FLIP element accidently rotated into a standard fuel region indicate an insubstantial increase in power generation in the FLIP element.

b. A maximum uranium content of 9 Wt-% in a standard TRIGA element is about 6% greater than the design value of 8.5 Wt-%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.8 will produce a maximum pressure within the clad during an accident well below 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 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 bundles positioned in the reactor grid plate.

b. The TRIGA core assembly may be standard, FLIP, or a combination thereof (mixed core) provided that any FLIP fuel be comprised of at least twenty-three (23) fuel elements, located in a contiguous, central region.

c. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

d. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Bases

a. Standard TRIGA cores have been in use for years and their characteristics are well documented. The Gulf Mark III all-FLIP core is operational and characteristics are available. Gulf has also performed a series of experiments using standard and FLIP fuel in mixed cores. In addition, studies performed at Texas A&M for a variety of mixed core arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR II).

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

c. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

d. 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 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-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 which have the same characteristics as the fuel region in which they are used.

b. The regulating control rod need not have scram capability and shall be a stainless rod or contain the materials as specified for shim-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. Since the regulating rod normally is a low worth rod, its function could be satisfied by using a solid stainless steel rod. 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. The use of fueled followers in the FLIP region has the additional advantage of reducing flux peaking in the water filled regions vacated by the withdrawal of the control rods. 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 continously 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 collection capability)
Function - Monitor concentration of radioactive particulate activity in building, alarm and readout at control console and readout in reception room.

Gas and Particulate Stack Radiation Monitor (gamma sensitive detector with air collection capability)

Function - Monitor concentration of radioactive particulate activity and radioactive gases in building exhaust, alarm and readout at control console and readout 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 designed to restrict leakage. The minimum free volume in the facility shall be 180,000 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 building and release them from a stack at a minimum of 85 feet from ground level.

c. Emergency shut down 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 atomsphere 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 startup, 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 (SAR, pg.56).

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. The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

c. Diffuser and skimmer pumps shall be located no more than 15 feet below the top of the reactor pool.

d. Pool water inlet and outlet pipe to the heat exchanger shall have emergency covers within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

e. A pool level alarm shall indicate loss of coolant if the pool level drops approximately 2 feet below normal level.

Bases

a. This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate in a safe manner at power levels up to 2,700 kW with natural convection flow of the coolant water. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores.

b. In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer system, the pool water level will drop no more than 15 feet from the top of the pool.

c. In the event of pipe failure and siphoning of pool water through the skimmer and diffuser water systems, the pool water level will drop no more than 15 feet from the top of the pool.

d. Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within

the reactor pool. Covers for these lines will be stored in the reactor pool. Time required to uncover the reactor core due to failure of a single pool coolant pipe system is 17 minutes.

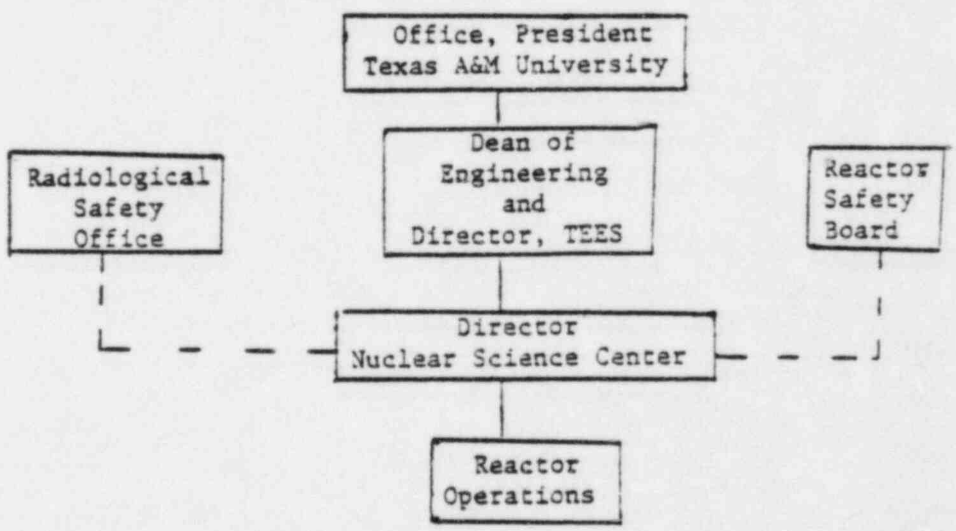
e. Loss of coolant alarm after 2 feet of loss required corrective action. This alarm is observed in the reactor control room and outside the reactor building.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

a. The facility shall be under the direct control of the Director (NSC) or a licensed senior operator designated by him to be in direct control. The Director shall be responsible to the Dean of the College of Engineering and Director of the Texas Engineering Experiment Station for safe operation and maintenance of the reactor and its associated equipment. The Director (NSC) 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 Nuclear Science Center Reactor shall be related to the University Administration as shown in the following chart.



6.2 REVIEW AND AUDIT

a. A Reactor Safety Board (RSB) 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 University Radiological Safety Officer shall be an ex-officio member of the Reactor Safety Board. The jurisdiction of the RSB shall include all nuclear operations in the facility and general safety standards.

b. The operations of the Reactor Safety Board 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

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

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

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

   (1) Review and approval of experiments utilizing the reactor facilities,

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

   (3) 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, and

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

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 AEC,

b. An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Board, and reports shall be made to the AEC 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 Safety Board for review and then submitted to the AEC when authorization is sought to resume operation of the reactor.

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 Director (NSC) or his designated alternate shall be notified and corrective action taken with respect to the operations involved,

b. The Director (NSC) or his designated alternate shall notify the Chairman of the Reactor Safety Board,

c. A report shall be made to the Reactor Safety Board 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 adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgement and action should the situation require such. Operating procedures shall be in effect for the following items:

a. Testing and calibration of reactor operating instrumentation and controls, control rod drives, 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 and experiment 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 and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes, and

h. Civil disturbances on or near the facility site.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Board. Temporary changes to the procedures that do not change their original intent may be made by the Director (NSC) or his designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Board.

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. Offsite 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 IV, 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 1.11 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 reoccurrence of the event;

(2) Any violation of a safety limit; and

(3) Any reportable occurrence as defined in Section 1.11 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 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 such 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.

<u>Liquid Waste</u> (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period.

(a) Total radioactivity released (in curies).

(b) The MPC used and the isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

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

(2) Total volume (in gallons) of effluent water (including dilutent) released during each period of release.

Gaseous Waste (summarized on a monthly basis)

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

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

(e) Average concentration of radioactive particulates with half lives greater than eight days 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.

   (2) Total volume (in gallons) of effluent water (including dilutent) during periods of release.

   Gaseous Waste (summarized on a monthly basis)

   Radioactivity discharged during the reporting period (in curies) for:

   (1) Argon-41

   (2) Particulates with half lives greater than eight days.

   Solid Waste (summarized on an annual basis)

   (1) The total amount of solid waste packaged (in cubic feet).

   (2) The total activity involved (in curies).

   (3) The dates of shipment and disposition (if shipped off site).

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

k. A description of any environmental surveys performed outside the facility.

GUIDANCE FOR

NON-POWER REACTORS

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The (Facility Director[1]) shall be responsible for overall facility operation. He shall delegate in writing the succession to this responsibility during his absence (or the succession shall be clearly presented in Figure 6.2-1).

6.2 ORGANIZATION

Facility Staff

6.2.1 The organization for facility management and operation shall be as shown in Figure 6.2 - 1.

a. Each on duty shift shall be composed of at least:

> List each member of the operating shift by position title or function, and identify those required to have an NRC license For example -
>
> Minimum staff when the reactor is not secured[2] shall include:
>
> 1. Senior Operator (SRO) on call but not necessarly on site;
> 2. Radiation Control Technician on call;
> 3. Reactor Operator (RO) at the controls;
> 4. Another personable to carry out emergency procedures in control room.

b. At least one licensed operator shall be at the controls when the reactor is not secured.[2]

c. An individual that is designated by the licensee as qualified to implement routine radiation protection procedures shall be present at the facility whenever the reactor is not secured or whenever any experiment or experimental facility is being serviced.

d. All core alternations that could affect reactivity of the reactor shall be supervised by a licensed Senior Operator.

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1. Brackets or boxes are used either to set off guidance from the technical specification or to identify areas of licensee option. Identifying position titles contained within the bracket, here and elsewhere in the report, are representative only. Each position title is used consistently throughout this guidance document.

2. A reactor is secured when (1) all poison control rods are fully inserted, (2) the console key switch is in the off position and the key is removed from the lock, and (3) no work is in progress involving fuel or incore experiments, or maintenance of the core structure, control rods or control rod drives.

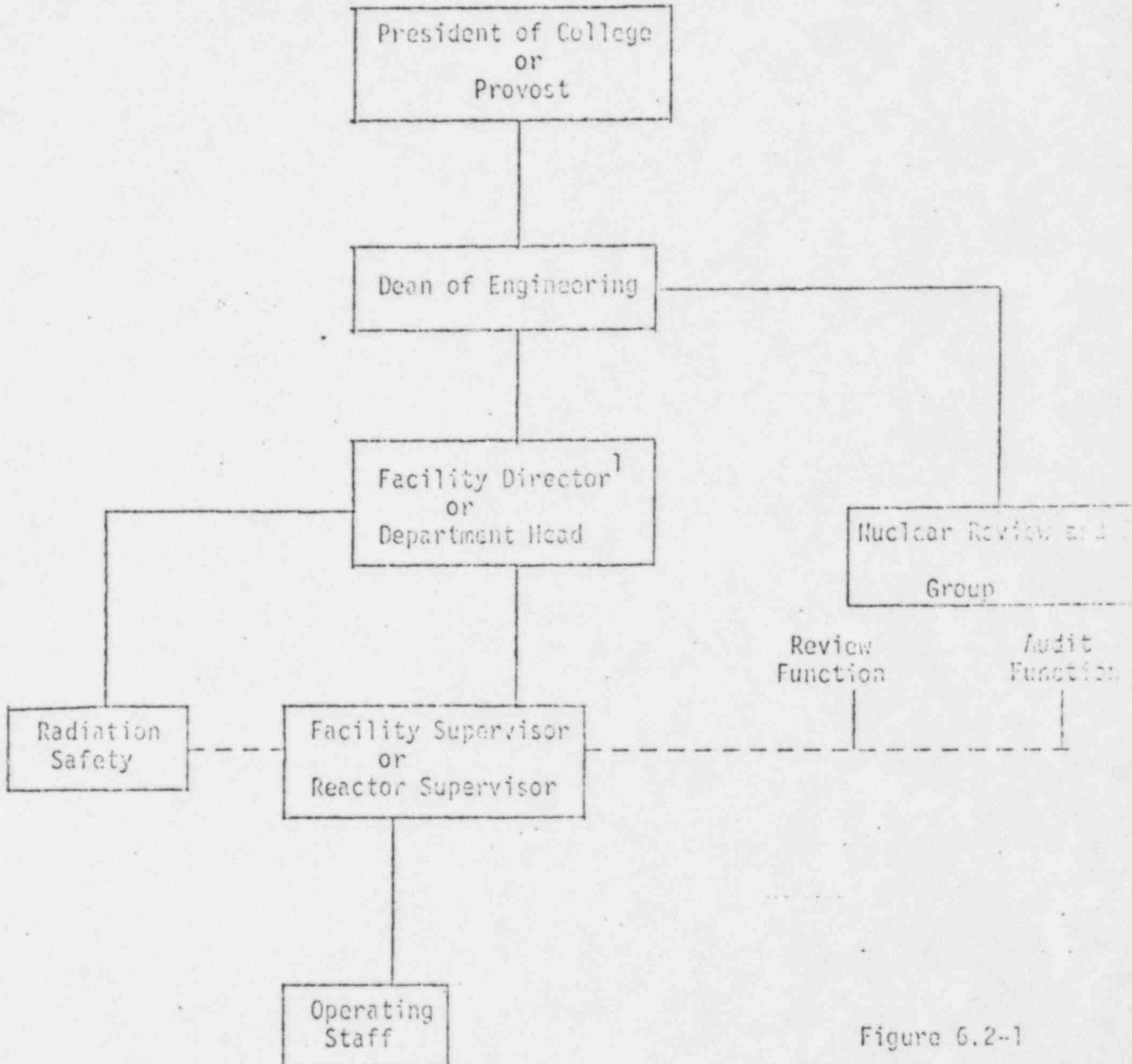

Figure 6.2-1
FACILITY ORGANIZATION

[1] Responsible for facility operation

6.3 FACILITY STAFF QUALIFICATIONS

Minimum qualifications for members of the facility staff may be specified by use of an overall qualification statement referencing an applicable standard (ANS 15 committee is developing a standard titled, 15.4, "Standard for Selection and Training of Personnel for Research Reactors") or alternately, by specifying individual position qualifications. Referencing an applicable standard would be preferably when a standard exists. Such a specification would be:

"Each member of the facility staff shall meet or exceed the minimum qualifications of (applicable standard) for comparable positions".

However, the following method should be used whenever a clear correlation cannot be drawn between a standard and the reactor organizational structure.

6.3.1(a) Facility Director/Reactor Supervisor

At the time of initial core loading or appointment to the active position, the reactor Administrator/Reactor Supervisor shall have a minimum of five years of nuclear experience. He shall have a baccalaureate or higher degree in an engineering or other scientific field. The degree will fulfill four years of experience on a one-for-one time basis. Equivalent education or experience may be substituted for a degree.

(b) Senior/Supervisory Reactor Operator

At the time of initial core loading or appointment to the active position, a supervisor shall have a minimum of a high school diploma or equivalent and should have four years of nuclear experience. A maximum of two years of experience may be fulfilled by related academic or technical training on a one-for-one time basis.

(c) Reactor Operator

At the time of initial core loading or appointment to the active position, operators shall have a high school diploma or equivalent.

6.4 TRAINING

6.4.1 The (position title) shall be responsible for the facility retraining and replacement training program.

6.5 REVIEW AND AUDIT

6.5.1 (NUCLEAR REVIEW AND AUDIT GROUP, NRAG)

The method by which independent review and audit of facility operations is accomplished may take one of several forms. The licensee may either assign this function to an organizational unit separate and independent from the group having responsibility for facility operation or may utilize a standing committee composed of individuals from within and outside the group having administrative responsibility.

Irrespective of the method used, the licensee shall specify the details of each functional element provided for the independent review and audit process for his particular facility as illustrated in the following example specifications.

FUNCTION

6.5.1.1 The (NRAG) shall function to provide independent review and audit of facility activities. Areas designated below shall be considered:

a. nuclear operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. tests and experiments

i. (other appropriate fields associated with the unique characteristics of the nuclear reactor)

COMPOSITION AND QUALIFICATIONS

6.5.1.2 The (NRAG) shall be composed of:

List all members by position title and indicate the Chairman. Do not use given names.

Chairman:	(Position Title)
Member:	(Position Title)
⋮	⋮
Member:	(Position Title)

The minimum qualifications for persons on the (NRAG) shall be 5 years of professional work experience in the discipline or specific field he represents. A baccalaureate degree may fulfill 4 years of experience.

If a seperate organizational unit is used, the supervisor of the unit shall have a minimum of 10 years of experience. A baccalaureate degree may fulfill 4 years of experience.

ALTERNATES

6.5.1.3 Alternate members may be appointed by the (NRAG Chairman) to serve on a temporary basis; each appointment shall be in writing. No more than two alternates shall participate on a voting basis in (NRAG) activities at any one time.

MEETING FREQUENCY

6.5.1.4 The (NRAG) or a subcommittee thereof shall meet at least once per calendar quarter. The (NRAG) shall meet at least semi-annually.

QUORUM

6.5.1.5 A quorum of (NRAG) for review shall consist of the Chairman or his designated alternate and (two) other members, or alternate members; however, a majority of those present shall be regular members. The quorum shall have representation experienced in reactor operations and radiation protection; however the operating staff shall not be a voting majority.

REVIEW

6.5.1.6 The (NRAG) shall review:

a. Safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments, conducted without NRC approval under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems that change the original intent or use, and are non-conservative, or those that involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which are significantly different from previous approved tests or experiments, or those that involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety.

g. Events which have been reported within 24 hours to the NRC in writing.

h. Audit reports.

AUDITS

6.5.1.7 Audits of facility activities shall be performed under the cognizance of the (NRAG) but in no case by the personnel responsible for the item audited. Individual audits may be performed by one individual who need not be an identified (NRAG) member. These audits shall examine the operating records and encompass:

a. The conformance of facility operation to the Technical Specifications and applicable license conditions, at least once per 12 months.

b. The performance training and qualifications of the entire facility staff, at least once per 12 months.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety, at least once per 6 to 12 months.

d. The Facility Emergency Plan and implementing procedures, at least once per 24 months.

e. The Facility Security Plan and implementing procedures, at least once per 24 months.

f. Any other area of facility operation considered appropriate by the (NRAG) or the (Facility Director).

AUTHORITY

6.5.1.8 The (NRAG) shall report to (Management at level above Facility Director) and advise the (Facility Director) on those areas of responsibility specified in Sections 6.5.1.6 and 6.5.1.7.

RECORDS

6.5.1.9 Records of (NRAG) activities shall be prepared and distributed as indicated below:

a. Minutes of each (NRAG) meeting shall be prepared, and forwarded to the (Facility Director) within 30 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.1.6 e, f, and g above, shall be prepared and forwarded to the (Facility Director) within 30 days following completion of the review.

c. Audit reports encompassed by Section 6.5.1.7 above, shall be forwarded to the (NRAG Chairman) and to the management responsible for the areas audited within 30 days after completion of the audit.

6.6 SAFETY LIMIT VIOLATION

6.6.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The reactor will be shut down immediately and reactor operation will not be resumed without authorization by the Commission.

b. The Safety Limit violation shall be reported to the Director of the appropriate NRC Regional Office of Inspection and Enforcement (or his designate), the (Facility Director) and to the (NRAG) not later than the next work day.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the (NRAG). This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the (NRAG) and the (Facility Director) within 14 days of the violation.

6.7 PROCEDURES

There shall be written operating procedures that cover the following activities. They shall be approved by the (Facility Director).

a. Conduct of irradiations and experiments that could affect the operation or safety of the reactor.

b. Startup, Operation, and Shutdown of the Reactor.

c. Fuel movement and changes to the core and experiments that can effect the reactivity.

d. Preventive or correction maintenance which could have an effect on the safety of the reactor.

e. Surveillance, testing and calibration of instruments, components and systems involving nuclear safety.

f. Review and approval of changes to procedures.

g. Personnel radiation protection consistent with 10 CFR Part 20.

h. Implementation of the Security Plan and Emergency Plan.

i. Administrative control of operation and maintenance.

Though substantive changes to the above procedures shall be made only approval by the (Facility Director) temporary changes to the procedures that do not change their original intent may be made by the (Reactor Supervisor). All such temporary changes shall be documented, and subsequently approved by the (Facility Director) within 14 days.

6.8 EXPERIMENTS

a. Prior to initiating any new reactor experiment, e.g., class of experiments that could affect reactivity of the reactor or result in release of radioactive materials, an experiment plan shall be prepared, reviewed by (NRAG or other identified committee), and approved by the (Reactor Supervisor).

b. Each experiment plan shall (1) identify the type of experiment (previously approved or recently reviewed per 6.8a), (2) identify the experimenters and (3) have been approved by the licensed senior operator in charge of reactor operation.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the appropriate NRC Regional Office unless otherwise noted.

6.9.1. Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the tests program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of power operation, (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e. initial criticality, completion of startup test program, and resumption or commencement of power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Operating Report. Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to (March 31) of each year.

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience having safety significance that was gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A brief narrative summary of:

(a) Changes in facility design, performance characteristics, and operating procedures related to reactor safety, that occurred during the reporting period.

(b) Results of major surveillance tests and inspections.

(2) A tabulation showing the energy generated by the reactor (in general, a monthly tabulation in megawatt-hours and/or hours the reactor is operating will be satisfactory).

(3) List of the unscheduled shutdowns, including the reasons therefore and corrective action taken, if any.

(4) Discussion of the major safety related corrective maintenance performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for the corrective maintenance required.

(5) A brief description of:

(a) Each change to the facility to the extent that it changes a description of the facility in the Safety Analysis Report.

(b) Changes to the procedures as described in the Safety Analysis Report.

(c) Any new or untried experiments or tests performed during the reporting period that are not described in the Safety Analysis Report.

(6) A summary of the safety evaluation made for each change, test, or experiment not submitted for Commission approval pursuant to 10 CFR 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no technical specification change was required.

(7) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the license as determined at or prior to the point of such release or discharge.

The specifications identified for this section should be used to the extent practical considering the nature of the effluents, the available equipment, and the specific facility.

(a) Liquid Waste (summarized on a 3 month basis)

(1) Total estimated quantity of radioactivity released (in curies) and Total volume (in liters) of effluent water (including diluent) released.

(2) An estimation of the specific activity for each detectable radionuclide present if the specific activity of the released material after dilution is greater than 1×10^{-7} microcuries/cc.

(3) Summary of the total release in curies of each nuclide determined in (2) above for the reporting period based on representative isotopic analysis.

(4) Estimated average concentration of the released radioactive material at the point of release for the reporting period in terms of microcuries/cc and fraction of the applicable MPC.

(b) Airborne Waste (summarized on a 3 month basis)

(1) Total estimated quantity of radioactivity released (in curies) determined by an approved sampling and counting method.

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

(3) 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.

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

(5) 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.

(c) Solid Waste (summarized on an annual basis)

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

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

(3) The dates of shipments and disposition (if shipped off site).

(8) A description of the results of any environmental radiological surveys performed outside the facility.

6.9.2 Reportable Occurrences

Reportable occurrences, including causes, probable consequences, corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, an amended licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate NRC Regional Office, or his designate no later than the first work day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

(2) Operation of the reactor or affected systems when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical specifications without taking permitted remedial action.

(3) Abnormal degradation discovered in a fission product barrier, i.e., fuel cladding, reactor coolant boundary, or containment.

(4) Reactivity balance anomalies involving:

(a) disagreement between expected and actual critical positions of approximately 0.3% Δ k/k;

(b) exceeding excess reactivity limit;

(c) shutdown margin less conservative than specified in technical specifications;

(d) unexpected short-term reactivity changes that cause a period of 10 seconds or less;

(e) if sub-critical, an unplanned reactivity insertion of more than approximately 0.5% Δ k/k or any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents, or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

(7) Unscheduled Conditions arising from natural or man-made events that, as a direct result of the event require reactor shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in

the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

SPECIAL REPORTS

Special reports may be required covering inspections, tests and maintenance activities. These special reports are determined on an individual basis for each facility and their preparation and submittal are designated in the Technical Specifications.

6.9.3 Special reports shall be submitted to the Director of the appropriate NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 Records to be Retained for a Period of at least five years:

(a) Operating logs or data which shall identify:

(1) Completion of pre-startup checkout, startup, power changes, and shutdown of the reactor.

(2) Installation or removal of fuel elements, control rods or experiments that could affect core reactivity.

(3) Installation or removal of jumpers, special tags or notices, or other temporary changes to reactor safety circuitry.

(4) Rod worth measurements and other reactivity measurements.

(b) Principal maintenance operations.

(c) Reportable occurrences.

(d) Surveillance activities required by technical specifications.

(e) Facility radiation and contamination surveys.

(f) Experiments performed with the reactor.

This requirement may be satisfied by the normal operations log book plus,

(1) records of radioactive material transferred from the facility as required by license.

(2) Records required by the (NRAG) for the performance of new or special experiments.

(g) Changes to operating procedures.

6.10.2 Records to be Retained for the life of the Facility.

(a) Gaseous and liquid radioactive effluents released to the environs.

(b) Appropriate off-site environmental monitoring surveys.

(c) Fuel inventories and fuel transfers.

(d) Radiation exposures for all personnel.

(e) Updated as-built drawings of the facility.

(f) Records of transient or operational cycles for those components designed for a limited number of transients or cycles.

(g) Records of training and qualification for members of the facility staff.

(h) Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

(i) Records of meetings of the (NRAG).