APPENDIX A

FACILITY LICENSE NO. R-87

TECHNICAL SPECIFICATIONS

FOR THE

PURDUE UNIVERSITY REACTOR

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1. DEFINITIONS

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

1.1 Channel Calibration - A channel calibration is an adjustment of the channel such that its output responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

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

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

1.4 Confinement - A closure on the reactor room that controls the movement of air into it and out through a controlled path.

1.5 Containment - A testable enclosure on the overall facility (for example, a reactor room) which is in the normally closed configuration and can support a defined pressure differential for functional purposes.

1.6 Core Experiment - A core experiment is one placed in the core, in the graphite reflector, or within six inches (measured horizontally) of the reflector. This includes any experiment in the pool directly above or below the core.

1.7 Experiment - An experiment shall mean:

a. any apparatus, device, or material installed in a core or experimental facility, or

b. any operation to measure reactor parameters or characteristics, or

c. any operation using the reactor as a source of radiation in conjunction with a) above.

1.8 Experimental Facility - Experimental facilities are:

a. those regions specifically designated as locations for experiments or

b. systems designed to permit or enhance the passage of a beam of radiation to another location.

1.9 Experiment With Movable Parts (Secured or Nonsecured) - An experiment with movable parts is an experiment that contains parts that are intended to be moved while the reactor is operating.

1.10 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in ""Dangerous Properties of Industrial Materials"" by N. I. Sax, Third Ed. (1968), or is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, 1966, ""Identification System for Fire Hazards of Materials,"" also enumerated in the ""Handbook of Laboratory Safety"" 2nd Ed. (1971) published by the Chemical Rubber Co.

1.11 Fueled Experiment - A fueled experiment is any experiment planned for irradiation of uranium 233, uranium 235, plutonium 239, or plutonium 241.

1.12 Measured Value - The measured value of a parameter is the value as it appears at the output of a measuring channel.

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

1.14 Movable Experiment - A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the core while the reactor is operating.

1.15 New Experiment - A new experiment is one whose nuclear characteristics have not been experimentally determined.

1.16 Nonsecured Experiment - Any experiment, experimental facility, or component of an experiment is considered to be unsecured when it is not secured as defined in part 1.28 of this section.

1.17 Operable - A system or component is operable when it is capable of performing its intended function in a normal manner.

1.18 Operating - A system or component is operating when it is performing its intended function in a normal manner.

1.19 Pool Experiment - A pool experiment is one positioned within the pool more than six inches (measured horizontally) from the graphite reflector.

1.20 Potential Reactivity Worth - The potential reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter an experiment's position or configuration.

The evaluation must consider possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes such as creating or filling of void spaces or motion of mechanical components. For removable experiments, the potential reactivity worth is equal to or greater than the static reactivity worth.

1.21 Reactor Facility - The reactor facility shall consist of that portion of the ground floor of the Duncan Annex of the Electrical Engineering Building occupied by the School of Nuclear Engineering. This consists of an area of approximately 5,000 square feet.

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

1.23 Reactor Secured - That overall condition where all of the following conditions are satisfied:

a. Reactor shutdown

b. Electrical power to the control rod circuits is switched off and the switch key is in proper custody.

c. No work in progress involving fuel movements, in-core components, experiments, or installed control rod drives.

1.24 Reactor Shutdown - That subcritical condition of the reactor where the negative reactivity, with or without experiments in place, is equal to or greater than the shutdown margin.

1.25 Readily Available on Call - Readily available on call shall mean the licensed senior operator shall insure that he is within a reasonable driving time (1/2 hour) from the reactor building, and the operator on duty is currently informed, and can contact him by phone.

1.26 Removable Experiment - A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.27 Reportable Occurrence - A reportable occurrence is any of the following:

a. 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.

b. 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.

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

d. Reactivity balance anomalies involving:

1. disagreement between expected and actual critical positions of more than 0.3% Δ k/k;

2. exceeding excess reactivity limit;

3. shutdown margin less conservative than specified in technical specifications;

4. unexpected short-term reactivity changes that cause a period of 10 seconds or less;

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

e. 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 Hazards Summary Report (HSR) and the Safety Analysis Report (SAR).

f. 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 HSR and the SAR.

g. 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.

h. 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.

i. 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 HSR, SAR, or technical specifications bases; or discovery during plant life of conditions not specifically considered in the HSR, the SAR, or the technical specification that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. An observed inadequacy in the implementation of either administrative or procedural controls which could result in operation of the reactor outside the limiting conditions for operation.

1.28 Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.29 Static Reactivity Worth - As used herein, the static reactivity worth of an experiment is the absolute value of the reactivity change which is measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment. For removable experiments, the terminal positions are fully removed from the reactor and fully inserted or installed in the normal functioning or intended position.

1.30 Surveillance and Test Intervals - These are intervals established for periodic surveillance and test actions. Established intervals shall be maintained on the average. Maximum intervals are allowed to provide operational flexibility, not to reduce frequency.

1.31 Tried Experiment - A tried experiment is:

a. An experiment previously performed in this facility, or

b. An experiment of approximately the same nuclear characteristics as an experiment previously tried.

1.32 True Value - The true value of a parameter is its exact value at any instant.

2. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit

Applicability - This specification applies to the steady state power level.

Objective - The objective is to define a power level below which it can be predicted with confidence that no damage to the fuel elements will occur.

Specification - The true value of the instantaneous power of the reactor shall not exceed 50 kW.

Basis - The Purdue University Reactor utilizes fuel of the same type as is used in several similar reactors, such as the reactor at the University of Missouri, Rolla. These reactors use natural convection cooling and are routinely operated at power levels exceeding 50 kW with no apparent damage to the fuel.

The steady state power of 50kW was chosen because calculations indicate that the average heat flux from fuel into coolant would be less then 0.5 watts/cm, and that no boiling would occur at this level. With fuel plate temperatures associated with this power level no damage to the fuel elements will occur. The aluminum alloy cladding does not melt below 1100^{O}F and is expected to maintain its integrity and retain essentially all of the fission fragments at temperatures below 1100^{O}F. For a step input of reactivity equal to the available excess in the core, combined with a postulated failure of the scram mechanisms such that all control rods jam out of the core, it is estimated that the coolant temperature would rise to less than 130^{O}F. This coolant temperature would restrict cladding temperatures well below 1100^{O}F thus assuring retention of all fission fragments.

2.2 Limiting Safety System Setting

Applicability - This specification applies to the reactor power level safety system setting for steady state operation.

Objective - The objective is to assure that the safety limit is not exceeded.

Specification - The measured value of the power level scram shall be no higher than 1.2 kW.

Basis - The LSSS has been chosen to assure that the reactor protective system will be actuated in such a manner as to prevent the safety limit from being exceeded during the most severe expected abnormal condition.

The safety margin between LSSS and the SL is sufficient to assure that the peak power achieved in a transient, starting at 1kW with a 1-second period and terminated by dropping a control rod, will not exceed 50kW. The 1-second period corresponds to a reactivity of .006 $\Delta k/k$, which is the maximum authorized to be loaded into the reactor.

The safety margin that is provided between the LSSS and the SL also allows for instrument uncertainties associated with measuring the above parameter.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability - These specifications apply to the reactivity conditions of the reactor, and the reactivity worths of control rods and experiments.

Objective - The objective is to assure that the reactor can be shut down at all times, that the safety limits will not be exceeded, and that operation is within the limits analyzed in the SAR.

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

a. The shutdown margin, relative to the cold xenon-free condition with the most reactive shim rod fully withdrawn, and the regulating rod fully withdrawn shall be at least 0.01 $\Delta k/k$.

b. The reactor shall be subcritical by more than 0.03 $\Delta k/k$ during core loading changes.

c. No shim-safety rod shall be removed from the core if the shutdown margin is less than 0.01 $\Delta k/k$ with the remaining shim-safety rod fully withdrawn.

d. The reactor shall be shutdown if the maximum positive reactivity of the core and any installed experiment exceeds 0.006 $\Delta k/k$.

e. The reactivity worth of each experiment shall be limited as follows:

Experiment	*Maximum Reactivity Worth*
Movable	.003 $\Delta k/k$
Unsecured	.003 $\Delta k/k$
Secured	.004 $\Delta k/k$

f. The total worth of all movable and unsecured experiments shall not exceed 0.003 $\Delta k/k$.

g. The total worth of all secured experiments shall not exceed 0.005 $\Delta k/k$.

Bases - The shutdown margin required by Specification 3.1.a assures that the reactor can be shut down from any operating condition and will remain shut down even if the control rod of the highest reactivity worth should be in the fully withdrawn position.

Specifications 3.1.b and 3.1.c provide assurance that the core will remain subcritical during loading changes and shim-safety rod maintenance or inspection.

Specification 3.1.d limits the allowable excess reactivity to the value assumed in the HSR. This limit assures that the consequences of reactivity transients will not be increased relative to transients previously reviewed, and assures reactor periods of sufficient length so that the reactor may be shutdown without exceeding the safety limit.

Specification 3.1.e limits the reactivity worth of secured experiments to values of reactivity which, if introduced as a positive step change, are calculated not to cause fuel melting. This specification also limits the reactivity worth of unsecured and movable experiments to values of reactivity which, if introduced as a positive step change, would not cause the violation of a safety limit. The manipulation of experiments worth up to 0.003 $\Delta k/k$ will result in reactor periods longer than 9 seconds. These periods can be readily compensated for by the action of the safety system without exceeding any safety limits.

A limitation of 0.003 $\Delta k/k$ for the total reactivity worth of all movable and unsecured experiments provides assurance that a common failure affecting all such experiments cannot result in an accident of greater consequences than the maximum credible accident analyzed in the HSR.

Specification 3.1.g along with 3.1.a assures that the reactor is capable of being shut down in the event of a positive reactivity insertion caused by the flooding of an experiment.

3.2 Reactor Safety System

Applicability - This specification applies to the reactor safety system and other safety-related instrumentation.

Objective - The objective is to specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

Specification - The reactor shall not be made critical unless the following conditions are met:

a. The reactor safety channels and safety-related instrumentation are operable in accordance with Tables I and II including the minimum number of channels and the indicated maximum or minimum set points.

b. Both shim-safety rods and the regulating rod shall be operable.

c. The time from the initiation of a scram condition in the scram circuit until the shim-safety rod reaches the rod lower limit switch shall not exceed one second.

TABLE I. SAFETY CHANNELS REQUIRED FOR OPERATION

Channel	*Minimum Number Required*	*Setpoint*	*Function*
Log count rate and period	1(a)	2 cps	2 cps rod withdrawal interlock
		12 sec. period	Setback
		7 sec. period	Slow scram
Log N and period	1(b)	12 sec. period	Setback
		7 sec. period	Slow scram
		7 sec. period	Fast scram
		120% power	Slow scram
Linear	1	110% range	Setback
		120% range	Slow scram
Safety	1(b)	110% power	Setback
		120% power	Fast scram
Manual Scram			
(console)	1		Slow scram
(hallway)	1		Slow scram

(a) Not required after Log N-Period channel comes on scale.
(b) Required to be operable but not on scale at startup.

TABLE II. SAFETY-RELATED CHANNELS (AREA RADIATION MONITORS)

Channel	*Minimum Number Required* (c)	*Setpoint*	*Function*
Pool top monitor	1	50 mR/hr or 2x full power background	Slow scram
Water process	1	7 1/2 mR/hr	Slow scram
Console monitor	1	7 1/2 mR/hr	Slow scram
Continuous air sampler	1		Air sampling

(c) For periods of time, not to exceed 12 hours of operation, a radiation monitor may be replaced by a gamma sensitive instrument which has its own alarm and is observable by the reactor operator.

Bases - The neutron flux level scrams provide redundant automatic protective action to prevent exceeding the safety limit on reactor power, and the period scram conservatively limits the rate of rise of the reactor power to periods which are manually controllable without reaching excessive power levels or fuel temperatures.

The rod withdrawal interlock on the Log Count Rate Channel assures that the operator has a measuring channel operating and indicating neutron flux levels during the approach to criticality.

The manual scram button and the "reactor on" key switch provide two methods for the reactor operator to manually shut down the reactor if an unsafe or abnormal condition should occur and the automatic reactor protection does not function.

The use of the area radiation monitors (Table II) will assure that areas of the Purdue University Reactor (PUR-1) facility in which a potential high radiation area exists are monitored. These fixed monitors initiate a scram whenever the preset alarm point is exceeded to avoid high radiation conditions.

Specifications 3.2.b and 3.2.c assure that the safety system response will be consistent with the assumptions used in evaluating the reactor's capability to withstand the maximum credible accident.

In specification 3.2.c. the rod lower limit switches are positioned to measure, as close as possible, the fully inserted position.

3.3 Primary Coolant Conditions

Applicability - This specification applies to the limiting conditions for reactor operation for the primary coolant.

Objective - The objective is to assure a compatible environment, adequate shielding, and a continuous coolant path for the reactor core.

Specification -

a. The primary coolant pH shall be maintained at 5.5 ± 1.0.

b. The primary coolant resistivity shall be maintained at a value greater than 330,000 ohm-cm.

c. The primary coolant shall be maintained at least 13 feet above the core.

Bases - Experience at the PUR-1 and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in specification 3.3.a and 3.3.b will minimize the amount and severity of corrosion of the aluminum components of the primary coolant system and the fuel element cladding.

The height of water in specification 3.3.c is enough to furnish adequate shielding as well as to guarantee a continuous coolant path.

3.4 Confinement

Applicability - This specification applies to the integrity of the reactor room.

Objective - The objective is to limit and control the release of airborne radioactive material from the reactor room.

Specification -

a. During reactor operation the following conditions will be met:

   1. The reactor room will be maintained at a negative pressure of at least 0.05 inch of water.

   2. All exterior doors in the reactor room shall remain closed except as required for personnel access.

b. All inlet and exhaust air ducts and the sewer vent shall contain an AEC #1 absolute filter or its equivalent.

c. Dampers in the ventilation system inlet and outlet ducts are capable of being closed.

d. The air conditioner can be shut off by the reactor operator.

Bases - The PUR-1 does not rely on a containment building to reduce the levels of airborne radioactive material released to the environment in the event of the design basis accident. However, in the event of such an accident, a significant fraction of the airborne material will be confined within the reactor room, and the specifications stated above will further reduce the release to the environment.

3.5 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, and to assure the safe operation of the reactor.

Specification - The reactor will not be operated unless the following conditions are met:

a. All experiments shall be constructed of material which will be corrosion resistant for the duration of their residence in the pool.

b. All experiments will follow procedures approved by the Committee on Reactor Operations.

c. Known explosive materials shall not be placed in the reactor pool.

d. Cooling shall be provided to prevent the surface temperature of an experiment from exceeding 100° C.

e. No experiment shall be placed in the reactor or pool that interferes with the safe operation of the reactor.

f. The radioactive material content, including fission products, of any singly encapsulated experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR 20. This dose limit applies to persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

g. The radioactive material content, including fission products, of any doubly encapsulated experiment or vented experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 Rem to the whole body or 1.5 Rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 Rem to the whole body or 30 Rem to the thyroid.

Bases - Specification 3.5.a through 3.5.e are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and serve as a guide for the review and approval of new experiments by the facility personnel and the Committee on Reactor Operations.

Specification 3.5.f and 3.5.g conform to the criteria set forth in Regulatory Guide 2.2 issued in November, 1973.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactivity Limits

Applicability - This specification applies to the surveillance requirements for reactivity limits.

Objective - The objective is to assure that the reactivity limits of Specification 3.1 are not exceeded.

Specification -

a. The shim-safety rod reactivity worths shall be measured and the shutdown margin calculated annually with no interval to exceed 15 months, and whenever a core configuration is loaded for which shim-safety rod worths have not been measured.

b. The shim-safety rods shall be visually inspected annually with no interval to exceed 15 months. If the rod is found to be deteriorated, it shall be replaced with a rod of equivalent or greater worth.

c. The reactivity worth of experiments placed in the PUR-1 shall be measured during the first startup subsequent to the experiment's insertion and shall be compared with the prior calculated value, and shall be verified if core configuration changes cause increases in experiment reactivity worth which may cause the experiment worth to exceed the values specified in Specification 3.1

Bases - Specification 4.1.a will assure that shim-safety rod reactivity worths are not degraded or changed by core manipulations which cause these rods to operate in regions where their effectiveness is reduced.

The boron stainless steel shim-safety rods have been in use at the PUR-1 since 1962, and over this period of time, no cracks or other evidence of deterioration have been observed. Based on this performance and the experience of other facilities using similar shim-safety rods, the specified inspection times are considered adequate to assure that the control rods will not fail.

4.2 <u>Reactor Safety System</u>

<u>Applicability</u> - This specification applies to the surveillance of the reactor safety system.

<u>Objective</u> - The objective is to assure that the reactor safety system is operable as required by Specification 3.2.

<u>Specification</u> -

a. A channel test of each of the reactor safety system channels listed in Table III shall be performed prior to each reactor startup following a shutdown in excess of 8 hours or if they have been repaired or de-energized. SPACE OUT

TABLE III.

SAFETY SYSTEM CHANNELS CHECKED AFTER PROLONGED SHUTDOWN

Log Count Rate (startup channel)
Log N-Period
Linear Level
Safety Channel

b. A channel check of each of the reactor safety system measuring channels in use or on scale shall be performed approximately every four hours when the reactor is in operation.

c. A channel calibration of the reactor safety channels shall be performed at the following average intervals:

   1. An electronic calibration will be performed annually, with no interval to exceed 15 months.

   2. A power calibration by foil activation will be performed annually, with no interval to exceed 15 months.

d. The operation of the radiation monitoring equipment shall be verified daily during periods when the reactor is in operation. Calibration of these monitors shall be performed semiannually, with no interval to exceed 7 1/2 months.

e. Shim-safety rod drop times will be measured annually, with no interval to exceed 15 months. These drop times shall also be measured prior to operation following maintenance which could affect the drop time or cause movement of the shim-safety rod control assembly.

Bases - A test of the safety system channels prior to each startup will assure their operability, and annual calibration will detect any long-term drift that is not detected by normal intercomparison of channels. The channel check of the neutron flux level channel will assure that changes in core-to-detector geometry or operating conditions will not cause undetected changes in the response of the measuring channels.

Area monitors will sound an alarm when they sense they are not operating correctly. In addition, the operator routinely records the readings of these monitors and will be aware of any reading which indicates loss of function.

The area monitoring system employed at the PUR-1 has exhibited very good stability over its lifetime, and semiannual calibration is considered adequate to correct long-term d:

The measured drop times of the shim-safety rods have been consistent since the PUR-1 was built. An annual check of this parameter is considered adequate to detect operation with materially changed drop times. Binding or rubbing caused by rod misalignment could result from maintenance; therefore, drop times will be checked after such maintenance.

4.3 Primary Coolant System

Applicability - This specification applies to the average surveillance schedules of the primary coolant system.

Objective - The objective is to assure high quality pool water, adequate shielding, and to detect the release of fission products from fuel elements.

Specification -

a. The pH of the primary coolant shall be recorded weekly.

b. The c nductivity of the primary coolant shall be recorded weekly.

c. The reactor pool water will be at or above the height of the skimmer trough whenever the reactor is operated.

d. Monthly samples of the primary coolant shall be taken to be analyzed for gross alpha and beta activity.

Bases - Weekly surveillance of pool water quality provides assurance that pH and conductivity changes will be detected before significant corrosive damage could occur.

When the reactor pool water is at the skimmer trough level, adequate shielding of more than 13 feet of water is assured.

Analysis of the reactor water for gross alpha and beta activity assures against undetected leaking fuel assemblies.

4.4 Containment

Applicability - This specification applies to the surveillance requirements for maintaining the integrity of the reactor room and fuel clad.

Objective - The objective is to assure that the integrity of the reactor room and the fuel clad is maintained, by specifying average surveillance intervals.

Specification -

a. The negative pressure of the reactor room will be recorded weekly.

b. Operation of the inlet and outlet dampers shall be checked semiannually, with no interval to exceed 7 1/2 months.

c. Operation of the air conditioner shall be checked semiannually, with no interval to exceed 7 1/2 months.

d. Representative fuel plates shall be inspected annually, with no interval to exceed 15 months.

Bases - Specification a, b, and c check the integrity of the reactor room, and d the integrity of the fuel clad. Based upon past experience these intervals have been shown to be adequate for insuring the operation of the systems affecting the integrity of the reactor room and fuel clad.

4.5 Experiments

Applicability - This specification applies to the surveillance of limitations on experiments.

Objective - The objective is to assure that the radioactive material content of experiments does not exceed the limits of parts f and g of specification 3.5.

Specification -

a. Calculations shall be made on samples of known composition to assure that the limits of specification 3.5.f and 3.5.g are not exceeded.

b. The mass of samples of unknown composition shall not exceed 10 grams.

Bases - The basis for the 10 gram limitation is the analysis of the hazards of different elements activated for a period of 30 minutes in the irradiation position with the maximum flux. The irradiation time of 30 minutes is three times longer than the maximum recommended time for unknown samples. Calculations show that elemental iodine presents the greatest hazard of non-radioactive elements. Iodine has only one stable isotope, I-127, which produces I-128 when activated. I-128 combines a relatively short half-life (25.0 mins.) with a high average beta energy per decay, high concentration factor in the body in a small organ, the thyroid, and a relatively large activation cross section. Using the assumptions that a 10 gram sample breaks and is instantly uniformly dispersed in the reactor room as either a gas or small particles and that the time to vacate the room is 90 seconds, the committed dose to the thyroid is 1.5 rem, 10% of the annual dose as stated in 10 CFR 20. Using the assumption that the ventilation system is turned off as the personnel exit the room, the leakage from the room to unrestricted areas, combined with the short half-life will not result in committed doses exceeding .15 rem during a two-hour period

5. DESIGN FEATURES

5.1 Site Description

5.1.1 The reactor is located on the ground floor of the Duncan Annex of the Electrical Engineering Building, Purdue University, West Lafayette, Indiana.

5.1.2 The School of Nuclear Engineering controls approximately 5000 square feet.

5.1.3 Access to this area is restricted except when classes are held here.

5.1.4 The reactor room remains locked at all times except for the entry or exit of authorized personnel.

5.1.5 The PUR-1 is housed in a closed room designed to restrict leakage.

5.1.6 The minimum free volume of the reactor room shall be 15,000 cubic feet.

5.1.7 The ventilation system is designed to exhaust air or other gases from the reactor room through an exhaust vent at a minimum of 50 feet above the ground.

5.1.8 Openings into the reactor room consist of the following:

a. Three personnel doors

b. Two locked transformer vault doors

c. Air intake

d. Air exhaust

e. Sewer vent

5.2 Fuel Assemblies

5.2.1 The fuel assemblies shall be MTR type consisting of aluminum clad plates enriched to approximately 93% in the U-235 isotope.

5.2.2 A standard fuel assembly shall consist of 10 fuel plates containing a maximum of 165 grams of U-235.

5.2.3 A control fuel assembly shall consist of 6 fuel plates containing a maximum of 99 grams of U-235.

5.2.4 Partially loaded fuel assemblies in which some of the fuel plates are replaced by aluminum plates containing no uranium may be used.

5.3 Fuel Storage

5.3.1 All reactor fuel assemblies shall be stored in a geometric array where k_{eff} is less than 0.8 for all conditions of moderation and reflection.

5.3.2 Irradiated fuel assemblies and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the temperature of the fuel assemblies or fueled devices will not exceed $100^{o}C$.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 The reactor facility shall be an integral part of the School of Nuclear Engineering of the Schools of Engineering at Purdue University as shown in Figure 6.1.

6.1.2 The Reactor Supervisor shall be responsible for the safe operation of the PUR-1. He shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, including the technical specifications and other applicable regulations.

6.1.3 In all matters pertaining to the operation of the reactor and the administrative aspects of these technical specifications, the Reactor Supervisor shall report to and be directly responsible to the Head of the School of Nuclear Engineering. In all matters pertaining to radiation safety he shall be responsible to the Radiological Control Committee.

6.1.4 Minimum Qualifications of Reactor Personnel - The minimum qualifications should be consistent with the American National Standard for the Selection and Training of Personnel for Research Reactors, ANSI/ANS 15.4, and includes the following:

a. Reactor Supervisor - At the time of appointment to the active position, the reactor supervisor shall have a minimum of five years of nuclear experience. He shall have a baccalaureate degree or equivalent experience in an engineering or other scientific field. The degree may fulfill four years of experience on a one-for-one basis. The reactor supervisor shall possess a valid Senior Operator License.

b. Licensed Senior Operator - At the time of appointment to the active position, a senior operator shall have a minimum of a high school diploma or equivalent and should have four years

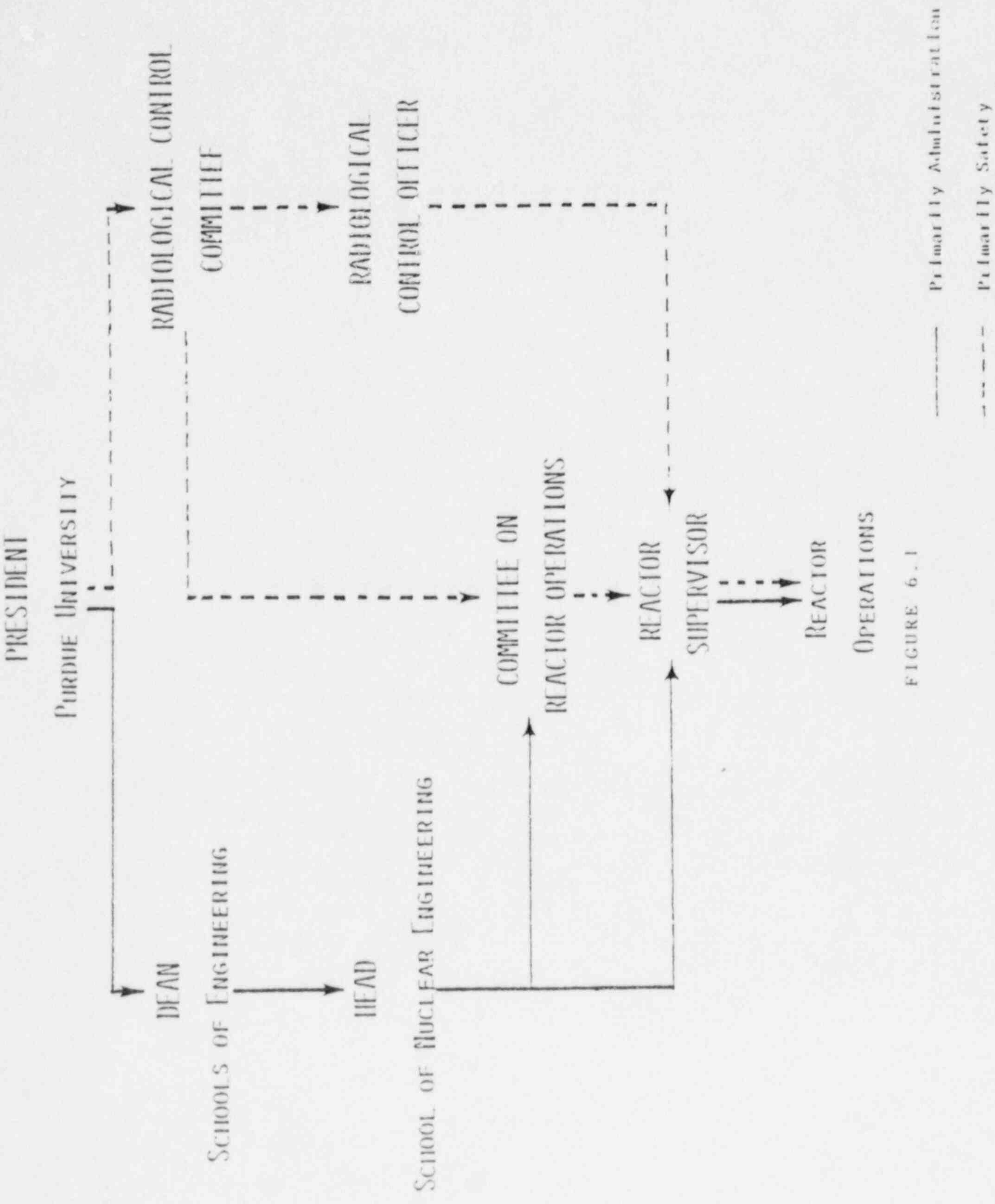
PRESIDENT
PURDUE UNIVERSITY
RADIOLOGICAL CONTROL COMMITTEE
RADIOLOGICAL CONTROL OFFICER
DEAN
SCHOOLS OF ENGINEERING
HEAD
SCHOOL OF NUCLEAR ENGINEERING
COMMITTEE ON REACTOR OPERATIONS
REACTOR SUPERVISOR
REACTOR OPERATIONS
Primarily Administration
Primarily Safety

FIGURE 6.1

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. He shall hold a valid NRC Senior Reactor Operator's license.

c. Licensed Operator - At the time of appointment to the active position, an operator shall have a high school diploma or equivalent. He shall hold a valid NRC Reactor Operator's license.

d. Operator Trainee - An operator trainee shall have all the qualifications to become a licensed operator except for possessing an operator's license.

6.1.5 A Radiological Control Officer who is organizationally independent of the PUR-1 operations group shall advise the Reactor Supervisor in matters concerning radiological safety. Minimum qualifications for the Radiological Control Officer (RCO) is a bachelor's degree or the equivalent in a science or engineering subject, including some formal training in radiation protection. The RCO should have at least five years of professional experience in applied radiation protection. A master's degree may be considered equivalent to one year of professional experience, and a doctor's degree equivalent to two years of professional experience where course work related to radiation protection is involved. At least three years of this professional experience should be in applied radiation protection work in a nuclear facility dealing with radiological problems.

6.1.5 A licensed reactor operator (RO) or licensed senior reactor operator (SRO) pursuant to 10 CFR 55 shall be present at the controls unless the reactor is shut down as defined in these specifications. During training operations an unlicensed operator may operate the controls but only under the direct supervision of an RO or an SRO.

6.1.7 An SRO shall be present or readily available on call at any time that the reactor is operating.

6.1.8 The identity of, and method for rapidly contacting, the SRO on duty shall be known to the reactor operator at any time that the reactor is operating.

6.1.9 The presence of an SRO shall be required at the reactor facility during recovery from unplanned or unscheduled shutdowns except in instances which result from the following:

a. A verified electrical power failure or interruption exclusive of internal power supply failures or interruption of the reactor instrumentation, control, and safety systems;

b. Accidental manipulation of equipment in a manner which does not affect the safety of the reactor;

c. A verified practice of the evacuation of the building initiated by persons exclusive of reactor operations personnel.

The SRO shall be notified of the shutdown and shall determine its cause. If due to one of the enumerated reasons above, he shall decide if his presence is necessary for a subsequent start up.

6.1.10 The presence of an SRO at the reactor faility is unnecessary for the initial daily start up, provided the core remains unchanged from the previous run.

6.1.11 The minimum crew for operating the reactor shall consist of 2 (two) persons, one of whom must be an NRC licensed member of the PUR-1 operations group. The second crew member must be instructed as to how to shut down the reactor in the event of an emergency.

6.1.12 During fuel changes and movement of large bulk experiments, an SRO will be present in the reactor room.

6.1.13 The Reactor Supervisor or his designated alternate shall be responsible for the facility retraining and replacement training program.

6.2 Review and Audit

6.2.1 A Committee on Reactor Operations (CORO) shall report to the Radiological Control Committee on matters of Radiation Safety and the Head, School of Nuclear Enginering on matters of administration and safety. CORO will advise the Reactor Supervisor on those areas of responsibility specified in Sections 6.2.5 and 6.2.6. The minimum qualifications for persons on the CORO shall be five years of professional work experience in the discipline or specific field he represents. A baccalaureate degree may fulfill four years of experience.

6.2.2 The CORO shall have at least 7 (seven) members of whom no more than a minority shall be directly concerned with the administration or direct use of the reactor. These members shall include the following:

a. The Chairman, a responsible, senior technical person, knowledgeable in the field of reactor technology, who does not have line responsibility for day-to-day operation of the reactor.

b. A senior radiological control officer.

c. The Purdue University Director of Safety and Security.

d. The Reactor Supervisor.

e. Three senior scientific staff members.

6.2.3 The CORO or a subcommittee thereof shall meet quarterly with no interval to exceed 4 months. The CORO shall meet semiannually, with no interval to exceed 7 1/2 months.

6.2.4 A quorum of CORO shall consist of not less than a majority of the full Committee and shall include the chairman or his designated alternate. No more than one-half of the voting members present shall be members of the reactor operations staff.

6.2.5 The CORO shall review and approve:

a. Safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments that may be conducted without prior NRC approval under the provision of Section 50.59, 10 CFR, to ascertain whether such actions would constitute an unreviewed safety question, or would require a change in Technical Specifications.

b. Proposed changes to procedures, equipment or systems that change the original intent or use, or those that might 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, and those that might 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 might affect nuclear safety.

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

h. Audit reports.

6.2.6 Audits

Audits of facility activities shall be performed under the cognizance of the CORO 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 CORO member. These audits shall examine the operating records and encompass:

a. The conformance of facility operation to the Technical Specifications and applicable license conditions, to be done annually with no interval to exceed 15 months.

b. The performance training and qualifications of the licensed facility staff, to be done annually with no interval to exceed 15 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, to be done annually with no interval to exceed 15 months.

d. The Facility Emergency Plan and implementing procedures, to be done biennially with no interval to exceed 2 1/2 years.

e. The Facility Security Plan and implementing procedures, to be done biennially with no interval to exceed 2 1/2 years.

f. Any other area of facility operation considered appropriate by the CORO or the Reactor Supervisor, to be done annually with no interval to exceed 15 months.

6.2.7 Records

Records of CORO activities shall be prepared and distributed as indicated below.

a. Minutes of each CORO meeting shall be prepared and forwarded to the Reactor Supervisor within 30 days following each meeting.

b. Reports of reviews encompassed by section 6.2.5 e, f, and g above, shall be prepared and forwarded to the Reactor Supervisor within 30 days following completion of the review.

c. Audit reports encompassed by Section 6.2.6 above, shall be forwarded to the CORO Chairman and to the management responsible for the areas audited within 30 days after completion of the audit.

6.3 Safety Limit Violation - The following actions shall be taken in the event the 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 designee), the Reactor Supervisor and to the CORO not later than the next work day.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the CORO. 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 CORO and the Reactor Supervisor within 14 days of the violation, in support of a request to the Commission for authorization to resume operations.

6.4 Operating Procedures - Written procedures, including applicable check lists reviewed and approved by the CORO, shall be in effect and followed for the following operations:

6.4.1 Startup, operation, and shutdown of the reactor.

6.4.2 Installation and removal of fuel elements and control rods.

6.4.3 Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes.

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

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

6.4.6 Experiment installation, operation, and removal.

6.4.7 Implementation of the Security Plan and Emergency Plan.

6.4.8 Calibration and preventive maintenance procedures on required instruments, systems, or components.

Non-routine operations which require the sequential performance of a series of sub-tasks shall be carried out with the written procedure at the console. To assure adherence to the documentation of the procedure, each step will be entered in the log book by the operator on duty as it is completed.

Substantive changes to the above procedures shall be made only with the approval of the CORO. The Reactor Supervisor may make changes to procedures which do not change the intent of the original procedure. All such changes to the procedures shall be documented and subsequently reviewed by CORO.

6.5 Operating Records

6.5.1 The following records and logs shall be prepared and retained for at least five years:

a. Normal facility operation and maintenance.

b. Reportable occurrences.

c. Tests, checks, and measurements documenting compliance with surveillance requirements.

d. Records of experiments performed.

e. Records of radioactive shipments.

f. Changes of operating procedures.

g. Facility radiation and contamination surveys.

6.5.2 The following records and logs shall be prepared and retained for the life of the facility:

a. Gaseous and liquid waste released to the environs.

b. Offsite environmental monitoring surveys.

c. Radiation exposures for all PUR-1 personnel.

d. Fuel inventories and transfers.

e. Updated, corrected, and as-built facility drawings.

f. Minutes of CORO meetings.

g. Records of transient or operational cycles for those components designed for a limited number of transients or cycles.

h. Records of training and qualification for members of the facility staff.

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

j. Annual operating reports.

6.6 Reporting Requirements - The following information shall be submitted to the USNRC in addition to the reports required by Title 10, Code of Federal Regulations.

6.6.1 Annual Operating Reports--a report covering the previous year shall be submitted to the Director of the Office of Nuclear Reactor Regulation with a copy to the NRC Regional Administrator by March 31 of each year. It shall include the following:

a. Changes in plant design and operation

1. changes in facility design

2. performance characteristics (e.g. equipment and fuel performance).

3. changes in operating procedures which relate to the safety of facility operations

4. results of surveillance tests and inspections required by these technical specifications

5. a brief summary of those changes, tests, and experiments which required authorization from the Commission pursuant to 10 CFR 50.59(a)

b. Power Generation--A tabulation of the thermal output of the facility during the reporting period.

c. Shutdowns--A listing of unscheduled shutdowns which have occurred during the reporting period, tabulated according to cause, and a brief discussion of the corrective and preventive actions taken to prevent recurrence.

d. Maintenance--A discussion of corrective maintenance (excluding preventive maintenance) performed during the reporting period on safety-related systems and components.

e. Changes, Tests, and Experiments--A brief description and a summary of the safety analysis and evaluation for those changes, tests, and experiments which were carried out without prior Commission approval, pursuant to the requirements of 10 CFR Part 50.59(b).

f. Radioactive Effluent Releases--A summary of the nature, amount, and maximum concentrations 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.

g. Occupational Personnel Radiation Exposure--A summary of radiation exposures greater than 25% of the appropriate limits of 10 CFR 20 received during the reporting period by facility personnel (faculty, students, or experimenters).

6.6.2 Non-Routine Reports

a. Reportable Occurrence Reports

In the event of a reportable occurrence (defined in 1.0) notification shall be made within 24 hours by telephone and telegraph* to the Regional Administrator, followed by a written report within 10 days to the Director, Office of Nuclear Reactor Regulation with a copy to the Regional Administrator. The written report on these reportable occurrences, and to the extent possible the preliminary telephone and telegraph notification shall: (a) describe, analyze, and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

b. Unusual Events

A written report shall be forwarded within 30 days to the Director, Office of Nuclear Reactor Regulation with a copy to the Regional Administrator in the event of:

1. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the HSR, the SAR or the bases for the Technical Specifications.

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* Telegraph notification may be sent on the next working day in the event of a reportable occurrence during a weekend or holiday period.

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications, in the HSR, or in the SAR.

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

4. Discovery of an inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.