APPENDIX A

TO FACILITY LICENSE NO. R-47

TECHNICAL SPECIFICATIONS

FOR THE

LYNCHBURG POOL REACTOR

DOCKET NO. 50-99

BABCOCK & WILCOX COMPANY

DATE: December, 1978

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1.0 DEFINITIONS

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Condition of Operation (LCO) are as defined in 50.36 of 10 CFR Part 50.

1.1 Reactor Safety System – That combination of safety channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

1.2 Safety Channel -- A measuring channel in the reactor safety system.

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

1.4 Logic Scram – A solid state electronic scram circuit capable of fast (<100 msec) response.

1.5 Relay Scram – An electro-mechanical scram circuit with a relatively slow (>400 msec) response.

1.6 Channel Calibration – 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.

1.7 Channel Test – The introduction of a signal into the channel to verify that it is operable.

1.8 Channel Check – A qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.9 Operating – A component or system is performing its intended function in its normal manner.

1.10 Operable – A component or system is capable of performing its intended function in its normal manner.

1.11 True Value – The True Value of a process variable is its actual value at any instant.

1.12 Measured Value -- The Measured Value of a process variable is the value of the variable as indicated by a measuring channel.

1.13 Reactor Operation – The control rods installed in the core are not fully inserted, or the magnet key is in the keyswitch. Reactor operation is not possible when there are less than six fuel elements loaded on the grid chosen for operation.

1.14 Routine Operation – Operation of a core configuration for experiments or tests other than annual surveillance required by the Technical Specifications.

1.15 Unscheduled Shutdown – Any unplanned shutdown of the reactor, after startup has been initiated, caused by actuation of the reactor safety system, operator error, equipment malfunctions, or a manual shutdown in response to conditions which could adversely affect safe operation.

1.16 Reactor Secured – The following conditions are satisfied:

a. The full insertion of all control rods has been verified.

b. The console key is removed, and

c. No operation is in progress which involves moving fuel elements in the LPR pool, moving reflector elements to or from the core, the insertion or removal of experiments from the core, or control rod maintenance.

1.17 Readily Available On Call – within a 15 (fifteen) mile radius of the reactor site. In the event of bad weather, the senior operator shall insure that he is within a reasonable driving time from the reactor building. The senior operator shall always keep the reactor operator informed of where he may be contacted and the telephone number.

1.18 Grid 1 – An 8x10 lattice with a pitch of 3.035" x 3.189". It is located in the north end of the LPR pool and is used in conjunction with the autoclave and beam ports. There is no provision for forced cooling on Grid 1.

1.19 Grid 2 – A 7x7 lattice with the same pitch as Grid 1. It is located approximately 6 inches southof Grid 1 and its west edge is aligned with the west edge of Grid 1. Grid 2 may be operated in forced or natural convection cooling.

1.20 Shim-Safety Rod – A control rod fabricated from borated stainless steel which is used to compensate for fuel burnup, temperature, and poison effects. A shim-safety rod is magnetically coupled to its drive unit allowing it to perform the function of a safety rod when the magnet is de-energized.

1.21 Regulating Rod – A control rod of lower reactivity worth than a shim rod fabricated from stainless steel which is used to control reactor power. The rod may be controlled by the operator with a manual switch or by an automatic controller.

1.22 Autoclave – A heated 3-foot-ID by 9-foot long cylindrical pressure vessel which penetrates the pool wall. Experiments within the vessel may be subjected to a 2-foot square neutron beam from the LPR core.

1.23 Autoclave Operation – The autoclave is at a pressure greater than the normal static water head or at a temperature above ambient.

1.24 An Experiment – (1) An apparatus, device or material, placed in the reactor core, in an experimental facility, or in line with a beam of radiation emanating from the reactor, (2) any operation designed to measure reactor characteristics.

a. 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 under tactile restraints 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.

b. Unsecured Experiment – Any experiment, experimental facility, or component of an experiment is deemed to be unsecured if it is not and when it is not secured. Moving parts of experiments are deemed to be unsecured when they are in motion.

c. Movable Experiment – An experiment which may be inserted, removed, or manipulated while the reactor is critical.

d. Removable Experiment – 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. The initial loading of a removable experiment must be performed in conjunction with inverse multiplication measurements, as in a new core loading. Removable experiments must be designed such that their removal from the reactor core is impossible without prior removal of fuel.

1.25 Experimental Facilities – Facilities used to perform experiments. They include horizontal and vertical beam tubes, thermal columns, void tanks, pneumatic transfer systems, incore facilities at single element positions, out-of-core irradiation racks, and the autoclave.

1.26 Reportable Occurrence – Any of the following:

a. A safety system setting less conservative than the limiting setting established in the Technical Specifications.

b. Operation in violation of a limiting condition for operation established in the Technical Specifications.

c. A safety system component malfunction or other component or system malfunction which could, or threatens to, render the safety system incapable of performing its intended safety functions.

d. Release of fission products from a failed fuel element.

e. An uncontrolled or unplanned release of radioactive material from the restricted area of the facility.

f. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Appendix B, Table 1 of 10 CFR 20.

g. An uncontrolled or unanticipated change in reactivity in excess of 0.005 $\Delta k/k$.

h. Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the facility.

i. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.27 Weekly – Once during each calendar week.

1.28 Monthly – Once during each calendar month.

1.29 Every Six Months – During the sixth calendar month following the preceding calibration.

1.30 Annually – Prior to commencing routine operation each calendar year, or within the last two months of the preceding calendar year.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limits

2.1.1 Safety Limits in the Forced Convection Mode

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the steady state with forced convection flow. These variables are:

P – Reactor Thermal Power

W – Reactor Coolant Flow through the Core

T_i – Reactor Coolant Inlet Temperature

H – Height of Water above the Top of the Core

Objective

To assure that the integrity of the fuel clad is maintained.

Specification

a. The true value of reactor power (P) shall not exceed the limit specified in Figure 2.1 corresponding to various true values of flow (W).

b. The true value of reactor coolant inlet temperature (T_i) shall not exceed 120°F (49°C).

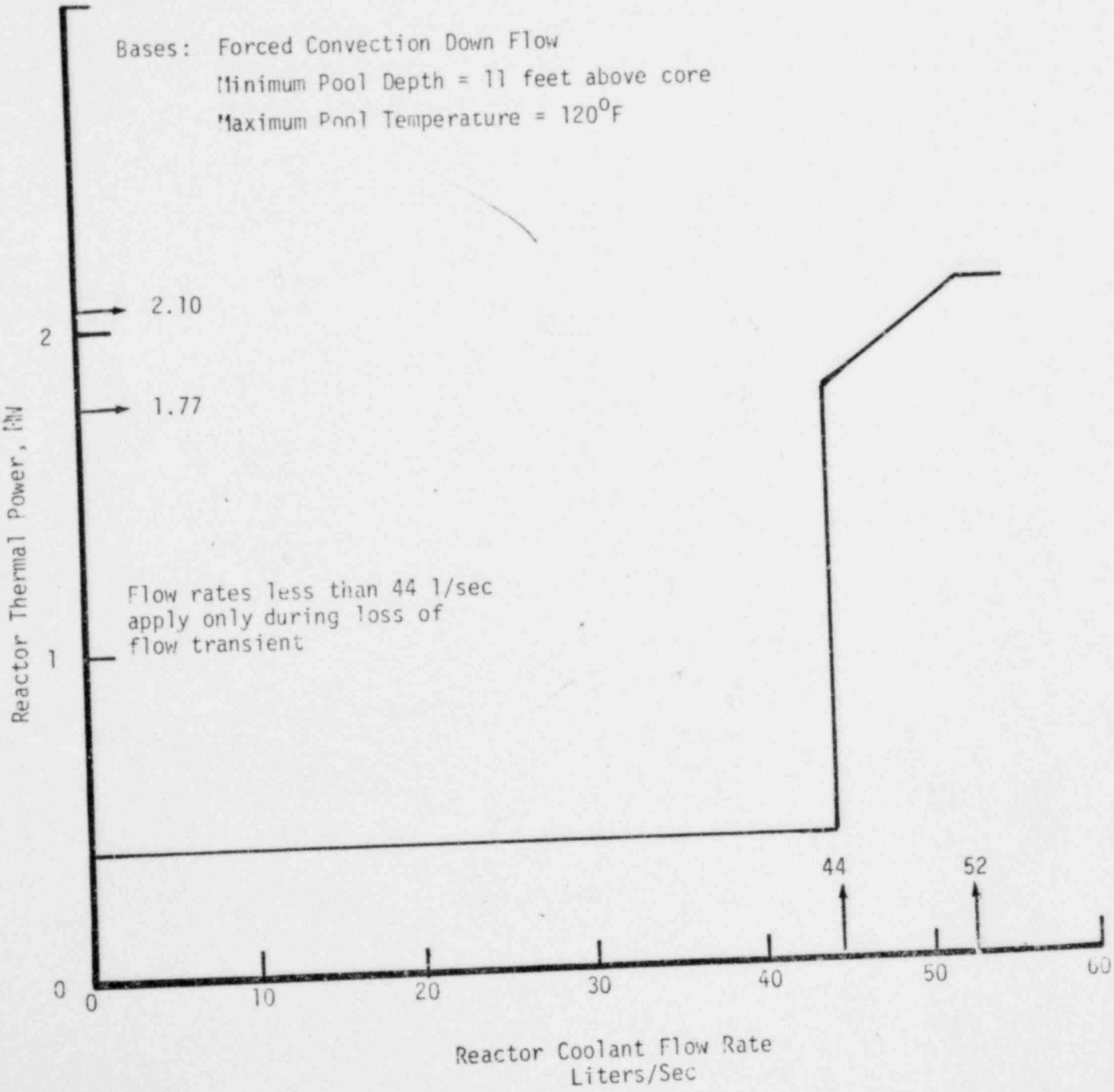
c. The true value of water height above the core (H) shall not be less than 11 feet while the reactor is operating.

Bases

A thermal hydraulic analysis has been performed to determine the conditions of power and coolant flow at which burnout and thereby cladding damage could occur. The analysis showed that for the Lynchburg Pool Reactor the possibility of an excursive flow instability exists at a lower power level than that required for critical heat flux (or departure from nucleate boiling) and therefore the avoidance of this flow instability has become the criterion used in establishing the safety limit in the region above 80% to full flow. The analysis is given in the Appendix to the Babcock & Wilcox letter dated February 27, 1973.

FIGURE 2.1

LPR SAFETY LIMIT
FOR FORCED CONVECTION



In the region below 80% of full flow, where under a loss of flow transient at power, the flow coasts down to zero, reverses, and then establishes natural convection, the criterion used in selecting a safety limit is that the fuel cladding temperature shall remain significantly below the melting point of the aluminum throughout the transient.

For initial conditions of 80% of full flow (44 liters/sec) and an operating power of 1.4 MW, the maximum cladding temperature reached under the conservative assumptions of the analysis was 634°F which is well below the temperature at which fuel cladding damage could possibly occur. The safety limit shown in Figure 2.1 for flow less than 80% of full flow is the steady state power, 400 kW, corresponding to the safety limit in the natural convection mode. The analysis supporting the safety limit for the loss of flow transient is given in the Appendix to the B&W letter dated May 17, 1973.

2.1.2 Safety Limits in the Natural Convection

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the natural convection mode of operation. These variables are:

P — Reactor Thermal Power

T_i — Reactor Coolant Inlet Temperature

H — Height of Water Above the Top of the Core

Specification

a. The true value of the reactor thermal power (P) shall not exceed 400 kW.

b. The true value of the reactor coolant inlet temperature (T_i) shall not exceed the limit specified in Figure 2.2 corresponding to various true values of the reactor thermal power (P)

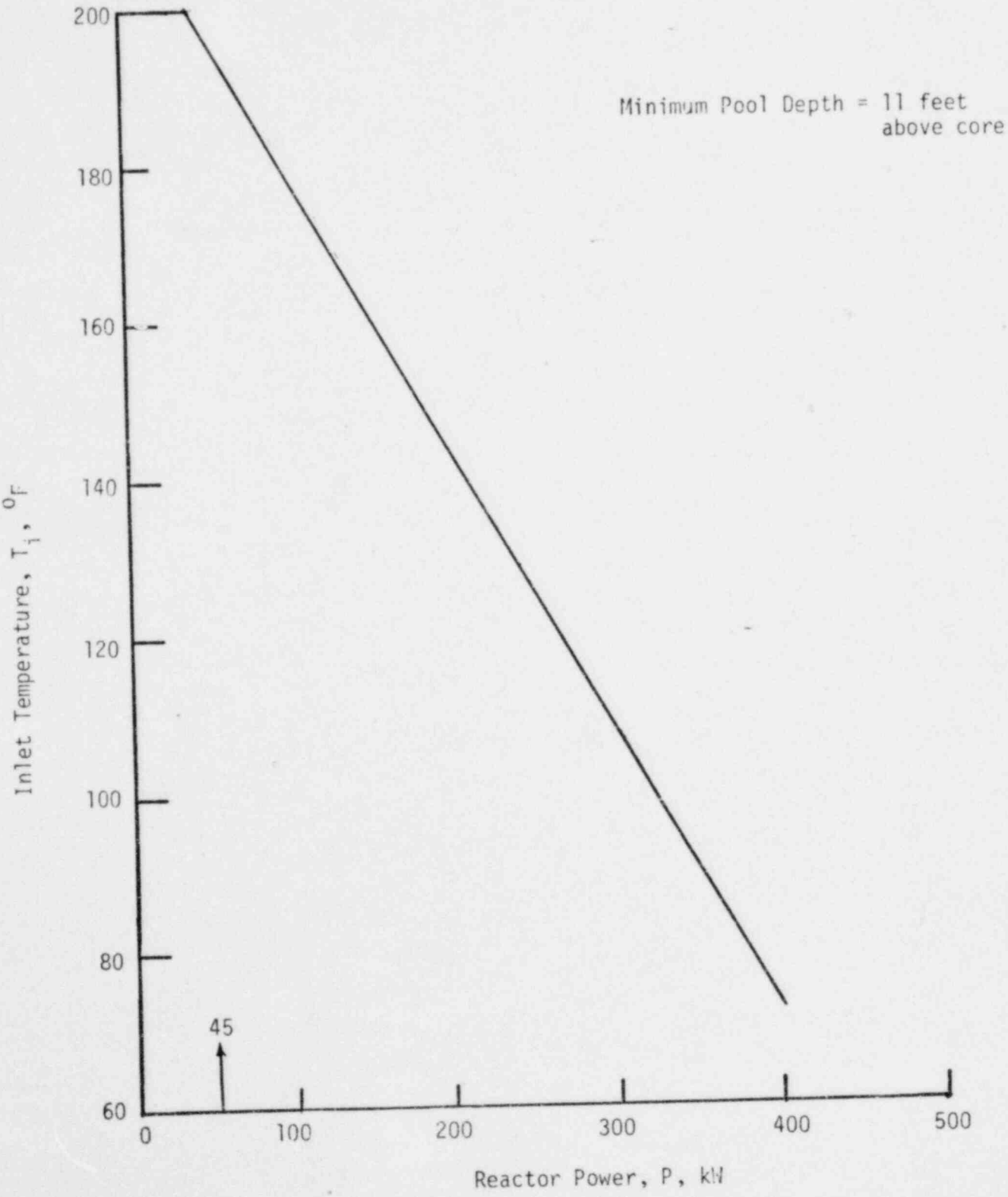
c. The true val[illegible] water height above the top of the core shall not be less than 11 feet while the reactor is operating.

Bases

Figure 2.2 represents the combination of reactor power and reactor coolant inlet temperature required to prevent the surface temperature of the plates from exceeding the water saturation temperature (225°F). No boiling will occur in the core since the saturation temperature is below the temperature at which nucleate boiling can occur. The analysis which constituted the curve in Figure 2.2 is contained in the Appendix to the Babcock & Wilcox letter dated May 17, 1973.

The above analysis is also in close agreement with the earlier experimental measurements reported in BAW-74 Supplement 4, "Application for Extension of AEC License R-47 to Permit 450 Kilowatt Operation of the Lynchburg Pool Reactor."

FIGURE 2.2
LPR SAFETY LIMIT FOR NATURAL CONVECTION



2.2 Limiting Safety System Setting (LSSS)

2.2.1 Limiting Safety System Setting in the Forced Convection Mode

Applicability

This specification applies to the set points for the safety channels monitoring reactor thermal power (P), primary coolant flow (W), height of water above the top of the core (H), and core inlet temperature (T_i).

Objective

To assure that automatic protective action is initiated to prevent a safety limit being exceeded.

Specification

a. The measured value of the limiting safety system setting for reactor thermal power (P), primary coolant flow through the core (W), height of water above the top of the core (H), and reactor coolant inlet temperature (T_i) shall be as follows:

Variable	LSSS
P (Max.)	1.4 MW
W (Min.)	48 liters/sec
H (Min.)	11.5 feet
T_i (Max.)	110°F

Bases

The limiting safety system settings specified above have been chosen to assure that automatic protective action will correct the most severe abnormal situation anticipated (loss of flow transient) before a safety limit is exceeded. The safety margin that is provided between the LSSS and the SL also allows for the most adverse combination of instrument uncertainties associated with measuring the above observable parameters.

These instrument uncertainties include a power level variation of seven percent, flow variation of twelve percent and a pool water variation of six inches, which are quite conservative.

The analysis given in the Babcock & Wilcox letter dated May 17, 1973, for a loss of flow transient indicates that, with the reactor thermal power at its LSSS of 1.4 MW and the height of water above the core and coolant inlet temperature at their respective LSSS namely 11.5 feet and 110°F, a LSSS of 48 liters/second is adequate to prevent the safety limits established in Specification 2.1.1 from being exceeded.

2.2.2 Limiting Safety System Settings in the Natural Convection Flow Mode

Applicability

These specifications apply to the set point for the safety channels monitoring the reactor thermal power (P), the height of water above the top of the core (H), and the reactor coolant inlet temperature (T_i)

Objective

To assure that automatic protective action is initiated to prevent a safety limit being exceeded.

Specification

a. The measured value of the limiting safety system setting for reactor thermal power (P), height of water above the top of the core (H), and the reactor coolant inlet temperature (T_i) shall be as follows:

Variable	LSSS
P (Max)	250 kW
H (Min)	11.5 feet
T_i (Min)	110°F

Bases

These LSSS have been chosen to assure that the reactor protective system will function in such a manner to prevent any safety limit from being exceeded during the most severe expected abnormal condition.

The safety margin between the LSSS and the SL for reactor thermal power is sufficient to assure that the power level will not exceed 400 kW for any anticipated reactivity transients including experiment malfunctions and the introduction of cold water into the core by improper manipulation of the coolant system.

The six inch margin between the LSSS and the SL for height of water above the top of the core is adequate to prevent the pool water level from dropping below the safety limit, while the reactor is operating, in the event of the most severe anticipated leak involving a pipe failure or beam tube failure.

There is a 14 degree margin between the reactor coolant inlet temperature LSSS and safety limit when the reactor thermal power is at its LSSS of 250 kW. Since it requires approximately 7 minutes to raise the pool temperature 1°F at this power level, a 14°F margin is considered adequately conservative.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objective

To assure that the reactor can be shut down at all times and that the safety limits will not be exceeded.

Specification

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

b. The reactor shall be subcritical by more than 0.0275 $\Delta k/k$ during loading changes.

c. A shim-safety rod shall not be removed from the core if the shutdown margin is less than 0.01 $\Delta k/k$ with the most reactive remaining shim-safety rod fully withdrawn.

d. The reactivity worth of each experiment and its rate of reactivity change shall be limited as follows:

Experiment	Maximum Reactivity Worth	Maximum Rate of Reactivity Insertion
Movable	0.0025 $\Delta k/k$	Optional
Unsecured (P>100 kW)	0.004 $\Delta k/k$	0.0025 $\Delta k/k$ sec
(P<100 kW)	0.006 $\Delta k/k$	0.0025 $\Delta k/k$ sec
Secured	0.015 $\Delta k/k$	Not applicable

e. The total reactivity worth of all movable and unsecured experiments shall not exceed 0.015 $\Delta k/k$.

f. The effective subcritical multiplication factor, k_{eff}, shall not exceed 0.95 for any lattice loaded into the autoclave.

g. Experiments which could increase reactivity by flooding shall not remain in or adjacent to the core unless the reactivity change caused by the flooding is measured or conservatively estimated and the shutdown margin required in Specification 3.1.a is satisfied after flooding.

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 shutdown after cool down and xenon decay even if the control rod of the highest reactivity worth should be in the fully withdrawn position.

Specification 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 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 0.0025 $\Delta k/k$ will result in reactor periods longer than 10 seconds. These periods can be readily compensated for by the action of the safety system without exceeding any safety limits.

A limitation of 0.015 $\Delta k/k$ for the total reactivity worth of all movable and unsecured experiments provide 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 1000 kW Hazard Evaluation Report, BAW-74, Supplement 8.

Specification 3.1.f assures that experimental configurations loaded in the autoclave will always remain subcritical by a margin of 0.05 $\Delta k/k$.

Specification 3.1.g assures that the shutdown margin required by specification 3.1.a will be met in the event of a positive reactivity insertion caused by the flooding of an experiment.

3.2 Reactor Safety System

Applicability

These specifications apply to the reactor safety system and other safety related instrumentation.

Objective

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:

a. The reactor safety systems and safety related instrumentation are operable in accordance with Tables 3.1 and 3.2 including the minimum number of channels and the indicated maximum or minimum set points,

b. All shim-safety rods are operable,

c. The delay time from the initiation of a scram condition in the logic scram circuit to the release of the shim-safety rods shall not exceed 100 milliseconds,

d. The time from the initiation of a scram condition in the logic scram circuit until the shim-safety rods are inserted shall not exceed 650 milliseconds, and

e. Positive mechanical means are provided to prevent the inadvertent lifting of fuel elements through movement of control rods.

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 primary coolant flow rate scram and natural convection header open scram provide redundant channels to assure, when the reactor is at power levels which required forced flow cooling, that an automatic shutdown of the reactor will occur to prevent exceeding a safety limit if sufficient flow is not maintained. The coolant mode switch selects either the forced flow mode or the natural convection mode of operation. The intermediate position of the mode switch scrams the reactor, thus preventing a change of coolant mode during operation.

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.

TABLE 3.1 REQUIRED SAFETY CHANNELS

Channel	Setpoint	Minimum Number Required	Function
Log Count Rate	2 cps	1[a]	2 cps rod withdrawal interlock
Log N		1	Wide range power level measurement and input for period scram
Period Scram	3 sec	1	Scram
Neutron level			
Forced convection	1.4 MW	2	Scram
Natural convection	250 kW		
Primary Coolant Flow Rate	48 liters per sec	1[b]	Scram
Natural Convection Header Open		1[b]	Scram
Loss of Primary Coolant Pump Power		1[b]	Scram
Coolant Mode Switch Intermediate Position		1	Scram
Reactor Coolant Inlet Temperature	110°F	1	Scram
Pool Level	11.5 ft	1	Scram
Manual Scram		1	Scram
Magnet Power Keyswitch		1	Scram
Autoclave Area Radiation Monitor	2 x operating background	1[d][e]	Scram
Beam Port Area Radiation Monitor	20 mR/hr	1[d][e]	Scram
Reactor Console Area Radiation Monitor	20 mR/hr	1[e]	Scram

TABLE 3.1 CONT'D.

Channel	Setpoint	Minimum Number Required	Function
Pool Top Air Monitor (20 MPC-hrs)		1	Ventilation System Shutdown
Loss of Water from Autoclave	5" from bottom of pressurizer	1[c]	Scram
High Pressure or Temperature in Autoclave	775 psig 515°F	1[c]	Scram

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(a) Required for reactor startups when power level is less than 5 watts and for operation with less than 1 decade overlap between the source range channel and the neutron flux level channels.

(b) Not required for natural convection operation.

(c) Required only for autoclave operation.

(d) Not required if no experiment is being performed in the area.

(e) For periods of time not to exceed 24 hours, one of the required area radiation monitors may be replaced by a gamma sensitive instrument which has its own alarm or is kept under visual observation.

TABLE 3.2 REQUIRED SAFETY RELATED INSTRUMENTATION

Instrumentation	Setpoint	Minimum Number Required	Function
Linear Level Channel		1	Linear power level measurement and input for the automatic control mode
Reactor Coolant Outlet Temperature		1[a]	Provide information for the heat balance determination
Pool Top Area Radiation Monitor	1000 mR/hr	1[b]	Alarm
Heat Exchange Area Radiation Monitor	10000 mR/hr	1[a][b]	Alarm

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(a) Not required for natural convection operation.

(b) For period of time not to exceed 24 hours one of the required area radiation monitors may be replaced by a gamma sensitive instrument which has its own alarm or is kept under visual observation.

The core inlet temperature channel and the pool level channel initiate a reactor shutdown before either of these two parameters reach their safety limit.

The manual scram button and the "magnet power" keyswitch provide two methods for the reactor operator to manually shutdown the reactor if an unsafe or abnormal condition should occur and the automatic reactor protection does not function.

The use of the above specified area radiation monitors will assure that areas of the LPR facility in which a potential high radiation area exists are monitored. These fixed monitors alarm at the LPR console whenever the preset alarm point is exceeded to alert the operator of high radiation conditions.

In the event of a failed experiment or fuel element leakage, the personnel within the LPR confinement area will be alerted well before the exposure limits set by 10 CFR Part 20 are reached. For operation of experiments which produce mainly Sr-90, the alarm set point shall be based on 20 MPC hours of Sr-90. For all other reactor operations, the alarm set point shall be based on 20 MPC hours of mixed fission products of which Sr-90 is expected to comprise less than 1%.

The scram circuits associated with the autoclave protect against continued operation of the reactor if abnormal temperature or pressure develops within the autoclave.

Specification 3.2.b through 3.2.d assures that the safety system response will be consistent with the assumptions used in evaluating the reactors capability to withstand the "maximum credible accident."

Specification 3.2.e provides assurance that the fuel elements which contain control rods will be restrained against inadvertent motion in the manner described in the January 26, 1961 Babcock and Wilcox submission.

3.3 LPR Confinement Area

Applicability

This specification applies to the LPR confinement area requirements.

Objective

To minimize the release of airborne radioactive materials from the LPR confinement area.

Specification

1. During reactor operation, all windows and doors on the perimeter of the LPR confinement area shall remain closed except as required for personnel access.

2. During reactor operation, the motor controlled intake louvre in the LPR wing heating and ventilation system shall be either closed or operable, and shall be closed automatically upon a signal from the LPR pool top air monitor.

Bases

Due to its remote location the LPR 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, the LPR staff does recognize that a significant fraction of the airborne material will be confined within the LPR wing and by requiring the doors and windows in the LPR area to be closed, the amount of airborne activity released will be reduced.

3.4 Limitations of Experiments

Applicability

This specification applies to experiments installed in the LPR.

Objective

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

Specification

a. Each experiment shall be designed so that the surface temperature shall be below the temperature calculated for the inception of nucleate boiling. Prior to insertion in the reactor, any capsule which is expected to operate with an internal pressure greater than its external pressure shall be tested at a pressure one and one-half times the calculated maximum differential pressure.

b. All experiments which are in contact with the reactor coolant shall be either corrosion resistant or encapsulated within corrosion resistant containers.

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

d. Neutron radiograph of explosives shall be conducted with the explosives contained in a blast-proof irradiation container which has been successfully prototype tested and demonstrated not to fail by detonation of at least twice the amount of explosive to be irradiated.

e. Inverse multiplication measurements shall be performed in the loading of all new fuel lattices into the autoclave.

f. The radioactive material content of any singly encapsulated experiment shall be limited such that the complete release of all gaseous, particulate, or volatile components directly to the reactor building will not result in exposures in excess of 10% of the equivalent annual exposures stated in 10 CFR 20 for persons continuously present in unrestricted areas for two hours or for persons present in the restricted area during the length of time required to evacuate the restricted area.

g. The radioactive material content of any doubly encapsulated experiment shall be limited such that the postulated complete release of all gaseous, particulate, or voltatile components from the encapsulation or confining boundary of the experiment will not result in exposures in excess of 500 mRem whole body or 1.5 Rem thyroid to persons continuously present in unrestricted areas for a period of two hours from the time of release, or exposures in excess of 5 Rem whole body or 30 Rem thyroid for persons located within the restricted area for the length of time required to evacuate the restricted area.

Bases

Specification 3.4.a through 3.4.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 and untried experiments by the facility personnel and the Safety Review Committee.

Neutron radiography at the LPR is conducted in a vertical beam tube which is located adjacent to the core outside the reflector elements. This beam tube does not penetrate the pool wall and it cannot cause a loss of coolant from the pool.

In the radiography of explosives, the explosive devices will be contained, during exposure inside a blast-proof box which will be located above the pool at the end of the beam tube. The box will not be coupled to the beam tube and will be constructed of an adequate thickness of material to fully contain any blast effects or missiles which might be generated by an accidental detonation. The blast-proof box will be tested prior to its use by detonating a device which contains twice the amount of explosive material contained in the device to be irradiated to demonstrate the strength of the blast-proof box.

Specification 3.4.f and 3.4.g will assure that the quantities of radioactive materials contained in experiments will be limited such that their failure will not result in restricted or unrestricted area doses which exceed the maximum annual exposures stated in 10 CFR 20.

3.5 Airborne Effluents

Applicability

This specification applies to the monitoring of Ar^{41} and other airborne effluents from the LPR.

Objective

The objective is to assure that exposure to the public resulting from the release of Ar^{41} and other airborne effluents will be as low as practicable.

Specification

1. The particulate air monitor shall alarm and secure the ventilation system if the integrated airborne exposure exceeds 20 MPC hours.

2. The integrated dose from Ar^{41} and direct radiation as measured by radiation monitors on the perimeter of the restricted area, shall not exceed 500 mRem in any one year.

Bases

In the event of a failed experiment or fuel element leakage, the personnel within the LPR confinement area will be alerted well before the exposure limits of 10 CFR Part 20 are reached. For operation of experiments which produce mainly Sr-90, the alarm set point shall be based on 20 MPC hours of Sr-90. For all other reactor operations, the alarm set point shall be based on 20 MPC hours of mixed fission products of which Sr-90 is expected to comprise less than 1%.

A uniform concentration of Ar^{41} at MPC (4×10^{-8} μCi/cc) will result in a dose of 500 mRem in a year in the LPR control room (based on a 168 hour week). By limiting the integrated dose on the perimeter of the restricted area to 500 mRem in any one year, the average Ar^{41} concentration is maintained below the MPC for release to an unrestricted area.

3.6 Liquid Effluents

Applicability

This specification applies to the monitoring of liquid effluents from the LPR.

Objective

The objective is to assure that exposure to the public resulting from the release of liquid effluents will be as low as practicable.

Specification

1. The activity of liquid effluents released to unrestricted areas shall be measured prior to release and shall not exceed 1 MPC at the time of release.

2. The total activity of all liquid effluents released to unrestricted areas from the facility shall not exceed 5 Ci/yr.

Bases

This specification is based on the activity limits specified in 10 CFR 20 for unrestricted areas. In actual practice, the time averaged release concentration is normally less than 10% of MPC.

The NRC staff has determined that the requirements for maintaining effluent releases at levels that are as low as practicable is met by maintaining releases to unrestricted areas below 5 Ci/yr.

3.7 Primary Coolant Conditions

Applicability

This specification applies to the limiting conditions for primary coolant pH, resistivity, radioactivity and flow distribution.

Objective

To maintain the primary coolant in such condition as to minimize the corrosion of the primary coolant system, fuel clad, and other reactor components.

Specification

a. The primary coolant pH shall be maintained between 4.5 and 7.5

b. The primary coolant resistivity shall be maintained at a value greater than 200,000 ohm-cm except for periods of time not to exceed 7 days when the resistivity may fall to 70,000 ohm-cm.

c. For operation in the forced convection mode all grid positions shall contain fuel elements, reflector elements, grid plugs or experimental facilities.

Bases

Experience at B&W and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in Specification 3.7.a and 3.7.b will minimize the amount and severity of corrosion of the aluminum components of the primary coolant system and the fuel element cladding.

The requirement that all grid positions be occupied will prevent the degradation of calculated flow rates due to flow bypassing the active fueled region through an unoccupied grid plate position.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactivity Limits

Applicability

This specification applies to the surveillance requirements for reactivity limits.

Objective

To assure that the reactivity limits of Specification 3.1 are not exceeded.

Specification

a. Shutdown margin shall be measured annually and whenever a core configuration is loaded for which the shutdown margin has not been measured. The shutdown margin shall be measured by the rod drop method.

b. Shim-safety rods shall be visually inspected after each 2500 megawatt hours of reactor operation but not less frequently than annually.

c. The reactivity worth of experiments inserted in the LPR shall be measured during the first startup subsequent to the experiments insertion, 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 LPR since 1961; and over this period of time, no cracks or 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.

The specified surveillance relating to the reactivity worth of experiments will assure that the reactor is not operated for extended periods before determining the reactivity worth of experiments. This specification will also provide assurance that experiment reactivity worths do not increase beyond the established limits due to core configuration changes.

4.2 Reactor Safety System

Applicability

This specification applies to the surveillance of the reactor safety system.

Objective

To assure that the reactor safety system is operable as required by Specification 3.2.

Specification

a. A channel test of each of the reactor safety system channels listed in Table 4.1 shall be performed prior to each reactor startup following a shutdown in excess of 6 hours or if they have been repaired or de-energized.

TABLE 4.1

Neutron Flux Level Safety Channels

Period Safety Channel

Primary Coolant Flow Rate Channel[a]

Natural Convection Header Open Channel[a]

Coolant Mode Switch Channel

Log Count Rate (startup channel)

- - - - -

(a) Not required for natural convection operation.

b. A channel calibration of the safety channels listed in Table 3.1, which can be calibrated, shall be performed at least annually.

c. A channel check of the neutron flux level safety channels shall be performed weekly if the reactor is operated at a power level above 100 kW. A channel check of the neutron flux level safety channels, comparing the channel outputs with a heat balance, shall be performed weekly if the reactor is operated in the forced convection mode at a power level above 200 kW. A power calibration shall be performed annually, and for each new core configuration.

d. The operation and set points of the radiation monitoring system and the pool top air monitor required in Specification 3.2 shall be verified prior to each reactor startup following a shutdown in excess of 6 hours or if the system has been repaired or de-energized.

e. The area radiation monitors and the pool top air monitor required in Specification 3.2 shall be calibrated at least every six months.

f. Shim-safety rod drop time and delay time shall be measured at least annually.

g. Shim-safety rod drop time shall be measured whenever the shim-safety rod's core location is changed or maintenance is performed which could affect the rod's drop time.

Bases

Prerun tests of the safety system channels 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.

The area monitors installed at the LPR are equipped with a failure alarm which senses a built-in background signal and alarms if a unit does not respond. In addition, the operator routinely records the readings of these monitors and will be aware of any reading which indicates loss of function.

The monitoring system employed at the LPR has exhibited very good stability over its lifetime, and semiannual calibration is considered adequate to assure that it is free from long-term drift.

The measured drop times and delay times of the shim-safety rods have been consistent for 20 years since the LPR was built. Annual check of these parameters is considered adequate to detect any deterioration which could change the drop time or delay time. 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 surveillance of the primary coolant system.

Objective

To assure high quality pool water and to detect the release of fission products from fuel elements.

Specification

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

b. The resistivity of the primary coolant shall be measured weekly.

c. The radioactivity of the primary coolant shall be analyzed monthly. This analysis shall be performed weekly if the reactor has been operated during the week.

d. A material balance shall be performed following each pool make-up to discover significant leakage of primary coolant from the facility.

Bases

Weekly surveillance of pool water quality and radioactivity provides assurance that pH and conductivity changes that could accelerate the corrosion rate of the primary coolant system would be detected before significant corrosive damage could occur, and that leaking fuel elements are not being used in the reactor.

A routine calculation which compares make up water volume with known modes of pool water loss will provide early warning of small pool water leaks in buried pipes and pool walls.

4.4 Fuel Elements

Applicability

This specification applies to the surveillance of the fuel elements.

Objective

To detect damage or deterioration of the fuel elements.

Specification

All fuel elements shall be visually inspected at least annually.

Bases

The above inspection interval is based upon past experience which has shown an annual inspection to be adequate for insuring fuel element integrity.

5.0 DESIGN FEATURES

5.1 Fuel Elements

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

The standard fuel element will have ten plates, each of which will have a nominal loading of 19 grams U-235.

Partially loaded fuel elements may also be used.

5.2 Site Description

Specification

1. The Lynchburg Pool Reactor (LPR) is located at the Lynchburg Research Center (LRC) in Campbell County, Virginia.

2. There is approximately a 600-foot radius exclusion area around the LPR. The area consists of land owned by the Babcock & Wilcox Company with the exception of a railroad right-of-way through the property at a distance of approximately 500 feet from the reactor.

3. The LRC controls access to restricted areas by chain-link fence, a five-strand wire fence, a gate, a doorway or temporary barricades and a roving patrol. Appropriate radiation warning signs are installed.

5.3 Fuel Storage

Specification

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

2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 The organizational structure of the Lynchburg Research Center relating to the Lynchburg Pool Reactor (LPR) shall be as shown in Figure 6.1.

6.1.2 The LPR Operations Supervisor shall be responsible for the safe operation of the LPR. 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 the regulations.

6.1.3 In all matters pertaining to the operation of the plant and these technical specifications, the LPR Operations Supervisor shall report to and be directly responsible to the Manager, Nuclear Physics Section.

6.1.4 Qualifications, LPR Operations Supervisor – The minimum qualifications for the LPR Operations Supervisor shall be at least four years of reactor operating experience and holder of a valid Senior Operator License for the LPR. Years spent in baccalaureate or graduate study may be substituted for operating experience on a one-for-one basis up to a maximum of two years.

6.1.5 A health physicist who is organizationally independent of the LPR operations group shall advise the LPR Operations Supervisor in matters concerning radiological safety.

6.1.6 An operator or senior operator licensed pursuant to 10 CFR 55 shall be present at the controls whenever the reactor is operating as defined in these specifications.

6.1.7 A licensed senior operator (LSO) shall be present or readily available on call at any time the reactor is operating.

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

6.1.9 The presence of a licensed senior operator shall be required during recovery from unplanned or unscheduled shutdowns or significant reduction in power 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 malfunction of an instrument or circuit; and

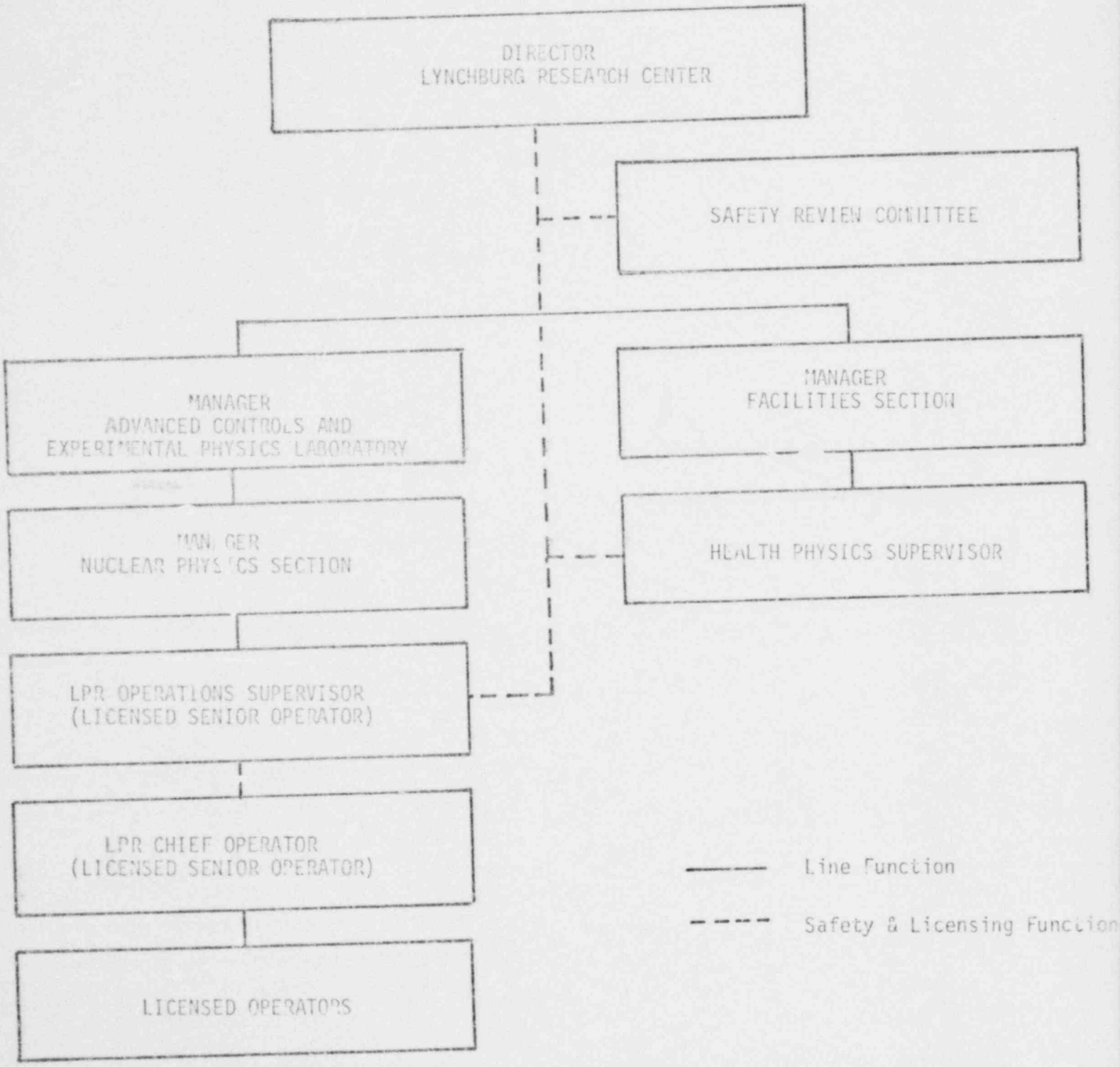
d. A verified practice of accidental evacuation of the building.

The licensed senior operator shall be notified of the shutdown or power reduction and shall determine that the shutdown was caused by one of the enumerated occurences and shall determine that his presence at the facility shall not be required.

6.1.10 All licensed operators at the facility shall participate in an approved operator requalification program as a condition of their continued assignment of operator duties.

FIGURE 6.1

LINE ORGANIZATION CHART FOR THE LPR



6.2 Review and Audit

6.2.1 A Safety Review Committee shall review and audit reactor operations and advise the Director of the Lynchburg Research Center in matters relating to the health and safety of the public and the safety of the facility operations.

6.2.2 The Safety Review Committee shall have at least five members of whom no more than the minority shall be from the line organization shown in Figure 6.1. The Committee shall be made up of senior personnel who shall collectively provide experience in reactor engineering, reactor operations, chemistry and radiochemistry, instrumentation and control systems, radiological safety, and mechanical and electrical systems.

6.2.3 The Committee shall meet at least once every four months.

6.2.4 The quorum shall consist of not less than a majority of the full Committee and shall include the chairman or his designated alternate.

6.2.5 Minutes of each Committee meeting shall be distributed to the Director of the Lynchburg Research Center, all Safety Review Committee members, and such others as the chairman may designate.

6.2.6 The Safety Review Committee shall:

a. Review and approve proposed experiments and tests utilizing the reactor facility which are significantly different from tests and experiments previously performed at the LPR.

b. Review reportable occurrences.

c. Review and approve proposed Standard Operating Procedures and proposed changes to Standard Operating Procedures.

d. Review and approve proposed changes to the Technical Specifications and proposed amendments to facility license and review proposed changes to the facility made pursuant to 10 CFR 50.59(c).

e. Audit reactor operations and reactor operational records for compliance with internal rules, procedures, and regulations and with licensed provisions including Technical Specifications.

f. Audit existing Standard Operating Procedures for adequacy and to assure that they achieve their intended purpose in light of any changes since their implementation.

g. Audit plant equipment performance with particular attention to operating anomalies, reportable occurrences, and the steps taken to identify and correct their causes.

6.3 Action To Be Taken In the Event of a Reportable Occurrence

In the event of a reportable occurrence, as defined in these Technical Specifications, the following action shall be taken:

6.3.1 The LPR Operations Supervisor shall be notified of the occurrence. Corrective action shall be taken to correct the abnormal conditions and to prevent its recurrence.

6.3.2 A report of such occurrences shall be made to the Safety Review Committee; the Manager, Advanced Controls and Experimental Physics Laboratory; the Manager, Nuclear Physics Section; and the Nuclear Regulatory Commission in accordance with Section 6.6.2. The report shall include an analysis of the causes of the occurrence, the effectiveness of corrective actions taken, and the recommendations of measures to prevent or reduce the probability or consequences of recurrence.

6.4 Operating Procedures

Written procedures, including applicable check lists, reviewed and approved by the Safety Review Committee 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, control rods, experiments, and experimental facilities.

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

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

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

6.4.6 Periodic surveillance of reactor instrumentation and safety systems, area monitors, and continuous air monitors.

6.4.7 Facility security plan.

Substantive changes to the above procedures shall be made only with the approval of the Safety Review Committee. Temporary changes to the procedures that do not change their original intent may be made with the approval of the LPR Operations Supervisor. All such temporary changes to the procedures shall be documented and subsequently reviewed by the Safety Review Committee.

6.5 Operating Records

6.5.1 The following records and logs shall be prepared and retained at the facility 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. Operator requalification program records.

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

a. Gaseous and liquid waste released to the environs.

b. Offsite enviornmental monitoring surveys.

c. Radiation exposures for all LPR personnel.

d. Fuel inventories and transfers.

e. Facility radiation and contamination surveys.

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

g. Minutes of Safety Review Committee meetings.

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 appropriate Regional Inspection and Enforcement Office by March 31 of each year. It shall include the following:

a. Operations Summary — A summary of operating experience occurring during the reporting period including:

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), and

6. changes in the plant operating staff serving in the following positions:

a) LPR Operations Supervisor

b) Health Physicist

c) Safety Review Committee members

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 preventive actions taken to prevent recurrence.

d. Maintenance — A discussion of corrective maintenance (excluding preventative 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 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 statement of the quantities of radioactive effluents released from the facility, with data summarized on a monthly basis following the general format of USNRC Regulatory Guide 1.21:

1. Gaseous Effluents

a) Gross Radioactivity Releases

1) Total gross radioactivity (in curies), primarily noble and activation gases.

2) Average concentration of gaseous effluents released during normal steady state operation. (Averaged over the period of reactor operation.)

3) Maximum instantaneous concentration of gaseous radionuclides released during special operation, tests, or experiments, such as beam tube experiments, or pneumatic tube operation.

4) Percent of technical specification limit.

NOTE: Where effective measurements of the above quantities cannot be made at or prior to the point of release, a conservative effluent release upper limit shall be estimated based upon calculation of the sources available within the facility for such releases.

b) Iodine Releases — (Required if iodine is identified in primary coolant samples, isotopic analysis required in (1)(a) above, or if fueled experiments are conducted at the facility.)

1) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

2) Percent of MPC.

c) Particulate Releases

1) Total gross radioactivity (β,γ) released (in curies) excluding background radioactivity.

2) Gross alpha radioactivity released (in curies) excluding background radioactivity. (Required if the operational or experimental program could result in the release of alpha emitters.)

3) Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.

4) Percent of MPC for particulate radioactivity with half-lives greater than eight days.

2. Liquid Effluents

a) Total gross radioactivity (β,γ) released (in curies) excluding tritium and average concentration released to the unrestricted area or sanitary sewer (averaged over period of release).

b) The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area.

c) Total alpha radioactivity (in curies) released and average concentration released to the unrestricted area (averaged over the period of release).

d) Total volume (in ml) of liquid waste released.

e) Total volume (in ml) of water used to dilute the liquid waste during the period of release prior to release from the restricted area.

f) Total radioactivity (in curies), and concentration (averaged over the period of release) by nuclide released, based on representative isotopic analyses performed for any release which exceed 1×10^{-7} μCi/ml.

g) Percent of technical specification limit for total radioactivity from the facility.

g. Environmental Monitoring — For each medium sampled, e.g., air, surface water, soil, fish, vegetation, include:

1. Number of sampling locations and a description of their location relative to the reactor.

2. Total number of samples.

3. Number of locations at which levels are found to be significantly above local backgrounds.

4. Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and the location of that point with respect to the site.

5. The maximum cumulative radiation dose which could have been received by an individual continuously present in an unrestricted area during reactor operation from:

   a) direct radiation and gaseous effluent, and

   b) liquid effluent.

If levels of radioactive materials in environmental media, as determined by an environmental monitoring program, indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10 CFR Part 20, estimates of the likely resultant exposure to individuals and to population groups and assumptions upon which estimates are based shall be provided.

If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

h. Occupational Personnel Radiation Exposure

A summary of radiation exposures greater than 500 mRem (50 mRem for persons under 18 years of age) received during the reporting period by facility personnel.

6.6.2 Non-Routine Reports

a. Reportable Occurrence Reports

Notification shall be made within 24 hours by telephone and telegraph to the Director of the appropriate Regional Inspection and Enforcement Office (copy to the Director of Licensing) followed by a written report within 14 days to the Director of the Regional Inspection and Enforcement Office in the event of a reportable occurrence as defined in Section 1.0. The written report on a reportable occurrence, 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 conditions 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 of the Regional Inspection and Enforcement Office 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 Safety Analysis Report or in the bases for the Technical Specifications.

2. Discovery of any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

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.

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