VIRGINIA POLYTECHNIC INSTITUTE

AND

STATE UNIVERSITY

NUCLEAR RESEARCH REACTOR

TECHNICAL SPECIFICATIONS

APPENDIX A

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1.0 Definitions

1.1 Reactor Shutdown - The reactor shall be considered shutdown whenever:

a. the coolant-moderator is dumped, and

b. the console switch is in the "off" position and the key withdrawn.

1.2 Reactor Operation - Reactor operation shall mean any condition wherein the reactor is not shutdown.

1.3 Operable - A system or component shall be considered operable when it is capable of performing its intended function in its normal manner.

1.4 Operating - A system or component shall be considered to be operating when it is performing its intended function in its normal manner.

1.5 An Experiment - An Experiment is an apparatus, device or material, placed in the reactor core, in an experiment facility, or in line with a beam of radiation emanating from the reactor, excluding normal reactor instrumentation.

A. Secured Experiment - Any experiment, experiment 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 core. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, 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. Movable Experiment - A movable experiment is one which may be inserted, removed, or manipulated while the reactor is operating.

1.6 New Experiment - A "new experiment" is an experiment which, in the opinion of the Reactor Supervisor, reactor radiation safety officer, and a senior reactor operator, differs from experiments previously carried out on the reactor facility.

1.7 Reactor Safety Channels - Reactor safety channels shall mean those circuits, including their associated input circuits, which are designed to initiate a reactor scram, or interlocks that provide reactor protection capability.

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

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

1.10 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

1.11 Safety Limits - Safety limits are limits upon important reactor variables which are necessary to reasonably protect against the uncontrolled release of radioactivity.

1.12 Limiting Safety System Settings - Limiting safety system settings are maximum or minimum settings for automatic protective devices so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

1.13 Limiting Conditions for Operation - Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

1.14 Unscheduled Shutdown - An unscheduled shutdown is any unplanned shutdown of the reactor, after startup has been initiated.

1.15 True Value - The true value of a parameter is its actual value at any instant.

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

1.17 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 parameter.

1.18 Reportable Occurrence - A reportable occurrence is any of those conditions described in Section 6.5.3 of this specification.

1.19 Experiment Facilities - An experiment facility is any structure, device or pipe system which is intended to guide, orient, position, manipulate, control the environment or otherwise facilitate a multiplicity of experiments of similar character.

1.20 Control Rod - A control rod is a rod fabricated from neutron absorbing material which is used to compensate for fuel burnup, temperature, and poison effects. A control rod is magnetically coupled to its drive unit allowing it to perform the safety function when the magnet is de-energized.

1.21 Readily Available on Call - Readily available on call means an individual who, (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where he may be rapidly contacted (e.g. by phone, etc.) (3) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 1 hr. or within a 30 mile radius).

1.22 Scram Time - is the elapsed time from the time a scram is initiated to the time the slowest control rod is fully inserted.

## 2.0 Safety Limits and Limiting Safety System Settings

### 2.1 Safety Limits of Reactor Operation

A. Applicability - This specification applies to the variables that affect the maximum fuel plate temperatures.

1. Steady State power in MW.
2. Maximum transient power in MW.
3. Reactor period in seconds.

B. Objective - To assure fuel cladding integrity.

C. Specification -

1. Maximum steady state power level is (reserved) .
2. Maximum transient power level is 14 MW.
3. Shortest reactor period of 90 msec.

D. Bases - It is shown in ref. 1 and ref. 2 that if the maximum positive excess reactivity available was simultaneously inserted into the reactor and all automatic safety functions fail, no operator action taken, the variables specified in section 2.1.C would be achieved. It is also shown in ref. 1 and ref. 2 that if these safety limits are reached no fuel or cladding melting, no water expulsion, and no core disassembly will occur. Reactor power and reactor period will be maintained well within safety limit specifications through limiting safety system scram settings (sec. 2.2.)

### 2.2 Limiting Safety System Settings

2.2.1 LSSS Related to the Safety Limits

A. Applicability - This specification applies to the setpoints of the safety channels.

B. Objective - To insure that automatic action is initiated that will prevent a safety limit from being exceeded and to limit excess reactivity available.

C. Specification

1. Automatic protective action will occur at:

a. ≤ 125% of maximum rated steady state power of 500KW.

b. ≥ 5 second reactor period.

D. Basis - It is shown in ref. 1 and ref. 2 that an instantaneous insertion of less than .8% ΔK/K with no protection action will result in not exceeding any safety limits, therefore any protective action initiated will terminate a transient prior to exceeding a safety limit.

2.2.2 LSSS Not Related to the Safety Limit

A. Applicability - the specification applies to the setpoint of the safety channel.

B. Objective - to insure that automatic action is initiated to shutdown the reactor.

C. Specificiation - automatic protective action will occur if core tank water level is 1" or less below the top of the tanks.

D. Basis - to prevent moderator from overflowing the core tanks and wetting of the graphite reflector.

## 3.0 Limiting Conditions for Operation

### 3.1 Reactivity Limitations

3.1.1 Shutdown Margin - The minimum shutdown margin provided by control rods in the cold, xenon-free condition with the highest worth safety or shim rod and regulating rod fully withdrawn, and with the highest worth non secured experiment in its most positive reactive state shall be less than 0.5% ΔK/K.

3.1.2 Excess Reactivity - The core shall not be loaded with an excess reactivity above cold clean critical of greater than or equal to .8% ΔK/K including positive reactivity from non-secured experiments.

3.1.3 Experiments - Reactivity limits on experiments are specified in para. 3.4.

3.2 Control and Safety Systems

3.2.1 Scram Time - The elapsed time from the time a scram is initiated to the time the slowest control rod is fully inserted shall not exceed 0.8 seconds.

3.2.2 Measuring Channels - The minimum number and type of measuring channels operable and providing information to the control room operator required for reactor operation are given in Table I, Instrument Arrays, Section A.

Bases - The power range power level instruments provide redundant information on reactor power in the range of 1% - 150% of the normal operating power level of 500KW.

The intermediate range power level instrument (Log "N") and a micro-microammeter provides usable reactor power information in the logarithmic range $10^{-3}$% - 300% of the normal power of 500 KW.

The startup range power level instrument (log count rate) covers the neutron flux range from the source level (∿ 1cps) to $10^6$ cps on a logarithmic scale. It enables the operator to start the reactor safely from a shutdown condition, and to bring the power to a level that can be measured by the Log N instrument.

Coolant flow rate and coolant inlet and outlet temperature instruments allow the operator to calculate reactor power and calibrate the neutron flux channel in terms of power.

Rod position indicators show the operator the relative positions of control rods, and enable rod reactivity calibrations to be made.

3.2.3 Safety Channels - The minimum number and type of channels providing automatic action that are required for reactor operation are given in Table I, Instrument Arrays, Section B.

Bases - The power level scram provides redundant automatic protective action to prevent exceeding the safety limit on reactor power.

The period scram limits the rate of increase in reactor power to values that are controllable without reaching excessive power levels or temperature.

The high water level in core tanks scram prevents water from overflowing the core tanks and wetting the surrounding graphite reflector.

TABLE I  INSTRUMENT ARRAYS

| A. Measuring Channel | No. Operable | Detector | Function/Action | Bypass Provisions |
|---|---|---|---|---|
| Power Level (power range) | A minimum of 2 channels operating on scale above the startup range | UCIC | - | None |
| Power Level (int. range) | | CIC | - | None |
| Power Level (startup range) | | Fission chamber | - | None |
| Power Level μμ ammeter | | CIC | - | None |
| Coolant Flow | 1 | Ultrasonic or flow orifice, D/P detector | - | None |
| Coolant Inlet Temp. | 1 | RTD | - | None |
| Coolant Outlet Temp. | 1 | RTD | - | None |
| Rod Position | 1/rod | Potentiometer | - | None |
| **B. Safety Channel** | | | | |
| Power Level (power range) | 2 | UCIC | Scram @ $\leq$ 125% | None |
| Power Level (int. range) | 1 | CIC | Scram @ $\geq$ 5 sec. period<br>Alarm @ $\geq$ 10 sec. period | None<br>None |
| Moderator Level in core tanks | 1 | Float switch | Scram @ $\leq$ 1" from top of core tanks | None |
| Manual Button | 4 | Contact in switch | | None |
| Neutron Count Rate | 1 | Fission chamber | Inhibit control rods & closing of dump valve @ $\leq$ 2 CPS. If dump valve closed, inhibit control rods | Interlock on dump valve closing may be bypassed by a Senior Reactor Operator |
| Reactor Room Ventilation Fan | 1 | Relay | Same as neutron count rate | None |
| Coolant Flow | 1 | Ultrasonic or flow orifice, D/P DETECTOR | Same as neutron count rate, if flow $\leq$ 15 GPM alarm | May be bypassed for a specific experiment with approval of Reactor Safety Committee |
| Coolant inlet temp. | 1 | RTD | Same as neutron count rate if temp. $\leq$ 70°F | Same as above |
| Dump Valve | 1 | Limit switch | Inhibit control rods if not fully closed | None |

TABLE I INSTRUMENT ARRAYS (Cont'd) pg. 2

| | No. Operable | Detector | Function/Action | Bypass Provision |
|---|---|---|---|---|
| B. Safety Channel | | | | |
| Core tank moderator level | 1 | Pressure switch | Inhibit control rods if if less than overflow pipe level | May be bypassed by a senior reactor operator for control rod tests with moderator coolant dumped |
| Safety Rod No. 1 | 1 | Limit switch | Inhibit Safety Rod No.2, Shim and Reg. rods if not fully withdrawn | May be bypassed by Senior Reactor Operator for control rod tests, such as drop tests |
| Safety Rod No. 2 | 1 | Limit switch | Inhibit Shim and reg. rod if not fully withdrawn | Same as above |
| Regulating Rod | 2 | Limit switch | Inhibit automatic and revert to manual operation when reg rod at upper or lower limit, alarm | None |
| Automatic Servo Control | 1 | Servo-Control Circuit | Inhibit automatic and revert to manual reg. rod control where $\geq \pm$ 15% deviation | None |
| Coolant outlet temp. | 1 | RTD | Alarm @ $\leq$ 180° F | None |
| Shield tank water level | 1 | Pressure switch | Alarm @ $\geq$ 2' below top of shield tank | May be bypassed by a Senior Reactor Operator for a specific experiment. |
| C. Radiation Monitoring | | | | |
| Fixed Area | 2 | GM tube | Detect radiation (γ) in key locations; alarm @ $\leq$15 MR/Hr, 1 monitor initiates building evacuation alarm and inhibits reactor room ventilation | May be bypassed for hot fuel transfers and specific experiments |

TABLE I INSTRUMENT ARRAYS (Cont'd) pg. 3

| C. Radiation Monitoring | No. Operable | Detector | Function/Action | Bypass Provisions |
|---|---|---|---|---|
| Primary Coolant | 1 | GM tube | Detect radiation (γ) in primary coolant outlet, alarm @ ≤ 1000 MR/HR | None |
| Exhaust Stack Radiation | 1 | GM tube | Detect radiation (γ) in key locations; alarm @ ≤ 15 MR/Hr, Initiate building evacuation alarm, inhibit reactor room ventilation | None |
| Exhaust Stack Fission Products | 1 | Scintillation Counter | Detect fission products in stack; alarms | None |

3.3 Radiation Monitoring System

3.3.1 The minimum acceptable monitoring instrumentation required for reactor operation is given in Table I, Instrument Arrays, Section C.

3.4 Limitations on Experiments

3.4.1 Experiments

A. Applicability - This specification applies to those experiments installed in the reactor and its experiment facilities.

B. Objective - The objective is to prevent damage to the reactor or excessive release of radioactive material in the event of an experiment malfunction.

C. Specification - Experiments installed in the reactor shall meet the following conditions:

1. The reactivity worth of any single movable experiment in the reactor core or experimental facilities shall not exceed 0.3% $\Delta K/K$.

2. No experiment shall be installed in the reactor in such a manner that:

   a. It could affect operation of the nuclear instrumentation system monitors,

   b. Failure of the experiment could interfere with the insertion of a control rod, or

   c. Failure of the experiment could credibly result in fuel element damage.

4. No experiment shall be performed involving materials which could:

   a. credibly contaminate the reactor coolant system causing corrosive action on reactor components or experiments

   b. Cause excessive production of airborne radioactivity, or

   c. Produce a violent chemical reaction.

5. Explosive materials such as gunpowder, dynamite, TNT, or nitroglycerine shall not be irradiated in the reactor irradiation facilities.

6. Each class of experiment irradiated in the reactor must have been previously reviewed and approved by the Reactor Safety Committee.

D. Bases

1. Movable experiments are defined as experiments which could conceivably be expelled during operation, accidentally or intentionally moved during operation, or flooded or collapsed in such a way as to affect reactivity. The LSSS for reactor period will not be exceeded allowing time for operator corrective action. Even without operator action the LSSS for reactor power will prevent any safety limit from being exceeded.

2. Ensures that no physical or nuclear interference with the safe operation of the reactor will occur.

3. This requirement guards against release of activation products in the primary coolant or chemical interaction with core components.

4. Explosive materials will not be handled since irradiation of these materials has not been evaluated.

5. Ensures that all experiments are evaluated by an independent group knowledgeable in the appropriate fields.

3.4.2 Fueled Experiments

A. Applicability - These specifications apply to experiments containing fissile material that are installed in the reactor or its experiment facilities.

B. Objectives - The objective is to prevent damage to the reactor, prevent excessive release of fission products in the event of an experiment failure, and also to ensure that safety limits are not exceeded.

C. Specifications - Experiments containing fissile material shall meet the following conditions:

1. Experiments applicable to this section shall conform to the specifications listed in Sect. 3.4.1.

2. The inventory of fissile and/or special nuclear material being irradiated in the shield tank experimental facility shall be limited to (reserved) grams.

3. The inventory of fissile and/or special nuclear material contained in one capsule and inserted in the reactor through a Rabbit Transfer System shall be limited to (reserved) grams.

4. The inventory of fissile and/or special nuclear material being irradiated in the south, north, vertical beam port, thermal column or offset stringers shall be limited to (reserved) grams.

D. Basis - These specification place limits on the fission product inventory such that a failure and hypothetical release of all contained fission products will not result in excessive exposure to personnel in restricted and unrestricted areas.

The detailed analyses that form the basis of these specifications are given in ref. 1.

3.5 Fuel

A. Applicability - This specification applies to the condition of the fuel elements present in the core.

B. Objective - To avoid excessive release of fission products.

C. Specification - Fuel elements exhibiting release of fission products due to cladding rupture shall, upon positive identification, be removed from the core. An increase in the normal fission product release by a factor of 10 shall constitute initial evidence of cladding rupture and require identification of the cause.

D. Basis - Release of fission products from the compact fuel elements used in this reactor (sect. 5.1) due to a localized cladding defect is confined to the defect locality. A relatively small defect thus cannot release large quantities of fission products. There is a normal small and variable background of fission product release due to trace amount of uranium on the fuel plates. It is thus safe to specify a recognizable and substantial increase in this background as a possible indication of cladding rupture. If the rupture were extensive, there would be no doubt at all of this condition.

3.6 Primary Coolant Water Quality

A. Applicability - This specification applies to primary water in contact with fuel elements.

B. Objective- To minimize corrosion of the aluminum cladding of fuel plates and activation of dissolved materials.

C. Specification - The primary coolant specific resistance shall not be less than 700,000 ohm - cm.

D. Basis - No excessive corrosion has been evident on fuel plate cladding during twenty years of operation by maintaining this specification.

3.7 Radioactive Releases

3.7.1 Airborne Stack Release Limit

A. Applicability - This specification applies to Argon - 41 released to unrestricted areas during normal reactor operation.

B. Objective - To prevent exceeding limits as specified in 10 CFR 20 for unrestricted areas.

C. Specification - The maximum release rate from the ventilation stack shall not exceed $1 \times 10^{-4}$ Ci/sec. and a total of 315 Ci/yr.

D. Basis

1. The total body dose to any individual in unrestricted areas due to Argon-41 released in gaseous effluents from the site shall be limited to the following expression:

$$3.17 \times 10^{4}\, M(\overline{\chi/Q})\tilde{Q}(D.F.) \leq 5 \text{mrem/yr.}$$

where $\tilde{Q}$ = The release of the noble gas Argon-41 (measured concentration x flow rate) in Ci. Releases shall be cumulative over the calendar year.

$M$ = The total body dose factor due to gamma emission for Argon-41, ($9.3 \times 10^{-3}$ mrad-m$^3$/pCi-yr.), Ref. 3

$\overline{\chi/Q}$ = Shall be calculated from measured values of Argon concentration sampled at the environmental monitoring station located at the exhaust fan outlet, in sec./m$^3$.

D.F. = Dilution factor, shall be calculated from measured values comparing the ratio of air flow rate at the exhaust fan system intake to the air flow rate at the discharge of the booster fan.

$3.17 \times 10^{4}$ = A conversion factor in, pCi-yr./Ci-sec.

3.7.2 Liquid Effluent Releases

A. Liquid waste from all radioactive operations shall be collected in holding containers.

B. Before release from the holding containers, the liquid waste shall be sampled and the activity level measured.

C. Liquid waste shall not be released from the site unless its activity concentration, including dilution with non-radioactive waste water, is below that specified in 10 CFR Part 20, appendix B, Table II, Column 2.

D. Records of and reports on liquid radioactive effluent releases shall be as specified in Section 6 of these Technical Specifications.

## 4.0 Surveillance Requirements

### 4.1 General

The requirements listed below generally prescribe tests or inspections to verify periodically that the performance of required systems is in accordance with specifications given above in sections 2 and 3. In all instances where the specified frequency is annual, the interval between tests is not to exceed 14 months; when semiannual, the interval should not exceed 7 months; when quarterly the interval should not exceed 14 weeks.

### 4.2 Safety Channel Calibration

A channel calibration of each safety channel shall be performed annually (see section 3.2.3).

### 4.3 Reactivity Surveillance

4.3.1 The reactivity worth of each control rod and the shutdown margin shall be determined whenever operation requires a reevaluation of core physics parameters, or annually, whichever occurs first. The rod worth will be determined using the reactivity period or rod-drop methods.

4.3.2 The reactivity worth of a new experiment shall be estimated and then determined experimentally at low power, before conducting the experiment.

### 4.4 Control and Safety System Surveillance

4.4.1 The scram time shall be measured quarterly. If a control rod mechanism is removed from the core temporarily, or if a new rod is installed, its scram time shall be measured before reactor operation.

4.4.2 A channel test of the power, intermediate and source range instruments shall be performed prior to each reactor run unless the preceding shutdown period is 8 hours or less. A channel test before startup is, however, required on any channel receiving maintanance during the shutdown period.

4.4.3 A channel check of each measuring channel in the reactor safety system shall be performed prior to each reactor run.

4.4.4 A visual inspection of control rod mechanisms shall be performed annually.

### 4.5 Radiation Monitoring System

4.5.1 The stack, area and fission product monitors shall be calibrated annually.

4.5.2 The stack, area and fission product monitors shall receive a channel check prior to each reactor run.

4.6 Experiment Limits

Surveillance to assure that experiments meet the requirements of section 3.4 shall be conducted prior to inserting each experiment into the reactor.

4.7 Reactor Fuel

One reactor fuel element shall be visually inspected annually for any pitting, blistering or corrosion on the fuel cladding.

4.8 Primary Water

The resistivity of primary water shall be checked prior to each reactor run.

4.9 Non-Exempt SNM

The quantity of non-exempt SNM will be determined semi-annually.

## 5.0 Design Features

Those design features relevant to operation, safety and to limits that have been previously specified are described below. These features shall not be changed without appropriate review.

### 5.1 Reactor Fuel

Standard fuel elements shall be of the flat-plate type, thirteen plates to each element. Each fully-loaded fuel plate shall be approximately 26-inches long by 3-inches wide by 0.080-inch thick, which includes 0.020-inch aluminum cladding on each side, containing a nominal 22 grams of U-235 as uranium-aluminum alloy. The fuel plates shall be separated by approximately 0.36 inch and mechanically joined at the top and bottom. Half and quarter-load fuel plates and aluminum dummy plates may be used to adjust the core loading.

### 5.2 Reactor Core

Twelve fuel elements, loaded six to each core tank, shall make up a core loading.

### 5.3 Nuclear Instrumentation (power level)

Design features of the components of this system (3.2.2, 3.2.3) that are important to safety are given below.

#### 5.3.1 Power Level (power range)

For this function two independent measuring channels are provided. Each channel covers reliably the range from about 1% to 150% (of 500 KW). Each channel comprises an uncompensated boron-coated ion chamber feeding an amplifier that controls bistables (electronic switches) whose output controls solid state relays that provide DC current that flows through safety rod no. 1, no. 2 and the shim safety rod electromagnets. Each channel controls and scrams the safety rods and shim-safety rod. Each channel is fail safe. Each channel indicates power level on a panel meter allowing channel checks to be done during reactor operation.

#### 5.3.2 Power Level (Intermediate Range)

For this function two channels are provided, covering reliably the range $10^{-3}$% to 300% (of 500 KW) with a logarithmic and linear output indication on a panel meter. To cover the range under all core conditions two gamma-compensated boron-coated ion chamber are used to supply a logarithmic amplifier and a micro-microammeter. Rate of change of power information is also derived, in the form of a period, that can produce a scram in the same way as in Sect. 5.3.1.

5.3.3 Power level (startup range)

A fission chamber is used to supply pulses to a pulse amplifier and logarithmic count rate circuitry. Pulse height discrimination selects pulse amplitudes that correspond to fission events and rejects those from alpha particles. Count rate on a logarithmic scale is displayed on a panel meter.

5.3.4 Neutron Source

For obtaining the reliable neutron information necessary for startup from a cold shut-down condition, a plutonium-beryllium neutron source is provided for insertion into the core as needed. The neutron source shall provide a minimum of $10^6$ neutrons per second and shall be positioned during reactor startup in the central graphite reflector region.

5.4 Rod Control System

5.4.1 Control Rods

Four 1/8" x 7" x 7" (nominal) boral control rods, 2 safety, 1 shim and 1 regulating of the windowshade type shall be positioned in slots machined in the graphite external reflector adjacent to the outside face and near each outside corner of the core.

5.4.2 Control Rod Drives

The two safety rods and shim rod are coupled to drive shafts through electromagnets that allow release of the rods after receiving a scram signal. The maximum time for insertion of the rods following initiation of a scram signal shall not exceed 0.8 seconds. Position indicators on the control console show the extent of withdrawal for each rod and a digital readout can be switched to any one rod. To limit the rate of reactivity increase upon startup, the rod drive speeds are limited to a maximum of 7 in./min. and no more than one safety or shim rod can be withdrawn simultaneously.

The regulating rod is provided to aid in fine control and maintenance of constant power for long periods. The rod can be either manually or servo-controlled. The maximum drive speed shall be limited to 30 in./min.

## 5.5 Cooling System

### 5.5.1 Primary Cooling System

Core cooling is affected by gravity flow of demineralized water from the parallel piped reactor core tanks overflow line to a below floor level dump-holdup tank that provides a delay to allow $^{16}N$ gamma activity partially to decay. The water is then pumped back to the core tanks through the primary side of the heat exchanger where heat is transferred to a secondary cooling system.

The dump-holdup tank is vented to the building exhaust duct. The driving force for the coolant is the fixed head between the core tanks overflow line and the water level in the dump holdup tank, the latter being fixed by the total water in the system. A 6" dump-inlet line leads from the dump-holdup tank to the bottom of the core tanks. A scram condition shall open the dump valve connecting the 6" dump-inlet line to the dump-holdup tank, draining the core tanks and bypassing the core tanks in the coolant loop. Primary flow is measured by a transducer downstream of the reactor coolant pump, with indication of flow on the control console. Temperature sensors in the core inlet and outlet lines allow the core $\Delta T$ to be measured. Annunciators are provided at the control console to indicate high outlet primary coolant temperature and loss of coolant flow.

### 5.5.2 Secondary Cooling System

Reactor power transferred through the heat exchanger is dissipated to the atmosphere via a cooling tower. To minimize corrosion, chemistry control will be used. To prevent water from entering the secondary system should a tube leak occur, the static pressure in the secondary is made higher than that of the primary through the relative elevations of the two systems.

## 5.6 Radiation Monitoring System

The following areas shall be monitored for radiation:

A. East and West walls of the reactor room for general area radiation levels.

B. Radiation levels of primary coolant upon exiting the core tanks prior to entering the dump-holdup tank.

C. Exhaust duct (stack) to roof. Two detectors are provided: one to monitor general radiation levels and one to monitor for an individual fission products release.

Indicators and alarms when setpoints are exceeded are provided at the control console.

5.7 Building Evacuation Alarm

A building evacuation alarm system consisting of radiation monitors described in paragraph 5.6 and warning horns in the corridors on each floor of Robeson Hall shall be operable except during required maintenance. The stack monitor and the area monitor located on the west wall of the reactor room shall actuate an alarm at the console, sound the warning horns, and shut off the reactor room ventilation system when pre-set radiation levels are reached. The alarm shall be reset by a key operated switch.

5.8 Fuel Storage

5.8.1 Two fuel storage pits shall be located in the reactor room floor and each shall be capable of storing 16 fully-loaded fuel elements. Each fuel element shall be stored in a separate cylindrical hole in the storage pit with appropriate shielding.

5.8.2 When each pit is fully-loaded and flooded with water, the $K_{inf}$ shall not exceed a calculated value of (reserved) .

6.0 Administrative Controls

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the reactor facility shall be as a minimum the structure shown in Fig. 1. Job titles shown are for illustration and may vary. Four levels of authority are provided, as follows:

Level 1: Individual responsible for the facility license and site administration.

Level 2: Individual responsible for the reactor facility operation and management.

Level 3: Individual responsible for daily reactor operations.

Level 4: Reactor operating staff.

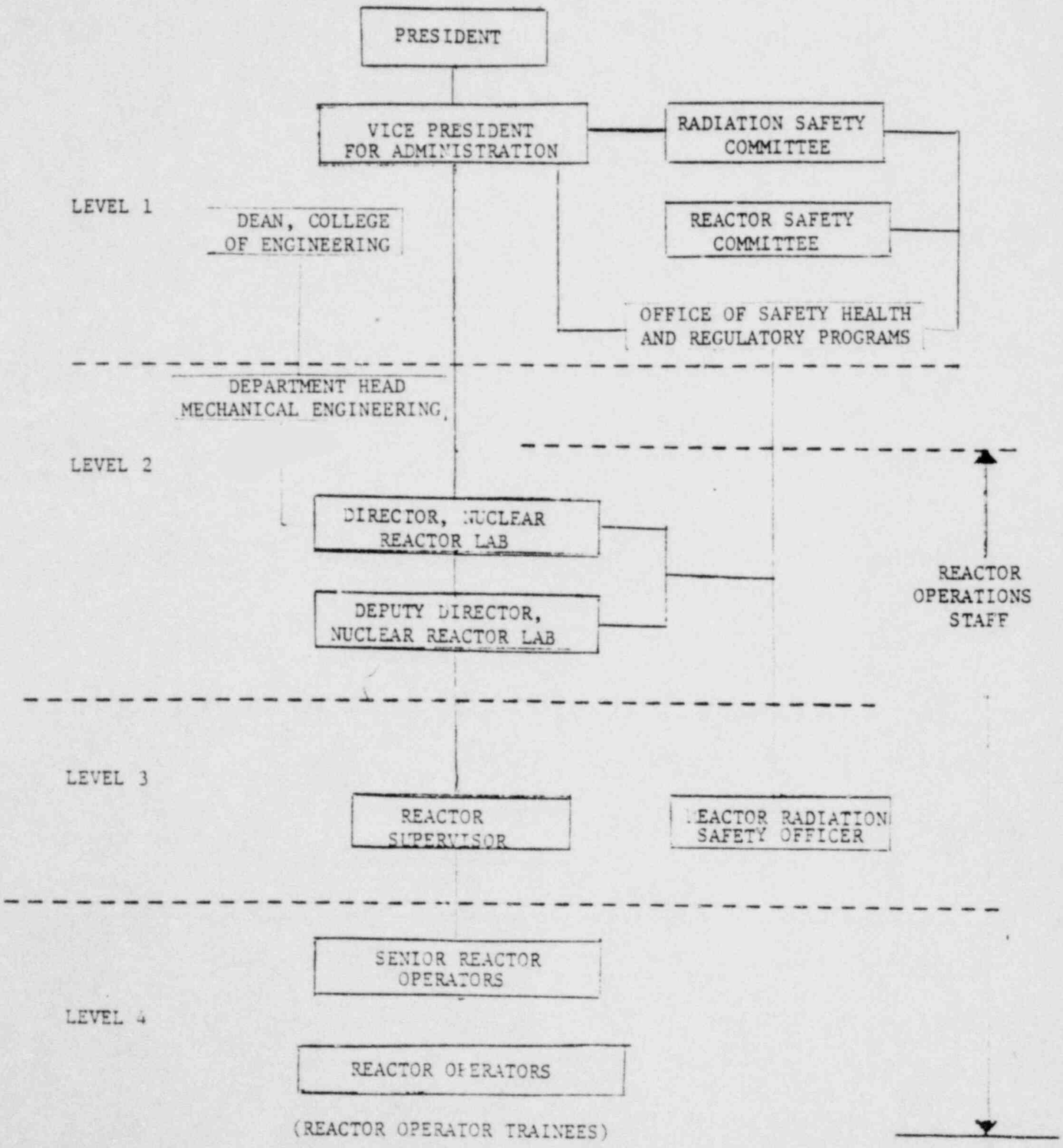
The Reactor Safety Committee shall report to Level 1. Radiation safety personnel shall report to Level 2 or higher.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be within the chain of command shown in Figure 2. Management levels in addition to having responsibility for the policies and operation of the reactor facility shall be responsible for safeguarding the public and facility personnel from undue radiation

FIGURE 1

ORGANIZATIONAL STRUCTURE



exposures and for adhering to all requirements of the operating license and technical specifications. In all instances responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon approproate qualifications.

6.1.3 Staffing

a. The minimum staffing when the reactor is not shutdown shall be:

1. A licensed Reactor Operator at the control console.

2. A reactor assistant shall be present in Robeson Hall.

3. A licensed Senior Reactor Operator shall be readily available on call.

4. Reactor Radiation Safety Officer readily available on call.

b. Events requiring the presence of a Senior Operator:

1. All fuel-element or control-rod alterations within the reactor core region;

2. Initial startup and approach to power;

3. Recovery from an unplanned or unscheduled shutdown;

4. Or a significant reduction in power.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of personnel shall meet the requirements of ref. 4 and ref. 5 and be in accordance with the requalification plan approved by the Commission.

6.1.5 Review and Audit

The independent review and audit of reactor facility operations shall be performed by the Reactor Safety Committee.

6.1.5.1 Composition and Qualifications

The Reactor Safety Committee shall be composed of a minimum of seven members. The members shall collectively provide a broad spectrum of expertise in the appropriate reactor technology. Members and alternates shall be appointed by and report to the Level 1 authority. They may include individuals from within and/or out-side the operating organization. Qualified and approved alternates may serve in the absence of regular members.

6.1.5.2 Charter and Rules

The committ e shall function under the following operating rules:

a. Meetings shall be held not less than semi-annually and more frequently as circumstances warrant consistent with effective monitoring of facility activities.

b. A quorum shall consist of half the membership, plus one.

c. Sub-groups may be appointed to review specific items.

d. Minutes shall be kept, and shall be disseminated to members and to the Level 1 authority within two months after the meeting.

e. The Committee shall appoint one or more qualified individuals to perform the Audit Function.

6.1.5.3 Review Function

The following items shall be reviewed by the review group or a subgroup therof:

a. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question.

b. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

c. Tests and experiments in accordance with section 6.3.

d. Proposed changes in technical specifications, license, or charter.

e. Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.

f. Operating abnormalities having safety significance, and audit reports.

g. Reportable occurrences listed in section 6.5.3.

6.1.5.4 Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with responsible personnel shall take place. In no case shall the individual or individuals conducting the audit be immediately responsible for the area being audited. The following items shall be audited:

a. the conformance of facility operations to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval not to exceed 18 months).

b. The retraining and requalification for the operating staff, at least once every other calendar year (interval not to exceed 30 months).

c. The results of actions taken to correct deficies occurring in reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval not to exceed 18 months).

d. The reactor facility Security Plan and implementing procedures at least once every other calendar year (interval not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Level 2 authority. A written report of the findings of the audit shall be submitted to the Level 1 authority and the Reactor Safety Committee members within 90 days after the audit has been completed.

6.2 Procedures

There shall be written procedures for, and prior to, initiating any of the activities listed in this section. The procedures shall be reviewed by the Reactor Safety Committee and approved by Level 2 or designated alternates, and such reviews and approvals shall be documented. Several of the following activities may be included in a single manual or set of procedures or divided among various manuals or procedures.

a. Startup, operation, and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Routine maintenance of major componentns of systems that could have an effect on reactor safety.

d. Surveillance tests and calibrations required by the technical specifications or those that may have an effect on reactor safety.

e. Personnel radiation protection, consistent with applicable regulations.

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of the Security Plan.

h. Emergency and Abnormal Procedures.

Substantive changes to the above procedures shall be made only after documented review by the Reactor Safety Committee and approval by Level 2 or designated alternates. Minor modifications to the original procedures which do not change their original intent may be made by the Level 3 authority (Reactor Supervisor) and must be approved by the Reactor Safety Committee at the next regularly scheduled meeting.

6.3 Experiment Review and Approval

a. All new experiments or classes of experiments that do not meet the requirements of section 3.4 shall be revieweed by the Reactor Safety Committee. This review shall assure that compliance with the requirements of the license, technical specifications, and applicable regulations has been satisfied, and shall be documented.

b. Prior to review, an experiment plan or proposal shall be prepared describing the experiment including any safety considerations.

c. Review comments of the Reactor Safety Committee setting forth any conditions and/or limitations shall be documented in Committee minutes and submitted to Level 2.

d. All new experiments or classes of experiments shall be approved in writing by Level 2 or designated alternates prior to their initiation.

e. Substantive changes to approved experiments shall be made only after review by the Reactor Safety Committee and written approval by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by the authority. Reactor Supervisor, Reactor Radiation Safety Officer and a senior reactor operator.

f. Approved experiments shall be carried out in accordance with established approved procedures.

6.4 Required Actions

6.4.1 Action to be Taken in Case of Safety Limit Violation

a. The reactor shall be shutdown, and reactor operations shall not be resumed until authorized by the Commission.

b. The safety limit violation shall promptly be reported to the Level 1 authority or designated alternates.

c. The safety limit violation shall be reported to the Commission in accordance with section 6.5.3.

d. A safety limit violation report shall be prepared. The report shall describe the following:

1. Applicable circumstances leading to the violation.

2. Effect of the violation upon reactor facility components, systems, or structures.

3. Corrective action to be taken to prevent recurrence.

The report shall be reviewed by the Reactor Safety Committee. A follow-up report describing extant activities shall be submitted to the Commission when authorization is sought to resume operation of the reactor.

6.4.2 Action to be taken in the event of an occurrence as defined in section 6.5.3, a. 1 through 3:

a. Corrective action shall be taken to return conditions to normal; otherwise, the reactor shall be shut down and reactor operation shall not be resumed unless authorized by the Level 2 authority or designated alternates.

b. All such occurrences shall be promptly reported to the Level 2 authority or designated alternates.

c. All such occurrences where applicable shall be reported to the Commission in accordance with section 6.5.3.

d. All such occurrences including action taken to prevent or reduce the probability of a recurrence shall be reviewed by the Reactor Safety Committee.

6.5 Reports

In addition to the requirements of applicable regulations, reports shall be made to the Commission as follows:

6.5.1 Startup Reports

Three months after completion of requisite startup and power-escalation testing of the reactor, or nine months after criticality, a written report shall be submitted to the Commission. The report shall include a summary of the following:

a. Description of measured values of operating conditions or characteristics obtained and comparison of these values with design predictions or specifications.

b. Descriptions of major corrective actions taken to obtain satisfactory operation.

c. Re-evaluation of safety analyses where measured values indicate substantial variance from those values used in the Safety Analysis Report.

6.5.2 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to the appropriate NRC Regional Office with a copy to the Director of Inspection & Enforcement within 3 months following the end of each prescribed year. Each annual operating report shall include the following information:

a. A narrative summary of reactor operating experience including the energy produced by the reactor.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. Tabulation of major changes in the reactor facility procedures, and new tests and/or experiments significantly different from those performend previously and which are not described in the Safety Analysis Report, including conclusions that no unreviewed safety questions were involved.

e. A summary of the nature and amount of radioactive effluents from the reactor facility released or discharged to the environs. The summary shall include where practicable an estimate of individual radionuclides present in the effluent if the estimated average release after dilution or diffusion is greater than 25% of the concentration allowed or recommended.

6.5.3 Special Reports (Reportable Occurrences)

a. There shall be a report not later than the following working day by telephone and confirmed by telegraph or similar conveyance to the Commission to be followed by a written report within 30 days of any of the following:

1. Release of radioactivity from the reactor above allowed limits, as provided by section 3.7 of this specification.

2. Violation of Safety Limits

3. Any of the following:

   a. Operation with actual safety-system settings less conservative than the limiting safety-system settings specified in the Technical Specifications.

   b. Operation in violation of Limiting Conditions for Operation established in the Technical Specifications.

   c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during tests or periods of reactor shutdowns.

      (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

   d. Abnormal and significant degradation in reactor fuel, and/or cladding or coolant boundary, where applicable which could result in exceeding prescribed radiation-exposure limits of personnel and/or environment.

   e. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused an unsafe condition with regard to reactor operations.

b. A written report within 30 days to the Commission of:

   1. Permanent changes in the facility organization structure

   2. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.6 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single, or multiple records, or a combination thereof.

6.6.1 Records to be retained for a period of at least two years or for the life of the component involved whichever is smaller.

a. Normal reactor facility operations (including scheduled and unscheduled shutdowns). Note: Supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least two years.

b. Principal maintenance operations.

c. Surveillance activities required by the Technical Specifications.

d. Reactor facility radiation and contamination surveys where required by applicable regulations.

e. Experiments performed with the reactor.

f. Approved changes in operating procedures.

g. Sealed Source leak test results.

6.6.2 Records to be retained for at least one requalification cycle or for the length of employment of the individual whichever is smaller:

a. Retraining and requalification of licensed operations personnel. However, records of the most recent complete cycle shall be maintained at all times the individual is employed.

6.6.3 Records to be retained for at least the lifetime of the reactor facility: (Note: Annual reports may be used where applicable as records in this section.)

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

b. Off-site environmental-monitoring surveys required by the Technical Specifications.

c. Radiation exposure for all personnel monitored.

d. Updated drawings of the reactor facility.

e. Reportable occurrences.

f. Special Nuclear Materials (SNM) inventories, receipts, and shipments.

g. Records of meeting and audit reports of the Reactor Safety Committee.

## 7.0 References


1. Safety Analysis Report for the Research and Training Reactor at VPI&SU (November, 1979).

2. The Power Excursion Safety Analysis of the VPI&SU Reactor at 500KW Model (August, 1976), by: K. D. Tuley.

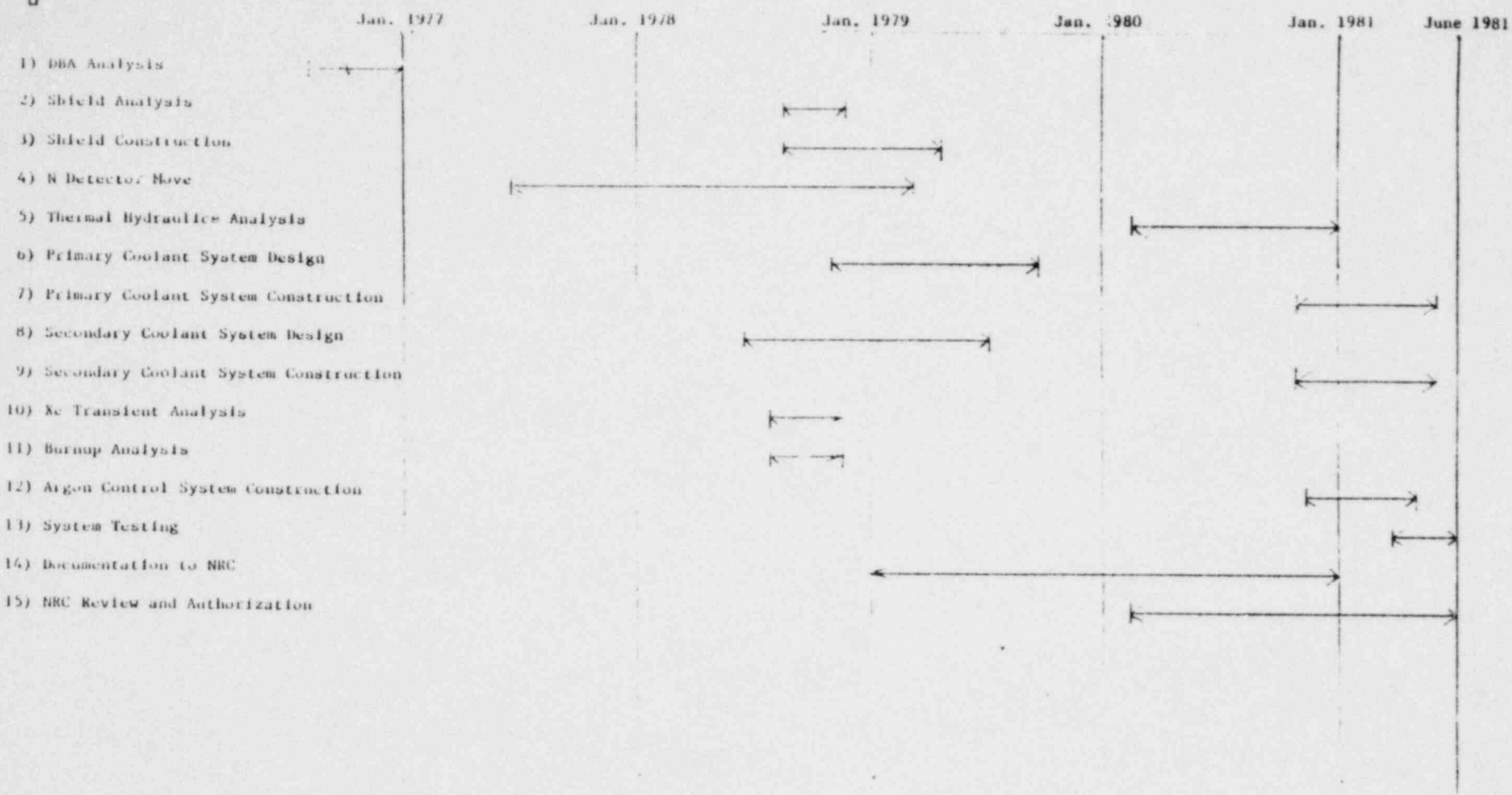
3. Regulatory Guide 1.109, Rev. 1, Table B-1.

4. ANS-15.4 Selection and Training of Personnel for Research Reactors.

5. Title 10-Chapter 1, Code of Federal Regulations, Part 55.

Contents

FIG. 1. SCHEDULE FOR POWER UPGRADE PROGRAM

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