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APPENDIX A

FACILITY LICENSE NO. R-119

PROPOSED

TECHNICAL SPECIFICATIONS

FOR THE

WESTINGHOUSE NUCLEAR TRAINING REACTOR

DOCKET NO. 50-87

0041N

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1.0 DEFINITIONS

The following terms are defined to aid in the uniform interpretation of the specifications.

1.1 Administrative Controls - the provisions related to organization and management, personnel requirements, procedures, record keeping, review and audit, and reporting that are considered necessary to assure operation of the facility in a safe manner.

1.2 Auxiliary Reactor Trip - consists of the dumping of the moderator-shield water through the ten-inch dump valve. The trip is initiated manually.

1.3 Channel Calibration - is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a Channel Test.

1.4 Channel Check - a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5 Channel Test - the introduction of a signal into the channel for verification that it is operable.

1.6 Excess Reactivity - is that amount of reactivity that would exist if all control rods (control, regulating, etc.) were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$).

1.7 Experiment - any operation, hardware, or target (excluding devices such as detectors, foils, etc.), which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, or any core loading which is not the normal core loading.

1.8 Measuring Channel - an arrangement of components and modules as required to measure the value of a process variable. The output of the measuring channel is the measured value of the process variable and may be considered the true value within the accuracy of the measuring channel.

1.9 Moderator-Shield Water - the water that is placed in the reactor tank.

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

1.11 Neutron Source - any neutron-emitting radioactive material, other than the reactor fuel, which is positioned in or near the reactor core to provide an external source of neutrons.

1.12 Operable - means a component or system is capable of performing its intended function. (Operating means it is performing its function.)

1.13 Readily Available on Call - normally means within a 20-mile radius of the facility and that the operator-on-duty knows the location and telephone number of the senior operator on duty.

1.14 Reactivity Worth of an Experiment - the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.15 Reactor Operational - means the reactor is not secured.

1.16 Reactor Safety System - those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.17 Reactor Secured - means that all control rods are in their down positions and the key is removed from the console lock, or when there is no fuel in the core.

1.18 Reactor Trip - consists of a gravity drop of control rods caused by the interruption of the electrical power to the magnet carriages. The trip can be initiated automatically by the safety system, manually by the manual reactor trip and manually by disconnecting the facility electrical power.

1.19 Safety Channel - an arrangement of components and modules as required to generate a single protective action signal when required by a facility condition.

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

1.21 Unsecured Experiment - any experiment that is not a secured experiment.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the reactor power level limitation and the annual integrated thermal power limitation.

Objective

The purpose of this specification is to establish the upper safety limit on power level for the reactor and the integrated thermal power produced annually.

Specifications

1. Although the license maximum operating power is 10 kilowatts (thermal), the safety limit established is 20 kilowatts (thermal).

Bases

The maximum licensed operating power level of the reactor is 10 kWt. In general, the operating power level is kept as low as possible, consistent with the training operation requirements, and normally is less than one hundred watts.

Reactor power levels are normally determined by the activation of gold foils. Errors considered to be possible in the use of this method are 20 percent in the absolute neutron flux measurements by activation methods and a maximum error of 25 percent in the nonlinearity of the monitoring instruments. These errors correspond to the Limiting Safety System Settings of 12 Kwt established in Section 2.2

Calculations and measurements have shown that during steady-state operation of 20 kWt the average moderator temperature increase is less than 20°F, and therefore, the clad and fuel temperatures remain significantly below their failure point.

A maximum power level of 20 Kwt provides adequate flexibility for the performance of training and irradiation operations and at the same time appropriately limit the quantity of radioactive material available for release.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the settings for instruments monitoring parameters associated with reactor safety limits.

Objective

The purpose of this specification is to assure protective action before safety limits are exceeded.

Specifications

The limiting safety system settings shall be as follows:

Maximum Power Level	12 kWt
Minimum Flux Level	2.5 neutron/cm^2-s
Minimum Period	3 seconds
Maximum Gamma-ray Exposure Rate (above the water shield)	R/hr value experimentally determined for 12 kWt power level operation with normal moderator-shield water height

Bases

The maximum power level trip setting of 12 kWt is established by estimating an error of 20 percent in the absolute neutron flux measurement by activation methods and a maximum error of 25 percent in the nonlinearity of monitoring instruments. The safety margin prevailing between the safety limit and the limiting safety system setting is adequate to allow for these errors.

The minimum flux level in the core has been established to prevent a source-out startup. A minimum interlock setpoint of 2 cps on a source level detector shall be used to assure this minimum neutron flux level.

The minimum 3-second period is specified so that there is sufficient time for the automatic safety system to respond before the power level safety limit is exceeded. The transient would be terminated in less than 200 milliseconds after the reactor trip.

The maximum gamma-ray exposure rate setting is established to correspond with the exposure rate above the top of the reactor sheild that would occur during reactor operation at 12 kWt with the moderator-shield water level at the normal (5 foot) water height. This setting assures that increasing radiation levels in the vicinity of the reactor room will be detected before they become excessive when the reactor is operated at moderator-shield water heights other than the normal level. An approximate value for this setting is estimated to be 500 mR/hr.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Control and Safety Systems

Applicability

These specifications apply to all methods of changing core reactivity available to the reactor operator.

Objective

The purpose of these specifications is to assure that an adequate shutdown method is available and that positive reactivity insertion rates are within those analyzed in the Safety Analysis Report.

Specifications

1. There shall be a minimum of five operable control rods. The maximum excess reactivity that can be loaded into the core including experiments shall be such that the reactor shall be subcritical by a margin greater than 1$ with the control rod having the largest reactivity worth fully withdrawn and with failure of the experiment.

2. Maximum excess reactivity loaded into the core will be 8.55$ (6.84 percent).

3. The maximum control rod and moderator-shield water reactivity insertion rate shall be less than 0.10$/s when keff is less than 0.99 and less than 0.035$/s when keff is greater than 0.99.

4. The total control rod drop time for each control rod from its full-out position to its full-in position shall be less than or equal to 1.2 seconds. This time shall include a maximum magnet carriage release time of 0.125 second.

5. Negative reactivity shall be available in operable cocked control rods prior to adding the moderator-shield water to the reactor. At least 1$ of negative reactivity shall be available when core loadings, capable of becoming critical, are to be filled with the moderator-shield water.

6. The auxiliary reactor trip (moderator-shield water dump) shall add negative reactivity within one minute of its activation. Remote auxiliary reactor trip controls shall be available at the console and in the reactor room.

7. The normal moderator-shield water level shall be established at a minimum of 5 feet above the top of the core. Reactor operations at water levels below this normal level shall be permitted only when the operating power is lowered accordingly. (Refer to Specification 2.2, "Maximum Gamma-ray Exposure Rate".) Moderator level and adjustments near criticality shall be made only after first establishing criticality by control rod manipulation.

8. A manual reactor trip shall be included in the reactor safety system. Controls for the reactor trip shall be available at the console and in the reactor room.

9. A manual electric switch shall be provided in the facility for the purpose of disconnecting the electrical power of the facility and causing a reactor trip.

10. The minimum safety system channels that shall be operating during reactor operation are listed in Table 1.

11. The interlocks that shall be operable during reactor operations are listed in Table 2.

Bases

A minimum number of five control rods and a maximum excess reactivity are specified to assure that there is adequate shutdown capability even for the stuck control rod condition. The only authorized core configurations are centered in the core structure utilizing all five control rods. The minimum obtainable critical fuel loading in the reactor, in the best right cylindrical configuration centered in the core structure, consists of sixteen fuel elements and five control rods surrounded with a single ring of twenty graphite reflector rods. With control rods withdrawn, there are effectively twenty-one fuel elements (considering control rod fuel followers as fuel elements) in the minimum critical loading.

The core loading consisting of nineteen fuel elements and five control rods in a hexagonal configuration, surrounded by a single ring of twenty graphite reflector rods, centered in the core structure represents the normal loading. The normal core loading is such that the maximum excess reactivity of the reactor will not be greater than 5.50$.

The maximum excess reactivity that can be loaded into the reactor core, including that associated with experiments or possible experiment failure, is limited in Section 4.0 such that the reactor can be made subcritical by at least 1.00$ with the highest worth rod stuck fully out of the core.

For example the current normal fuel loading has a 4.4 percent Δk/k (5.50$) of excess reactivity. The shutdown margin with all control rods fully inserted is 17.7 percent Δk/k (22.13$). Their sum yields a total control rod worth of 22.1 percent Δk/k (27.63$). The highest worth rod is worth 10.06 percent Δk/k (12.58$). Thus, under these conditions, the shutdown margin with the highest worth rod fully withdrawn is 7.64 percent Δk/k (9.56$). Therefore, with the current configuration a maximum excess reactivity of 6.84 percent Δk/k (8.556) will still guarantee the reactor will be made subcritical by greater than 1$ with the most reactive rod stuck fully out of the core.

The maximum reactivity insertion rates, far from and near criticality, are specified to assure that the reactivity addition rate is less than that analyzed in the maximum credible accident (MCA). The maximum control rod withdrawal rate and the moderator-shield water addition rate are controlled by these limitations.

The insertion time of less than 1.2 seconds for each control rod from its fully withdrawn position is specified to assure that the insertion time does not exceed that assumed when establishing the minimum period specified in specification 2.2 as a limiting safety system setting.

The required control rod withdrawal prior to adding the moderator-shield water is specified to assure that reactor trip will have the capability of adding negative reactivity during reactor startup.

The auxiliary reactor trip is specified to assure that there is a secondary mode of shutdown available during reactor operations. The requirement that negative reactivity be introduced in less than one minute following activation of the trip is established to limit the consequences of a potential power transient. By including a remote auxiliary reactor trip control in the reactor room, the trip may be activated readily by individuals in the room under emergency conditions.

The normal moderator-shield water level of the reactor is established at a minimum of five feet above the top of the core to assure that an adequate shield is provided at the maximum power level of the reactor. When reactor operations require a lower moderator-shield water height (down to a one foot level), the operating power must be lowered accordingly so that the gamma-ray exposure rate limit setting is not exceeded over the core. When moderator water level reactivity control is to be utilized (water level below one foot from the top of the core), the gamma-ray exposure rate limit setting is lowered accordingly - to approximately 1/10 of its maximum permissible value - to further reduce the possibility of operating with a high neutron and gamma radiation field in the vicinity of the reactor room. Controlling the reactor by moderator level near criticality is permitted only when the reactor is first made critical by control rod movements. This assures that the control rod is the primary mode of reactivity control in a critical reactor.

A manual reactor trip assures that a reactor trip can be readily initiated by an operator at his demand. Including a manual reactor trip control in the reactor room assures that the trip may be activated readily by individuals in the reactor room under emergency conditions.

By providing a method of disconnecting the facility electrical power, an additional mode is established to manually cause a reactor trip.

The safety system channels listed in Table 1 provide a high degree of redundancy to assure that human or mechanical failures will not endanger the reactor facility of the general public.

The interlock system listed in Table 2 assures that only authorized personnel can operate the reactor, that the proper sequence of operations is performed, that no one can accidentally enter the reactor room when the reactor is operating, and that the reactor room is entered with proper conditions prevailing when the master console key is on.

3.2 Reactor Parameters

Applicability

These specifications apply to core nuclear parameters and moderator-shield water physical parameters.

Objective

The purpose of the specifications on reactivity coefficients is to assure that the inherent reactivity feedback mechanisms of the water moderator are safe. The purpose of the specification on purity of the water moderator is to assure adequate corrosion control in a room temperature, open air aluminum-water system.

Specifications

1. At temperatures greater than 80°F, the temperature coefficient of reactivity shall be negative.

2. The moderator-shield water shall have a pH within the range of 4.5-8, inclusive, and a resistivity greater than 200,000 ohm-cm when averaged over a two-month period.

Bases

The minimum absolute value of the temperature coefficient of reactivity is specified to assure that an adequate inherent negative reactivity effect takes place when the reactor temperature increases above the value where the coefficient becomes negative. At lower temperatures, where the coefficient together with the slow rate at which controlled temperature increases may be effected provide assurance that positive reactivity insertion due to controlled temperature increases will be small enough and slow enough as to be safely controllable. Uncontrolled heatups will quickly raise the moderator temperature to the range where the coefficient is negative while providing only a negligible positive reactivity effect.

The moderator-shield water quality is specified to assure adequate corrosion control in a room temperature, open air, aluminum-water system. This corrosion is a long-term reaction. Based on years of experience, a two-month period for averaging has proven adequate to avoid quality degradation which would result in appreciable corrosion.

3.3 Radiation Monitoring

Applicability

These specifications apply to the minimum radiation monitoring requirements for reactor operations.

Objective

The purpose of these specifications is to assure that adequate monitoring is available to preclude undetected radiation hazards or uncontrolled releases of radioactive material.

Specifications

1. The minimum radiation monitoring systems for reactor operation shall include:

   A. A critical detector system which monitors the main fuel storage area and also functions as an area monitor. This system shall have an audible alarm in the reactor room.

   B. An area gamma monitor in the console room with an audible alarm.

2. Instruments to permit the periodic sampling and measuring of radioactivity in the air and the moderator-shield water shall be provided.

3. Portable detection and survey instruments shall be provided.

Bases

The continuous monitoring of radiation levels in the reactor room and the console room assures the warning of the existence of any abnormally high radiation levels. The availability of instruments to measure the amount of radioactivity in the air and moderator-shield water will assist in monitoring fuel clad integrity and assures continued compliance with the requirements of 10 CFR Part 20. The availability of the required portable monitors provides assurance that personnel will be able to monitor potential radiation fields before an area is entered.

4.0 EXPERIMENTS

Availability

These specifications apply to all experiments placed in the reactor tank.

Objective

The objective of these specifications is to define a set of criteria for experiments to assure the safety of the reactor and personnel.

Specifications

1. No experiment shall be performed until a written program, which has been developed in sufficient detail to permit good understanding of the safety aspects, is reviewed by the Reactor Safeguards Committee (RSC) and approved by the Facility Manager.

2. No experiments shall be conducted if the associated experimental equipment could interfere with the control rod functions or could adversely affect the nuclear instrumentation.

3. The maximum reactivity change for withdrawal and insertion of movable experiments shall be 0.25$.

4. The maximum reactivity worth of any individual unsecured experiments shall be limited such that the failure of any experiment or associated equipment will not result in a positive reactivity addition greater than 0.80$. In addition the reactivity worth of all unsecured experiments is limited such that a common mode failure of all such experiments and their associated equipment will not result in a positive reactivity addition greater than 0.80$.

5. Experiments shall not contain explosives or other material which may produce a violent chemical reaction and/or airborne radioactivity.

Bases

The experiments to be performed in the reactor programs are discussed in the Safety Analysis Report (SAR). The present programs are oriented almost exclusively toward fundamental reactor technology training. Other special programs may involve the use of the reactor as an irradiation facility. To assure that experiments are well planned and evaluated prior to being performed, detailed written procedures for all new experiments must be prepared, reviewed by the RSC and approved by the Facility Manager.

Since the control rods enter the core by gravity and are required by other Technical Specifications to be operable, no experiment should be allowed to interfere with their functions. To assure that specified power limits are not exceeded, the nuclear instrumentation must be capable of accurately monitoring core parameters.

All reactor experiments are reviewed and approved prior to their performance to assure that the experimental techniques and procedures are safe and proper, and the hazards from possible accidents are minimal. A maximum reactivity change is established for movable experiments to assure that the reactor controls are readily capable of controlling the reactor.

A positive reactivity addition of 0.80$ due to failure of an unsecured experiment or associated equipment would cause the reactor power to rise on a stable period greater than one second. Thus, the reactor safety systems would be able to trip the reactor before an excessive power level is reached.

Restrictions on irradiations of explosives and highly flammable materials are imposed to minimize the possibility of explosions or fires in the vicinity of the reactor.

To minimize the possibility of exposing facility personnel or the public to radioactive materials, no experiments will be performed with materials that could result in a violent chemical reaction and/or produce airborne radioactivity.

5.0 SURVEILLANCE REQUIREMENTS

5.1 Reactor Control and Safety

Applicability

These specifications apply to the surveillance of the safety and control apparatus and instrumentation of the facility.

Objective

The purpose of these specifications is to assure that the safety and control equipment is operable and meets the criteria established in the design bases.

Specifications

1. The total control rod drop time and magnet release time shall be measured semiannually to verify that the requirement of specification 3.1, item 3, is met.

2. The moderator-shield water dump time shall be measured semiannually to verfiy that the requirement of specification 3.1, item 5, is met.

3. The maximum control rod and moderator-shield water reactivity insertion rates shall be validated annually to verify that the requirements of specification 3.1, item 2, are met.

4. The following shall be performed each day prior to initial reactor operation, except when continuous reactor operations are scheduled, then they shall be performed once each day.

   A. A visual inspection of reactor components.

   B. An operability check or test of safety system channels.

   C. An operability check of the interlock system.

   D. An operability check of the radiation monitors and alarm setpoints.

5. The safety system channels shall be calibrated semiannually.

6. Fuel elements shall be visually inspected anually for signs of degradation and other physical changes.

Bases

Past performance of control rods and control rod drives, and the moderator-shield water fill-and-dump valve system have demonstrated that testing at intervals of six months is adequate to assure compliance with specification 3.1, items 3 and 5, and validation at intervals of twelve months assures compliance with item 2 of specfication 3.1.

Visual inspection of the reactor components, including the control rods, prior to operation is to assure that the components have not been damaged and that the core is in the proper condition. Since redundancy of all safety channels is provided, random failures should not jeopardize the ability of the overall system to perform its required functions. The interlock system for the reactor is designed so that its failure places the system in a safe or non-operating condition. However, to assure that failures in the safety channels and interlock system are detected as soon as possible, frequent surveillance is desirable and thus specified. The frequent checks of the area radiation monitors and their alarm points assures the availability of the system to perform its required functions.

Past experience has indicated that, in conjunction with the daily check, calibration of the safety channels at intervals of six months assures that proper accuracy is maintained.

Past experience with inspection of fuel elements has confirmed their continued integrity. Annual inspections will continue to confirm the absence of indications of physical changes.

Surveillance testing intervals shall also contain maximum intervals as set out below to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term.

a. Five years (intervals not to exceed six years)
b. Two years (intervals not to exceed two and one-half years)
c. Annual (intervals not to exceed 15 months)
d. Semiannual (intervals not to exceed seven and one-half months)
e. Quarterly (intervals not to exceed four months)
f. Monthly (intervals not to exceed six weeks)
g. Weekly (intervals not to exceed ten days)
h. Daily (must be done during the calendar day)

5.2 Reactor Parameters

Applicability

These specifications apply to the verification of control rod reactivity worths, temperature and void coefficients of reactivity, and reactor power level, which are pertinent to the reactor control and transient analysis and to water quality.

Objective

The purpose of these specifications is to assure that the analytical bases are and remain valid and that the reactor is safely operated.

Specifications

1. The following parameters shall be determined during the initial physics measurements of each new reactor core configuration or composition and validated periodically with the stated frequency:

   A. Individual control rod reactivity worths (semiannually).

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B. Temperature and void coefficients of reactivity (annually).

C. Reactor power calibration (semiannually).

2. The reactor moderator-shield water quality shall be determined monthly.

Bases

Measurements of the above core parameters are made when a new reactor configuration or composition is assembled. Whenever the core configuration or composition is altered, the core parameters are evaluated to assure that they are within the limits of these specifications and the values analyzed in the SAR. During the initial startup test period of the reactor, measurements and determinations of the core parameters will be made for all standard assemblies which are to be utilized in the reactor's operational programs. Verification of these parameters are made periodically to assure that changes have not taken place.

Past experience indicates that monthly measurements of the moderator-shield water quality are adequate to comply with the requirement of specification 3.2, item 3.

5.3 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the radiation monitoring equipment and activities of the facility.

Objective

The purpose of these specifications is to assure the continued validity of radiation protection standards in the facility.

Specifications

1. The area radiation monitors and the portable radiation survey instruments shall be calibrated semiannually.

2. The air in the reactor room shall be sampled and measured for particulate activity monthly.

3. The water in the reactor tank and dump tank shall be sampled and measured for radioactive contaminants monthly.

4. The reactor facility shall be surveyed for radioactive contamination semiannually.

Bases

Experience has demonstrated that calibration of the area radiation monitors and the portable survey instruments semiannually is adequate to assure that significant deterioration in accuracy does not occur.

The specified frequencies for monitoring radioactive contamination in the air and water in the reactor room as well as in the overall reactor facility is based on previous experience.

6.0 DESIGN FEATURES

6.1 Site

The NTR site shall be located on Commonwealth Edison property adjacent to the exclusion area for the Zion Station.

6.2 Reactor

6.2.1 Reactor Tank

The aluminum reactor tank shall have a capacity of approximately 7000 gallons of water with no core components in place. The tank nominal dimensions shall be 8 feet in diameter and 19 feet high. The reactor tank shall fit inside the dump tank in an eccentric fashion and shall hang on a floor level steel structure which lies over the dump tank. A horizontal support shall be provided near the bottom of the tank. The tank has openings for the inlet water line of the water fill system, the drain line at the bottom of the tank, and the water dump line. An annular support plate shall be provided to support the core structure.

6.2.2 Reactor Core

The aluminum reactor core structure shall be comprised of upper and lower grid plates, core shroud tubes for fuel elements, graphite reflector rods and control rod locations and tie-down rods. The two horizontal grid plates are separated by approximately 44 inches. The bottom plate is approximately 48 inches in diameter and varies in thickness from 2 inches at its edge to a nominal 6 inches in the center portion. The upper grid plate is an elongated hexagon, 1 inch thick. The distance between the flats along the two short axes is approximately 31 inches and along the long axis 34 inches. The center-to-center spacing of the position holes is 3.125 inches. There is a total of ninety-three position holes, five of which contain control rod shroud tubes, seven are experimental hole positions normally containing fuel element insert adapters and the remainder contain fuel element shroud tubes. The core structure is radially centered in the reactor tank and positioned vertically so that the top of the reactor core is approximately 11.5 feet from the floor level.

6.2.3 Standard Fuel Elements

A standard fuel element shall be composed of 3 concentric tubes of an aluminum-clad fuel alloy that are held together and positioned at each end by aluminum brackets. The fuel alloy (13 w/o U-Al, 93.5 percent U-235 enriched) has a thickness of 0.053 inch and the clad thickness is a nominal 0.036 inch. the aluminum grade used is no. 1100 (2S) or its equivalent.

Each standard element nominally contains 200 gms of U-235 (outer tube - 82.4 gms, middle tube - 66.6 gms and inner tube - 50.9 gms). The length of the fuel alloy is 36 inches and is centered lengthwise in the 42-inch fuel tube. The total fuel element length, including the top and bottom aluminum brackets, is 47-5/16 inches. The outside diameter of the outer, middle and inner tubes is 2.50 inches, 2.06 inches and 1.62 inches, respectively. The nominal water gap between the fuel tubes is 0.094 inch.

When the fuel elements are disassembled and reassembled in any combination of one or two fuel tubes, the resulting element shall be considered a special, non-standard partial fuel element.

6.2.4 Control Rods

The control rods shall consist of a cadmium neutron absorbing section, clad with aluminum and an attached standard fuel element follower. The length of the poison section of the rod is 36 inches. The maximum diameter of the rod is 2.5 inches. The aluminum-clad cadmium tube and the fuel element follower are linked by an axial stainless steel support rod. This linkage extends to and includes a shock absorber piston at its lower end, a stop cap above the piston section, and a magnet carriage armature at its upper end.

The control rods are guided through their respective core position holes by aluminum shroud or guide tubes. The shroud tubes extend above and below the core grid plates. The enclosed lower section with appropriate holes serves as a water dash pot and the upper section provides a physical stop to prevent further downward motion of the control rod. There are five control rod locatons in the core structure. The control rod in the location lying closest to the center of the structure is the safety rod and those in the four outer locations are shim rods.

6.2.5 Control Rod Drive Assembly

The drive assembly for the control rods shall consist of a magnet carriage, a vertical drive, a position indicator and suitable drive motors. Indications of magnet carriage "up", magnet carriage "down" and control rod "down" are provided. The position indicator gives an indication of the control rod or its magnet carriage position throughout its distance of travel (maximum 41 inches).

The drive assemblies shall be mounted on an overhead platform located approximately ten feet above the floor level and centered over the reactor tank.

6.2.6 Graphite Reflector Rods

A graphite reflector rod shall consist of a type G83 graphite rod, 2.625 inches in diameter, forty-eight inches long with a 0.5 inch hole bored axially the full length of the rod. A 0.5 inch aluminum support rod shall be inserted in the axially bored hole and have support spacers bolted on each end. A ball joint handling adapter shall be screwed and pinned to the top of the aluminum support rod.

The normal core loading shall consist of twenty (20) graphite reflector rods loaded in core positions immediately adjacent to and completely surrounding the core positions occupied by fuel elements and control rods arranged in a normal core configuration.

6.3 Water Handling System

The water handling system shall allow remote filling and emptying of the reactor tank. It shall provide for a water dump in the event that an auxiliary reactor trip is necessary. The filling system shall be controlled by the operator who must satisfy the sequential interlock system before adding water to the reactor tank. A sump pump shall be provided to add the moderator-shield water from the storage-dump tank into the reactor tank. Slow and fast fill rates of about 35 gpm and 120 gpm shall be possible. A nominal 10-inch valve shall be installed in the dump line and have the capability of emptying the reactor tank on demand of the operator. A valve shall be installed in the bottom drain line of the reactor tank to provide for completely emptying the reactor tank.

6.4 Fuel Storage

When not in use, the fuel elements shall be stored in the reactor room in facilities which will provide adequate isolation and radiation safety. No more than six fuel elements shall be out of the core or the approved storage areas at one time. The following facilities shall be considered approved storage areas:

1. Reactor tank storage tubes, a maximum number of six, shall be located inside the reactor tank and each shall be capable of holding not more than one fuel element. These tubes shall be located approximately nine feet below the reactor tank top and next to the reactor tank wall in a single slab array.

2. The main fuel storage area shall consist of a single slab array with each storage location in the array capable of holding not more than one fuel element. The main fuel storage area shall be located along the west interior wall of the reactor room and provide adequate shielding from stored fuel.

3. When the control rods are unloaded from the core and the fuel element followers are not removed, the rods shall be hung in the control rod storage rack in the reactor room. This rack shall have storage locations with spacing equal to or greater than 5-3/4" in a single line and shall have no more than 9 positions.

7.0 ADMINISTRATIVE CONTROLS

7.1 Organization

The Manager of the Westinghouse Nuclear Training Reactor Facility shall be responsible for the safe operation of the reactor. The Manager reports through the normal line of management as indicated in Table 3 to higher levels of management. The facility organization shall consist of the Manager, a Training Reactor Coordinator, a Reactor Lead

Engineer and a staff of reactor operators. All members of the facility operating staff shall be licensed operators or operators in training.

7.2 Personnel Requirements

7.2.1 When the reactor is not secured, the reactor console shall be under the surveillance of a licensed operator.

7.2.2 A licensed senior operator shall be "readily available on call" at all times during reactor operations. A licensed senior operator shall be present at the reactor facility during initial loading and approach to critical and power following each configuration change, recovery from an unplanned shutdown or significant reduction in power, and fuel handling and refueling. However, a licensed senior operator's presence will not be required during recovery from an unplanned shutdown or significant reduction in power when the cause has been clearly established and corrected. The identity of an a method for rapidly contacting the senior operator on duty shall be known to the operator on duty.

7.3 Review and Audit

7.3.1 The Reactor Safeguards Committee (RSC) shall include at least four scientists or engineers who are not in the line organization responsible for reactor operations (i.e., Nuclear Training Services Group - NTS) and shall represent at least one half of the Committee membership. the minimum qualifications of the RSC members with regard to nuclear experience shall be:

1. Each member must have a minimum of five years industrial experience in nuclear and related fields and must have a minimum of three years of active participation in his nuclear oriented discipline.

2. The experience and knowledge of each member must be applicable to or pertain to the Committee's responsibility to properly review the facility and its operation from the standpoint of safety.

3. Each member must be capable and willing to exercise his individual judgement in regard to all Committee reviews and decisions.

4. The "nuclear oriented discipline" of a minimum of two Committee members must lie in the areas of reactor physics and reactor operations.

7.3.2 The RSC shall meet at least once each six months. A quorum of the RSC shall consist of at least four members and at least half of those present shall be from organizations outside the line organization responsible for reactor operations.

7.3.3 The RSC shall review activities and advise the Manager of NTR and/or whatever echelon of management it feels appropriate on all matters pertaining to the safe operation of the NTR. The reviews shall cover:

1. Proposed experiments, tests and operations not described in the Safety Analysis Report.

2. Proposed changes or modificatioans to the facility not described in the Safety Analysis Report.

3. Proposed changes to the Technical Specifications.

4. Proposed normal operating procedures and emergency procedures, and proposed changes thereto.

5. Facility operation for compliance with internal rules, procedures and regulations, and with license provisions.

6. Performance of facility apparatus and equipment.

7.3.4 Recording and reporting requirements for the RSC shall include:

1. Minutes of each meeting.

2. Special reports on experiments reviewed and facility inspections, including the Committee's findings.

3. Special reports on facility radiation safety practices and records made semiannually.

4. All Committee reports and meeting minutes shall be transmitted through the line management up to and including the Manager, NTS.

7.4 Operating Procedures

7.4.1 Personnel entry into the reactor room shall be subject to the following procedures and conditions:

1. No person shall be allowed in the reactor room unless the reactor is subcritical by at least 3$.

2. No person shall be allowed in the reactor room if remote changes are being made to the reactor which may produce positive reactivity effects.

3. When personnel enter the reactor room to perform activities which may affect the reactor's condition in any way, the following requirements shall be met:

   A. An inter-communication system shall be operational providing voice communications between the reactor room and the control room.

   B. An audible signal of the reactor source multiplication shall be heard in the reactor room.

7.4.2 Approved written operating procedures shall be followed for the following items.

1. Facility security alarms.

2. Routine reactor operations.

3. Radiation safety procedures.

4. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

5. Non-routine operations and emergency situations.

6. Fuel handling, storage and changes in the core.

7.4.3 New procedures and changes in the operating procedures shall require review by the RSC and the approval of the Facility Manager.

7.4.4 Temporary changes in the operating procedures which do not change the intent of the original procedures may be made by the Facility Manager. Such changes shall be recorded in the operating records and reported to the RSC.

7.5 <u>Actions to be Taken in the Event of a Reportable Occurrence</u>

7.5.1 Reportable occurrences shall include but not necessarily be limited to the following:

1. A violation of a limiting safety system setting.

2. A violation of a limiting condition for operation.

3. An engineered safety system component malfunction or other component or system malfunction which could render the reactor safety system incapable of performing its intended safety function.

4. An uncontrolled and unanticipated change in reactivity.

5. A personnel action that may cause an unsafe condition in connection with the operation of the reactor.

7.5.2 In the event of a reportable occurrence, reactor operation shall not be resumed until the cause is known and appropriate corrective measures are taken.

7.5.3 The occurrence shall be reported to the NRC in accordance with Section 7.7.1 of the specifications.

7.5.4 A report shall be prepared which shall include an analysis of the causes of the occurrence and recommendations for action to prevent or reduce the probability of recurrence. The report shall be submitted to the RSC for review and shall be maintained as part of the facility records.

7.6 Action to be Taken in the Event a Safety Limit is Exceeded

7.6.1 If a safety limit is exceeded, the reactor shall be secured or otherwise placed in a safe condition and reactor operation shall not be resumed until authorized by the NRC.

7.6.2 An immediate report of the occurrence shall be made to the NRC in accordance with Section 7.7.1 of the specifications.

7.6.3 A complete analysis of the incident together with recommendations for preventing or reducing the probability of recurrence shall be prepared and submitted to the RSC and to the NRC when authorization to resume operation is sought.

7.7 Reporting Requirements

In addition to reports otherwise required by applicable federal regulations, the licensee shall report the following occurrances to the NRC in accordance with the requirements of 10CFR50.73d.

7.7.1 For each reportable occurrence or violation of a safety limit, the licensee shall notify, within 24 hours by telephone or telegraph, the NRC and shall submit within ten days a report in writing.

7.7.2 The licensee shall report to the NRC in writing within 30 days of any observed occurrence that would result in a significant change in the transient or accident analyses as described in the Safety Analysis Report, as ammended, or any changes in the facility organization structure.

7.7.3 The licensee shall submit in writing, to the NRC an annual operating report within 60 days after the end of each calendar year, providing the following information:

1. A narrative summary of operating experience (including experiments performed) and changes in facility design, performance characteristics and operating procedures related to reactor safety.

2. The energy generated by the reactor and the number of hours the reactor was operational.

3. The number of inadvertent reactor trips, including the reasons therefor.

4. Discussion of the major maintenance operations performed during the reporting period, including the effect, if any, on the safe operation of the reactor and the reasons for any corrective maintenance required.

5. A summary description of changes in the facility or procedures, and tests and experiments carried out under the conditions of Section 50.59 of 10 CFR 50..

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

7. A summary of reportable occurrences.

7.8 Records

Records maintained by the licensee shall include, but not necessarily be limited to, the following:

7.8.1 Reactor operating records, including power levels and periods or operation at each power level.

7.8.2 Records of inadvertent reactor trips, including reasons therefor.

7.8.3 Records of experiments, including any unusual events involved in their performance and in their handling.

7.8.4 Records of reportable occurrences.

7.8.5 Records of tests and measurements performed pursuant to the Technical Specifications.

7.8.6 Records of maintenance operations involving substitution or replacement of reactor equipment or components.

7.8.7 Records of fuel inventories and transfers.

7.8.8 Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

7.8.9 Records of facility contamination and radiation survey results.

7.8.10 Records of radiation exposures for all facility personnel and visitors.

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

Items 1 through 6 shall be retained for at least five years; items 7, 8, 9 and 11 shall be retained for the life of the facility, and item 10 records shall be retained indefinitely or until the Commission authorizes their disposal.

TABLE 1

MINIMUM SAFETY SYSTEM CHANNELS

Reactor Conditions and Ranges	Channels	Minimum Number	Functions
Source Range	Linear or Log Neutron Level	1	High Neutron Level Reactor Trip
(keff < 99)	Linear or Log Neutron Level	1	
Startup and Power	Linear or Log Neutron Level	2	High Neutron Level Reactor Trip
			(Associated with Kwt)
Range	Period	1	Period Trip
(keff $\geq$ 0.99)	Linear or Log Gamma Level	1	High Gamma Level Reactor Trip

TABLE 2

MINIMUM INTERLOCKS

INTERLOCKS	ACTION IF INTERLOCK NOT SATISFIED
Console Master Key "On"	Reactor Trip
Reactor Room Door Closed (a)	Reactor Trip and prevents control rod withdrawal and moderator insertion
Neutron Flux Up (a)	Prevents control rod withdrawal (b) and moderator insertion
Safety Rod Cocked (a)	Prevents control rod withdrawal (b) and moderator insertion
Water Level Up (a)	Prevents control rod withdrawal (b)
Reactor Room Access Key "On"	Prevents control rod withdrawal (b)
Count rate cutout (high and low)	Prevents bank control rod withdrawal (b) and "fast fill" mode of moderator insertion

(a) During maintenance checks, special operations and "console master key on" reactor room entry, these interlocks may be temporarily bypassed using special individual key switches.

(b) With following exception: safety rod is moved to cocked position prior to any other positive insertion operation.

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TABLE 3

ORGANIZATIONAL CHART

(W) Nuclear Training Reactor Facility

Nuclear Training Center, Zion, Illinois