UNITED STATES NUCLEAR REGULATORY COMMISSION

NOTICE OF FINAL FINDING OF NO SIGNIFICANT ENVIRONMENTAL IMPACT

REGARDING PROPOSED AMENDMENT TO

FACILITY OPERATING LICENSE NO. R-81

UNION CARBIDE SUBSIDIARY B, INC.

DOCKET NO. 50-54

The Nuclear Regulatory Commission (the Commission) is considering issuance of an Amendment to Facility Operating License No. R-81 for the Union Carbide Subsidiary B, Inc. (UCS) research reactor located in Sterling Forest, New York.

The amendment will renew the Operating License until June 30, 2000, in accordance with the licensee's application dated May 23, 1980, as supplemented. Opportunity for hearing was afforded by the Notice of Proposed Issuance published in the Federal Register on August 1, 1980 at 45 FR 51320.

Continued operation of the reactor will not require alteration of buildings or structures, will not lead to changes in effluents released from the facility to the environment, will not increase the probability or consequences of accidents, and will not involve any unresolved issues concerning alternative uses of available resources. Based on the foregoing and on the Environmental Assessment, the Commission concludes that renewal of the license will not result in any significant environmental impacts.

Finding of No Significant Impact

The Commission has prepared an Environmental Assessment of this action and has concluded that the proposed action will not have a significant effect on the quality of the human environment. Therefore, the Commission has determined not to prepare an Environmental Impact Statement for the proposed action.

Summary of Environmental Impacts As Described in the Environmental Assessment

The proposed action would authorize the licensee to continue operating the reactor in the same manner that it has been operated since 1961. The environmental impacts associated with the continued operation of the UCS facility are discussed in an Environmental Assessment associated with this action. The Assessment concluded that continued operation of the UCS reactor for an additional 16 years will not result in any significant environmental impacts on air, water, land or biota in the area, and that an Environmental Impact Statement need not be prepared. These conclusions were based on the following:

a) the excess reactivity available under the Technical Specifications is insufficient to support a reactor transient generating enough energy to cause overheating of the fuel or loss of integrity of the cladding;

b) the expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those likely to cause loss of integrity of fuel-element cladding. The staff performed conservative analyses of the most serious credible accidents and determined that the calculated potential radiation doses in unrestricted areas are small fractions of 10 CFR Part 20 allowable doses;

c) the systems provided for control of radiological effluents can be operated to ensure that releases of radioactive wastes from the facility are within the limits of 10 CFR Part 20 and are as low as is reasonably achievable (ALARA); and

d) the licensee's Technical Specifications, which provide limiting conditions for the operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably.

For further details with respect to this proposed action, see the application for license renewal dated May 23, 1980, as supplemented, the Environmental Assessment, and the Safety Evaluation Report prepared by the staff (NUREG-1059).

The Environmental Assessment is available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D. C. 20555. A copy may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, ATTENTION: Director, Division of Licensing.

Copies of NUREG-1059 may be purchased by calling (301) 492-9530 or by writing to the Publication Services Section, Document Management Branch, Division of Technical Information and Document Control, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555; or purchased from the National

Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161.

Dated at Bethesda, Maryland, this 14th of Sept., 1984

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161.

Dated at Bethesda, Maryland, this

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

DL:SSPB	DL:SSPB	DL:SSPB	DL:SSPB	OELD	DL:AD/SA	DL:DIR
PAnderson:jc	HBernard	HBerkow	CThomas		DCrutchfield	DEisenhut
8/13/84	8/12/84	8/13/84	8/13/84	8/14/84	9/12/84	9/14/84

ENVIRONMENTAL ASSESSMENT

FOR THE

TRAINING AND RESEARCH REATOR OF THE

UNION CARBIDE SUBSIDIARY B, INC.

LICENSE NO. R-81

DOCKET NO. 50-54

Description of Proposed Action

This Environmental Assessment is written in connection with the proposed renewal for 20 years of the operating license of the research reactor at the Union Carbide Subsidiary B, Inc. (UCS) facility located in Sterling Forest, New York, in response to a timely application from the licensee dated May 23, 1980, as supplemented. The proposed action would authorize continued operation of the reactor in the manner that it has been operated since facility license No. R-81 was issued in 1961. Currently, there are no plans to change any of the structures or operating characteristics associated with the reactor during the renewal period requested by the licensee.

Need for the Proposed Action

The operating license for the facility was due to expire in June 1980. The proposed action is required to authorize continued operation so that the facility can continue to be used in the licensee's mission of education and research.

Alternatives to the Proposed Action

As required by Section 102(2)(E) of NEPA (42 U.S.C.A. §4332(2)(E)), the staff has considered possible alteratives to the proposed action. The only reasonable alternative to the proposed action that was considered was not renewing the operating license. This alternative would have led to cessation of operations, with a resulting change in status and a likely small impact on the environment. From the standpoint of environmental impact, there are no appropriate alternatives to the proposed action.

Environmental Impact of Continued Operation

The UCS reactor operates in an existing shielded water tank inside an existing multiple-purpose building. No new construction is associated with continued operation of the reactor and there is no change in reactor operating conditions or practices. Therefore, this licensing action would lead to no change in the physical environment.

Based on the review of the specific facility operating characteristics that are considered for potential impact on the environment, as set forth in the

Commission's Safety Evaluation Report (SER)[1] for this action, it is concluded that renewal of this operating license will have an insignificant environmental impact. These conclusions were based on the following:

a) the excess reactivity available under the Technical Specifications is insufficient to support a reactor transient generating enough energy to cause overheating of the fuel or loss of integrity of the cladding,

b) the expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those likely to cause loss of integrity of fuel-element cladding. The staff performed conservative analyses of the most serious credible accidents and determined that the calculated potential radiation doses in unrestricted areas are small fractions of 10 CFR Part 20 allowable doses;

c) the systems provided for control of radiological effluents can be operated to ensure that releases of radioactive wastes from the facility are within the limits of 10 CFR Part 20 and are as low as is reasonably achievable (ALARA); and

d) the licensee's Technical Specifications, which provide limiting conditions for the operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably.

Agencies and Persons Consulted

The Commission has obtained the technical assistance of the Los Alamos National Laboratory in performing the safety evaluation for continued operation of the UCS facility.

Conclusion and Basis for Final No Significant Impact Finding

Based on the foregoing considerations, the staff has concluded that there will be no significant environmental impact attributable to this proposed license renewal. Having reached this conclusion, the staff determined not to prepare an Environmental Impact Statement for the proposed action and that a Final No Significant Impact Finding is appropriate.

Dated: September 28, 1984

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1 NUREG-1059, "Safety Evaluation Report Related to the Renewal of the Operating License for the Union Carbide Subsidiary B, Inc. Research Reactor."

APPENDIX A

FACILITY LICENSE NO. R-81

TECHNICAL SPECIFICATIONS

FOR

UNION CARBIDE

DOCKET NO. 50-54

DATE: JULY 1984

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1.0 DEFINITIONS

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

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

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

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.

Experiment: An experiment is (1) any apparatus, device, or material placed in the reactor core region (in an experimental facility associated with the reactor, or in line with a beam or radiation emanating from the reactor) or (2) any incore operation designed to measure reactor characteristics.

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

Limiting Safety System Setting (LSSS): Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

Measured Value: The measured value of a process variable is the value of the variable as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Movable Experiment: A movable experiment is one that may be inserted, removed, or manipulated while the reactor is critical.

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

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

Reactor Operating: The reactor is considered to be operating when it is performing its intended function in a normal manner.

Reactor Secured: The reactor is secured when (1) the core contains insufficient fuel to attain criticality under optimum conditions of moderation and reflection; (2) the moderator has been removed; (3) the minimum number of control rods are fully inserted as required by Technical Specifications; (4) the console key switch is in the OFF position and the key is removed from the lock; (5) no work is in progress involving core fuel, core structure, installed control rods or control rod drives unless they are physically decoupled from the control rods; and (6) no in-core experiments are being moved or serviced with a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar, whichever is smaller.

Reactor Safety System: The reactor safety system is a combination of safety channels and associated circuitry that forms the automatic protective system for the reactor, or provides information that requires the initiation of manual protective action.

Reactor Shutdown: The reactor is shut down when the negative reactivity of the cold, clean core, including the reactivity worths of all experiments, is equal to or greater than the shutdown margin.

Readily Available on Call: Readily available on call is applicable to an individual who (1) has been specifically designated and the designation has been made 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 hour or within a 30-mile radius).

Reporting Interval: 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 monthly, the interval shall not exceed 6 weeks; when weekly, the interval shall not exceed 10 days; and when daily, the interval shall not exceed 3 days.

Reportable Occurrence: A reportable occurrence is any of those conditions described in Section 6.5.2 of these specifications.

Safety Channel: A safety channel is a measuring or protective channel in the reactor safety system.

Safety Limit: Safety limit is a limit on important process variables that is found to be necessary to reasonably protect the integrity of certain physical barriers that guard against release of radioactivity. The principal physical barrier is the fuel cladding.

Scram Time: Scram time is the elapsed time between the instant a limiting safety system setpoint is reached and the instant that the slowest control rod is fully inserted.

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

True Value: The true value of a process variable is its actual value at any instant in terms of the instrument reading.

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

Untried Experiment: An untried experiment is an experiment or class of experiments that has not been previously evaluated and approved by the Nuclear Safeguards Committee.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits in Forced Cooling Mode of Operation

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

(1) power in MW
(2) flow in gpm
(3) height of water above the core

Objective: The objective is to ensure that the integrity of the fuel cladding is maintained.

Specifications: In forced cooling flow mode of operation

(1) the maximum steady power level under various flow conditions shall be less than or equal to 5 MW (nominal)

(2) the pool water level shall not be less than 20 ft above the top of the core

(3) flow shall be not less than 1,800 gpm for power above 250 kW

Basis: The superposition method of Gambill and Bundy (Final Hazards Summary Report, April 1977) is used to derive the burnout heat flux as a function of primary flow rate. A safety factor of 1.25 is applied to allow for uncertainties in the correlation. The analysis is given in Section A1 of Appendix 2 of the Safety Analysis Report (SAR).

Pool temperature (or core inlet temperature) is not included in the specification because this variable changes very slowly and has only a minor effect: e.g., a 10F° change results in only a 5% variation in burnout flux. The latter, however, is evaluated conservatively near the high end of the pool temperature range which is well below the temperature expected during operation. A derating factor can be applied for pool temperatures in excess of 120°F. The relationship between total power and peak heat flux is derived for the core situation with the greatest peaking factors, namely, a new fuel element adjacent to a central in-core flux trap. Reactor power, primary flow rate, and water level will be maintained well within safety limit specifications through limiting safety system scram settings (see Section 2.2.1 of these specifications).

2.1.2 Limits in Natural Convection Mode of Operation

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

(1) power in MW
(2) height of water above the core

Objective: The objective is to ensure that the integrity of the fuel cladding is maintained.

Specifications: In the natural convection mode of operation

(1) the maximum reactor power level shall be 500 kW

(2) the pool water level shall not be less than 20 ft above the top of the core.

Basis: The analysis is given in Section A2 of Appendix 2 of the SAR. The homogeneous method of Gambill and Bundy (Final Hazards Summary Report, April 1977) used in this analysis has been employed successfully to predict natural convection burnout in Oak Ridge Reactor (ORR) and High Flux Irradiation Reactor (HFIR) fuel to be 6.7 MW. The former fuel is close in design to Union Carbide fuel. A safety factor of 1.24 is applied to account for random variations and uncertainties. A pool temperature near the high end of the operating range (120°F) is assumed. The safety system settings on power and pool level ensure adherence to these specifications (see Section 2.2.2 of these specifications). Maintaining a maximum power level during convection flow of 0.5 MW permits a significant safety factor between operation and burnout.

2.2 Limiting Safety System Settings

2.2.1 Safety Channel Setpoints in Forced Cooling Mode

Applicability: This specification applies to the setpoints of the safety channels.

Objective: The objective is to ensure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specifications: For operation in the forced cooling mode, the limiting safety system settings shall be

(1) power level at any flow rate not to exceed 7.5 MW (nominal)
(2) steady-state power levels not to exceed 5 MWt
(3) coolant flow shall not be less than 1,800 gpm for powers above 250 kW.
(4) pool level shall not be less than 20 ft above the top of the core

Basis: Safety limits have been shown previously (Section 2.1 and SAR) to lie at a low flow-to-power ratio. To provide adequate assurance that these limits are not approached too closely, the LSSS are chosen conservatively so as to minimize the chance of boiling in the core. This results in a much larger flow/power ratio. In Section A3 of Appendix 2 of the SAR, power levels derived using conservative correlations for incipient boiling are tabulated for various values of pool temperatures and flow rates to illustrate the resulting temperature margins.

For a reactivity transient the case considered is the step insertion of 0.25% $\Delta k/k$ positive reactivity with the reactor operating at a steady power of 7.5 MW (SAR, May 1980). The analysis given in Section B3 of Appendix 2 of the SAR shows that the power at the end of 0.75 second (the scram time) will be no more than 11 MW. This is well below the safety limit for this mode of operation. No automatic scram is associated with pool temperature as this parameter changes very slowly allowing ample time for appropriate operator action.

2.2.2 Safety Channel Setpoints in Natural Convection Mode

Applicability: This specification applies to the setpoints of the safety channels.

Objective: The objective is to ensure that automatic action is initiated that will prevent a safety limit from being exceeded.

Specifications: For operation in the natural convection mode, the limiting safety system settings are

(1) power level $\leq$ 250 kW
(2) pool level $\geq$ 20 ft above the core

Basis: The setpoints are chosen to avoid boiling in the core during routine operation with natural convection cooling. The analysis given in Section A4 of Appendix 2 of the SAR shows that a power level of 0.35 MW is needed for incipient boiling to occur. To allow for uncertainties, a safety factor of 1.3 is applied to this, resulting in a safety system setpoint of 0.25 MW. The latter is well below the safety limit of 6.7 MW given in Section A2 of Appendix 2 of the SAR. In the case of reactivity transient, a step insertion of 0.25% $\Delta k/k$ positive reactivity at an initial power level of 0.25 MW will, following the analysis of Section B3 of Appendix 2 of the SAR, result in a transient power of 0.38 MW after 1 second. The latter is well below the safety limit of 6.7 MW for the natural convection mode.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Shutdown Margin

Applicability: This specification applies to minimum shutdown margin for reactor operation.

Objective: The objective is to ensure that the reactor can be safely shut down at all times and to ensure that fuel temperature safety limits will not be exceeded.

Specification: The minimum shutdown margin provided by control rods in the cold, xenon-free condition with the highest-worth rod fully withdrawn and with the highest-worth non-secured experiment in its most positive reactive state shall not be less than 0.5% $\Delta k/k$.

Basis: This specification ensures that the reactor can be shut down from any operating condition and remain shut down after cooldown and xenon decay even if the highest-worth control rod is stuck in its fully withdrawn condition.

3.2 Excess Reactivity

Applicability: This specification applies to the maximum excess reactivity that can be loaded into the core.

Objective: The objective is to ensure that the core will never have excess reactivity beyond the capabilities of the control rods to safely shut down the reactor.

Specification: The core shall not be loaded with an excess reactivity of more than 10.0% $\Delta k/k$ when located in the stall position and 8.2% $\Delta k/k$ when the core is located in the open pool position.

Basis: These limits on excess reactivity assure a negative shutdown margin of 0.5% or greater upon a scram even if the rod with the maximum worth is stuck out of the core.

3.3 Control and Safety Systems

Applicability: This specification applies to the control and safety systems.

Objective: The objective is to ensure that the safety systems respond with adequate negative reactivity in a minimum time to ensure safe control and shutdown of the reactor when required.

Specifications: The specifications are as follows:

3.3.1 Control Rods

See Section 3.1, Shutdown Margin.

3.3.2 Regulating Rod

The integral worth of the regulating rod shall not exceed 0.6% $\Delta k/k$. This ensures that a malfunction of the control system cannot make the reactor prompt critical.

3.3.3 Scram Time

The scram time shall not exceed 0.75 second and the control rod magnet release time shall not exceed 0.05 second. In the transient analysis given in Section B3 of Appendix 2 of the SAR, these values were assumed.

3.3.4 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 3.1.

Table 3.1 Minimum number and operating mode of measuring channels

Channel	Minimum no. operable	Operating mode required
Power level (normal)	2	All
Power level (intermediate)	1	All
Period channel	1	All
Count rate	1*	Startup
Coolant flow	1	Forced cooling
Core ΔT	1	Forced cooling
Rod position	1/rod	All
Pool temperature	1	All
Pool level	1	All

*Operable below 50 W.

Basis: The normal power level instruments (level safeties) provide redundant information on reactor power in the range 25% - 150% of the normal operating power level of 5 MW.

The intermediate power instrument (log N) provides usable reactor power information in the logarithmic range 10^{-4}% - 300% of the normal power of 5 MW.

The count rate channel covers the neutron flux range from the source level ($\cong$1 cps) to 10^4 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 ΔT instruments allow the operator to calculate reactor power and calibrate the neutron flux channels in terms of power.

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

Pool temperature information allows the operator to adjust the cooling system to keep pool temperature within a preferred range, and to adjust the overpower reverse setpoint (see Section 3 3.4).

3.3.5 Safety Channels

The minimum number and type of channels providing automatic action that are required for reactor operation are given in Table 3.2.

Table 3.2 Minimum number, function, and operating mode of safety channels

Channel	Minimum no. operable	Function	Operating mode required
Power level (normal)	2	Scram @ 7.5 MW	All
Power level (intermediate)	1	Scram @ <3-sec period	All
		Reverse @ <10-sec period	All
		Inhibit @ <30-sec period	All
Count rate	1[a]	Inhibit @ <2 cps	Startup
		Inhibit @ <30-sec period	Startup
Pool water level	1	Scram @ <22 ft.	All
Pool temperature	1	Alarm @ >120°F	All
Coolant flow	1[b]	Scram @ <1800 gpm	Forced circ.
Manual button	1	Scram	All
Bridge lock	1	Scram	All
Guide tube lift	1/rod	Scram	All
Flapper valve	1	Scram (above 250 kW w/ valve open)	All
Keyswitch	1	Scram	All

[a]Operable below 50 W.

[b]Operable above 250 kW.

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

The period scram, assisted by the intermediate level period reverse and rod inhibit, limits the rate of increase in reactor power to values that are controllable without reaching excessive power levels or temperature. These functions are not limiting safety system settings.

The two inhibits on the count rate channel prevent inadvertent criticality during cold startup that could arise from lack of neutron information or from too rapid reactivity insertion by control rods.

The scram pool on level provides an adequate head of water above the core and guards against loss of coolant and loss of building containment.

The coolant flow and flapper valve scrams ensure adequate coolant flow to prevent boiling in the core.

The scrams on bridge lock and guide tubes prevent unplanned reactivity changes that could occur through core and control element movements, respectively.

The keyswitch scram prevents unauthorized operation of the reactor.

Bypass is permitted on those parameters that can be monitored by alternate means if the initiating circuit malfunctions.

3.4 Radiation Monitoring Systems

Applicability: This specification applies to the radiation monitoring systems required for safe operation of the reactor, operating personnel protection, and protection of the public.

Objective: The objective is to ensure that operation of the reactor is within the goal of ALARA and to detect the release of fission products within Union Carbide Subsidiary (UCS) set limits.

Specifications: The specifications for the minimum acceptable monitoring instrumentation required for reactor operation are given in Table 3.3.

Basis: These setpoints produce less personnel exposure than indicated in 10 CFR Part 20, and permit early identification of fuel cladding or process system failure.

3.5 Engineered Safety Features

Applicability: These specifications apply to required equipment for the confinement of activity through controlled release of reactor building air to the atmosphere.

Objective: The objective is to ensure personnel are notified about high radiation incidents and to initiate engineered safeguards systems for safe shutdown of the reactor and mitigation of release of radioactivity to the environment.

Table 3.3 Radiation monitoring systems

Type	No. operable	Max. alarm setpoint	Function
Excursion monitor	1	5 R/hr	Detect high radiation: Alarm and isolate at >5 R/hr
Exhaust duct monitor ("stack monitor")	1	*	Detect particulate, gas, and iodine activities; alarm in control room
Building continuous air monitor (CAM)	1	**	Detect particulate activity in reactor building; alarm in control room
Fixed area monitor	3	50 mr/hr	Detect radiation (γ) in key locations; alarm in control room
Evacuation switch	1	--	Alarm and initiate evacuation sequence (manual)

Note: For maintenance or repair, required radiation monitors (except for excursion monitor) may be replaced by portable or substitute instruments for periods up to 24 hours provided the function will still be accomplished. Interruption for brief periods to permit checking or calibration is permissible.

*The alarm setpoint for the stack gas monitor shall not be set above a value that would result in an exposure greater than 2 mrem per hour assuming a dilution factor of 2000 and the isotope mixture determined by the annual environmental report or most recent analysis. The alarm setpoint for the stack I-131 and stack particulate monitor shall not be set above a value corresponding to that listed in Appendix B, Table II, Column I of 10 CFR Part 20, assuming a dilution factor of 2000 and averaging over one week.

**25% of the maximum permissible concentration at restricted areas according to Appendix B of 10 CFR Part 20.

Specifications: The specifications are as follows:

3.5.1 Excursion Monitor

Specification: See Section 3.3.5.

Basis: This monitor senses excessive radiation at the reactor bridge and automatically initiates the "evacuation sequence," which consists of a distinctive alarm, closure of damper valves in the building ventilation system and hold-up tank vent, and starting of the emergency exhaust fan (see Section 5.5.2).

3.5.2 Emergency Electric Generator

Specification: The specifications are as follows:

Equipment: Electric generator

No. Operable: 1

Function: Upon loss of utility power, start automatically and supply emergency power to the exhaust fan and ventilation system controls. A 6-day supply of fuel shall be maintained.

Basis: Upon loss of utility power the reactor scrams automatically. Controlled release confinement requires the ability to run the emergency exhaust fan and to close building damper valves. The latter are pneumatically operated but are electrically controlled.

3.5.3 Confinement

Specifications:

(1) The emergency exhaust fan shall be capable of sustaining a negative pressure within the reactor building of at least 0.01 in. wg at an exhaust flow rate of not greater than 200 cfm.

(2) Filters in the emergency exhaust shall be HEPA filters and charcoal filters, with tested efficiencies of a minimum of 99.5% for particle removal and 95% for iodine removal, respectively.

(3) Depth of water in the canal shall be at least 10 ft. This is equivalent to a water height above the core of 22 ft.

(4) At least one door of the double airlock doors and the truck doors shall be closed while the reactor is operating.

Basis: To effect controlled release of gaseous activity present in the building's atmosphere under accident conditions, a negative pressure is required so that any building leakage will be inward. Section C2 of Appendix 2 of the SAR contains an analysis of a hypothetical accident resulting in release of airborne activity to unrestricted areas. The assumed exhaust rate is 200 cfm and the filter efficiency for elemental iodine is 95%. In the design of the containment building the water seal in the canal is effected when the water depth is >10 ft.

3.6 Limitations on Experiments

3.6.1 Experiments

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

Objective: The objective is to prevent damage to the reactor or excessive release of radioactive material in the event of an experiment failure and also to prevent the safety limits from being exceeded.

Specifications: Experiments installed in the reactor shall meet the following conditions:

(1) The combined total worth of all experiments in the core at one time which can add positive reactivity to the core from a common-mode failure shall not exceed 2% $\Delta k/k$.

(2) The combined total worth of all non-secured experiments which can add positive reactivity to the core from a common-mode failure shall not exceed 1.7% $\Delta k/k$.

(3) The reactivity of any single experiment shall not exceed 0.5% $\Delta k/k$.

(4) An experiment with a greater negative worth than 0.25% $\Delta k/k$ shall not be moved when the reactor is critical.

(5) An experiment worth more than 0.25% $\Delta k/k$ but less than 0.5% $\Delta k/k$ may be moved with the reactor subcritical by at least 0.75% $\Delta k/k$.

(6) All material to be irradiated in the reactor shall either be corrosion resistant or encapsulated within corrosion-resistant containers.

(7) Where failure of the pressure-containing walls of an experiment container can cause a hazard to personnel or to the reactor, the container shall be designed and tested in accordance with the applicable pressure vessel codes.

(8) In-core experiments exposed to reactor water shall be designed to prevent surface boiling.

(9) Experimental apparatus, material, or equipment to be inserted in the reactor shall not interfere with the safe operation of the reactor.

(10) The total primary coolant flow utilized by all in-core experiments shall meet the following requirement: (fraction of core in experiments) $\leq$ 1-5/6 (fraction of rated power produced in fuel element).

(11) Experiments on the grid-plate extension are limited to a total reactivity of 0.2% $\Delta k/k$.

(12) Each class of experiment irradiation in the reactor must have been previously reviewed and approved by the Nuclear Safeguards Committee (see Section 6.8 below).

Basis:

(1) See Section G5c of "Final Hazards Summary Report UCNC Research Reactor" (May 1960).

(2) It is shown in "Reactor Power Excursion Tests in the SPERT IV Facility," IDO-17000 (August 1964) that the reactor can safely self-limit a step reactivity insertion of \$2.14. This corresponds to an insertion of $2.14 \times 0.81 = 1.73\%$ $\Delta k/k$.

(3) The method of Section B3 of Appendix 2 of the SAR, shows that a step insertion of 0.5% $\Delta k/k$ with the reactor critical of 5 MW (or 0.25 MW, in natural convection mode) will result at the end of 0.75 second in a power of not more than 14 MW and 0.4 MW, for natural convection. Each of these power levels does not exceed the corresponding safety limit.

(4) Similarly it is shown that a step increase of 0.25% $\Delta k/k$ will produce a power level at the end of the scram time that is much less than the safety limit in either mode of operation. In addition, 0.25% $\Delta k/k$ is well within the automatic control capability of the reactor control system.

(5) This specification ensures that, even with a 45% error in estimation of the reactivity of an experiment, the reactor will not be made critical. Even if the reactor were critical, the resulting period ($\sim$3 seconds) would automatically initiate corrective control action.

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

(7) This specification ensures that there will be no mechanical damage to the reactor core nor hazards to personnel should experiment containers fail where pressure exists or builds up during irradiation. In the case of fueled experiments, it further ensures against hazardous and uncontrolled release of fission products into the reactor building or the environment from the same cause.

(8-9) Ensures that no physical or nuclear interference with the safe operation of the reactor will occur.

(10) This condition is assumed in the analysis given in Section A of Appendix 2 of the SAR.

(11) These limits ensure that movement of these experiments will not result in reactivity changes in excess of that in specification (4) above.

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

3.6.2 Fueled Experiments

Applicability: These specifications apply to experiments containing nuclear fuel that are installed in the reactor or its experiment facilities.

Objective: 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.

Specifications: Fuel-bearing experiments in the reactor shall meet the following conditions:

(1) All fueled experiments are to conform to the specifications listed above in Section 3.6.1.

(2) The inventory of solid fuel-bearing materials in a single irradiation capsule shall be limited to 200 g of source and/or 50 g of special nuclear material.

(3) The fission power of an irradiation capsule containing special nuclear material shall be limited to 13 kW.

(4) The iodine inventory of a singly encapsulated capsule shall be limited to a maximum of 1000 Ci I-131 dose equivalent.

Basis: These specifications place limits on the fission product inventory in a fueled capsule so that capsule failure and the hypothetical release of all contained fission products to the reactor coolant will not result in excessive exposure to personnel on and off site.

It has also been established (see License Amendment No. 10) that failure of a single capsule will not initiate failure in other neighboring capsules.

3.7 Fuel

Applicability: These specifications apply to the number and condition of the fuel elements present in the core.

Objective: The objective is to ensure that power is distributed in the core among a sufficient number of fuel elements to avoid excessive peak/average ratio, and to avoid excessive release of fission products.

Specifications: Fuel elements present in the core shall meet the following conditions:

(1) The total number of fuel elements in the core shall be limited to the total excess reactivity and shutdown value specified in Sections 3.2 and 3.1, respectively.

(2) The minimum number of fuel elements in the core shall be 30. Each control element shall count as 1/2 fuel element for this purpose.

(3) Each control rod shall be kept within ±10% of the average of all the five control rod positions, whenever the reactor power exceeds 500 kW.

(4) Fuel elements exhibiting release of fission products from cladding rupture shall, upon positive identification, be removed from the core. An increase in the normal gaseous fission product release (from system contamination) by a factor of 100 shall constitute initial evidence of cladding rupture and require identification of the cause.

(5) Fuel element loading and distribution in the core shall be such that peak/average thermal flux will not exceed 3.3.

(6) The fuel plates are composed of enriched uranium-aluminum sandwiched between high purity aluminum clad. Fuel plates may be fabricated by alloying the uranium-aluminum or by the powder metallurgy method where the starting ingredients (uranium-aluminum) are in the fine powder form. Burnup of the fuel assemblies shall be limited to 0.94×10^{21} fission/cm^3.

Fuel plates may also be fabricated from uranium oxide-aluminum (U_3O_8-Al) using the powder metallurgy process, and the burnup shall be limited to 1.5×10^{21} fission/cm^3.

Basis:

(1) Limitations on excess reactivity and shutdown margin ensures safe shutdown for all core configurations.

(2) A minimum of 30 elements is assumed in the analysis given in Section A1 and A2 of Appendix 2 of the SAR.

(3) Although it maintains the limits on excess reactivity and shutdown margin, this specification minimizes flux tilts that could cause concentrations or shifts in power distribution across the core. Such shifts are only significant in power operation, and thus this limitation is restricted to power levels above 10% of the normal 5 MW.

(4) Release of fission products from the compact fuel plates used in this reactor (Section 5.1), from 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 from uranium contamination in the coolant and on 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.

(5) This peak/average value is used in Appendix 2 of the SAR analysis.

(6) See License Amendment No. 12 and Supplement No. 3 to Final Hazard Summary Report (December 1977).

3.8 Pool Water Quality

Applicability: This specification applies to primary cooling system water in contact with fuel elements.

Objective: The objective is to minimize corrosion of the aluminum cladding of fuel plates and activation of dissolved materials.

Specifications: Pool water in contact with fuel elements shall meet the following conditions:

(1) Bulk pool water temperature will not exceed 130°F.

(2) Pool water specific resistance is to be not less than 200,000 ohm-cm, except that for periods no longer than 14 days it may be 70,000 ohm-cm.

(3) The pH of the pool water shall be maintained between 5.0 and 7.5.

3.9 Radioactive Releases

Applicability: These specifications relate to activity and doses from release of radioactive emissions and liquid wastes generated from reactor operations. When combined with hot laboratory releases, the specifications relate to activity and doses from mixed releases.

Objective: The objective is to ensure that releases are as low as reasonably achievable.

Specifications: The specifications are as follows:

3.9.1 Airborne Stack Release Limit

Maximum yearly release rates for noble gases, radioiodines and particulates of half-life greater than 8 days shall be limited by the following expression:

$$\sum_i Q_i\ (\overline{\chi/Q})/MPC_i < 1/6$$

where:

Q_i = The average release rate for any 12 consecutive months of radionuclide, i, in gaseous effluent from the stack in Ci/sec.

MPC_i = Activity concentration of radionuclide, i, as given in Table II, Column 1 of Appendix B to 10 CFR Part 20, in μCi/cc.

$\overline{\chi/Q}$ = Shall be calculated monthly from measured values of iodine concentration sampled at or above the tree line 380 m NE of the exhaust stack.

3.9.2 Dose in Unrestricted Areas

(1) Total body dose attributable to noble gas releases and dose from radioiodines in gaseous effluents for the critical individuals in unrestricted areas should be calculated at least once per calendar quarter and reported in the annual report (see Section 6.6.1(5)).

(2) The total body dose to any individual in unrestricted areas due to noble gases released in gaseous effluents from the site shall be limited to the following expressions:

(a) During any calendar quarter:

$$3.17 \times 10^{-8} \sum_i M_i\ (\overline{\chi/Q})\ \tilde{Q}_i \leq 2.5 \text{ mrem}$$

(b) During any calendar year:

$$3.17 \times 10^{-8} \sum_i M_i\ (\overline{\chi/Q})\ \tilde{Q} \leq 5 \text{ mrem}$$

where:

$\tilde{Q}_i$ = The release of noble gas radionuclide, i (measured concentration x flow rate), in μCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

M_i = The total body dose factors from gamma emissions for each identified noble gas radionuclide, mrem/year per $\mu Ci/m^3$ from Regulatory Guide 1.109 (Revision 1), Table B-1.

$\overline{\chi/Q}$ = Shall be calculated from measured values of iodine concentration sampled at the environmental monitoring station in Laurel Ridge. This measured value shall be increased by a factor of 2 when calculating the body dose limits.

(3) The dose to an individual from radioiodines in gaseous effluents released to unrestricted areas shall be limited as follows:

(a) During any calendar quarter:

$$3.17 \times 10^{-8} \sum_i (R_i W Q_i) \leq 7.5 \text{ mrem, and}$$

(b) During any calendar year:

$$3.17 \times 10^{-8} \sum_i (R_i W Q_i) \leq 15 \text{ mrem}$$

where:

Q_i = The release of radioiodines in gaseous effluents, i, in μCi. Release shall be cumulative over the calendar quarter or year, as appropriate.

R_i = The dose factor for each radioiodine, i, in mrem/year per $\mu Ci/m^3$ (from Regulatory Guide 1.109) except for I-125 which is determined to be as follows:

Adult thyroid (inhalation): 1.2×10^{-3} mrem/pCi

Infant thyroid (inhalation): 4.0×10^{-3} mrem/pCi

Child thyroid (inhalation): 1.8×10^{-3} mrem/pCi

*W = The average dispersion parameter for estimating the dose to an individual in the controlling location from radioiodines in gaseous effluents released to unrestricted areas.

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*The present controlling dose pathway is via infant inhalation at the Laurel Ridge residential site. If the land use census (Section 3.10 of these specifications) identifies a location or pathway which yields a calculated dose or dose commitment greater than via the presently calculated dose pathway, the dispersion parameter ($\overline{\chi/Q}$ or $\overline{D/Q}$) and dose factor (R_i) for this more restrictive pathway shall be used in this specification.

$W = (\overline{\chi/Q})$ for the inhalation pathway, in sec/m^3 (as determined in Section 3.8.2(1))

$W = (\overline{D/Q})$ for the food pathways, in m^{-2}

3.9.3 Liquid Effluent Releases

(1) Liquid waste from all radioactive operations shall be collected in hold tanks.

(2) Before release from the hold tanks, the liquid waste shall be sampled and the activity level measured.

(3) 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. This activity concentration shall be determined at least once per month by an analysis of a composite sample of all tanks released during that period.

(4) Total radioactivity released in liquid effluents shall not exceed 0.01 Ci (Sr-90 equivalent) in any 12-consecutive-month period.

(5) Records of and reports on liquid radioactive effluent releases shall be as specified in Section 6.0 of these technical specifications.

3.10 Radiological Environmental Monitoring

The radiological environmental monitoring program shall be conducted as specified in Table 3.4. The results of analyses performed on the radiological environmental monitoring samples shall be summarized in an annual radiological environmental report.

3.11 Land Use Census

A land use census shall be conducted at least once per 12 months between June 1st and October 1st, and shall identify the location of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 square feet producing fresh, leafy vegetables in each of the 16 meteorological sectors within a distance of 5 miles.

3.12 Basis for Environmental Specifications

(1) Specification 3.9.1 is provided to ensure that the dose at the exclusion area boundary from gaseous effluents from the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the restricted area boundary to $\leq$500 mrem/year, to the total body. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background via the inhalation pathway to $\leq$1500 mrem/year.

Table 3.4 Radiological environmental monitoring program

Exposure pathway and/or sample	No. of samples and sample locations	Sampling and collection frequency	Type and frequency of analysis
<u>Airborne</u>			
Radioiodine and particulates	1 sample from 380 meters NE of stack 1 sample from Laurel Ridge area	Continuous operation of sampler with sample collection as required by dust loading, but at least once per 7 days	Radioiodine canister analyze at least once per 7 days for I-131* Particulate sampler. Analyze for gross beta radioactivity 24 hours following filter change. Perform gamma isotopic analysis on each sample when gross beta activity is 10 times the mean of control samples for any medium. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
<u>Direct Radiation</u>	1 sample from 380 meters NE of stack	At least once per 31 days	Gamma dose: At least once per 31 days.
	1 sample from Laurel Ridge area	At least once per 92 days (readout frequencies are determined by type of dosimeters selected).	Gamma dose: At least once per 92 days.
<u>Ingestion</u>			
Food products	Location to be determined from land use census	One sample of broad-leaf vegetation, at time of harvest	I-131* analysis

Table 3.4 (Continued)

Exposure pathway and/or sample	No. of samples and sample locations	Sampling and collection frequency	Type and frequency of analysis
<u>Ingestion</u> (Continued)			
Water	Indian Kill inlet	Monthly	Gross beta: Monthly
	Indian Kill outlet	Monthly	Gross beta: Monthly
	Warwick Brook	Monthly	Gross beta: Monthly
	Sterling Lake outlet	Monthly	Gross beta: Monthly
	Ramapo River	Monthly	Gross beta: Monthly

*The maximum values for the lower limit for I-131 are 7×10^{-2} pCi/m^3 airborne concentration and 60 pCi/kg, wet weight leafy vegetables.

(2) Specification 3.9.2 is provided to demonstrate compliance with 10 CFR §20.1(c) which requires that radioactive materials released to unrestricted areas be kept to as low as reasonably achievable. The action statements provide the operating flexibility and at the same time implement the design objective of minimizing the release to unrestricted areas to as low as reasonably achievable. The specifications for noble gas releases are based on limiting the total body dose at the limiting populated area to less than 5 mrem/year. The specification for radioiodine is based on the assumption that the limiting dose pathway for these radioisotopes is via infant inhalation at the Laurel Ridge residential site, and limits the infant thyroid dose to less than 15 mrem/year.

(3) The radiological monitoring program required by specification 3.10 provides measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. This monitoring program may change based on operational experience and results of the land use census.

(4) Specification 3.11 is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the technical specification limit of dose, via the most restrictive dose pathway and the monitoring program, can be made if required by the results of this census.

4.0 SURVEILLANCE REQUIREMENTS

Applicability: The requirements listed below generally prescribe tests or inspections of systems and components.

Objective: The ojective is to periodically verify that the performance of required systems is in accordance with specifications given above in Sections 2.0 and 3.0.

Specifications: The specifications are as follows:

4.1 Safety Channel Calibration

A channel calibration of each safety channel shall be performed annually (see Section 3.3.4).

4.2 Reactivity Surveillance

(1) The reactivity worth of each control rod (including the regulating 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.

(2) The reactivity worth of an experiment shall be estimated, or measured at low power, before conducting the experiment.

(3) Boron carbide rods shall be gauged quarterly and any dimensional changes reported promptly to the Commission. Silver/indium/cadmium control rods shall be gauged annually, or, in the case of newly installed rods, at the end of the first 6 months. If any Ag/In/Cd or boron aluminum alloy rod does not meet the acceptance criteria, it shall be removed from service. In addition, all other rods manufactured from the same batch shall be inspected.

4.3 Control and Safety System Surveillance

(1) The scram time shall be measured annually. If a control rod is removed from the core temporarily, or if a new rod is installed, its scram time shall be measured before reactor operation. If the bridge is moved, the scram time will be measured before subsequent reactor operation.

(2) A channel test of each measuring channel in the reactor safety system shall be performed monthly or prior to each reactor operating period, whichever occurs first, unless the preceding shutdown period is 8 hours or less. A channel test before startup is, however, required on any channel receiving maintenance during the shutdown period.

(3) A channel check of each measuring channel (except for the pool level) in the reactor safety system shall be performed daily when the reactor is in operation.

4.4 Radiation Monitoring System

(1) The excursion, stack, and area monitors shall be calibrated annually.

(2) The excursion, stack, and area monitors shall receive a channel test monthly.

(3) The excursion, stack, and area monitors shall receive a channel check and a setpoint verification daily during reactor operating periods.

4.5 Engineered Safety Features

4.5.1 Excursion Monitor: See Section 4.4, above.

4.5.2 Emergency Generator

(1) The ability of the emergency generator to start, to run normally, and to generate 440 V AC shall be checked weekly.

(2) The generator shall be tested for its ability to accept, via the automatic transfer switch, the reactor electrical load once every 6 months. A commercial power outage and subsequent pickup of load by the emergency generator will count as a successful load test.

4.5.3 Confinement

(1) The efficiency of the charcoal filters and of the absolute filters in the emergency exhaust system shall be measured annually and the flow rate verified.

(2) The operability of the evacuation alarm and containment isolation system shall be tested, and negative pressure verified, semiannually. A utility power outage may be used to initiate such tests.

4.6 Reactor Fuel

(1) Upon receipt from the fuel vendor, all fuel elements shall be visually inspected and the accompanying quality control documents checked for compliance with specifications.

(2) Each new fuel element will be inspected for damage and flow obstructions prior to insertion into the core.

4.7 Sealed Sources

The antimony-beryllium sealed source shall be leak tested in accordance with the procedures described in the application for License Amendment No. 5, except that the frequency of leak testing will be in accordance with 10 CFR 34.25(b).

4.8 Pool Water

(1) The pH and specific resistance of the pool water shall be determined each week.

(2) An analysis of the pool water for radioactive isotope identification shall be done at semiannual intervals. This analysis is to include Sb-124 as an indicator of Sb-Be neutron source integrity.

(3) Activity of the pool water will be measured weekly.

4.9 Core Spray

The core spray in the reactor operating position shall be tested for operability semiannually.

4.10 Flux Distribution

In order to verify that power gradients among fuel element do not cause peaking factors to exceed those used in the basis of Section 2.1, the radial neutron flux distribution will be determined whenever a significant core configuration change is made.

5.0 DESIGN FEATURES

Applicability: This specification applies to design features relevant to operation safety and to limits that have been previously specified.

Objective: The objective is to ensure that these design features shall not be changed without appropriate review.

Specifications: The specifications are as follows:

5.1 Reactor Fuel

Fuel elements shall be of the general MTR/ORR type with thin plates containing uranium fuel enriched to about 93% U-235 and clad with aluminum. The fuel matrix may be fabricated by alloying high-purity aluminum-uranium or by the powder metallurgy method where the starting ingredients (uranium-aluminum) are in fine powder form. Fuel matrix may also be fabricated from uranium oxide-aluminum (U_3O_8-Al) using the powder metallurgy process. Elements shall conform to these nominal specifications:

Overall size:	3 in. x 3 in. x 34 in.
Clad thickness:	0.015 in.
Plate thickness:	0.050 in.
Water channel width:	0.12 in.
No. of plates:	Standard element - 16 fueled plates (min.) Window element - 16 fueled plates (min.) Control element - 9 fueled plates (min.) Partial element - 9 fueled plates (min.)
Plate attachment:	Swagged or pinned.
Fuel content (total):	200 g U-235 nominal.
Fuel burnup:	The fuel burnup shall not exceed 0.94×10^{21} fission/cm^3 except for U_3O_8-Al which shall not exceed 1.5×10^{21} fission/cm^3.

5.2 Control and Safety Systems

Design features of the components of this system (see Sections 3.3.4 and 3.3.5) that are important to safety are given below.

5.2.1 Power Level (Normal Channels)

For this function three independent measuring channels are provided, two of which are required to be operable as a minimum. Each channel covers reliably the range from about 25% to 150% of (5 MW). Each channel comprises an uncompensated boron-coated ion chamber feeding an amplifier that controls electronic switches in the DC current that flows through each control rod electromagnet. Each channel controls and scrams all control rods. Each channel is fail-safe. The "fast" scram (~5 ms) from each channel also produces, and is backed up by a "slow" scram (~20 ms) through interruption of the AC supply to the rod electromagnet DC power supply. Each channel indicates power level on a panel meter allowing channel checks to be done during reactor operation. Each chamber can be changed in position, over a limited range, so as to allow the channel reading to be standardized against reactor thermal power.

5.2.2 Power Level (Intermediate) Channel

For this function, a single channel is provided, covering reliably the range 10^{-3}% to 300% (of 5 MW) with a logarithmic output indication on both a panel meter and a chart recorder. To cover the range under all core conditions, a gamma-compensated boron-ion chamber is used to supply a logarithmic amplifier. The chamber can be changed in position, over a limited range, so as to allow the channel reading to be standardized against reactor thermal power. Rate of change of power information is also derived, in the form of a period, that can produce a fast scram (and backup slow scram) in the same way as in Section 5.2.1. Control and inhibit actions [viz., bypassing of count rate channel functions, bypassing of flow and flapper scrams] are also derived from this channel. To negate the effect of overcompensation in the ion chamber, which can occur under certain conditions, even in an initially undercompensated chamber, provision is made to supply an adjustable small current to the channel amplifier (up to 1.5×10^{-10} amps) so as to facilitate startup.

5.2.3 Count Rate Channel

A fission chamber is used to supply pulses to a linear 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 and a chart recorder. In addition, count rate period information is derived and similarly displayed. The channel covers a range of 1 to 10^4 cps, corresponding roughly to a range of 0.25 MW to 2.5 W, but the upper limit can be increased many decades by repositioning the chamber. The motor-operated chamber drive is operated from the control room, the drive position being indicated on a meter. To prevent control-rod withdrawal when the neutron count rate information may not be reliably indicated, inhibits are provided on count rate and period, and when the fission chamber is being repositioned. All except the latter are bypassed at a power of >50 W. A scaler is also provided for obtaining accurate values at low count rates if needed (e.g., approach to critical with new fuel or new core configuration).

5.2.4 Neutron Source

For obtaining the reliable neutron information necessary for startup from a cold shutdown condition, an antimony-beryllium neutron source is provided for insertion into the core as needed. This source, nominally 50 Ci of Sb-124, is renewed by neutron activation in the core. Its presence in the core is not essential except after extended shutdowns. Integrity of the source is checked by periodic sampling of pool water (see Section 4.8).

5.3 Rod Control System

5.3.1 Control Rods

The number of control rods shall be determined by the required shutdown margin and flux uniformity requirements. Up to five control rods are provided for the control of core reactivity. These rods may be fabricated of either boron-carbide, silver-indium-cadmium, or boron aluminum alloy (see Section 4.2(3)). Individual integral worths vary from about 1 to 4% $\Delta k/k$, depending on position and core configuration. The rods are coupled to drive shafts through electromagnets that allow release of the rods within 50 ms after receiving a scram signal. 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 5 in./minute and no more than two rods can be withdrawn simultaneously. Switches on the guide tubes attached to the control fuel elements are arranged to produce a scram if any guide tube is lifted. This guards against lifting of the attached fuel element.

5.3.2 Regulating Rod

One regulating rod is provided to aid in fine control and maintenance of constant reactor power for long periods. The rod is non-fueled, is limited to a total worth of 0.6% $\Delta k/k$ for operational reasons predominantly (Section 3.1.4), and can be either manually or servo-controlled. The drive speed is 24 in./minute. Coarse and fine position readouts are provided.

5.4 Cooling System

5.4.1 Primary Cooling System

Core cooling is effected by gravity flow of demineralized water from the reactor pool to an underground holdup tank that provides an approximate 10-minute delay to allow N-16 activity to decay. The water is then pumped back to the pool through the primary side of a heat exchanger where heat is transferred to a secondary cooling system. The holdup tank is vented to the building exhaust duct. The driving force for the coolant is the fixed head between the pool overflow gutter and the water level in the holdup tank, the latter being fixed by the total volume of water in the system. Flow is adjusted to a desired amount with a valve in the core exit line. Core cooling is not immediately affected by pump failure as a flow will continue until the water levels equalize; neither will the pool be drained. To prevent leakage of water through the pool walls, a continuous steel shell is located within the concrete pour of the pools. All embedments penetrating the pools are welded to this shell. To eliminate corrosion of inaccessible piping, the embedded portion of the reactor primary cooling piping under the pools is stainless steel. To change over automatically to natural convection cooling at low flow rates, a weighted flapper valve seals the core exit plenum. This valve, held closed by the core pressure drop, opens by gravity when the flow drops to approximately 700 gpm. Leakage at the flapper valve seat, or in the plenum, is monitored by a plenum leak detector that senses increase in plenum pressure and alarms in the control room. Primary flow is measured by taking the pressure drop across an orifice plate in the core exit line; indication is both in the pump room and on a recorder in the control room. Temperature sensors in the pool (above the core) and in the core exit line allow the core ΔT to be measured. These are resistance thermometers having alternative recorder and digital readout in the control room. Float switches are provided to monitor pool level. Normal pool level (at the overflow gutters) provides 24 ft of water over the top of the fuel.

5.4.2 Secondary Cooling System

Reactor power transferred through the heat exchanger is dissipated to the atmosphere via a cooling tower. To minimize corrosion, the exchanger has stainless steel shell and tubes. 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 elevation of the two systems.

5.4.3 Core Spray

For backup in the event a hot core is exposed, two spray nozzles are located at the two alternative operating positions of the reactor. They are controlled from a manually operated valve located outside the reactor building.

5.5 Confinement System

5.5.1 Physical Features

The confinement structure consists of the reactor building, with a free air volume of about 7700 m^3. This building houses the reactor, the primary cooling system including holdup tank, and the heat exchanger. Personnel gain access via double airlock doors or sliding doors with inflatable seals. A 12-ft-deep water-filled canal penetrates the building and confinement is provided by a 25-in.-deep water-seal weir. A water-tight gate with inflatable seals can be used to shut the canal off from the reactor pools when needed. The building is ventilated through pneumatically operated damper valves that can also be used to seal the building. These dampers are fail-shut when air pressure is reduced.

5.5.2 Emergency Sequence

Although negative pressure within the confinement building is not a requisite for reactor operation, it is required in the event of a release for the controlled-release confinement of airborne radioactive material. The emergency sequence is initiated either automatically by the excursion monitor (see Section 3.6.1) or manually by the console operator. The sequence follows: All air-supply ducts and the pool sweep dampers are closed immediately, followed later by the exhaust duct damper as soon as negative pressure (-1 in. wg) is attained in the building (but not more than 7 seconds later). Closing the pool sweep damper prevents activity released above the core from reaching the exhaust duct before it closes. Also closed immediately are the isolation valves in the vent line and the air purge to the holdup tank. This prevents activity in this tank from reaching the exhaust duct. Upon closure of the exhaust duct damper, the emergency exhaust fan starts and maintains a nominal negative pressure in the confinement building. This fan exhausts building air at a rate of $\leq$200 cfm through absolute and charcoal filters before connecting into the normal exhaust duct. The latter discharges to the atmosphere through a stack at a high elevation. The entire evacuation sequence is fail-safe upon loss of utility electric power. It will operate with either utility or emergency generator power.

5.5.3 Exhaust Duct Monitor ("Stack Monitor")

Air in the exhaust duct is continuously sampled for particulate iodine and gaseous activities, each being read by separate detectors. The relative proportions of each type of activity can thus be determined. The results are indicated on chart records, with repeaters in the control room. Detection or indication of a release is not dependent on all three detectors being operational, for any release will have associated with it all three types of activity or will affect each detector to some extent. When setpoints are exceeded, alarms are given at the monitor and repeated in the control room.

5.6 Fuel Storage

5.6.1 New Fuel

Unirradiated new fuel elements are stored in a vault-type room security area equipped with intrusion alarms in accordance with the security plan. Elements are stored upright in metal racks in which the separation between elements is a minimum of 2 in. With such an arrangement, subcriticality is assured ("Critical Experiments With SPERT-D Fuel Elements" (July 14, 1965)).

5.6.2 Irradiated Fuel

Irradiated fuel is stored upright under water in the storage pool within the reactor building in criticality-safe racks. Each rack accommodates 16 elements in wells with center-to-center spacing of 6 in. "Supplementary Information to Final Hazards Summary Report" (April 28, 1961) states that an infinite number of elements so stored would be subcritical. Each well has a hole at the bottom to allow the water to circulate for cooling.

6.0 ADMINSTRATIVE 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 Figure 6.1. Job titles are shown for illustration and may vary. Four levels of authority are provided, as follows:

Level 1: individual responsible for the facility's license and site administration

Level 2: (a) individual responsible for the reactor facility operation and management

(b) individual responsible for site radiation safety and environmental affairs

Level 3: (a) individual responsible for daily reactor operations

(b) individual responsible for radiation protection within the license facility

Level 4: reactor operating staff.

The Nuclear Safeguards Committee shall report to the Level 1 authority. Radiation safety personnel shall report to either the Level 1 or Level 2 authority.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be within the chain of command shown in Figure 6.1. 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 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 appropriate qualifications.

6.1.3 Staffing

(1) The minimum staffing when the reactor is not secured shall be:

(a) a licensed reactor operator in the control room

(b) a second licensed reactor operator present at the reactor facility (unexpected absence for 2 hours is acceptable provided immediate action is taken to obtain a replacement)

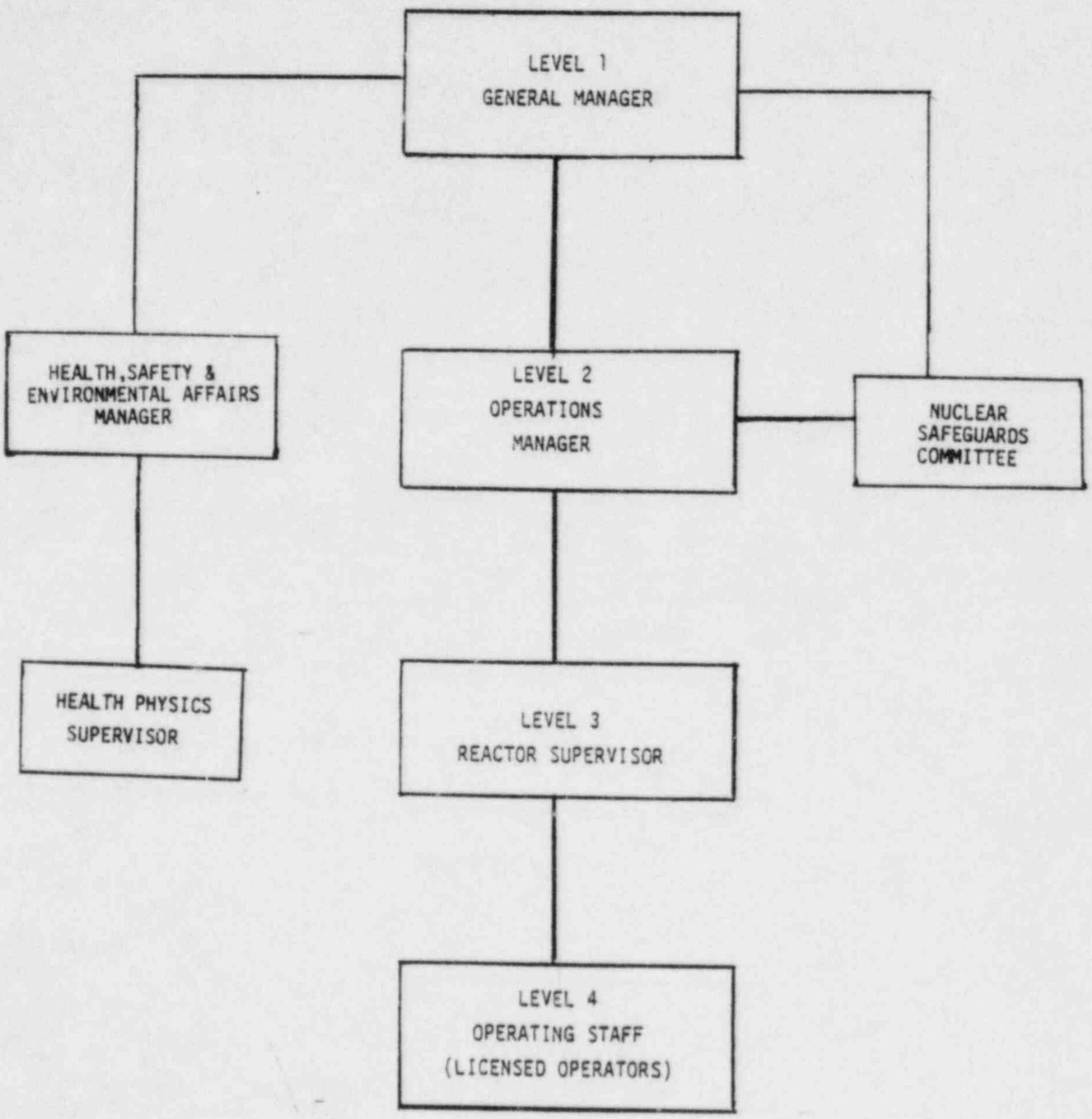

Figure 6.1 Organization structure of Union Carbide relating to reactor facility

(c) a licensed senior reactor operator readily available on call

(d) a member of the operating shift designated by Level 2 management as knowledgeable in radiation control

(2) Events requiring the presence of a senior operator:

(a) all fuel-element or control-rod alterations within the reactor core region

(b) relocations of any experiments with reactivity worth greater than or equal to one dollar

(c) recovery from unplanned or unscheduled shutdowns unless they are of a type excluded by the Level 2 authority. Such exclusions shall be posted in the control room or placed in the appropriate procedures. Furthermore, the presence of a senior operator at the facility shall not be required during recovery from unplanned or unscheduled shutdown or significant reduction in power in instances which result from:

(i) electrical power interruptions from internal or external failures exclusive of power supply failures of the reactor instrumentation, control, and safety systems;

(ii) signals which, in the opinion of the senior operator, were properly verified to be either false or readily explainable and to have resulted from monitoring, experimental, or control equipment, or from personnel error; and

(iii) intentional shutdowns made by the reactor operator which are not related to the safety of the reactor; provided that prior to the initiation of such recovery, the senior operator shall be notified of the shutdown or power reduction, and shall determine that the shutdown was caused by one of the enumerated occurrences, and shall determine that his presence at the facility during recovery is not required.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of personnel shall meet or exceed the requirements of ANS-15.4/N380 and Appendix A of 10 CFR Part 55 and be in accordance with the requalification plan approved by the Commission.

6.2 Review and Audit

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

6.2.1 Composition and Qualifications

The Nuclear Safeguards Committee shall be composed of a minimum of five 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 the operating organization but shall include members from outside the operating organization. Qualified and approved alternates may serve in the absence of regular members.

6.2.2 Charter and Rules

The committee shall function under the following operating rules:

(1) Meetings shall be held not less than semiannually or more frequently as circumstances warrant, consistent with effective monitoring of facility activities.

(2) A quorum shall consist of not less than one-half the membership, where the operating staff does not constitute a majority.

(3) Subgroups may be appointed to review specific items.

(4) Minutes shall be kept, and shall be disseminated to members and to the Level 1 authority within 1 month after the meeting.

(5) The committee shall appoint one or more qualified individuals to perform the audit function.

6.2.3 Review Function

The following items shall be reviewed by the Nuclear Safeguards Committee, or a subgroup thereof:

(1) determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question

(2) all new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance

(3) tests and experiments in accordance with Section 6.4

(4) proposed changes in technical specifications, license, or charter

(5) violations of technical specifications, license, or charter; violations of internal procedures or instructions having safety significance

(6) operating abnormalities having safety significance, and audit reports

(7) reportable occurrences listed in Section 6.6.2

6.2.4 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:

(1) 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 14 months)

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

(3) the results of actions taken to correct deficiencies 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 14 months)

(4) 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 to the Nuclear Safeguards Committee members within 90 days after the audit has been completed.

6.3 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 Nuclear Safeguards Committee and approved by the Level 2 authority 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.

(1) startup, operation, and shutdown of the reactor

(2) fuel loading, unloading, and movement within the reactor

(3) routine maintenance of major components of systems that could have an effect on reactor safety

(4) surveillance tests and calibrations required by the technical specifications or those that may have an effect on reactor safety

(5) personnel radiation protection, consistent with applicable regulations

(6) administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity

(7) implementation of the security plan and the emergency plan

Substantive changes to the above procedures shall be made only after documented review by the Nuclear Safeguards Committee and approval by the Level 2 authority 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 Level 2 authority or designated alternates within 14 days. Temporary changes to the procedures that do not affect reactor safety may be made by a senior reactor operator and are valid for a period of 1 month. Such temporary changes shall be documented and reported to the Level 2 authority or designated alternates.

6.4 Experiment Review and Approval

(1) All new experiments or classes of experiments that could affect reactivity or result in release of radioactive materials shall be reviewed by the Nuclear Safeguards 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.

(2) Before review, an experiment plan or proposal shall be prepared describing the experiment including any safety considerations. Where experiment-engineered design features are required to maintain technical specifications limits or prevent the release of radioactivity, the need for a surveillance program to monitor these engineered design features shall be included in the experiment review.

(3) Review of comments of the Nuclear Safeguards Committee setting forth any conditions and/or limitations shall be documented in Committee minutes and submitted to the Level 2 authority.

(4) All new experiments or classes of experiments shall be approved in writing by the Level 2 authority or designated alternates prior to their initiation.

(5) Substantive changes to approved experiments shall be made only after review by the Nuclear Safeguards Committee and written approval by the Level 2 authority or designated alternates. Minor changes that do not significantly alter the experiment may be approved by the Level 3 authority (reactor supervisor).

(6) Approved experiments shall be carried out in accordance with established approved procedures.

6.5 Required Actions

6.5.1 Action To Be Taken in Case of Safety Limit Violation

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the Commission.

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

(3) The safety limit violation shall be reported to the Commission in accordance with Section 6.6.2.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

    (a) applicable circumstances leading to the violation

(b) effect of the violation upon reactor facility components, systems, or structures

(c) corrective action to be taken to prevent recurrence

The report shall be reviewed by the Nuclear Safeguards Committee. A followup report describing extant activities shall be submitted to the Commission when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken in the Event of a Reportable Occurrence (as defined in Section 6.6.2(1)(a) and (c)):

(1) 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.

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

(3) All such occurrences where applicable shall be reported to the Commission in accordance with Section 6.6.2.

(4) All such occurrences, including action taken to prevent or reduce the probability of a recurrence, shall be reviewed by the Nuclear Safeguards Committee.

6.6 Reports

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

6.6.1 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 copies to the Directors of the Office of Nuclear Reactor Regulations and the Office of Inspection and Enforcement within 3 months following the end of each prescribed year. Each annual operating report shall include the following information:

(1) a narrative summary of reactor operating experience, including the energy produced by the reactor

(2) the unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence, but excluding those of the types listed in Section 6.1.3(2)(c), above

(3) tabulation of major preventive and corrective maintenance operations having safety significance

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

(5) 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) a summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended

(7) a summary of the calculated doses to a critical individual in the unrestricted area due to the airborne releases of noble gases and radioiodines

6.6.2 Special Reports (Reportable Occurrences)

(1) 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 14 days of any of the following:

(a) release of radioactivity from the reactor above allowed limits, as provided by Section 3.9.1 of these specifications

(b) violation of safety limits

(c) any of the following:

(i) operation with actual safety-system settings less conservative than the limiting safety-system settings specified in the technical specifications

(ii) operation in violation of limiting conditions for operation established in the technical specifications

(iii) a malfunction of a component in the reactor safety system 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.)

(iv) an unanticipated or uncontrolled change in reactivity greater than or equal to 1% $\Delta k/k$

(v) abnormal and significant degradation in reactor fuel, and/or cladding, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel and/or environment

(vi) 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

(2) A written report within 30 days to the Commission of:

(a) permanent changes in the facility organization structure

(b) significant changes in the transient or accident analysis as described in the Safety Analysis Report

(c) exceeding the liquid effluent limit as specified in Section 3.9.3(4), above

(3) A report within 30 days to the Commission and to New York State if the limits of items listed in Sections 3.9.2(2), 3.9.2(3), and 3.11 are exceeded

6.7 <u>Records</u>

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. Recorder charts showing operating parameters of the reactor (i.e., power level, flow, temperature, etc.) for unscheduled shutdown and significant unplanned transients shall be maintained for a minimum period of 2 years.

6.7.1 Records To Be Retained for a Period of at Least Five Years (or for the life of the component involved, whichever is smaller)

(1) 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 2 years.

(2) principal maintenance operations

(3) reportable occurrences

(4) surveillance activities required by the technical specifications

(5) reactor facility radiation and contamination surveys where required by applicable regulations

(6) experiments performed with the reactor

(7) special nuclear materials (SNM) inventories, receipts, and shipments

(8) approved changes in operating procedures

(9) records of meeting and audit reports of the Nuclear Safeguards Committee

(10) sealed source leak test results.

6.7.2 Records To Be Retained for a Period of at Least One Requalification Cycle or for the Length of Employment of the Individual, Whichever Is Smaller

(1) 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)

6.7.3 Records To Be Retained for the Lifetime of the Reactor Facility (Note: Annual reports may be used where applicable as records in this section.)

(1) gaseous and liquid radioactive effluents released to the environs

(2) offsite environmental-monitoring surveys required by the technical specifications

(3) radiation exposure of all personnel monitored

(4) updated drawings of the reactor facility

7.0 REFERENCES

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