UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

GEORGIA INSTITUTE OF TECHNOLOGY

DOCKET NO. 50-160

AMENDED FACILITY OPERATING LICENSE

Amendment No. 1
License No. R-97

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for license, as amended, filed by the Georgia Institute of Technology (the licensee) complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies and bodies have been duly made;

   B. Construction of the facility has been completed substantially in conformity with Construction Permit No. CPRR-57, as modified by CPRR-116, and the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensee is a nonprofit educational institute and will use the facility for the conduct of educational research and development activities, and has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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G. The issuance of this amended facility operating license will not be inimical to the common defense and security or to the health and safety of the public, and

H. The receipt, possession, and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23 and 70.31.

2. Facility Operating License No. R-97 issued to the Georgia Institute of Technology is hereby amended in its entirety to read as follows:

A. This license applies to the heavy-water moderated, tank-type nuclear reactor (the facility) owned by the Georgia Institute of Technology. The facility is located on the licensee's campus in Atlanta, Georgia, and is described in the application dated February 1, 1960, and subsequent amendments thereto, including the application received on March 11, 1968, and amendments thereto dated July 13, 1971, October 22, 1971, June 23, 1972, October 30, 1972, and November 13, 1972 (the application).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Georgia Institute of Technology:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use and operate the facility at the designated location in Atlanta, Georgia, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material", to receive, possess and use in connection with operation of the reactor up to 33 kilograms of contained uranium 235;

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to the Licensing of Byproduct Material", (1) to possess and use a 50-curie antimony-beryllium sealed neutron source for reactor startup, and (2) to possess, but not to separate, such byproduct material as may be produced by operation of the reactor.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50 and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The reactor shall not be operated at steady state power levels in excess of 5 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A attached hereto are hereby incorporated in this license. The licensee shall operate the reactor in accordance with these Technical Specifications.

D. This amended license is effective as of its date of issuance and shall expire at midnight, June 6, 1994.

FOR THE ATOMIC ENERGY COMMISSION

Karl R. Goller

Karl R. Goller
Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Appendix A - Technical Specifications

Date of Issuance: June 6, 1974

APPENDIX A

TO FACILITY LICENSE NO. R-97

TECHNICAL SPECIFICATIONS

FOR THE

GEORGIA TECH RESEARCH REACTOR

DOCKET NO. 50-160

GEORGIA INSTITUTE OF TECHNOLOGY

DATE: June 6, 1974

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1.0 DEFINITIONS

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

1.1 Safety Channel - A safety channel is a measuring channel in the reactor safety system.

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

1.3 Operable - Operable means a component or system is capable of performing its intended function in its normal manner.

1.4 Operating - Operating means a component or system is performing its intended function in its normal manner.

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

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

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

1.8 Unscheduled Shutdown - An unscheduled shutdown is defined as any unplanned shutdown of the reactor, after startup has been initiated, caused by actuation of the reactor safety system, operator error, equipment malfunctions, or a manual shutdown in response to conditions which could adversely affect safe operation.

1.9 Reactor Shutdown - Reactor shutdown means that the shim-safety blades are fully inserted and the control rod power is off. The reactor is considered to be operating whenever this condition is not met and there are six or more fuel elements loaded in the core.

1.10 Reactor Secured - Reactor secured is defined as follows:

a. The reactor is shutdown as defined in Definition 1.9.
b. Subcriticality of the cold xenon free core by at least one dollar has been confirmed.
c. No operation is in progress which involves moving fuel elements within the reactor vessel, the insertion or removal of experiments from the core, or control rod maintenance.

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

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

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

1.14 Abnormal Occurrence - An abnormal occurrence is any of the following:

a. A safety system setting less conservative than the limiting setting established in the Technical Specifications.
b. Operation in violation of a limiting condition for operation established in the Technical Specifications.
c. A safety system component malfunction or other component or system malfunction which could, or threatens to, render the safety system or the engineered safeguard systems incapable of performing their intended safety functions.
d. Release of fission products from a failed fuel element.
e. An uncontrolled or unplanned release of radioactive material from the restricted area of the facility.
f. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Appendix B, Table 1 of 10 CFR 20.
g. An uncontrolled or unanticipated change in reactivity in excess of 0.005 $\Delta k/k$.
h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the plant.

1.15 Experiment - An experiment, as used herein, is any of the following:

a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;
b. An evaluation or test of a reactor system operational, surveillance, or maintenance technique;
c. An experimental or testing activity which is conducted within the confinement or containment system or the reactor; or
d. The material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

1.16 Experimental Facility - An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

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

1.18 Movable Experiment - A movable experiment is one which may be inserted, removed, or manipulated while the reactor is critical.

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

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

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

1.21 Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, buoyant, pneumatic, or other forces which are normal to the operating environment of the experiment, or which might arise as a result of credible malfunctions.

1.22 Unsecured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 1.21 above. Moving parts of experiments are deemed to be unsecured when they are in motion.

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

1.24 Fast Scram - A fast scram is the spring assisted gravity insertion of all shim-safety blades that begins within 100 milliseconds after introduction of a delay scram signal into the safety system.

1.25 Delay Scram - A delay scram is the spring assisted gravity insertion of all shim-safety blades that begins within 10 seconds after introduction of a delay scram signal into the safety system.

1.26 Containment Integrity - Containment integrity exists when all of the following conditions are met:

a. One door on each personnel airlock is closed and sealed.
b. The truck door is closed and sealed.
c. Controls, equipment and interlocks for isolation of the containment building are operable or the containment is isolated.

1.27 Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

1.28 Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 SAFETY LIMITS IN THE FORCED CONVECTION MODE

APPLICABILITY

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the steady state with forced convection flow. The variables are reactor thermal power, reactor coolant flow, reactor coolant inlet temperature, and the moderator level in the reactor vessel.

OBJECTIVE

To maintain the integrity of the fuel element cladding and prevent the release of significant amounts of fission products.

SPECIFICATION

a. The reactor power shall not exceed the limit specified in Figure II-1 corresponding to values of reactor coolant flow.
b. The reactor coolant inlet temperature shall not exceed 123°F
c. The moderator level shall be at overflow.

BASIS

Gross fuel element failure and concomitant fission product release will not occur unless there is departure from nucleate boiling. The integrity of the fuel element cladding can be assured by control of the reactor power, the reactor coolant flow rate, and reactor coolant outlet (or inlet) temperature.

The basis for establishing the safety limits on reactor power, coolant flow, and outlet temperature is a thermal hydraulic analysis to calculate the values of these parameters at which departure from nucleate boiling occurs.

This analysis[1] establishes that departure from nucleate boiling will not occur at power levels up to 11.5 MW with the coolant outlet temperature and coolant flow at their respective limiting safety system settings. The analysis is not extended below 760 GPM because the orifices are not designed for extremely low flow.

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(1) Letter, R. S. Kirkland to USAEC, June 23, 1972, Enclosure 5.

2.1.2 SAFETY LIMITS IN THE NATURAL CONVECTION MODE

APPLICABILITY

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

SPECIFICATION

The reactor thermal power shall not exceed two (2) kW.

BASIS

Experience with the GTRR has shown that no damage to the core and no boiling occurs without forced convection coolant flow at power levels up to two kW.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 LIMITING SAFETY SYSTEM SETTINGS IN THE FORCED CONVECTION MODE

APPLICABILITY

Applies to the settings of those instruments monitoring the safety limits.

OBJECTIVE

To assure automatic protective action is initiated before a safety limit is exceeded.

SPECIFICATION

The safety system trip settings shall be as follows:

Thermal Power	5.5MW
Reactor Coolant Flow	1625 GPM
Reactor Outlet Temperature	139 °F

BASIS

The trip settings are chosen so that the reactor is operated with no incipient boiling. An analysis was made showing that at 1800 gallons per minute total coolant flow, five MW thermal power an inlet reactor coolant temperature of 114°F and the application of all the engineering uncertainty factors, a maximum fuel surface temperature 8°F less than the local D_2O saturation temperature might occur.[1]

REFERENCE

(1) Letter, R. S. Kirkland to USAEC, October 22, 1971, Response No. 10.

2.2.2 LIMITING SAFETY SYSTEM SETTINGS IN NATURAL CONVECTION MODE

APPLICABILITY

Applies to the values of safety system settings when operating in the natural convection mode.

OBJECTIVE

To assure the reactor is not operated at a power level sufficient to cause fuel damage.

SPECIFICATION

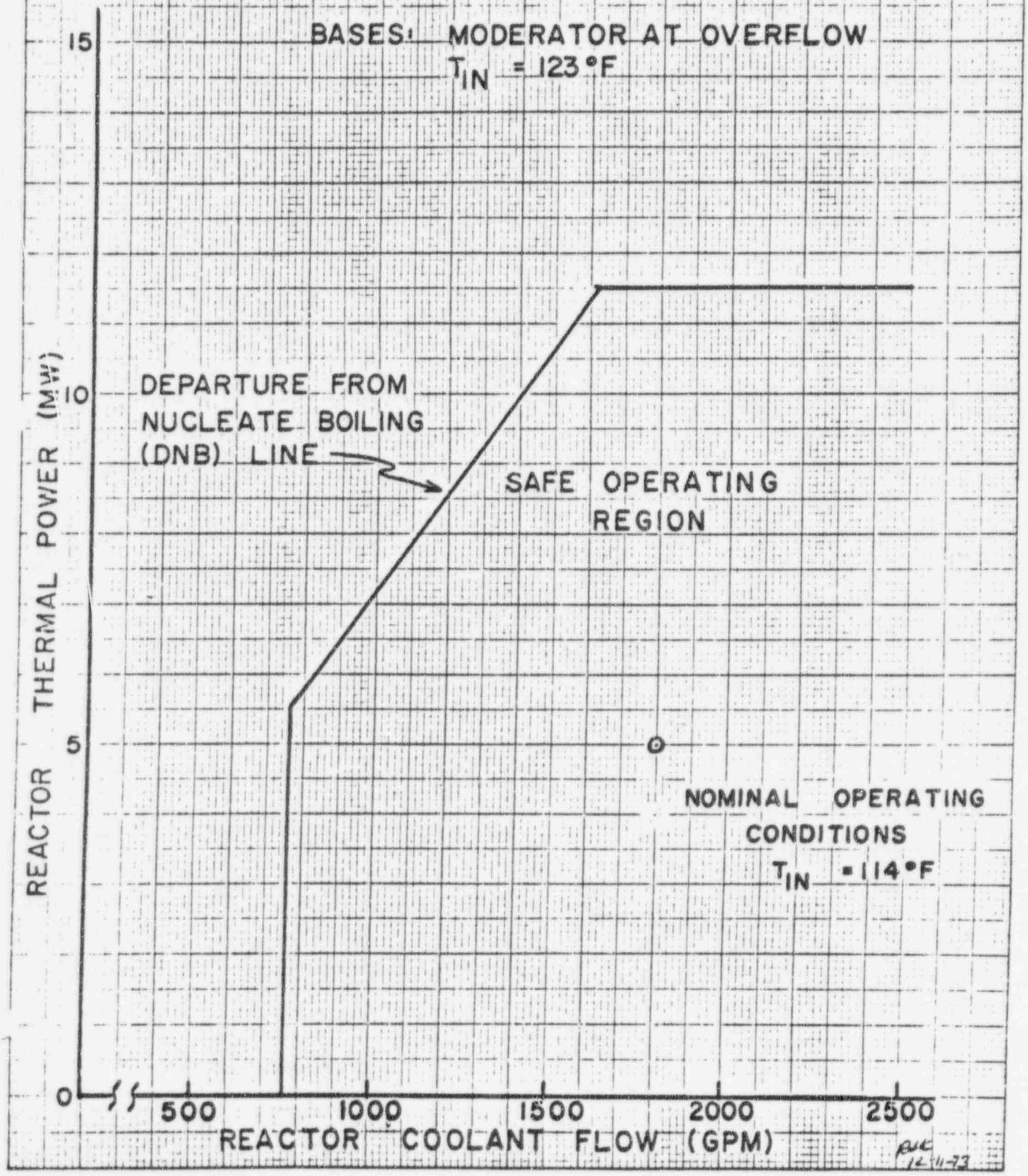
The reactor thermal power safety system setting shall not exceed 1.1 kW when operating in the natural convection mode.

BASIS

In the natural convection mode of reactor operation the main coolant pumps are not operating. The reactor isolation valves may be closed so that only internal, natural convection is available for cooling. Experience with the GTRR has shown that the reactor can be operated at one kW indefinitely without exceeding a bulk reactor temperature of 123°F.

FIG. II-1

GTRR SAFETY LIMIT FOR FORCED CONVECTION



3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTIVITY LIMITS

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

a. The shutdown margin relative to the cold xenon free critical condition shall be at least 0.01 $\Delta k/k$ with the most reactive shim-safety blade and the regulating rod fully withdrawn.
b. The reactor shall be subcritical by more than 0.0275 $\Delta k/k$ during loading changes.
c. A shim-safety blade shall not be removed from the core if the shutdown margin is less than 0.01 $\Delta k/k$ with the most reactive remaining shim-safety rod fully withdrawn.
d. Each shim-safety blade shall be withdrawn at least 10° above the fully inserted position prior to criticality.

BASIS

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

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

The restriction on shim blade position for criticality assures that, in the event of a shim blade failure which results in the shim blade passing through its normal insertion limit to a position which results in a positive reactivity insertion, a negative period will be generated by the first 10° insertion that will cause the three remaining shim-safety blades to scram.

3.2 REACTOR SAFETY SYSTEM

APPLICABILITY

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

OBJECTIVE

To specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

SPECIFICATION

The reactor shall not be made critical unless:

a. The reactor safety systems and safety related instrumentation are operable in accordance with Tables 3.1 and 3.2 including the minimum number of channels and the indicated maximum or minimum set points.
b. All shim-safety blades are operable.
c. The most recent drop time of each shim-safety blade is less than 0.5 second from fully withdrawn to 90% worth inserted.
d. The delay time from the introduction of a fast scram signal into the safety system to the release of the shim-safety blades is less than 100 milliseconds.
e. At power levels greater than 50 kW the shim-safety blades shall be banked within 5°.

BASIS

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

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

TABLE 3.1

REQUIRED SAFETY CHANNELS

Channel	Setpoint	Minimum No. Required	Function
Start up	2 cps	1[a]	Minimum countrate permissive rod withdrawal interlock
Period trip	$\leq$10 sec (pos or neg)	2[c]	Scram
Power trip	5.5 MW	2[c]	Scram
Low D_2O flow	1625 gpm	2[b][c]	Scram
High D_2O Temperature	139°F	2[c]	Scram
Low D_2O Level	$\leq$12" below over flow	2[c]	Isolate reactor vessel Scram Initiate ECCS
No D_2O Overflow	-	1	Scram
Manual scram	-	1	Scram
Reflector drain	-	1	Backup scram
Containment doors open	-	1 per airlock	Scram
Reactor isolation valves closed	-	2[c] per valve	Scram

[a] Required during startup and for operation with less than 1 decade overlap between the startup channel and the pico-ammeter channel.

[b] Not required for natural convection operation

[c] One of the twelve required safety channels may be bypassed for a period not to exceed 8 hours for test, repair, or calibration

The primary coolant flow rate scrams and provides redundant channels to assure, when the reactor is at power levels which require forced flow cooling, that an automatic shutdown of the reactor will occur to prevent exceeding a safety limit if sufficient flow is not maintained.

Two redundant D_2O temperature scrams provide automatic protection against exceeding the core inlet temperature safety limit in the event of a failure of the secondary coolant system to adequately dissipate the heat removed from the primary coolant system.

A loss-of-coolant accident is detected by two low D_2O level scrams which also isolate the reactor vessel to prevent coolant loss through the primary coolant piping and initiate the emergency core coolant system.

In addition to the automatic protective systems, the manual scram and the reflector drain provide backup methods to shut the reactor down by operator action. The reflector drain provides a shutdown capability of 2.75% $\Delta k/k$.

The containment doors open scram assures that the reactor is not made critical with both doors open on an airlock.

Two switches which sense the position of each of the reactor isolation valves initiate a reactor scram upon closure of these valves to prevent damage to the cladding due to a loss of coolant flow.

The pico ammeter channel provides a narrow range indication of reactor power level over all ranges of reactor operation above the range of the start up channel thereby assuring that the operator has an accurate monitoring channel available for comparison to the safety channels.

The building area radiation monitors assure that areas throughout the facility in which high radiation areas could occur due to improper sample handling, equipment or sheilding movements, etc.

The gas monitor, filter assembly monitor, Kanne chamber and the particulate monitor provide diverse and redundant channels which monitor particulate and gaseous releases from the reactor building. These monitors provide readout and alarm functions in the control room and initiate a containment isolation in the event that their present alarm points are exceeded. In addition the D_2O leak

TABLE 3.2

SAFETY RELATED INSTRUMENTATION REQUIRED FOR OPERATION

Instrumentation	Setpoint	Minimum No. Required	Function
Pico ammeter channel	-	1	Linear power level measurement and input for the automatic control mode
High building radiation	$\leq$10 mr/hr or 2x normal 5Mw Background	5(a)	Alarm and prevents startup
Gas monitor		1(b)	Initiates containment isolation
Filter assembly monitor		1(d)	Initiates containment isolation
Kanne chamber		1(b)(c)	Initiates containment isolation
D_2O Leak detection system	-	1(c)	Initiates containment isolation
Particulate monitor		1(d)	Initiates containment isolation
Emergency cooling tank level	$<$280 gal.	2(e)	Alarm and prevents startup

(a) Area monitors shall be located on the experimental level, the reactor top, in the reactor basement, and in an area that will be allow changes in reactor coolant radioactivity to be detected.

(b)(c)(d)(e) Either channel may be bypassed for a period not to exceed 8 hours for test, repair or calibration.

detector system senses small leaks in the primary coolant system and provides an anticipatory building isolation to prevent the release of tritium.

The emergency cooling tank level alarm provides redundant channels which alarm if there is an insufficient quantity of D_2O available in the emergency cooling system tank.

Requiring that at power levels greater than 50 kW all shim-safety blades are banked within 5° of each other assures that peaking factors less conservative than those assumed in Safety Analyses Report do not occur due to uneven shim blade placement.

Specifications 3.2.b through 3.2.d assure that the safety system response will be consistant with the assumptions used in evaluating the reactor's capability to withstand the design basis accident in the Safety Analysis Report.

3.3 CONTAINMENT BUILDING

APPLICABILITY

This specification applies to the GTRR containment building requirements.

OBJECTIVE

To minimize the release of airborne radioactive materials from the GTRR.

SPECIFICATION

Containment integrity shall be maintained when any of the following conditions exist:

a. The reactor is operating.
b. Maintenance or operational activities which could change core reactivity are in progress.
c. Movement of irradiated fuel is in progress, except movement of irradiated fuel contained in the fuel transfer cask to or from the containment building.
d. The reactor has been shutdown from a power level greater than 1 MW for less than eight hours.

BASIS

Building containment is a major engineered safety feature which serves as the final physical barrier to contain radioactive particles and gases following an accident. Containment integrity is therefore required during all operations which could result in significant radioactive releases.

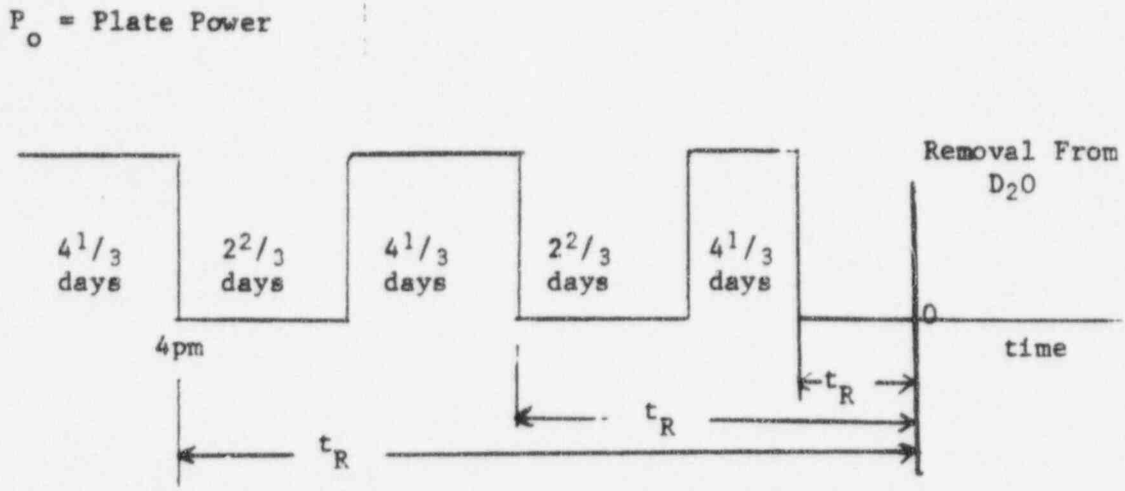
Any maintenance or operational activities that change the core reactivity could reduce the shutdown margin of the reactor. Therefore containment integrity is required during such activities.

The movement of irradiated fuel containing significant fission product inventories poses a potential hazard should its clad fail. Therefore containment integrity is required except when the transfer cask is being moved to or from the containment building.

Fuel melting and the subsequent release of fission products could result from a loss-of-coolant accident following reactor shutdown if sufficient decay heat is present. Containment integrity is therefore required until such time as the decay heat generation rate is less than that required to melt the fuel plates. A limit of 450°C was set as the upper value for a fuel element plate temperature to preclude melting of the plates. The decay time needed to assure that this temperature would not be reached was calculated.

The analysis used a method developed and tested at MIT.(1) It was carried out on a fuel element which has only natural convective cooling applied in the transfer cask and in the reactor vessel. The analysis assumes a standard operating history as shown below and relies upon experimentally-proven values for the product of convection heat transfer coefficient, h, and fuel element heat transfer area, A. It is applied to a fuel element which has been subjected to a flux peaking factor of 1.5 during operation and is subsequently either removed after some cooling period, t_R, or else is subjected to a loss of coolant in the reactor vessel.

Standard Operating History



The results for loss of coolant from the reactor vessel after eight hours showed a maximum plate temperature of 426°C. This maximum occurred 45 minutes after loss of coolant. This is for the case where the element remains in the vessel with its large convection volume, but is subjected to gamma heating from adjacent elements.

For the more confined heat transfer situation, without gamma rays from other elements, but with a restricted heat transfer volume, the maximum after a 12 hour cooldown is 361°C, occurring one hour after removal. This is the waiting time proposed for fuel element transfers out of the reactor.

It is therefore concluded that a fuel element in the reactor will not melt if it has been cooled for eight hours following operation at reactor power levels greater than one MW. If the reactor has not operated at power levels in excess of one MW, this eight hour cooling time is not required. This is because the fuel elements will not reach temperatures high enough to melt the clad should a loss-of-coolant accident occur.[2] Therefore containment integrity is not necessary.

REFERENCES

(1) MITR Operations Memo No. 98 dtd. December 25, 1965.

(2) Final Safeguards Report for the GTRR, February 1963, Section 8.3, pp. 106-111.

3.4 LIMITATIONS OF EXPERIMENTS

APPLICABILITY

These specifications apply to experiments performed at the GTRR.

OBJECTIVE

To prevent damage to the reactor and to limit radiation dose to facility personnel and the public in the event of experiment failure.

SPECIFICATIONS

a. The potential reactivity worth of each secured removable experiment shall be limited to 0.015 $\Delta k/k$.
b. The magnitude of the potential reactivity of each unsecured experiment shall be limited to 0.004 $\Delta k/k$.
c. The rate of change of reactivity of any unsecured experiment, any movable experiment, or any combination of such experiments having a total reactivity worth in excess of 0.0025 $\Delta k/k$ introduced by intentionally setting the experiment(s) in motion relative to the reactor shall not exceed 0.0025 $\Delta k/k$-sec.
d. The sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist shall not exceed 0.015 $\Delta k/k$.
e. The surface temperature of the material which bounds or supports any experiment shall not exceed the lowest of the following, where applicable:
   (1) The saturation temperature of liquid reactor coolant at any point of mutual contact.
   (2) A temperature consevatively below that at which the corrosion rate of the boundary material at any surface would lead to its failure, or,
   (3) A temperature conservatively below that at which the strength of the boundary material would be reduced to a point predictably leading to failure.
f. Materials of construction and fabrication and assembly techniques utilized in experiments shall be so specified and used that assurance is provided that no stress failure can occur at stresses twice those anticipated in the manipulation and conduct of the experiment or twice those which could occur as a result of unintended but credible changes of, or within, the experiment.

g. The radioactive material content, including fission products, of any singly encapsulated experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR Part 20. This dose limit applies to persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.
h. The radioactive material content, including fission products, of any doubly encapsulated or vented experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 rem to the whole-body or 1.5 Rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of five rem to the whole-body or 30 Rem to the thyroid.
i. Explosive materials in excess of 25 milligrams of TNT equivalent shall not be irradiated or stored within the reactor containment building.
j. Explosive materials in amounts up to 25 milligrams TNT equivalent may be irradiated and stored within the containment only if they are encapsulated in such a manner to assure compliance with Specification 3.4.f in the event of detonation of the explosive material.
k. Experiments which could increase reactivity by flooding, shall not remain in or adjacent to the core unless measurements are made to assure that the shut down margin required in Specification 3.1.a would be satisfied after flooding.

BASIS

Limiting the potential reactivity worth of secured removable experiments to 0.015 $\Delta k/k$ assures that any transient arising from the instantaneous removal of such experiments will not result in cladding failure and concomitant release of radioactive material which could lead to doses in excess of the limits set forth in 10 CFR Part 20.

A positive step change caused by the ejection or insertion of unsecured experiments worth less than 0.004 $\Delta k/k$ would not result in a transient behaviour exceeding the Safety Limits established in Section 2.1 of these Specifications.

Manipulations of movable experiments within the limits established in Specification 3.4.c will result in asymptotic periods longer than 20 seconds. Periods of this magnitude are easily accommodated by automatic response of the reactor safety system or by operator action. Prior to the manipulation of movable experiment the reactor power level will be reduced as required to accomodate the calculated prompt jump associated with the step insertion of the potential reactivity worth of the experiment.

Conformance with Specification 3.4.d assures that common mode failures resulting in the insertion of the total reactivity worth of all unsecured experiments will not result in accident consequences more severe than those evaluated for the failure of a single secured experiment.

Specifications 3.4.3 and 3.4.f provide assurance that experiments will not fail due to the pressure or temperature effects of operation under anticipated operating conditions. For the purposes of this specification the reactor shall be assumed to be operating at the Limiting Safety System Settings established in Section 2 of these Technical Specifications.

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

Adherence to Specification 3.4.i will prevent large quantities of explosives from being present within the reactor containment building and thereby preclude damage to the safety system and safety related equipment. Small quantities of explosive material may be safely used and stored as long as the encapsulation used has been shown to withstand the detonation of twice the quantity of explosive to be used.

Specification 3.4.k assures that the shutdown margin required by Specification 3.1.a will be met in the event of a positive reactivity insertion caused by the flooding of an experiment.

3.5 RADIOACTIVE EFFLUENTS

APPLICABILITY

This specification applies to the controlled release of radioactive liquids and gases from the reactor site.

OBJECTIVE

To define the limits and conditions for the release of radioactive effluents to the environs to assure that any radioactive releases are as low as practicable and would not result in radiation exposures greater than a few percent of natural background exposures and, in any event, within the limits of 10 CFR Part 20 for instantaneous release rates.

SPECIFICATIONS

a. Liquid Effluents

(1) The concentration of gross radioactivity, above background, in liquid effluents discharged from the Reactor Building to the sanitary sewer shall not exceed 3×10^{-6} µCi/ml, excluding tritium, unless the discharge is controlled on a radionuclide basis in accordance with Appendix B, Table II, Column 2 and Note 1 thereto of 10 CFR Part 20.

(2) The concentration of tritium in liquid effluents discharged to the sanitary sewer shall not exceed 1×10^{-1} µCi/ml.

(3) If any of the limits of Specification a(1) or (2) are exceeded, normal orderly shutdown of the liquid waste system shall be initiated and liquid discharge from the facility shall not be resumed until the cause of the excessive discharge rate is identified and corrected.

(4) The annual total quantity of gross radioactivity to be released in liquid effluents from the reactor facility shall not exceed one curie to the sanitary sewerage system.

(5) During release of liquid radioactive effluents, two independent samples of each tank shall be taken, one prior to release and one during release. An independent sample shall be taken from the discharge line during release.

(6) Equipment installed for the control and treatment of liquid effluents shall be maintained and operated.

b. Gaseous Effluents

(1) The maximum release rates of gross radioactivity in gaseous effluents shall not exceed 585 μCi per second of Ar-41 equivalent.
(2) The maximum release rate of I-131 shall not exceed 8.3×10^{-4} μCi/sec.
(3) The maximum release rate of particulates with half lives longer than eight days in gaseous effluents shall be limited in accordance with the following equation:

$$Q \leq 2.7 \, \overline{MPC_a} \quad (Ci/sec)$$

where Q is the release rate (Ci/sec) of particulates with half lives longer than eight days and $\overline{MPC_a}$ is the composite maximum permissible concentration in air as defined in Appendix B, Table II, Column 1 and Notes thereto of 10 CFR Part 20.
(4) If the maximum release rate for any of the above gaseous effluents is exceeded, normal orderly shutdown of the gaseous waste system shall be initiated and gaseous discharge from the facility shall not be resumed until the cause of the excessive discharge is identified and corrected.
(5) During release of gaseous radioactive effluents, the following conditions shall be met:
   (a) One of the gross radioactivity monitors, the charcoal filter cartridge and particulate monitor shall be operable.
   (b) Both gross radioactivity monitors shall be set to alarm and automatically isolate the gaseous waste releases prior to exceeding the release rates in Specification b.1.
(6) When the containment building is not isolated, at least one exhaust effluent monitoring channel with readout in the control room shall be operable and capable of initiating building isolation. The time from initiation of closure to isolation valve closure shall not exceed five seconds.

BASIS

a. Liquid Effluents

The liquid waste handling system is described in the SAR dated December 1967. Radioactive effluents released to the sewage on the basis of gross radioactivity are assumed not to contain I-129 and radium. The maximum amount of tritium in the discharge is limited to the value given in 10 CFR 20. The total quantity of radioactivity limit is in accord with 10 CFR 20 for disposal to a sewage system. The independent samples taken prior to and during liquid effluent release shall determine the radioactivity concentration in the liquid released from the tanks and the radioactivity concentration in the discharge line to the sanitary sewers. The equipment installed for control and treatment of liquid effluents shall be maintained and operated as required by 10 CFR 50.

b. Gaseous Effluents

The release-rate limit for gross radioactivity takes into account onsite meteorological data developed by the licensee and diffusion assumptions appropriate to the site. The AEC Staff determined the annual average diffussion parameters (X/Q) to be 5.2×10^{-4} sec/m^3 in the most critical sector, ENE at 40 meters. The method utilized by the AEC staff is described in Sections 7.4 and 7.5 of "Meteorology and Atomic Energy - 1968," equation 7.21 being used for the beta dose and equation 7.63 being used for the gamma dose (whole body). Based on these calculations using Ar-41 as the primary dose contributor, the skin dose due to the gamma plus beta was determined to be controlling rather than the whole body dose from the gamma. A maximum release rate limit of gross radioactivity in the amount of 585 μCi/sec will not result in annual doses to unrestricted areas in excess of the limits specified in 10 CFR 20.

The release rate limit for iodines and particulates with half lives longer than eight days takes annual average atmospheric dilution into account and ensures that at any point on or beyond the restricted area fence the requirements of 10 CFR 20 will be met. The limit is based on the annual average diffusion value of X/Q which is 4.1×10^{-4} sec/m^3, for the 22.5° sector having the least diffusion on an annual average. The release rate for I-131 was determined on the basis of the method given in Regulatory Guide 1.42, the nearest worst real cow as permitted by this guide was not utilized but may be applied if appropriate data is collectod for determining the nearest real cow. The formula prescribed in the specification for release of particulates with half lives greater than eight days takes into consideration the additional reduction of MPC limits by the factor of 700 for possible ecological chain effects.

Isolation of the exhaust effluent stack is initiated by high radiation in the off-gas system. Such isolation is required for abnormally high gross radioactivity releases either due to abnormal reactor operation or reactor accident.

3.6 PRIMARY COOLANT SYSTEM

APPLICABILITY

This specification applies to the limiting conditions for the primary coolant system pH, resistivity, flow distribution, level and D_2 concentration.

OBJECTIVE

To assure adequate reactor core cooling and to protect the integrity of the primary coolant system.

SPECIFICATIONS

The reactor shall not be critical unless:

a. The primary coolant pH is between 4.5 and 7.5.
b. The primary coolant resistivity is at a value greater than 200,000 ohm-cm except for periods of time not to exceed seven days when the resistivity may fall to 70,000 ohm-cm.
c. All grid positions contain fuel elements, grid plugs or experimental facilities for operation in the forced convection mode.
d. The reactor vessel coolant level is less than four inches below the overflow standpipe level, except for operation at power levels up to 1.0 kW in the natural convection mode.
e. The D_2 concentration in the helium sweep system is less than 2% by volume.
f. The concentrations of radioactive materials in the secondary coolant system are less than the values listed in 10 CFR 20, Appendix B, Table II, Column 2.

BASIS

Experience at GTRR and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in Specification 3.6.a and 3.6.b will minimize the amount and severity of corrosion of the aluminum components of the primary coolant system and the fuel element cladding.

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

The limiting value for reactor vessel coolant level is somewhat arbitrary since the core is in no danger of melting so long as it is covered. However, a drop of vessel coolant level indicates a malfunction of the reactor system and possible approach to uncovering the core. Therefore a measurable value, well above the core is chosen. Determination of the worth of the top reflector drain requires operation of the reactor at low power levels without a restriction on reactor vessel coolant level.

Because of radiolytic disassociation of D_2O, deuterium and oxygen gas will build up in the helium gas sweep system. If this D_2 gas were to reach an explosive concentration (about 7.8% by volume at 25°C in helium),[1] damage to the primary system could occur. To assure a substantial margin of safety, a limit of 2% is set.

REFERENCES

(1) "Flammability of Deuterium in Oxygen-Helium Mixtures," USAEC Report No. TID 20898, Explosives Research Center, Bureau of Mines, June 1964.

3.7 EMERGENCY COOLING SYSTEM

APPLICABILITY

These specifications apply to the emergency core cooling system.

OBJECTIVE

To assure that the fuel elements are adequately cooled to prevent fission product release in the event of loss of primary coolant from the reactor vessel.

SPECIFICATIONS

The reactor shall not be operated at power levels in excess of one MW unless:

a. The D_2O Emergency Coolant System is operable.
b. A source of make-up water to the Emergency Coolant Tank is available.
c. The water level in the irradiated fuel storage pool is within 12 inches or less of normal overflow.

BASIS

In the event of a loss of coolant accident on this reactor, the emergency coolant system provides a means of retaining sufficient cooling water in the reactor vessel or, if necessary, a means of supplying additional cooling water to prevent melting of the reactor core and the associated release of fission products.[1][4] Therefore operability of this system is required for reactor operation. The Emergency Coolant System Tank provides supply of D_2O to the core spray nozzles for 30 minutes. It is then necessary to supply makeup water to the tank from the city water supply to the Nuclear Research Center building. Therefore this source of water is also required for reactor operation. Should city water not be available, cooling water from the irradiated fuel storage pool will be used. For this reason, a minimum level in the pool is required.

When the reactor is operated at one MW or less, the emergency cooling system is not required because the fuel can dissipate the decay heat with only natural convection cooling in air.[2][3]

REFERENCES

(1) Safety Analysis Report for the 5 MW GTRR, GT-NE-7, December 1967, Section 4.4.8.3, pp. 79-81.

(2) Final Safeguards Report for the GTRR, February 1963, Section 8.3, pp. 106-111.

(3) Thompson, T. J. and Beckerley, (eds.), The Technology of Nuclear Reactor Safety, Volume I, p. 692, The MIT Press, Cambridge, Massachusetts, 1964.

(4) Response to Question D.1, Letter to USAEC, Docket 50-160, dated 6/23/72.

3.8 FUEL HANDLING AND STORAGE

APPLICABILITY

Applies to the handling and storage of fuel elements.

OBJECTIVE

To prevent inadvertant criticality outside of the reactor vessel and to prevent overheating of irradiated fuel elements.

SPECIFICATIONS

a. All fuel elements outside of the reactor shall be stored and handled such that the calculated k_{eff} is less than 0.85 under optimum conditions of water moderation and reflection.
b. No more than four unirradiated fuel elements shall be together in any one room outside of the reactor, shipping containers or fuel storage racks.
c. An irradiated fuel element shall not be removed from the reactor within 12 hours of a reactor shutdown from a power level in excess of one MW.

BASIS

Criticality of stored or handled fuel elements outside of the reactor can be prevented if the fuel elements are maintained in a geometry that assures an adequate margin below criticality. This margin is established as a k_{eff} of 0.85.

The irradiated fuel storage racks in the storage pool will accommodate up to 40 fuel elements stored in a linear array along the pool walls. Experiments at ORNL have demonstrated the sub-criticality of such an array of similar elements.(1)

Fresh fuel elements will be stored in the fuel storage vault. Calculations have indicated that the k_{eff} of an array of fresh elements in a flooded condition to be less than 0.85.(2)

Calculations have shown that four unirradiated fuel elements cannot achieve criticality.(3) Therefore, grouping of fresh fuel outside of the reactor, shipping containers or normal storage will be limited to this number.

An analysis was made to determine the time-fuel temperature relationship that occurs following removal of a fuel element from the core into the fuel transfer flask. These results, detailed in the basis for specification of 3.3 indicate that the maximum fuel plate temperature reached following 12 hours of cooling before removal into the flask will be 361°C. This provides a sufficient margin to assure that no fuel plate melting will occur.

REFERENCES

(1) Final Safeguards Report for the GTRR, February 1963, Section 7.7, pp. 96-7.

(2) See correspondence relative to Change No. 10 to the Technical Specification of Operating License No. R-97, Docket 50-160.

(3) Safety Analysis Report for the 5 Mw GTRR, GT-NE-7, December 1967, Section 4.4.10.

4.0 SURVEILLANCE REQUIREMENTS

4.1 REACTIVITY LIMITS

APPLICABILITY

This specification applies to the surveillance requirements for reactivity limits.

OBJECTIVE

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

SPECIFICATIONS

a. Shim-safety blade reactivity worths shall be measured and the shutdown margin calculated annually and whenever a core configuration is loaded for which shim-safety blade worths have not been measured.
b. The reactivity worth of experiments inserted in the GTRR shall be measured during the first startup subsequent to the experiments insertion, and shall be verified if core configuration changes cause increases in experiment reactivity worth which may cause the experiment worth to exceed the values specified in Specification 3.4.

BASIS

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

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

4.2 REACTOR SAFETY SYSTEM SURVEILLANCE

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATIONS

a. The channels listed in Tables 4.1 and 4.2 shall be tested and calibrated as indicated.
b. A channel check of the power trip channels and the pico-ammeter channels, comparing the channel outputs to a heat balance, shall be made weekly when the reactor is operated at a power level at or above one Mw and after the installation of a new core configuration.
c. The drop time and withdrawal rate of each shim-safety blade shall be measured monthly.
d. The withdrawal and insertion rate of the regulating rod shall be measured monthly.
e. The charcoal cartridge sampler on the containment building exhaust shall have a radioisotopic analysis performed bi-weekly.
f. Grab samples of the exhaust stack effluent shall be obtained and have a radioisotopic analysis performed monthly.

BASIS

Calibration of the safety system and other safety related instrumentation means to measure the performance as guided by vendors instructions and performance specifications of the instrument in its response to accurately prescribed input signals. Past experience and maintenance records attest to the reliability of these channels, and justifies the calibration frequency in Tables 4.1 and 4.2.

Instrument channel tests are made to obtain assurance that the critical parameter trip channels(1) are operating correctly. These tests involve the observation of trip operation in both

TABLE 4.1

SURVEILLANCE REQUIREMENTS FOR REACTOR CONTROL AND SAFETY SYSTEMS

	Surveillance Requirements		
Channel	Test Prior to Startup (1)	Test Weekly	Calibrate Semi-annlly
1. Power trip	x	x	x
2. Period trip	x	x	x
3. Start up channel	x	x	x
4. Logic and magnet amplifier channel	x	x	x
5. Pico ammeter channels			x
6. Reactor D_2O level channels		x	x
7. D_2O temperature channels		x	x
8. D_2O flow rate channels		x	x

(1) Performed if reactor has been shutdown for eight or more hours or any listed system has been deenergized for one or more hours.

TABLE 4.2

SURVEILLANCE REQUIREMENTS FOR SAFETY RELATED INSTRUMENTATION

Channel	Surveillance Requirement			
	Daily Check(1)	Weekly Test	Source Calibration Monthly	Known Parameter Source Calibration Annually
Kanne exhaust gas	x		x	x
GM gas monitor	x	x	x	x
Moving filter particulate	x		x	NA
Cooling water gamma monitor	x	x	x	NA
Area radiation monitors	x	x	x	NA

(1) Applicable only when reactor operation staff is on duty.
NA = Not Applicable

the electronic and the electromechanical circuits of the trip channels. Loss of control blade magnetic clutch current shall be the indicator of correct trip operation. The testing frequency as prescribed in Tables 4.1 and 4.2 has been shown to be adequate.

In addition, an operating cycle begins in accordance with written startup procedures including a check-list to establish that all instrumentation channels are operable and scrams and important interlocks have been tested prior to startup. The eight-hour time limit in Table 4.1 permits reactor restart following a brief shutdown for maintenance or because of a spurious scram without retesting instrumentation unrelated to the scram or maintenance. Instrument drift is not significant during this period of time if the instruments remain energized. If an instrument is deenergized for an hour or more, or is repaired, it is considered prudent to test it before it is returned to service. Furthermore, redundant instrumentation is provided and channel performance can readily be checked by intercomparison.

A check of the readings of the neutron channels can readily be accomplished by comparing them to the results of a primary system heat balance. Because the primary coolant temperature rise across the core is small, these measurements will be performed with the reactor power level at or above one Mw to improve the accuracy of the measurements. Based on past experience monthly thermal power calibration is justified.

The shim-safety blades are provided to control large amounts of reactivity and to insure adequate shutdown margin. The reactivity worth of the shim-safety blade can vary due to changes in core configuration, changes in the number and type of experiments, or burnup of the neutron absorber. The absolute reactivity worth of all installed unsecured experiments is limited to 0.015 $\Delta k/k$ (see Section 3.4) and should significantly affect control rod reactivity worth. Burnup of the cadmium neutron absorber is a long-term effect. Consequently, annual verification of shim-safety blade reactivity worths is considered adequate.

The shim-safety blade drives are constant speed mechanical devices. Withdrawal and insertion rates should not vary except due to mechanical wear. The surveillance frequency was chosen to provide a significant margin over the expected failure or wear rates of these devices. Because of its importance scram times from the full out blade position are tested as often as the driven withdrawal times.

It is not expected that any radioactive iodine or particulate radioactive material will be released through the exhaust gas system. The charcoal cartridge and the moving filter will be analyzed on a monthly basis to demonstrate this. Grab samples of the stack exhaust will be analyzed to verify that the controlling radionuclide is Ar-41.

REFERENCES

(1) Letter, R. S. Kirkland to USAEC, October 22, 1972, Question No. 2.

4.3 CONTAINMENT BUILDING

APPLICABILITY

This specification applies to the surveillance of the containment building.

OBJECTIVE

To verify containment building integrity and to determine and record the building leakage rate under test conditions.

SPECIFICATION

a. The containment building isolation initiating system shall be tested twice a year at approximately six month intervals.
b. An integrated leakage rate test of the containment building shall be performed annually at a pressure of at least 2.0 psig. Leakage from the building shall not exceed 1.0% of the building air volume in 24 hours at 2.0 psig overpressure.
c. All additions, modifications, or maintenance of the containment building or its penetrations shall be tested to verify that the building can maintain its required leak tightness.

BASIS

Containment building isolation is initiated by a signal from either the gaseous waste monitor, Kanne chamber, the moving-air-particulate monitor or a manual push button on the reactor console. Operability and trip point checks semi-annually are considered standard frequency for isolation initiating systems in nuclear facilities.

The containment has been leak tested annually since 1963 and the leak rate of 0.5% of the building volume per 24 hours at two psi overpressure has never been exceeded.(1) No trend has developed which would indicate a gradual deterioration of the containment building. An annual leak rate test frequency is therefore consistant with past experience.

Any additions, modifications, or maintenance to the building or its penetrations will be tested to verify that such work has not adversely affected the leak tightness of the building.

REFERENCE

(1)Safety Analysis Report for the 5 MW GTRR, GT-NE-7, December 1967, Section 7.7.2, p. 133.

4.4 PRIMARY COOLANT SYSTEM

APPLICABILITY

This specification applies to the surveillance of the primary coolant system.

OBJECTIVE

To assure high quality primary water, to detect the release of fission products from fuel elements and to detect leakage from the primary coolant system to the secondary coolant system.

SPECIFICATIONS

a. The pH of the primary coolant shall be measured weekly.
b. The resistivity of the primary coolant shall be measured weekly.
c. A radionuclide analysis of the primary coolant system shall be performed monthly.
d. Samples of the secondary coolant system shall be analyzed for tritium on a monthly basis.

BASIS

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

Samples of the primary coolant water will be analyzed to detect the possible presence of fission products as well as miscellaneous corrosion products. The presence of tritium in the secondary coolant water would indicate a leak in the D_2O system. Therefore, a routine check of the H_2O system will be performed.

4.5 EMERGENCY COOLING SYSTEM

APPLICABILITY

This specification applies to the surveillance of the emergency core cooling system.

OBJECTIVE

To assure that the emergency core cooling system will function properly if required.

SPECIFICATION

a. The emergency core cooling system shall be tested for operability following any maintenance or modification of the system which could affect its performance but at least monthly. This test shall include closing of the reactor isolation valves, initiation of flow on drop of reactor tank coolant level and verification of system flow rate.
b. Flow to each element shall be verified semiannually at approximately six month intervals and to each element when changed.
c. The light water supply to the Emergency Coolant Tank shall be verified initially and annually thereafter.
d. The light water pump that supplies water from the irradiated fuel storage pool shall be tested monthly.
e. The level instrumentation on the D_2O Emergency Coolant Tank shall be calibrated semi-annually.

BASIS

Testing of the system in the manner prescribed will assure that it is operable. The frequency chosen is consistent with the importance of this system.

In addition to establishing closure of the isolation valves and verifying system flow, it is necessary to be certain that each individual element is receiving flow. Since the flow distribution system is fixed and generally inaccessible and unalterable, a semi-annual surveillance requirement is satisfactory except when a fuel element is moved or added to the core.

Initial testing of the entire system will verify light water flow through the entire system from city water supply to the Emergency Coolant Tank. Thereafter, annual tests will be made of the light water system to verify flow capability. A monthly surveillance interval for the pump supplying light water from the irradiated fuel storage pool is considered adequate.

The level instrumentation on the Emergency Coolant Tank provides continuous monitoring of the tank D_2O level as well as providing the means for determining system flow during a test. It should therefore be calibrated on a semi-annual basis.

REFERENCES

(1) Safety Analysis Report for the 5 MW GTRR, GT-NE-7, December 1967, Section 7.7.1, p. 133.

5.0 SITE DESCRIPTION

5.1 SPECIFICATION

a. The reactor facility is located on the Georgia Institute of Technology campus in the city of Atlanta, Georgia.
b. The restricted area is formed by the six-foot security fence on the east, south and west of the containment building and the laboratory building on the north. The closest unrestricted area is 40 meters from the reactor stack exhaust.
c. The exclusion area is the area inside the circle formed by a 100 meter (328 foot) radius centered at the reactor.
d. The low population zone outer boundary is formed by a 400 meter (1312 foot) radius from the containment building.
e. The population center distance for the GTRR is established as a radius of 523 meters (1750 feet) from the containment building.

5.2 FUEL ELEMENTS

SPECIFICATIONS

The fuel elements shall be of the MTR type consisting of 16 aluminum clad fully enriched fuel plates each of which will have a nominal loading of 11.75 grams of U-235.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

a. The organization for the management and operation of the reactor shall be as indicated in Figure 6.1. The Director, Nuclear Research Center shall have overall responsibility for direction and operation of the reactor facility, including safeguarding the general public and facility personnel from radiation exposure and adhering to all requirements of the operating license and Technical Specifications.

b. A Radiological Safety Officer, who is organizationally independent of the GTRR operations staff, shall advise the Director, Nuclear Research Center in matters pertaining to radiological safety at the GTRR.

c. The minimum qualifications with regard to education and experience backgrounds of key supervisory personnel in the Reactor Operations group shall be as follows:

(1) Reactor Supervisor

The Reactor Supervisor must have a college degree or equivalent in specialized training and applicable experience, and at least five-years experience in a responsible position in reactor operations or related fields including at least one year experience in reactor facility management or supervision. He shall hold a Senior Reactor Operator's license for the GTRR.

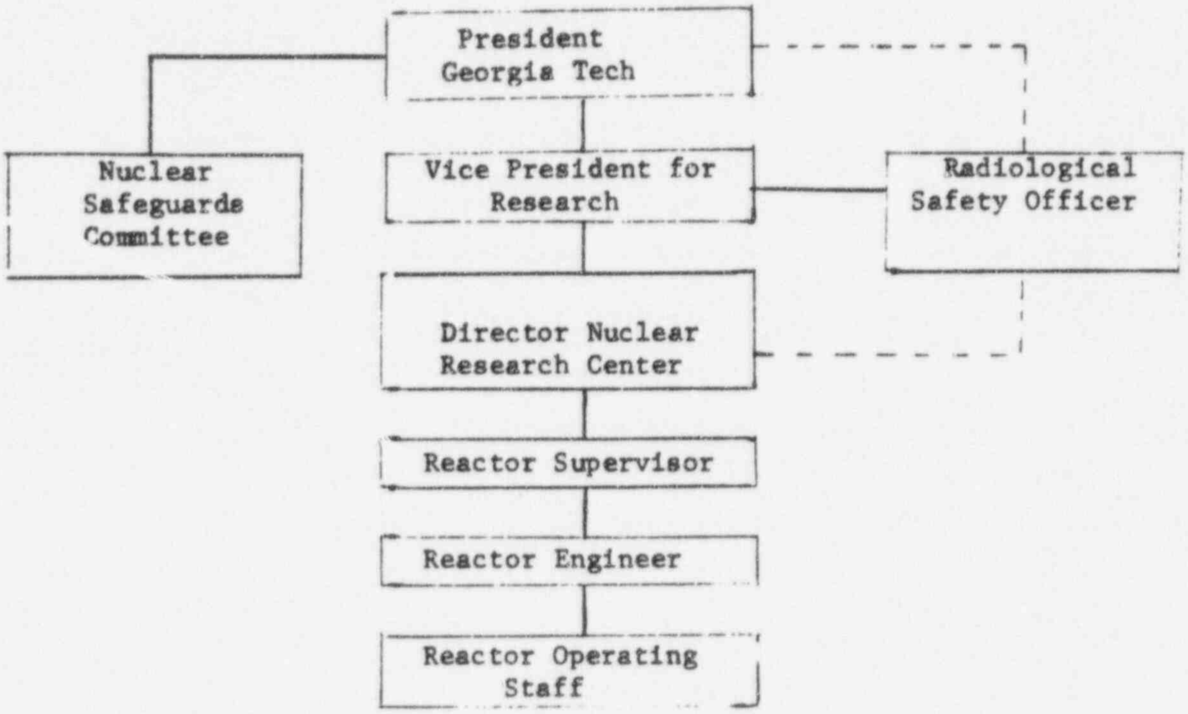
(2) Reactor Engineer

The Reactor Engineer must have a combined total of at least seven years of college level education and/or nuclear reactor experience with at least three-years experience in reactor operations or related fields. He shall be qualified to hold a Senior Reactor Operator's license.

d. Senior Reactor Operator's License

Whenever the reactor is not secured, the minimum crew complement at the facility shall be two persons, including at least one senior operator licensed pursuant to 10 CFR 55.

Figure 6.1


President
Georgia Tech
Nuclear
Safeguards
Committee
Vice President for
Research
Radiological
Safety Officer
Director Nuclear
Research Center
Reactor Supervisor
Reactor Engineer
Reactor Operating
Staff

e. An operator or senior operator licensed pursuant to 10 CFR 55 shall be present at the controls unless the reactor is shut down as defined in these specifications.

6.2 NUCLEAR SAFEGUARDS COMMITTEE

a. A Nuclear Safeguards Committee shall be established by the President of the Institute(1) and shall be responsible for maintaining health and safety standards associated with operation of the reactor and its associated facilities.

b. The Nuclear Safeguards Committee shall be composed of five or more senior technical personnel who collectively provide experience in reactor engineering, reactor operations, chemistry and radiochemistry, instrumentation and control systems, radiological safety, and mechanical and electrical systems. The Reactor Supervisor and Radiological Safety Officer shall be ex officio members of the Committee. No more than a minority of the Committee members shall be from the GTRR staff.

c. The Committee shall meet quarterly and as circumstances warrant. Written records of the proceedings, including any recommendations or concurrences, shall be distributed to all Committee members and the President, Georgia Tech.

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

e. The Nuclear Safeguards Committee shall:

(1) Review and approve proposed experiments and tests utilizing the reactor facility which are significantly different from tests and experiments previously performed at the GTRR.

(2) Review abnormal occurrences.

(3) Review and approve proposed operating procedures and proposed changes to operating procedures which change the original intent of the operating procedure in a non-conservative manner.

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

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

(6) Audit existing operating procedures for adequacy and to assure that they achieve their intended purpose in light of any changes since their implementation.
(7) Audit plant equipment performance with particular attention to operating anomalies, abnormal occurrences, and the steps taken to identify and correct their causes.

6.3 ADMINISTRATIVE CONTROLS OF EXPERIMENTS

a. Evaluation by Safety Review Group

(1) No experiment shall be performed without review and approval by the Nuclear Safequards Committee. Repctitive experiments with common safety considerations may be reviewed and approved as a class.
(2) Criteria for review of an experiment or class of experiments shall include (a) applicable regulatory positions including those in 10 CFR Part 20 and the technical specifications and (b) in-house safety criteria and rules which have been established for facility operations, including those which govern requirements for encapsulation, venting, filtration, shielding, and similar experiment design considerations, as well as those which govern the quality assurance program required under § 50.34.
(3) Records shall be kept of the Nuclear Safeguards Committee's review and authorization for each experiment or class of experiments.

b. Operations Approval

(1) Every experiment shall have the prior explicit written approval of the Licensed Senior Operator in charge of reactor operations.
(2) Every person who is to carry out an experiment shall be certified by the Licensed Senior Operator in charge of reactor operations as to the sufficiency of his knowledge and training in procedures required for the safe conduct of the experiment.

c. Procedures for Active Conduct of Experiments

(1) Detailed written procedures shall be provided for the use or operation of each experimental facility.

(2) The Licensed Operator at the console shall be notified just prior to moving any experiment within the reactor area and should authorize such movement.

(3) Each experiment removed from the reactor or reactor system shall be subject to a radiation monitoring procedure which anticipates exposure rates greater than those predicted. The results of such monitoring should be documented.

d. Procedures Relating to Personnel Access to Experiments

(1) There shall be a documented procedure for the control of visitor access to the reactor area to minimize the likelihood of unnecessary exposure to radiation as a result of experimental activities and to minimize the possibility of intentional or unintentional obstruction of safety.

(2) There shall be a written training procedure for the purpose of qualifying experimenters in the reactor and safety related aspects of their activities, including their expected responses to alarms.

e. Quality Assurance Program

There shall be a Quality Assurance Program covering the design, fabrication, and testing of experiments, including procedures for verification of kinds and amounts of their material contents to assure compliance with the technical specifications in Section 3.4.

6.4 PROCEDURES

a. All procedures and major changes thereto shall be reviewed and approved by the Nuclear Safeguards Committee prior to being effective. Changes which do not alter the original intent of a procedure may be approved by the Reactor Supervisor. Such changes shall be recorded and submitted periodically to the Nuclear Safeguards Committee for routine review.

b. Written procedures shall be provided and utilized for the following:

(1) Normal startup, operation and shutdown of the reactor and of all systems and components involving nuclear safety of the system.

(2) Installation and removal of fuel elements, control blades, experiments and experimental facilities.

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.
(4) Emergency conditions involving potential or actual release of radioactivity.
(5) Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.
(6) Radiation and radioactive contamination control.
(7) Surveillance and testing requirements.
(8) A site emergency plan delineating the action to be taken in the event of emergency conditions and accidents which result in or could lead to the release of radioactive materials in quantities that could endanger the health and safety of employees or the public. Periodic evacuation drills for facility personnel shall be conducted to assure that facility personnel are familiar with the emergency plan.
(9) Physical security of the facility and associated special nuclear material.

6.5 OPERATING RECORDS

a. The following records and logs shall be prepared and retained at the facility for at least five years:

   (1) Normal facility operation and maintenance.
   (2) Abnormal occurrences.
   (3) Tests, checks, and measurements documenting compliance with surveillance requirements.
   (4) Records of experiments performed.

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

   (1) Gaseous and liquid waste released to the environs.
   (2) Offsite environmental monitoring surveys.
   (3) Radiation exposures for all GTRR personnel.
   (4) Fuel inventories and transfers.
   (5) Facility radiation and contamination surveys.
   (6) Updated, corrected, and as-built facility drawings.
   (7) Minutes of Nuclear Safeguards Committee meetings.
   (8) Records of radioactive shipments.

6.6 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE

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

a. All abnormal occurrences shall be promptly reported to the Reactor Supervisor or his designated alternate.
b. All abnormal occurrences shall be reported to the U. S. Atomic Energy Commission in accordance with Section 6.7 of these specifications.
c. All abnormal occurrences shall be reviewed by the Nuclear Safeguards Committee.

6.7 REPORTING REQUIREMENTS

The following information shall be submitted to the USAEC in addition to the reports required by Title 10, Code of Federal Regulations.

a. Annual Operating Reports

A report covering the previous year shall be submitted to the Director of the Regional Regulatory Operations Office by March 1 of each year. It shall include the following:

(1) Operations Summary

A summary of operating experience occurring during the reporting period including:

(a) changes in facility design,
(b) performance characteristics (e.g., equipment and fuel performance),
(c) changes in operating procedures which relate to the safety of facility operations,
(d) results of surveillance tests and inspections required by these technical specifications,
(e) a brief summary of these changes, tests, and experiments which required authorization from the Commission pursuant to 10 CFR 50.59(a), and
(f) changes in the plant operating staff serving in the following positions:

1. Director, Nuclear Research Center

2. Reactor Supervisor

3. Reactor Engineer

4. Radiological Safety Officer

5. Nuclear Safeguards Committee members

(2) Power Generation

A tabulation of the thermal output of the facility during the reporting period.

(3) Shutdowns

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

(4) Maintenance

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

(5) Changes, Tests and Experiments

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

(6) Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from the plant, with data summarized following the general format of USAEC Regulatory Guide 1.21:

(a) Gaseous Effluents

1. Gross Radioactivity Releases

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

b. Average concentration of gaseous effluents released during normal steady state operation. (Averaged over the period of reactor operation.)
c. Maximum instantaneous concentration of gaseous radionuclides released during special operations, tests, or experiments, such as beam tube experiments, or pneumatic tube operation.
d. Percent of technical specification limit.

2. Iodine Releases

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

a. Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.
b. Percent of technical specfication limit.

3. Particulate Releases

a. Total gross radioactivity (β,γ) released (in curies) excluding background radioactivity.
b. Gross alpha radioactivity released (in curies) excluding background radioactivity. (Required if the operational or experimental program could result in the release of alpha emitters.)
c. Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.
d. Percent of MPC for particulate radioactivity with half-lives greater than eight days.

(b) Liquid Effluents

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

2. The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area.
3. Total alpha radioactivity (in curies) released and average concentration released to the unrestricted area (averaged over the period of release).
4. Total volume (in ml) of liquid waste released.
5. Total volume (in ml) of water used to dilute the liquid waste during the period of release prior to release from the restricted area.
6. Total radioactivity (in curies), and concentration (averaged over the period of release) by nuclide released, based on representative isotopic analyses performed for any release which exceed 1×10^{-7} μCi/ml.
7. Percent of technical specification limit for total radioactivity from the site.

(7) Environmental Monitoring

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

(a) Number of sampling locations and a description of their location relative to the reactor.
(b) Total number of samples.
(c) Number of locations at which levels are found to be significantly above local backgrounds.
(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and the location of that point with respect to the site.
(e) The maximum cumulative radiation dose which could have been received by an individual continuously present in an unrestricted area during reactor operation from:

1. direct radiation and gaseous effluent, and

2. liquid effluent.

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

(8) Occupational Personnel Radiation Exposure

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

b. Non-Routine Reports

(1) Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (copy to the Director of Licensing) followed by a written report within 10 days to the Director of the Regional Regulatory Operations Office in the event of the abnormal occurrences as defined in Section 1.0. The written report on these abnormal occurrences, and to the extent possible, the preliminary telephone and telegraph notification shall:

(a) describe, analyze, and evaluate safety implications,
(b) outline the measures taken to assure that the cause of the condition is determined,
(c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and
(d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

(2) Unusual Events

A written report shall be forwarded with 30 days to the Director of the Regional Regulatory Operations Office in the event of:

(a) Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.
(b) Discovery of any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.
(c) Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.