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Operations Division

Reactor Operations Section

TECHNICAL SPECIFICATIONS FOR THE BULK SHIELDING REACTOR

Operations Division Staff

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Operations Division

Notice: This document contains information of a preliminary nature. It is subject to revision or correction and, therefore, does not represent a final report.

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TECHNICAL SPECIFICATIONS FOR THE BULK SHIELDING REACTOR

1. GLOSSARY OF TERMS

The following list is a glossary of terms frequently used in the discussions in the Technical Specifications.

1.1 ABNORMAL OCCURRENCE

An abnormal occurrence is any of the following:

1. Any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings (LSSS).

2. Operation in violation of a Technical Specification.

3. Incidents or conditions which prevented or could have prevented the performance of the intended safety function of an engineered safety feature or the reactor safety system.

4. Release of radioactive materials of a magnitude to indicate a failure of the principal physical boundary.

5. An uncontrolled or unexplained change in reactivity.

6. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

7. An uncontrolled or unanticipated release of a source of radioactivity in excess of the limits defined in DOE Order 5480.1, Attachment XI-I.

1.2 ACCIDENT

An accident is defined as those transients and excursions which are not expected to occur during the life of the reactor but which have been analyzed in the safety documents.

1.3 ANTICIPATED TRANSIENT

An anticipated transient is defined as any transient or excursion which could reasonably be expected to occur at least once during the life of the reactor.

1.4 BURNOUT

Burnout is a fuel cladding breach or degradation due to excessive heat fluxes which causes exposure of the enriched fuel and fission products.

1.5 CALIBRATION

Calibration is an adjustment of electronic, mechanical, pneumatic, or other sensing systems so that the output of the system responds, within acceptable range and with acceptable accuracy, to known values of the parameter that the system measures or to known input signals when access to the primary element is limited.

1.6 CERTIFIED OPERATOR (RO or SRO)

A certified operator is any individual who has successfully completed the training, examination, and certification for reactor operator (RO) or senior reactor operator (SRO) pursuant to DOE Order 5480.1A.

1.7 CHANNEL CALIBRATION

A channel calibration is an adjustment of the channel such that output responds, with acceptable range and acceptable accuracy, to known values of the parameter that the channel measures or to known input signals when access to the primary element is limited.

1.8 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 expected values or other independent channels or methods of measuring the same variable.

1.9 CHANNEL TEST

A channel test is the introduction of an appropriate signal into a channel to verify that it is operable.

1.10 CORE CONFIGURATION

The core configuration is the arrangement of reactor core elements within the reactor, including position, number, and fuel mass.

1.11 CRITICAL HEAT FLUX

The critical heat flux or burnout heat flux is defined as the heat flux at which burnout is likely to occur.

1.12 ENGINEERED SAFETY FEATURES

Engineered safety features are those features of a unit, other than reactor trip or those used only for normal operation, that are provided to prevent, limit, or mitigate the release of radioactive material.

1.13 EXCESS REACTIVITY

Excess reactivity is the excess k_{eff} above unity that would exist if all the shim-safety rods were fully withdrawn.

1.14 EXPERIMENT

An experiment is any apparatus (other than the reactor components and structure), device, or material placed in the reactor core region, in an experimental facility, or in line with a beam of radiation emanating from the reactor or any operation designed to measure reactor characteristics.

1.15 FLIGHT TIME

Flight time is the time interval between the introduction of a fast scram signal and the actuation of the seat switch of the shim-safety rod being tested. Flight time measurements are normally made with the shim-safety rod withdrawn to the upper limit of its travel.

1.16 FUEL ELEMENT

Fuel elements are core pieces that can be inserted into the BSR grid and that contain a fissionable material for use in the production of energy.

1.17 FUNCTIONAL TEST

A functional test is a test of a component or system of components to assure that the component or system of components perform as per design when called upon.

1.18 GANGED-ROD POSITION

Ganged-rod position is the term used to describe the relationship between shim-safety rod positions. Any number of shim-safety rods are said to be at ganged-rod position if they are withdrawn equidistant from their seats.

1.19 LIMITING CONDITIONS FOR OPERATION

Limiting Conditions for Operation (LCO) are those administratively established constraints on operation of the facility.

1.20 LIMITING SAFETY SYSTEM SETTINGS

Limiting Safety System Settings (LSSS) are those limiting values for settings of the safety channels that require initiation of protective action.

1.21 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.

1.22 MOVABLE EXPERIMENT

A movable experiment is one in which it is intended that the entire experiment or part of it may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.23 NOMINAL FULL POWER

Nominal full power is 2 MW.

1.24 OPERABLE

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

1.25 OPERATING

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

1.26 OPERATING MODES

1. Natural-convection mode - The natural-convection mode is defined as operation at power levels equal to or less than 1.2 MW in which natural convection cooling is adequate.

2. Forced-flow mode - The forced-flow mode is defined as operation at power levels in excess of 1.2 MW in which forced convection cooling is required. Forced flow may be used at power levels equal to or less than 1.2 MW.

1.27 POWER OPERATION

Power operation is operation in excess of 100 kW (sometimes referred to as "operation at power").

1.28 REACTOR SAFETY SYSTEM

The reactor safety system is that combination of measuring channels and associated circuitry which forms the automatic protective system of the reactor.

1.29 REACTOR SECURED

The reactor is secured if all of the following conditions are satisfied:

1. Reactor is shutdown.

2. Electrical power to the control rod circuits is switched off, and switch key is in proper custody.

3. No work is in progress involving in-core components, experiments, or installed control rod drives.

1.30 REACTOR SHUTDOWN

The reactor is shut down when all shim-safety rods are fully inserted or when it is in a reactivity condition equivalent to that in which all shim-safety rods are fully inserted.

1.31 RELEASE TIME

The release time of a shim-safety rod is the time interval between the initiation of a fast scram signal and the first detectable movement of the magnet armature.

1.32 SAFETY CHANNEL

A safety channel is a measuring channel in the reactor safety system.

1.33 SAFETY LIMIT

Safety limits are limits on important process variables which are necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

1.34 SHALL

The word "shall" is used to denote a requirement.

1.35 SHOULD

The word "should" is used to denote a recommendation.

1.36 SHIM-SAFETY RODS

Shim-safety rods (control rods) are the neutron-absorbing core components used to control the reactivity of the core.

1.37 SHUTDOWN MARGIN

The shutdown margin is the amount of reactivity required to attain criticality from a given subcritical condition.

1.38 SHUTDOWN MECHANISMS

The shutdown mechanisms consist of any of the following:

1. Scram - Gravity insertion of all withdrawn shim-safety rods.

2. Setback - A reduction of reactor power by the automatic control system.

3. Reverse - Motor driven insertion of all shim-safety rods at their normal insertion speed.

1.39 SPECIFICATIONS FOR BSR FUEL ELEMENTS AND/OR BSR SHIM-SAFETY ROD ELEMENTS

The specifications for BSR fuel elements and/or BSR shim-safety rod elements are detailed material and fabrication specifications, quality assurance requirements, and associated engineering drawings approved by ORNL Operations Division management.

1.40 STARTUP OPERATION

Startup operation is that period of operation prior to attaining 100 kW.

1.41 SYPHON BREAK

The syphon break is a pipe system that establishes a vacuum break to prevent syphoning pool water and uncovering the reactor core should a primary water line fail.

1.42 TIME INTERVALS

1. Biennially - To be performed once every two years at intervals not to exceed 30 months.

2. Annually - To be performed once each year at intervals not to exceed 15 months.

3. Semiannually - To be performed twice each year at intervals not to exceed eight months.

4. Quarterly - To be performed four times each year at intervals not to exceed four months.

5. Monthly - To be performed once each month at intervals not to exceed six weeks.

6. Weekly - To be performed once each week at intervals not to exceed 10 days.

7. Daily - To be performed once each day at intervals not to exceed 32 hours.

1.43 TECHNICAL SPECIFICATIONS

A technical specification is a safety document approved by DOE which in a specified format defines the conditions and safety boundaries under which activities are to be carried out at a reactor.

1.44 TRIED EXPERIMENT

A tried experiment is:

1. an experiment previously approved and performed in this reactor, or

2. an experiment of similar size, shape, composition, and location as an experiment previously approved and performed in this reactor.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 Safety Limit for Operation in Mode 1 (Natural Convection)

Applicability. This specification applies to the reactor power associated with core thermal and hydraulic performance when operating in the natural convection mode.

Objective. The objective is to establish limits for the critical process variable which, if not exceeded, will provide confidence that the integrity of the fuel-element cladding will not be breached.

Specification. When operating in the natural convection mode, the reactor power shall not exceed 3.5 MW.

Basis. The safety limit set forth in this specification has been established to ensure operation at or below a power level at which there is a wide burnout margin. The critical heat flux for natural convection cooling and 120°F inlet temperatures as interpolated from the experimental data of Gambill and Bundy[1] is about 1.25×10^5 Btu/(ft^{2}h). The maximum heat flux in the BSR has been very conservatively estimated to be 2.51×10^4 Btu/(ft^{2}h)/MW (Refs. 2,3). Hence the burnout power level is about 5 MW under natural convection conditions.

The inlet water temperature is normally automatically controlled at 100°F and operation of the reactor is administratively prohibited if the inlet temperature exceeds 120°F. Since the pool contains approximately 130,000 gal (1.06×10^6 lb) of water, nearly 2 hours operation at 3.5 MW is required to raise the pool water temperature 20°F to the administrative limit. Rapid changes in coolant inlet temperature, therefore, need not be considered since there is adequate time for supervisory action to effect administrative control.

2.1.2 Safety Limit for Operation in Mode 2 (Forced Convection)

Applicability. This specification applies to the process variables of Reactor Thermal Power and Reactor Primary Coolant Flow.

Objective. The objective is to establish limits for the critical process variables which, if not exceeded, will provide confidence that the integrity of the fuel-element cladding will not be breached.

Specification. When operating in the forced convection mode, the relationship between total reactor primary coolant flow rate and reactor power shall lie to the right of and under curve B of Fig. 2.1.

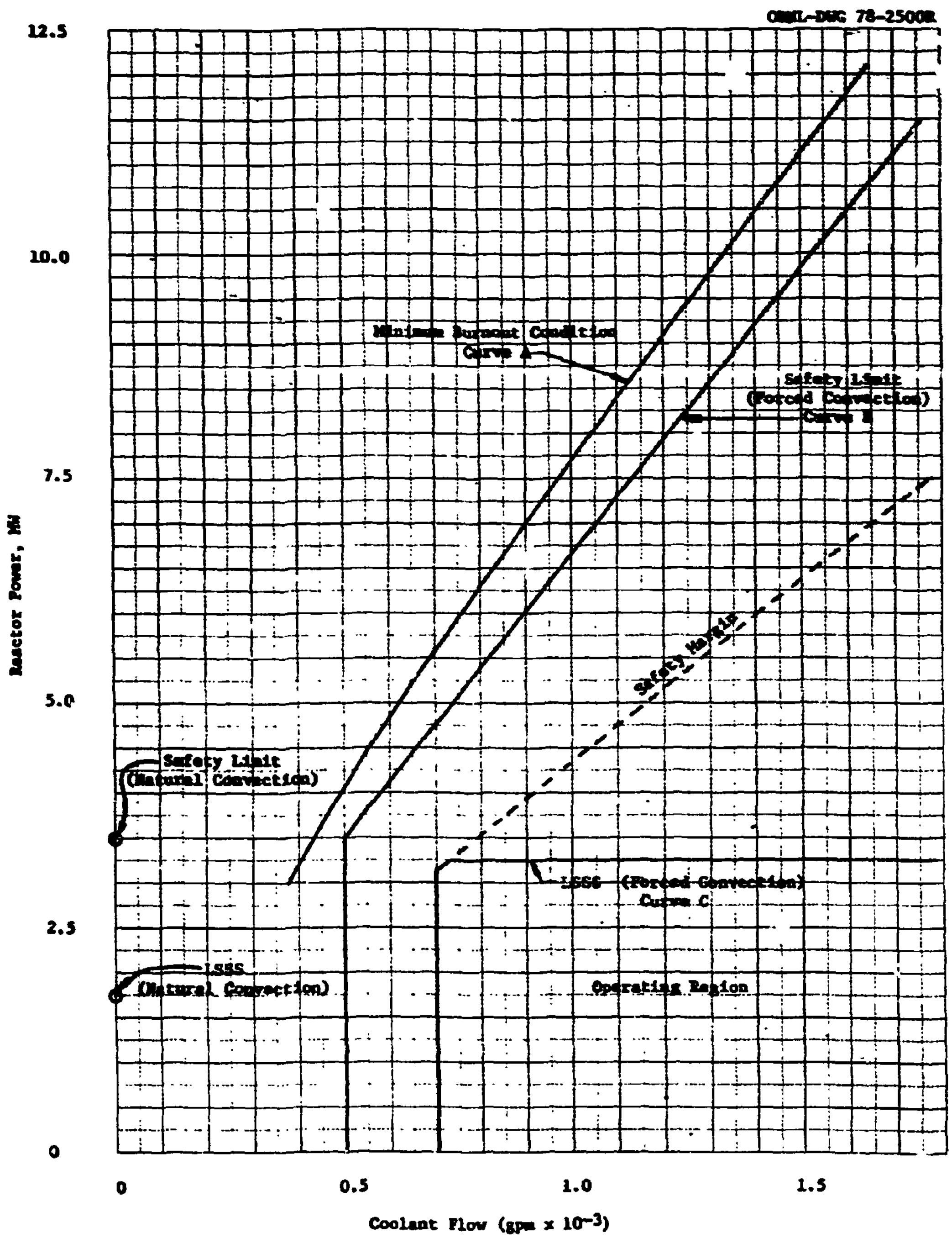


Fig. 2.1. Safety limits and limiting safety system settings for the BSR.

Basis. The safety limit set forth in this specification has been established to guarantee against operation in a region which can cause damage to the fuel cladding.

The parameters which are of importance in determining the critical heat flux for subcooled boiling burnout are coolant velocity and degree of subcooling. Since neither of these can be measured directly, an analysis[3] of typical BSR cores was undertaken in order to relate these parameters to the measured variables: flow, power level, and temperature.

Curve A of Fig. 2.1 is a very conservative estimate of the conditions required to produce a critical heat flux somewhere in the core. It is based upon the Gambill minimum critical heat flux correlation[4] and assumes a coolant inlet temperature of 120°F and a maximum-to-average heat flux ratio of 3. (The highest observed value of this ratio is 2.23.) The safety limit (curve B of Fig. 2.1) has been established by requiring a 10% margin on power level below the estimated burnout curve. The lower limit on flow is established to prevent operation in the region where a combination of low velocity downward forced convection and upward natural convection could produce flow instabilities.

2.2 LIMITING SAFETY SYSTEM SETTINGS (LSSS)

Applicability. These specifications apply to the safety system action set points for the reactor process variables related to the safety limits.

Objective. The objective of these specifications is to provide limiting values of the set points of the safety channels so that automatic protective action, sufficient to provide confidence that the criteria set forth in the safety limits (Section 2.1) are satisfied, will occur when required.

Specifications.

1. For Mode 1, natural convection - Reactor power: The LSSS on the power of the reactor in the natural convection mode shall be 1.75 MW.

2. For Mode 2, forced convection - The relationship between the reactor primary coolant flow rate LSSS and the reactor power LSSS shall lie below and to the right of Curve C of Fig. 2.1.

Bases. The limiting safety system settings were chosen such that when measurement uncertainties and anticipated transient conditions are considered there is confidence that the criteria set forth in the safety limits are at all times satisfied.

1. The maximum power level LSSS for the natural convection mode has been set at the nucleate boiling level, a factor of about 2.9 below burnout.

2. For the forced convection flow mode, the LSSS have been chosen to provide a factor of 1.75 between the LSSS and the lowest occurrence of a critical heat flux.[3] The specific limitation on flow provided at the lower end is to ensure that the limits are within the range of validity of the calculations.

3. LIMITING CONDITIONS FOR OPERATION

3.1 SYSTEM REACTIVITY

Applicability. These specifications apply to the reactivity condition of the reactor and its associated components.

Objective. The objective is to ensure that the reactor can be made subcritical and maintained in a subcritical condition whenever required, and to prevent the occurrence of any condition that might add a significant amount of uncontrolled reactivity.

3.1.1 The Reactor Core

Specifications.

1. Excess reactivity loaded in the core - The core excess reactivity shall be less than or equal to 50% of the reactivity worth of the shim-safety rods when they are withdrawn in the ganged position. In addition, the reactor core shall be so arranged that the shutdown margin with the highest worth shim-safety rod fully withdrawn shall be greater than 0.8% $\Delta k/k$ relative to the cold clean condition. The total mass of fuel in the core shall be adjusted to achieve this before criticality for power operation is attained. (Failure to meet these requirements during criticality tests of a new core configuration or for other investigative purposes shall not be regarded as a violation of the specification provided corrective action is taken prior to startup for power operation.)

2. Reactivity worth of experiments - The limitations on reactivity worth of experiments either within the core or sufficiently close to the reactor to produce significant interactions are specified in Section 3.7.

3. Minimum distance - The minimum distance between the BSR core and the PCA core shall be more than 10 in. and shall be maintained by means of mechanical stops.

Bases.

1. The limitation on excess reactivity is expressed in terms of relative rod worths rather than in absolute terms because the absolute worths of the shim-safety rods are a function of the specific fuel mass and configuration of the core. The requirements that no one rod can produce criticality guarantees that the rod worths are distributed in a manner that minimizes the consequences of the uncontrolled withdrawal of a single rod.

   To ensure that the worths of the shim-safety rods are adequate to meet the requirements of Section 3.1.1, Specification "1" (see

Sections 4.1.1.1 and 4.3.2.4, and page 4-1), they are calibrated, individually and ganged, whenever a significant change in core configuration is introduced.

Normally the total worth of the shim-safety rod complement is about 11% $\Delta k/k$ so that the shutdown margin is about 5.5% $\Delta k/k$.

2. Reactor experiments are examined to determine their reactivity effects and also their potential for increasing or decreasing the reactivity of the system; and specific requirements are imposed to ensure that reactor experiments cannot, by failure or misoperation, significantly affect the reactivity balance of the reactor. This is discussed further in Section 3.7.

3. It has been determined that this distance is adequate to prevent interaction between the two cores.[5]

3.2 CORE CONFIGURATION

Applicability. This specification applies to the determination of core configurations which may be used in the reactor.

Objective. The objective is to ensure that, under all permissible operating conditions, the maximum heat flux in the reactor is sufficiently below the burnout heat flux to provide an adequate margin of safety.

Specification. The reactor core configuration shall be such that the maximum power generated in any fuel element shall not exceed 0.09 MW per MW of reactor power.

Basis. At normal flow (1100 gpm) the minimum burnout heat flux in the BSR core has been conservatively estimated to be 1.75×10^5 Btu/(ft^2h) (Ref. 3). This specification which utilizes an axial hot spot factor of 1.5 provides a margin of 2.73 between the normal power level of 2 MW and the burnout power level. Should the flow be reduced to the LSSS of 700 gpm the margin would become 1.75. Should both the power level and the flow reach their respective LSSS values simultaneously, a margin of 1.13 would remain. In the natural convection mode with the power level at the LSSS, the margin is 2.85.

3.3 REACTOR CONTROL AND SAFETY SYSTEMS

Applicability. These specifications apply to the safety, control, and surveillance instrumentation and to the mechanical components of these systems which are required for startup and operation.

Objective. The objective is to ensure that an adequate complement of safety, control, and surveillance instrumentation together with the requisite mechanical components are available during startup and operation.

Specifications.

3.3.1 Minimum Reactor Instrumentation Requirements

The minimum complement of reactor safety and measuring instrumentation required for startup and operation is specified in Table 3.1.

Table 3.1. Minimum instrumentation required for reactor operation

Description	Number Required: Mode 1 natural convection	Number Required: Mode 2 forced convection
Safety or protective channels		
Safety level channels	2	2
Log N channels	1[a]	1[a]
ΔT channels	None	1[b]
Primary coolant flow channels[c]	None	2
Manual scram	1	1
Measuring channels		
Fission chamber channel	1[a]	1[a]
Flapper valve position channels	1	1
Pool water level channels	1	1

[a]Startup only, not needed with the reactor at power.

[b]ΔT channels serve as backup for the reactor safety level channels and therefore do not need to be redundant.

[c]Flow channels may utilize core ΔP instrumentation.

3.3.2 Mechanical Control System

1. Number of shim-safety rods - The reactor shall not be operated with fewer than four shim-safety rods.

2. Maximum release time - The release time of the shim-safety rods shall be less than 30 ms.

3. Maximum flight time - The maximum flight time shall be 800 ms.

4. Maximum rate of positive reactivity addition - The maximum rate of positive reactivity addition (by movement of the shim-safety rods) shall be limited to 0.10% $\Delta k/k$ per second.

5. Maximum reactivity available to the servo system - The maximum reactivity available to the servo control system shall be limited to 0.70% $\Delta k/k$.

Bases. The reactor control and safety system is designed to prevent the only credible cause for a massive release of fission products from the BSR fuel, namely, overheating to an extent sufficient to cause melting and destruction of the integrity of the fuel cladding. Such a condition could be brought about either by an increase in power beyond the capacity for heat removal, a decrease in the heat-removal capability of the system, or a combination of both.

The requirement that the above-listed minimum instrumentation be operable and that the associated mechanical components be operable is sufficient to guarantee against these conditions.

The release and flight times prescribed are those which could reasonably be expected when the shim-safety rods are free to fall within their guides and thus ensure that the rods are unrestricted and free to fall when called upon to do so.

Based upon SPERT data cited in the BSR Safety Analysis report,[2] restriction of the positive reactivity addition rate to 0.1% $\Delta k/k$ per second guarantees that during a startup runaway the maximum fuel surface temperature would not exceed 320°F, which is well below the melting point.

During normal operation, the reactor is operated by means of a servo system which provides automatic shimming to compensate for small positive and negative reactivity changes. The BSR has a delayed neutron fraction equivalent to approximately 0.81% $\Delta k/k$, and the positive reactivity which is available to the servo is maintained at less than this value to prevent the possibility of prompt criticality as a result of servo action.

3.4 COOLING SYSTEM AND COOLANT LEAKS

3.4.1 Cooling System

Applicability. These specifications apply to the conditions which must exist in order for the reactor pool to supply cooling water to the reactor and provide adequate shielding for the reactor core.

Objective. The objective is to ensure an adequate supply of cooling water to provide heat removal, shielding, and moderation to the reactor.

Specifications.

1. Coolant flow - When operating in the Forced-Convection Mode, the flow rates through the core are established to ensure adequate cooling as required by the LSSS specified in Section 2.1.2.

2. Syphon break - The pool coolant system is a syphon device equipped with a syphon break so that rupture of the primary coolant lines cannot drain the pool and expose the reactor core.

3. Coolant temperature - During reactor operation, the bulk coolant temperature shall not exceed 120°F.

4. Shielding - The height of the water in the pool shall be maintained at a level sufficient to protect against radiation exposures in excess of the limits specified in DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection."

5. Radioactivity level - The radioactivity in the cooling water shall be maintained at a level low enough to provide protection against radiation exposures in excess of those specified in DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection."

6. Cooling system integrity - The integrity of the cooling system piping and other components shall be maintained. However, expected minor failures, such as valves and heat exchanger tubes, shall not constitute a violation of these technical specifications.

Bases.

1,2. Forced-convection flow through the BSR core is provided by a syphon system which takes water from the reactor core and leads it to a decay tank located at a lower elevation. The coolant is then pumped through a heat exchanger and returned to the pool. The available coolant flow rate depends upon the head provided by the depth of the water in the pool, and this must be adequate to satisfy the LSSS flow requirements.

3. The critical heat flux calculations and resulting safety limit and LSSS values were determined using an inlet temperature of 120°F.

Thus, if this value is not exceeded, these limits remain conservative with respect to the inlet temperature.

4,5. Normally the radiation levels in the occupied areas near the reactor pool are considerably less than those specified by DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection"; however, in order to allow for abnormal operations and experimental operations, the legislated limits are specified.

6. The adequacy of the primary cooling system components were ensured initially by codes, tests, and inspections during the time they were being assembled. Examinations of the components at frequencies specified according to those set forth in Section 4.4 and by the in-service inspection plan will ensure that their original capabilities are maintained.

3.4.2 Coolant Leaks

Applicability. This specification applies to leaks from the pool or primary water cooling system.

Objective. The purpose of this specification is to ensure that the loss of water from the pool is within acceptable limits.

Specification. The reactor shall not be operated without a functioning syphon break system. The reactor shall not be operated with a leak or leaks in the pool or primary coolant system which exceed 30 gpm.

Basis. It has been determined[6] that should the design basis accident occur while the reactor coolant system is leaking at a rate of 30 gpm, even under the most pessimistic assumptions, the consequences would not result in any significant hazard to the public.

3.5 GASEOUS WASTE SYSTEM

Applicability. These specifications apply to those systems which are required to remove and dispose of gaseous effluents from the reactor.

Objective. The objective is to ensure that sufficient capacity for the removal of gaseous effluents is available whenever the reactor is operating.

Specifications.

1. Normal Off-Gas System (NOG)

   a. The NOG system shall be in operation and shall be adequate to prevent outleakage at all times whenever one of the following conditions exist:

(1) there are experiments in the reactor which require the use of the NOG, or

(2) the reactor is operating at a power in excess of 300 kW.

b. When the NOG is in operation, the negative pressure shall be monitored and shall be greater than 15 in. of water.

3.6 EMERGENCY SYSTEMS

3.6.1 Confinement System

Applicability. These specifications apply to the operating status of the reactor building and the associated exhaust system.

Objective. The objective is to ensure containment integrity when it is needed.

Specifications. The BSR shall not be operated unless the building confinement system is operable. The minimum requirements are as follows:

1. Flow through the exhaust duct must be in the normal range, i.e., between 3,025 cfm and 5,500 cfm.

2. The filter bank for the exhaust system shall be operable and the filters therein shall meet removal efficiency specifications, i.e., particulate filters shall show an efficiency of $\geq$99.5% for removal of particles $\geq$0.3 micron in size, and the iodine absorbers shall show an efficiency of $\geq$97% for removal of elemental iodine.

3. All building doors and any other openings in the building must be properly closed, equipped with automatic closers, or be under the direct control of qualified operators.

4. Automatic activation of the containment mode of the confinement system shall occur when either of the following conditions exist:

   a. the general radiation background in the vicinity of the monitor located in the upper half of the reactor building exceeds 150 mR/h; or

   b. the radiation background at the monitor on the building exhaust duct exceeds 50 mR/h.

Bases. The BSR containment system, in principle, is described on page 34 of the description and safety analysis report.[2] The restrictions imposed by Specification 3.5.1 will ensure that the requirements for containment are met when needed at any time when the reactor is operating.

Since the establishment of a negative pressure differential is related to the exhaust rate, the requirements on exhaust rate have been selected to guarantee a negative pressure.

The filter efficiencies are maintained at or above values that can be practically achieved with commercial equipment. Based on the maximum hypothetical accident at the BSR (Ref. 2, page 50), a minimum iodine decontamination factor of 10 is required. The iodine filters are normally operated with a decontamination factor of 100 or more. Many years of experience have shown that these values are more than adequate to reduce the normal effluents to trivial levels, and under the conditions of the maximum hypothetical accident, they provide a reduction in the gaseous fission product release to well below the maximum permitted by guidelines of 10CFR100.[10]

3.7 RADIATION MONITORING EQUIPMENT

Applicability. These specifications apply to the monitoring of radiation levels and airborne activity in the reactor building.

Objective. The objective is to ensure that a sufficient number and variety of operable instruments to adequately monitor radiation levels throughout the building are available.

Specifications. The reactor shall not be operated unless the following minimum requirements for operable area radiation monitors are met or backup measures approved by the Reactor Supervisor or his designated alternate are provided:

1. At least two operable constant air monitors shall be located at appropriate points within the reactor building.
2. At least two operable monitrons shall be located at appropriate points within the building.
3. Readouts from the required equipment shall be displayed in the reactor control room and at the remote control console.
4. At least one β,γ stack monitor is operable.

Bases. The minimum required number of radiation monitors provides protection and warning of elevated levels of radiation so that there will be sufficient time to evacuate the building or portions of the building and to take necessary steps to prevent the spread of radioactivity to the surroundings.

Indication in the control room will warn of excessive radioactivity levels within the building.

The beta-gamma stack monitor serves as an independent monitor which provides backup protection and warning of elevated levels of radiation entering the stack from the BSR/ORR cell vent and NOG systems and other protective systems.

3.8 LIMITATIONS ON EXPERIMENTS

Applicability. These specifications apply to all experiments installed in the BSR.

Objective. The objective of these specifications is to establish general limits for BSR experiments that will protect the reactor from damage and shall limit radiation exposure to both on-site and off-site personnel to as low as practicable levels.

Specifications. The following limitations apply to all in-reactor experiments:

1. Reactivity - The following limits apply to individual experiments:

   a. Experiments which can cause a positive reactivity change greater than 0.5% $\Delta k/k$ and which are movable experiments shall be equipped with a mechanical insertion-and-removal system having the same reliability as the reactor control-system drives. The reactivity insertion rate by these mechanisms shall not exceed 0.01% $\Delta k/k$ per second.

   b. Experiments which can cause a positive reactivity change greater than 0.5% $\Delta k/k$ due to motion but which need not be moved while the reactor is critical shall be so firmly supported that no credible circumstance can cause them to be moved while the reactor is critical.

   c. Experiments which can cause a positive reactivity change greater than 0.5% $\Delta k/k$ due to being damaged by temperature or pressure shall be instrumented to cause an appropriate reactor power or reactivity reduction if such temperature or pressure is approached. This is in addition to the safeguards built into the experiment to control the temperature and/or pressure.

   d. The combined reactivity worth of all experiments which can add positive reactivity to the core due to a common-mode failure shall be $\leq$50% of the shutdown margin of the shim-safety rod control system.

2. Hydraulic stability - All in-core experiments shall be designed to withstand the hydraulic forces and each assembly shall be staked, welded, or adequately held to avoid loss into the reactor.

3. Temperature - Heat developed in any experiments by gamma absorption, fissions, electric heaters, etc., shall be dissipated by an adequate coolant flow. Under normal conditions, the temperature of the outer container shall not exceed the saturation temperature of the reactor coolant.

4. Explosives - No explosives or mixtures of material that under credible circumstances can detonate shall be irradiated in the reactor.

5. Pressure containment - Where failure of pressure-containing walls of an experiment can cause a hazard to personnel or to the reactor, the container shall meet the intent of applicable pressure vessel codes. The design for each such container shall be reviewed by a competent engineer and written approval obtained from the Operations Division and the Reactor Experiment Review Committee (RERC).

Bases.

1. Reactivity

   a, b, and c. The upper limit of 0.5% $\Delta k/k$ for a reactivity change by an experiment ensures that such change will not result in prompt criticality. It also ensures that the servo system can immediately compensate for such a change. The limit of 0.01% $\Delta k/k$ per second for an intentional, controlled reactivity insertion rate ensures adequate compensation by the servo system.

   d. The restriction on the combined reactivity worth of all experiments guarantees that the shutdown margin is always adequate to make the reactor subcritical.

2. Hydraulic stability - The reactor components must be protected from damage that might result from the movement of an experiment in an uncontrolled manner due to hydraulic forces. The requirements of 3.7.2 will ensure this.

3. Temperature - As with reactor components, the surfaces of experiment rigs are maintained below the saturation temperature of the reactor coolant to avoid steam blanketing and possible burnout.

4. Explosives - The irradiation of explosives is avoided to protect reactor components from possible damage.

5. Pressure containment - Depending on location, failure of pressure-containing walls could affect reactivity or damage adjacent reactor components. Failure of pressure-containing walls external to the reactor may endanger personnel directly by impact injury or release of radioactive gases. The use of applicable pressure vessel codes ensures that adequate safety margins are maintained.

4. SURVEILLANCE REQUIREMENTS

The surveillance requirements in this section apply to continuous system and facility operation. During periods of extended shutdown (periods of time greater than the minimum frequency listed), the stated requirements may be suspended. However, in this case, the requisite surveillance must be performed before the system is again considered operable.

4.1 SYSTEM REACTIVITY

Applicability. These specifications apply to the surveillance requirements for system reactivity.

Objective. The objective is to ensure compliance with the specifications in Section 3.1 relative to system reactivity.

4.1.1 The Reactor Core

Specifications.

1. Prior to the startup following significant changes to the reactor configuration or loading, including experiment and reflector changes, or annually, whichever of the above shall occur first, the ganged-rod position that represents 50% of the shim-safety rod worth shall be determined. If the reactor reaches criticality with the rods withdrawn to a lesser extent than this, the reactor shall be shut down and the situation corrected.

   Prior to the startup following significant changes to the reactor configuration or loading, including experiment and reflector changes, or annually, whichever of the above shall occur first, each shim-safety rod shall be withdrawn to its upper limit to ensure that the reactor is sufficiently subcritical with each rod withdrawn while the others are fully inserted. (This check is normally accomplished in conjunction with the flight time measurements.) Should the test fail, the reactor shall be shut down and the situation corrected.

2. When required, calculations shall be performed to ensure that the requirements of Section 3.1.1, Specification "2" are met. Where uncertainty exists, experiments to determine the worth of experiments shall be performed. Additional specifications concerning experiments are listed in Section 4.7.

3. The shim-safety rods shall be calibrated for reactivity worth whenever a significant change in core loading or experiental load is made.

Bases. These surveillance requirements for reactivity limitations are based upon more than 20 years of experience with the BSR which indicates that adherence to the required checks at the specified frequency is adequate to guarantee compliance with Section 3.1.1.

4.2 CORE CONFIGURATION

Applicability. This specification applies to the procedures for determining permissible core configurations.

Objective. The objective is to ensure compliance with the specifications in Section 3.2 relative to reactor core configurations.

Specification. All fuel loadings shall be specified by a qualified member of the Technical Staff and, prior to power operation, all reactor core configurations shall be examined by the Operations Division Technical Staff either by comparison with previously analyzed cores or by calculation and/or experiment to verify that the criterion set forth in Section 3.2 is satisfied.

Basis. Methods[3] have been developed to estimate the maximum heat flux in the reactor fuel elements. Moreover, data on the operating power level and maximum heat flux in a large number of representative core configurations operated in the past are available for comparison.

4.3 REACTOR CONTROL AND SAFETY SYSTEM

Applicability. These specifications apply to the surveillance requirements for the safety and measuring instrumentation and for the mechanical components of the control systems required for startup and operation.

Objective. The objective is to ensure the continued operability of the control and safety systems.

4.3.1 Surveillance of Reactor Instrumentation

Specifications

1. Testing

   a. The safety-level channels, the Log-N channel, and the fission chamber channel shall be tested:

      (1) prior to each startup following refueling or other changes in the reactor core configuration or a startup following an extended shutdown ($\geq$8 h), and

(2) prior to startup after any change or maintenance involving any component of these channels.

2. Calibration

a. The safety level, Log-N, and fission chamber channels shall be calibrated quarterly or following major maintenance.

b. The ΔT and primary-coolant flow channels shall be calibrated annually or following major maintenance.

c. The flapper-valve position channels shall be subjected to a functional test quarterly or following major maintenance.

d. The pool water level channels shall be calibrated annually or following major maintenance.

3. Channel checks

a. A channel check of those safety or protective channels required during operation shall be made prior to each startup following refueling or other changes in the reactor configuration or a startup following an extended shutdown ($\geq$8 h).

4.3.2 Surveillance of Mechanical Control System

Specifications

1. The shim-safety rods shall be visually and dimensionally inspected biennially.

2. The release time and flight time for each shim-safety rod shall be measured quarterly or any time that the shim-safety rod is moved to a new position. It shall also be measured following maintenance to a shim-safety rod or its related drive mechanisms.

3. The maximum rate of reactivity addition by the shim-safety rods shall be determined following a significant increase in total worth of the shim-safety rods, whenever there is a significant change in core configuration, or annually.

4. A measurement of the positive reactivity assigned to the servo-control system shall be made following a significant increase in the worth of the control rod as determined from shim-safety rod worth measurements or annually.

Bases. The specified channel checks, tests, and calibrations will provide assurance that the various measuring channels are operating properly with reliable and correct output signals. The time intervals between checks, tests, and calibrations are based on over many years of experience with similar instrumentation in the BSR and other ORNL reactors.

Many years of experience with the 1.5% natural boron-stainless steel shim-safety rods has revealed no significant corrosion, cracking, growth, or other changes. Movement of, or maintenance to, the shim-safety rod or related drive parts could result in changes which could produce improper release or scram times, thus necessitating a measurement following any such activity. In the absence of maintenance or changes, records show that release time and scram time change very little over a period of years.

Shim-safety rod calibration data have been gathered on a number of occasions during the lifetime of the rods, as well as following each significant change in core configuration. The data indicate that during the lifetime of a typical rod the composite reactivity worth of all the shim-safety rods experiences no significant change. Therefore, recalibration following significant changes in core configuration provides adequate information to predict the shutdown margin.

4.4 COOLING SYSTEM AND COOLANT LEAKS

4.4.1 Cooling System

Applicability. These specifications apply to the surveillance requirements of the primary coolant system.

Objective. The objective is to ensure the continuing integrity of the primary cooling system and to ensure compliance with the specifications set forth in Section 3.4.

Specifications.

1. When the reactor is operating, the height of the water above the reactor shall be checked once per shift.

2. The reactor shall not be operated without a syphon-break system. The syphon break system will be functionally tested at ten-year intervals.

3. The temperature of the pool water shall be checked:

   a. at intervals not to exceed three hours when operating in the natural convection mode or

   b. once per shift when operating in the forced convection mode.

4. The radioactivity of the water is continuously monitored by an instrument which provides an alarm in the control room. During

operation, an independent check of the radioactivity of the pool water shall be made daily.

5. All accessible parts of the reactor cooling system shall be subject to nondestructive examination for cracks and reduction in wall thickness in accordance with a repeating ten-year inspection plan. The plan sets forth the method of inspection to be used, the area, and the percentage and frequency of each area to be examined.

Bases. Because of the large volume of water (approximately 130,000 gal) in the reactor pool, it would require an extremely large leak to lower the water level rapidly (approximately 55 gpm/ft per hour). The forced convection system is a syphon device equipped with a syphon break so that rupture of the primary coolant line cannot drain the pool. Moreover, the pool water level is constantly monitored and displayed in the control room by the pool water-level channel (Section 3.3.1), hence the surveillance provided is adequate.

Even at 2 MW operation with no secondary cooling, the pool water temperature cannot rise at a rate in excess of 6°F per hour. Thus, the surveillance is adequate.

Continuous monitoring of the pool water radioactivity and the daily check are supplemented by the complement of radiation protection instruments in the building. Hence, the surveillance is adequate to fulfill the requirements of Section 3.4.

The nondestructive examinations performed during the past ten years of reactor operation have verified the adequacy of the components. Both the methods and frequencies of inspection, as specified in the in-service inspection plan, have been found to be adequate to detect incipient damage to the components.

4.4.2 Coolant Leaks

Applicability. This specification applies to the surveillance requirements necessary to evaluate leaks from the primary cooling system.

Objective. The objective is to ensure that leaks from the primary coolant system in excess of those specified in Section 3.5.2 are detected.

Specification. Based upon the surveillance requirements of Section 4.4.1, water losses from the primary coolant system shall be determined once per shift.

Basis. Recording the pool water level once per shift ensures that immediate attention will be given to water losses and will provide for the implementation of the requirements of Specification 3.4.2.

4.5 GASEOUS WASTE SYSTEM

Applicability. These specifications apply to the surveillance requirements for the gaseous waste systems.

Objective. The objective is to ensure proper operation of the systems and to meet specifications set forth in Section 3.5.

Specifications.

1. Normal Off-Gas System (NOG)

   a. When the NOG system is in operation:

      (1) checks shall be made prior to each startup to ensure that the system is operating normally, and

      (2) pressure drop across the filters shall be measured at least semiannually.

   b. The NOG system vacuum shall be under continuous surveillance of at least one instrumentation channel.

4.6 EMERGENCY SYSTEMS

4.6.1 Confinement System

Applicability. These specifications apply to the surveillance requirements for the confinement system.

Objective. The objective of these specifications is to ensure continued reliability of the confinement and the containment actuation systems.

Specifications.

1. The complete confinement closure system, including all instruments and indicators, shall be subjected to a thorough functional test quarterly.

2. A test to confirm in-leakage to the building in the containment mode shall be conducted quarterly.

3. The efficiency of the HEPA filters and charcoal absorbers in the building containment system shall be measured semiannually or after major maintenance or filter replacement.

4. A functional test of the system shall be conducted at least semi-annually and following major maintenance. This functional test

shall include automatic activation of the containment mode as specified in Section 3.6.1.4.

Bases. The safety limits, specified in Section 2, protect the fuel element cladding. Should a release of activity into the building occur, analysis has shown (see Ref. 2) that the confinement system is adequate to prevent a significant release of activity to the environment.

An analysis of the design and performance history indicates that the stated minimum frequencies of testing ensure that the system will be operable if needed.

4.7 RADIATION MONITORING EQUIPMENT

Applicability. These specifications apply to the surveillance requirements for the area radiation monitoring equipment.

Objective. The objective is to ensure that the radiation monitoring equipment is operable.

Specifications.

1. A calibration and a test of the radiation monitoring equipment for operability shall be conducted at least semiannually and after major maintenance.

2. The stack monitor shall be calibrated at least semiannually, or following maintenance or change involving components of the system.

Bases. Area radiation monitors are simple radiation detection devices, and experience has indicated that they operate quite reliably for long periods of time. Nevertheless, their failure at any time cannot be discounted or predicted. Therefore, all the radiation monitors in the plant-wide radiation detection system are continuously monitored electronically to detect the most common types of failures expected. These devices are primarily used to detect the presence of radiation where none had previously existed. Their alarm points are set at a small value above background and absolute accuracy is not essential. Hence, the semiannual test for operability combined with the continuous electronic monitoring is considered adequate to ensure that the equipment remains in proper operation.

4.8 LIMITATIONS ON EXPERIMENTS

Applicability. These specifications apply to the surveillance requirements for experiment limitations.

Objective. The objective is to ensure compliance with the specifications set forth in Section 3.7 regarding limitations on experiments.

Specifications.

1. The Technical Section of the Operations Division shall review and examine all experiments proposed for insertion into the reactor for safety, potential hazards, and compatibility with the operation of the reactor and other experiments.

2. New or unusual experiments shall be submitted with recommendations of the Technical Section to the Reactor Experiment Review Committee (RERC) for review in accordance with the provisions of Section 6.10 of these Technical Specifications.

3. Tried experiments may be approved by the Technical Section without further review by the RERC.

Bases. Review of experiments in accordance with the above specifications adequately ensures compliance with the specifications set forth in Section 3.7.

5. DESIGN FEATURES

5.1 REACTOR FUEL

Applicability. These specifications apply to the fuel elements used in the reactor core.

Objective. The objective is to ensure that the fuel elements are of such design and are fabricated in such a manner as to permit their use with a high degree of confidence in their reliability with respect to their physical and nuclear characteristics.

Specifications.

1. Reactor fuel elements

   a. The fuel elements for the reactor consist of "MTR" plate-type elements with cores containing uranium highly enriched in ^{235}U. The material requirements shall be as follows:

      (1) Fuel plates: U-Al alloy, U_3O_8-Al cermet, or U-Al_x intermetallic, Al cladding

      (2) ^{235}U enrichment: up to 97%

      (3) Number of plates per element: 10 to 25

      (4) Nominal weight of ^{235}U per element: $\leq$350 g .

   b. Fuel element design and core loading shall be correlated by experiment and calculation to ensure that the maximum operating heat flux will always be below the critical heat flux by a factor of at least 1.6 (see Ref. 3).

   c. All fuel elements fabricated for normal use as reactor fuel shall conform to the ORNL "Specifications for BSR Fuel Elements," and in addition, a quality assurance program equivalent to ANSI NQA-1 shall be developed and implemented by the supplier.

   d. Before awarding a contract to fabricate fuel elements, an ORNL site-inspection team shall evaluate the prospective vendor's operation and make a positive determination that the vendor is capable of producing elements which have an acceptably low probability of failure in the reactor.

2. Control-rod element

   a. The control-rod elements consist of a partial fuel element containing MTR type fuel plates similar to those used in the fuel elements with a ... ..t aluminum housing and guide for the control rod. The materi l requirements shall be as follows:

(1) Fuel plates: U-Al alloy, U_3O_8-Al cermet, or U-Al_x intermetallic, Al cladding

(2) ^{235}U enrichment: $\leq$97%

(3) Number of plates: 8 to 20

(4) Nominal weight of ^{235}U per element: $\leq$250 g .

b. Same as Section 5.1, Specification "1.b."

c. Same as Section 5.1, Specification "1.c" except "Specifications for BSR Control-Rod Elements."

d. Same as Section 5.1, Specification "1.d."

Basis. Calculations during the design and experience during operation of the reactor indicate that fuel control-rod elements and control rods having these characteristics meet the criteria for reliability and performance.

5.2 REACTOR BUILDING

Applicability. This specification applies to the building that houses the reactor.

Objective. The objective is to ensure that provisions are made to restrict the amount of airborne radioactivity released to the atmosphere.

Specification. The reactor is housed in a building designed to be maintained at negative pressure with respect to the outside ambient pressure.

Basis. The confinement system is designed to exhaust air from the reactor building at a rate to ensure leakage into the building and to maintain a slight negative pressure within the building with respect to outside ambient pressure. The air exhausted from the reactor building when in the containment mode passes through a series of filters described in Section 3.5.1 before being discharged through a stack to the atmosphere. The confinement system is in operation during all modes of reactor operation and during in-pool movement of irradiated fuel.

5.3 FUEL STORAGE AND HANDLING

Applicability. These specifications apply to storage facilities for new and irradiated fuel.

Objective. The objective is to store fuel in such a manner as to ensure subcritical conditions and ensure adequate cooling and shielding for irradiated fuel.

Specifications.

1. For storage and handling outside the reactor

   a. Fuel elements shall be stored in storage facilities that have been reviewed and approved by ORNL's Criticality Committee. The criterion used is that k_{eff} shall not exceed 0.95.

   b. The transfer of fuel elements between approved storage facilities shall be only by authorization of the Reactor Supervisor or his authorized representative.

   c. Irradiated fuel shall be stored in a manner which will provide natural convection cooling and adequate shielding.

2. For handling within the reactor:

   a. All transfers shall be authorized by the Reactor Supervisor or his authorized representative.

   b. All transfers shall be by written request which provides for identification of all fuel elements handled.

Bases. All fuel units used in the BSR are fabricated according to specifications that have been developed to ensure dimensional control to guarantee that the elements will fit into a fixed, nonadjustable grid, to ensure that the fuel content is within rigid limits, to ensure that the cladding will safely contain fission products for the life of the elements, and to guarantee structural integrity of the assembled elements.

Stored fuel elements shall be maintained in a geometry that will prevent criticality. The methods of preventing and/or handling criticality hazards outside of the reactor core must be approved by the ORNL Criticality Committee, and this approval is documented in completed Nuclear Safety Requests. Critical experiments have been performed and these data are used in evaluating requests.

The handling of all fuel elements to, from, and/or within the reactor shall result in approved core configurations. These core configurations are based on data from boiling experiments, previous fuel loadings, and empirical calculations. All fuel loadings are prepared by technical personnel based on these evaluations.

6. ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

The Oak Ridge National Laboratory, which is owned by the Department of Energy and operated under contract by Martin Marietta Energy Systems, Inc., shall be responsible for operation and supervision of the facility. The Operations Division shall be directly responsible for the operation of the facility. The relationship of the reactor staff to the Laboratory's organization structure is shown in Fig. 6.1.

6.2 PERSONNEL QUALIFICATION

1. The reactor shall be operated by personnel examined and certified under the general provisions of DOE Order 5480.1A, Chapter VI, "Safety of Department of Energy Owned Reactors," and approved by the Operations Division Director.

2. Jobs requiring an operator or a senior operator certification are indicated in Fig. 6.1 (organization chart).

6.3 STAFF REQUIREMENTS

1. The minimum personnel on duty during reactor operation shall be:

   a. one certified senior operator and

   b. one certified operator.

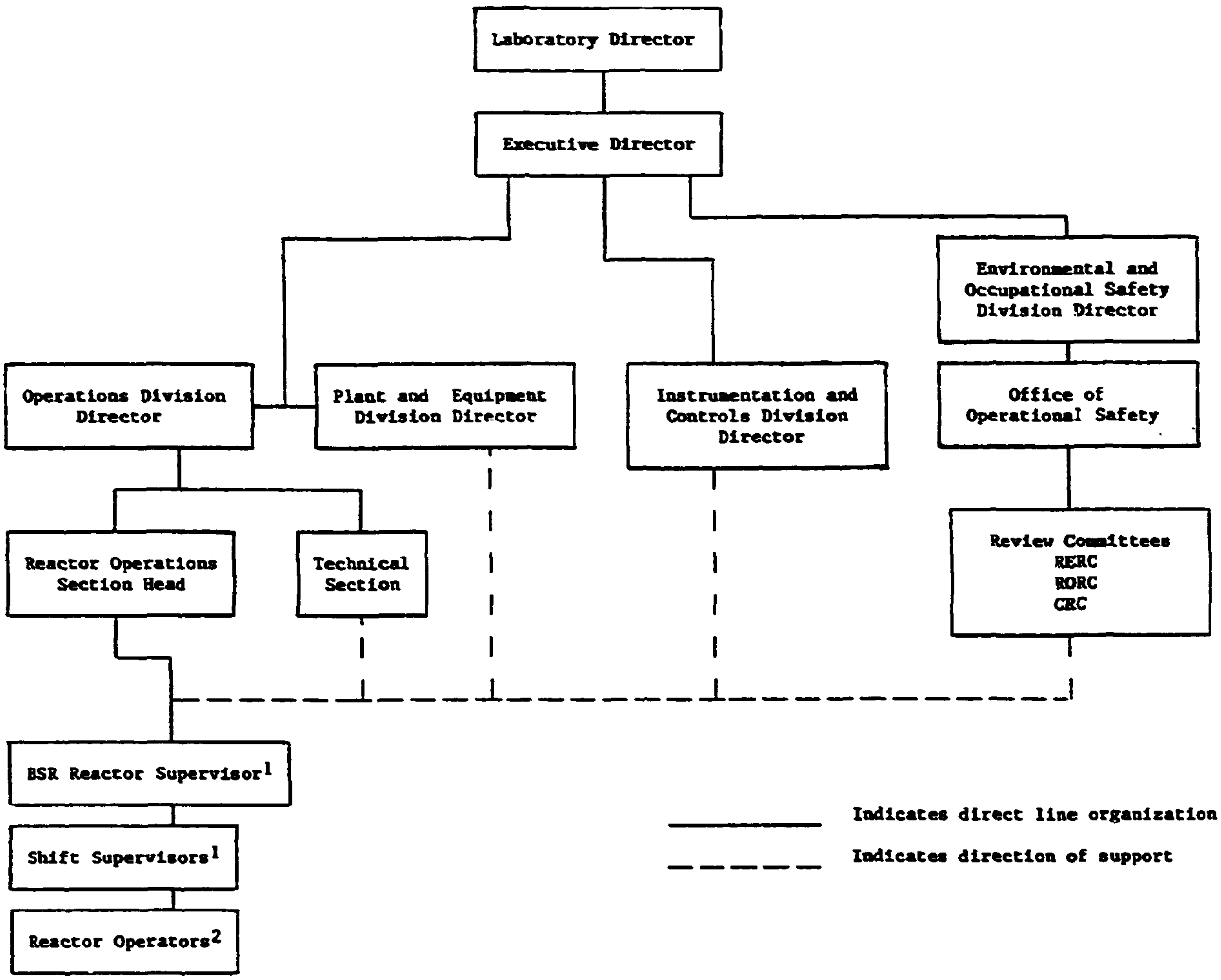
2. The control room or the remote control panel shall be attended by at least one certified operator at all times during reactor operation.

3. At least one certified senior operator shall be present in the control room during reactor startup.

6.4 FACILITY MODIFICATIONS

1. It shall be the responsibility of the Division Director to ensure that changes in technical specifications or modifications to the plant protection system, reactivity control systems, or engineered safety features, or changes that involve a safety question not reviewed in the Safety Analysis Report shall receive prior review and authorization by the RORC and DOE.

2. Certain mechanical and instrumentation and control design changes may be made by the laboratory without RORC and DOE approval provided the effect of the change does not involve a change in the technical



[1]Shall be certified senior reactor operator.
[2]Shall be certified reactor operator.

Fig. 6.1. Organization chart.

specifications or an unreviewed safety question. Formal procedures shall be followed for documenting important mechanical and instrumentation and control design changes.

6.5 OPERATING PROCEDURES

1. The reactor shall be operated in accordance with documented operating procedures. In no instance shall the operating procedures designate authorization to operate the reactor outside the bounds of any specifications listed in this document. The procedures shall be adequate to ensure safe operation of the reactor but should not preclude the use of independent judgement and action should the situation require such. Detailed written procedures shall be provided for, but not limited to, the following:

   a. emergency and abnormal conditions including evacuations;

   b. reactor startup, operation, and shutdown;

   c. installation and removal of fuel elements, control rods, and all non-permanent reactor components; and

   d. routine maintenance of control rod drives and reactor safety systems.

2. A standard method shall be used to change operating procedures as necessary to ensure that all persons concerned are notified of the change and that a permanent record is made. Permanent procedure changes must be formally written and approved by at least two of the following senior staff members:

   a. Operations Division Director,

   b. Reactor Operations Section Head,

   c. Reactor Operations Department Head,

   d. BSR Supervisor,

   Temporary changes that do not alter the original intent of a procedure shall be made, when required, by issuing special operating instructions. Such special operating instructions shall be approved by at least two of the senior staff members listed below:

   a. Operations Division Director,

   b. Reactor Operations Section Head,

   c. Reactor Operations Department Head,

d. BSR Supervisor,

e. Day Supervisor

3. Radiation control procedures shall be maintained and made available to all Operations personnel.

6.6 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a safety limit is exceeded, the following actions shall be taken:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by DOE.

2. An immediate report shall be made to the Laboratory Executive Director and the Office of Operational Safety.

3. A written report shall be made to DOE no later than the next work day.

4. A written report shall be made which shall include an analysis of the causes and the extent of possible resultant damage, effectiveness of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be sent to the Reactor Operations Review Committee (RORC) and a similar report submitted to DOE when authorization to resume operation of the reactor is sought.

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

In the event of an abnormal occurrence, as defined in Section 1, the following actions shall be taken:

1. The Reactor Supervisor and other appropriate management personnel shall be notified and corrective action taken.

2. A report shall be made that shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence in accordance with DOE Order 5000.3, "Unusual Occurrence Reporting System."

3. A report shall be submitted to DOE per Section 6.8, Reporting Requirements.

6.8 REPORTING REQUIREMENTS

1. A notification shall be made no later than the next work day to the Environment, Safety and Health Division, and to the Contracting Officer Technical Representative (COTR), DOE, Oak Ridge Operations, of the following conditions:

    a. any release of radioactivity to the environment above the permissible limits specified in DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection,"

    b. any violation of a safety limit (Section 2.1) (Note: An immediate report shall be made to the Laboratory Executive Director and the Office of Operational Safety.),

    c. any exposure to personnel in controlled and uncontrolled areas that exceed the standards in DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection," and

    d. any violation of technical specifications.

2. A written report shall be made within ten work days to DOE-ORO for any abnormal occurrence as defined in Section 1.

6.9 OPERATING RECORDS

In addition to the requirements of applicable regulations, and in no way substituting therefor, records and logs shall be prepared of at least the following items and retained for a period of at least five years:

1. normal plant operation;

2. principal maintenance activities;

3. abnormal occurrences;

4. equipment and component surveillance activities required by the Technical Specifications;

5. fuel inventories and transfers;

6. experiments performed with the reactor; and

7. updated, corrected, and as-built drawings of the facility (these shall be retained for the lifetime of the facility).

6.10 REVIEW COMMITTEES

6.10.1 Reactor Operations Review Committee

There shall be a Reactor Operations Review Committee (RORC) responsible for periodically conducting an independent safety review of the reactor facility. The members of the RORC shall be appointed by the Director of the Laboratory and shall not be directly involved in the operation of the reactor. The committee members shall collectively possess expertise in all areas of reactor operations and safety.

The RORC shall meet with the operating personnel as frequently as it deems necessary to keep informed of any operational problems or potential hazards. The committee shall conduct at least one formal review annually, and the minutes of this review shall be reported in writing to the Director of the Laboratory. In compliance with the requirements of DOE Order 5480.1A, the RORC shall review any proposed modifications that have safety significance. A detailed description of the committee's functions and method of review is presented in Ref. 7.

6.10.2 Reactor Experiment Review Committee

There shall be a Reactor Experiment Review Committee (RERC) responsible for reviewing any new or unusual experiments proposed for insertion into the reactor. The RERC shall review on request proposed experiments in such detail as to ensure that no credible failure or malfunction of the experiment could create a positive change in reactivity greater than the reactor safety system was designed to accommodate. The committee shall review experiments from the standpoint of personnel and equipment safety and the continuity of reactor operations. The committee shall, as it deems necessary, place limits upon any material, systems, components, effluents, or operations that may present a hazard to personnel or to the reactor. The committee shall, in executing its responsibility, make recommendations or establish conditions on design, construction, and operation of an experiment. A detailed description of the committee's functions and method of review is presented in Ref. 8.

6.10.3 Criticality Committee

There shall be a Criticality Committee responsible for the review of operations which involve handling, storage, transportation, and disposal of significant quantities of fissile material. The committee shall, on request, serve as a consulting group and provide assistance in problems involving criticality. The committee shall conduct an annual review of all areas containing significant amounts of fissile material to ensure that approved procedures are being followed. A detailed description of the committee's functions and method of review is presented in Ref. 9.

7. MONITORING OF AIRBORNE EFFLUENTS

Normally, five different radiation monitors, operable during reactor operation, provide the operator with information related to potential or actual release of radioactive effluents. These are:

1. the continuous air monitors,
2. the beta-gamma monitrons,
3. a high gamma monitor,
4. a decay tank off gas beta-gamma monitor, and
5. a beta-gamma monitor in the cell ventilation duct.

Any one of these is sufficient to provide the operator with information related to potential or actual release of radioactive effluents. An alarm is actuated in each of these five instrument channels by radiation levels exceeding a level no greater than ten times "current normal" levels. In the event an alarm is received on any of these channels, an immediate investigation shall be initiated by Reactor Operations supervision to ascertain the cause and to determine the appropriate necessary remedial actions. In addition, the output signals from the continuous air monitors are continuously recorded.

8. REFERENCES

1. W. R. Gambill, and R. D. Bundy, **Burnout Heat Fluxes for Low-Pressure Water in Natural Circulation**, ORNL-3026 (December 1960).

2. L. E. Stanford, T. P. Hamrick, and F. T. Binford, **Description and Safety Analysis of Significant Change of the BSR for 2-MW Operation**, ORNL/TM-2231 (May 1968).

3. F. T. Binford, "Burnout Conditions in BSR Cores Under Forced Convection Flow," Supplement No. 1 to **Description and Safety Analysis of Significant Change of the BSR for 2-MW Operation**, ORNL/TM-2231 (March 1968).

4. W. R. Gambill, **Design Curves for Burnout Heat Flux in Forced-Convection Subcooled Light Water Systems**, ORNL/TM-2421 (November 1968).

5. E. P. Blizard, **Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960**, ORNL-3016, pp. 18-26.

6. Internal Correspondence from F. T. Binford to J. A. Cox, "Significance of Operating with a Leak in the ORR Primary Coolant System" (September 4, 1973).

7. Internal Correspondence from G. H. Burger to Distribution, "Revised Charter for the ORNL Reactor Operations Review Committee" (May 3, 1985).

8. Internal correspondence from G. H. Burger to Distribution, "Revised Charter for the ORNL Reactor Experiment Review Committee" (May 3, 1985).

9. Internal correspondence from G. H. Burger to Distribution, "Revised Charter for the ORNL Criticality Committee" (May 3, 1985).

10. F. T. Binford, **The Oak Ridge Research Reactor - Safety Analysis**, ORNL-4169, Vol. II (March 1968).

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